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A Study of Air-Operated Valves in U.S. Nuclear Power Plants

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A Study of Air-Operated Valves in U.S. Nuclear Power Plants

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ABSTRACT

A study of air-operated valves in nuclear power plant applications was conducted for the NRC Office of Research (the project was initiated by NRC/AEOD). The results of the study were based on visits to seven nuclear power plant sites, literature studies, and examinations of event records in databases available to the NRC. The purpose is to provide information to the NRC staff concerning capabilities and performance of air-operated valves (AOVs).

Descriptions of air systems and AOVs were studied along with the support systems and equipment. Systems and equipment that contain AOVs and SOVs were studied to determine their dependencies. Applications of AOVs and SOVs were listed along with current NRC requirements.

Observations and conclusions included:

- Licensees may not know if the design basis loads or environmental conditions can be met with acceptable margins for the AOVs in their plants. AOVs may have reduced operating margins caused by such factors as aging, load mechanisms not understood or considered in the original design, or previously contaminated air. Calculations or valve descriptions were found that included mistakes or inaccurate information.

- Air systems and solenoid-operated valves (SOVs) have been and continue to be sources of common-cause failures of AOVs. Accumulators are potential sources of AOV failures. Air-operated dampers have been and continue to be potential sources of safety system failures.

- Accident sequence precursor (ASP) analyses performed by the NRC indicated that there have been a number of risk significant events involving AOVs.

- Generic probabilistic risk assessments performed by INEEL indicated that changes in the failure probabilities of AOVs and SOVs can result in proportional changes in system unreliability and that AOVs and SOVs can have an important role in system reliability. It was also noted that some plants were using low generic probability values to estimate AOV failure probabilities in their PRA calculations.

Examples were observed during the plant visits of events and conditions, including common-cause events and conditions, involving AOVs and SOVs which were under-reported or not reported to the NRC. This may have resulted in computing lower estimates of the risk and safety significance associated with AOV failures.

- Many of the licensees' individualized plant reviews for Maintenance Rule evaluations and for the plants' AOV programs included reviews of PRA findings by expert panels. The licensees found that by using their plants' operating experience and plant-specific probabilistic risk

assessments, certain AOVs had high risk importance, high risk achievement worth, or were important to preventing large early releases.

- The nuclear industry and several licensees are preparing or have prepared AOV program plans which include design basis reviews, margin calculations, and use of diagnostic systems to ensure the operability of AOVs.

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EXECUTIVE SUMMARY

The purpose of this study was to provide information to the NRC staff concerning air-operated valve (AOV) capabilities and performance in nuclear power plants, in order to help them determine if and how the NRC needs to focus additional attention on the design, qualification, testing and/or maintenance of AOVs. One of the primary objectives of this study was to determine if AOVs can perform their designed functions.

Seven nuclear plant sites were visited, a literature search was conducted, and the NRC's databases were consulted to identify pertinent events and information. Descriptions of air systems and AOVs were studied along with the support systems and equipment. Systems and equipment that contain AOVs and solenoid-operated valves (SOVs) were studied to determine their dependencies. Applications of AOVs and SOVs were listed along with current NRC requirements.

There are several hundred to over a thousand AOVs in each plant and typically over 2000 SOVs.

Events involving AOVs and reported conditions of AOVs indicate that many AOVs have reduced operating margins caused by such factors as aging, load mechanisms not understood or considered in the original design, and common-cause failures caused by currently or previously contaminated air or solenoid-operated valve defects. Licensees may not know that the design basis loads or environmental conditions can be met, with acceptable margins, for some of the AOVs in their plants.

Licensees are relying on either manufacturers' information and calculations or, more recently, design reviews of the AOV design bases. Several manufacturers and nuclear plant architects/engineers' AOV calculations or valve descriptions were found that included mistakes or inaccurate information. The knowledge on the part of licensees of the design bases and margins in AOVs to meet the design basis demands is a matter of safety significance.

Air systems are a source of common-cause failures of AOVs. A reliable supply of clean, dry, oil-free air, at specified pressure is essential to proper function of AOVs and SOVs. Licensees must know that the air supply is reliable at all times in order to justify the assumptions that AOVs will move to or remain in their fail-safe positions.

Accumulators that supply air to safety-related and important non-safety related AOVs can be sources of contamination, reduced capacity, and subsequent AOV inoperability unless it is verified that the accumulators are of sufficient capacity, are free of contamination, do not contain trapped water, and the accumulator check valves are functioning. Licensees must know that the accumulators are capable of functioning properly in order to justify the assumptions that AOVs will move or remain in their fail-safe positions.

Air-operated dampers have been and can be a potential source of failure of emergency diesel generators, control room ventilation systems and other safety-related systems that they may serve.

SOVs are a source of common-cause failure of AOVs. Root causes of AOV failure include potential contamination from the air system, design, qualification, and maintenance. These causes have been reported previously and need to be systematically addressed by licensees based on the safety status and risk significance of the SOVs.

Generic probabilistic risk assessments performed by the INEEL indicated that changes in the failure probabilities of AOVs and SOVs can result in proportional changes in system unreliability and that AOVs and SOVs can have an important role in system reliability. It was also noted that some plants were using low generic probability values to estimate AOV failure probabilities in their PRA calculations. Individual AOVs may be found to be of low risk significance but the common-cause failure (CCF) of two or more AOVs performing the same function may have considerable risk significance.

Many of the licensees' individualized plant reviews for Maintenance Rule evaluations and for the plants' AOV programs included reviews of PRA findings by expert panels. The licensees found that by using their plants' operating experience and plant-specific probabilistic risk assessments, certain AOVs had high risk importance, high risk achievement worth, or were important to preventing large early releases.

Among other observations, the site visits and reviews of events revealed that at some plants common-cause AOV failures, degradations, or precursors were under-reported or not reported to the NRC. This may have resulted in lower estimates of the risk and safety significance of AOVs in several studies. The under-reporting of common-cause AOV events could lead to lower computed beta factors and lower estimates of core damage frequency and large early releases.

The number and scope of NRC generic communications and studies of AOV and air systems provide an important indicator of the overall safety significance of these components and systems. Over 100 NRC generic communications related to AOV events and problems were identified. Accident Sequence Precursor (ASP) analyses included a number of risk significant events involving AOVs.

The nuclear industry and several licensees are preparing or have prepared AOV program plans, including testing, that are in varying stages of development. The plans are not, as yet, fully implemented. Diagnostic systems for AOVs and SOVs are available but manufacturers' claims regarding accuracy and attributes have yet to be verified.

Tables of pertinent events to support the conclusions are included, along with trip reports from the visits to the sites.

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A Study of Air-Operated Valves in U.S. Nuclear Power Plants

1. INTRODUCTION AND PURPOSE

The NRC, with assistance from the INEEL, studied the design, qualification, operation, maintenance, and testing of air-operated valves (AOVs) in commercial nuclear power plants. The study was conducted in accordance with the Program Plan dated October 22, 1997, (INEEL Letter to H. Ornstein, NRC, from J. Bryce, October 23, 1997, Job Code E8238, Task

Order 15 - JHB-167-97 and INEEL Letter from T. Ryan to H. Ornstein, NRC, Transmittal of Revised Cost Estimate, Schedule, and Spending Plan for Revision 2 of JCN 8328, Task Order No. 15, Investigation of Air-Operated Valves, March 25, 1999, RYAN-55-99).

Seven nuclear plant sites were visited as part of this study.

The purpose of this study was to provide information to the NRC staff concerning AOV capabilities and performance in nuclear power plants. The goal was to help the NRC determine if and how the NRC needs to focus additional attention on the design, qualification, testing (initial and in-service), and/or maintenance of AOVs in order to reduce plant vulnerabilities associated with individual or common-cause failures. Design, qualification, applications, maintenance, and testing of AOVs were studied in relation to their safety significance in nuclear power plant applications.

2. BACKGROUND

AOVs are used in all U.S. nuclear power plants. The population of AOVs in each plant varies widely. The number of AOVs per plant can be over a thousand and the number of safety-related AOVs per plant can be several hundred. Many plants have large numbers of "important" AOVs that are not necessarily designated "safety-related." Solenoid-operated valves (SOVs) in most nuclear power plants number in the thousands.

"Important" AOVs are those that could interfere with the function of safety-related equipment or can cause scrams or trips, or, if they should fail may result in release of radioactive contamination. For example, such a failure was part of the initial sequence of the TMI-2 event. "The loss of the main feedwater pumps which initiated the turbine trip followed by a reactor trip, has been attributed to the presence of water in the instrument air system that caused the condensate polisher air outlet valves to close. It is postulated that water at 100 psig in the condensate polisher entered the service air system, which is at 80-100 psig, through a failed open check valve." (TMI Report, Vol. II, Part 2, pp. 470-471)

The NRC and the nuclear industry have been interested in the safety significance and performance of AOVs for many years and have conducted a number of studies of various applications of these components and the air systems that provide motive power. Studies by both the NRC and industry are referenced herein. In addition, a list of generic communications was compiled (Appendix A) related to AOVs, SOVs, or air systems. There are undoubtedly other pertinent studies or references that could also have been included.

The NRC's Office for the Analysis and Evaluation of Operational Data (AEOD) published two case studies, NUREG-1275, Volume 2 on air system problems and NUREG-1275, Volume 6, on SOV problems. These studies did not focus directly on AOVs but they contained relevant information on two of the dominant contributors to AOV failures and demonstrated plant vulnerability to common-cause failures of AOVs.

Relatively recent experience in the nuclear industry with motor-operated valves (MOVs), specifically the focus on determining design basis demands vs. capabilities (margins) has led to an increased awareness by all concerned of the operational requirements of AOVs. Various groups of licensees, nuclear industry organizations, and vendors are currently involved in efforts to ensure the reliable performance of AOVs in nuclear power plants. These efforts include pilot programs and individual plant initiatives concerning diagnostic testing and design verification. The NRC has not, as yet, requested that licensees construct programs for AOVs similar to the efforts described in Generic Letter 89-10 and its supplements with respect to motor-operated valves.

3. OBJECTIVES

One of the primary objectives of this study was to determine if safety-related air-operated valves are designed, qualified, installed, maintained, and tested so that there is a reasonable assurance that they can perform their design basis functions. Important non-safety-related AOVs (see the SCOPE section) were also studied to determine if they could function so as not to cause or compound events that might compromise safety.

Sources of common-cause failures of AOVs, such as the air system and SOVs, were studied because of their potential impact on safety.

Operating experience over the last 14 years (plus) was reviewed to help the NRC draw conclusions about the effectiveness of current regulations and guidance on AOVs. The safety significance of the operating experience was evaluated.

Regulatory requirements regarding AOVs were reviewed and assessed against the operating experience.

Domestic activities in the nuclear industry regarding AOVs, including development of AOV programs and use of diagnostic testing, were studied in an effort to assess their effectiveness in ensuring AOV operability.

Three common assumptions regarding AOV operability were critically reviewed as part of this study. These assumptions are that AOVs will "fail-safe," that providing clean, dry, oil-free air, at proper pressure, is all that is necessary to ensure the satisfactory performance of AOVs, and that the American Society of Mechanical Engineers (ASME) stroke-timing test (the test currently mandated by the NRC for safety-related AOVs) provides reasonable assurance of the operability of AOVs without augmentation. Results of the review are included in this study. The study results indicated that the "fail-safe" assumption that many AOVs are designed for (i.e., to move to a specified position in the event of a loss of air) depends on proper function of valves, valve operators, and components associated with AOVs or in proximity to them, and is not valid unless verified. Also, a clean, dry, oil-free air (or nitrogen) supply at proper pressure is necessary for the proper function of AOVs; but it is not sufficient, by itself, to ensure the acceptable function of AOVs. Finally, the ASME stroke-timing test is generally performed as a no-load test that does not provide sufficient information regarding future operability of AOVs used in nuclear power plant applications.

4. SCOPE

Safety-related and important non-safety-related AOVs (including directly attached components such as operators, solenoids, springs, and diaphragms) were included in this study. AOVs are defined as valves that use air or inert gas as the motive power source to change the position of valve, valve operator, or a component of the valve. Other parts or mechanisms such as pneumatic regulators, pneumatic controllers, pneumatic boosters, pneumatic reducers, or solenoids are considered part of the valve/operator if they serve a particular valve, even if they are not directly within its casing or mounted on it.

Solenoid-operated valves (SOVs) may be piece parts of AOVs, serve as pilots for AOVs, or in some cases, directly act as valves. SOVs are considered to be air-operated valves in this study even though the device that actuates an SOV is an electric powered solenoid. SOVs that are used as pilots or piece parts of AOVs are closely tied to the operation of the AOVs and use the same air or nitrogen that operates AOVs as their process fluid. The process fluid is very closely related to the operation of an SOV and in some cases provides some of the force to actuate or control the motion of an SOV. SOV operability depends so heavily on the quality of the process fluid (air or nitrogen) that classifying SOVs as a category of AOVs is reasonable. The phrase, "quality of the process fluid (air or nitrogen)" refers to the necessity that air or nitrogen be clean, dry, oil free, and delivered at pressures within the range specified for the SOV.

Safety-related AOVs are valves and operators that:

- Must remain functional during and following design basis events;
- Ensure the integrity of the reactor coolant pressure boundary;
- Ensure the capability to shut down the reactor and maintain it in a safe-shutdown condition; or
- Ensure the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR 100 guidelines.

Important non-safety-related AOVs valves and operators are:

- Relied upon to mitigate accidents or transients;
- Used in plant emergency operating procedures (EOPs);
- Those whose failure could prevent safety-related structures, systems, and components (SSCs) from fulfilling their safety-related functions; or
- Those whose failure could cause a reactor scram (trip) or actuation of a safety-related system.

The above categorizations are in accordance with the information in 10 CFR 50.2, 50.49, and 50.65.

Safety-related and important non-safety-related AOVs are expected to perform their intended function under design-basis conditions, normal operating conditions, and, in some cases, shutdown conditions.

Design, qualification, maintenance, and testing of AOVs were studied in nuclear power plant applications.

Air or inert gas supply systems and components were included in this study because they are a source of common-cause failures in the equipment served. The air or inert gas operating fluid might be supplied from the plant's pneumatic systems such as instrument air/service air systems, accumulators, receivers, emergency tanks, or from the process fluid.

Electric power, as well as air, is usually required for AOVs in order to provide power for control components such as limit switches and SOVs associated with the AOVs. The electrical supply system for AOVs was not included in this study. However, electrical components that serve AOVs directly, such as limit switches and SOVs were noted in some of the events.

In addition to a review of the NRC's Licensee Event Report (LER) database, searches for sources of other pertinent AOV related events were conducted in other available industry and NRC databases. Final Safety Analysis Report (FSAR) reviews were conducted for the seven plants visited in order to learn about the individual air systems and plant characteristics.

PLANTS VISITED FOR AOV STUDY

Trip Number and Plant Name	Dates Visited	Plant Description / A&E / Year Commercial Operations Started
1. Palo Verde 1, 2, and 3	October 28–29, 1997	Combustion Engineering, 2 loop, System 80 (no PORVs) PWR/Bechtel/1986
2. Fermi 2	November 3–4, 1997	General Electric BWR 4/Detroit Edison/1988
3. Palisades	November 18–19, 1997	Combustion Engineering, 2 loop PWR/Bechtel/1971
4. LaSalle 1 and 2	December 17–18, 1997	General Electric BWR 5/Sargent & Lundy/1984
5. Three Mile Island 1	February 12–13, 1998	Babcock and Wilcox, lowered loop PWR/Gilbert/1974
6. Indian Point 3	March 10–11, 1998	Westinghouse, 4 loop PWR/United Engineers & Constructors/1976
7. Turkey Point 3 and 4	March 24–25, 1998	Westinghouse, 3 loop PWR/Bechtel / 1972

A trip report was prepared after each visit and these are included in Appendix C.

This report includes consideration of public comments received on the draft dated April 26, 1999.

5. DESCRIPTION OF AIR SYSTEMS AND AOVS

The purpose of air and/or nitrogen systems that supply power to AOVs is to provide clean, dry, oil-free air and/or nitrogen at a specified pressure. Generally, the instrument air (IA) system and its backups perform this task. Air systems in nuclear plants are unique to each plant or site and no two sites have the same system. Air systems, in general, are designed and maintained as non-safety-related systems, although portions of air systems and separate air systems have been designated as safety-related in some plants. General descriptions of IA (and backup nitrogen) systems are to be found in the introductory parts of:

- NUREG-1275, Volume 2, "Operating Experience Feedback Report - Air Systems Problems;"
- NUREG/CR-2796, "Compressed-Air and Backup Nitrogen Systems in Nuclear Power Plants;" and
- NUREG/CR-5419, "Aging Assessment of Instrument Air Systems in Nuclear Power Plants."

A comprehensive and detailed overview of nuclear power plant IA systems and their major components was found in the Instrument Air Systems Guide for Power Plant Maintenance Personnel, NMAC NP-7079. Descriptive remarks concerning the particular air/nitrogen systems, including IA systems, encountered in the visits to plant sites are included in the trip reports in Appendix C.

Air-operated valves can range from "simple" diaphragm actuators to complicated, solenoid-piloted designs. The design options for AOVs and air-operators are virtually unlimited. Conceptually, some AOVs may appear to be straightforward devices but the details of design, operation and maintenance indicate that even the "simple" diaphragm AOV is a complex device. Those complexities include materials and material interactions, tolerances, and an almost unlimited variety of design details. Further, if the additional complexities of support systems are considered, the challenge of ensuring reliable AOV operation is considerable. A number of vendors have provided AOVs for nuclear power plant applications, including Fisher, Copes-Vulcan, WKM, Anchor Darling, and Target Rock, to name a few. Fermi 2, for (perhaps an uncommon) example, has AOVs from approximately 38 manufacturers.

The following references provide general descriptions of AOVs and their associated control mechanisms:

- NUREG/CR-6016, "Aging and Service Wear of Air-Operated Valves Used in Safety-Related Systems at Nuclear Power Plants;"
- Electric Power Research Institute (EPRI) report EPRI NP-7412, "Maintenance Guide for Air-Operated Valves, Pneumatic Actuators, and Accessories;" and
- EPRI NP-7412, Revision 1, "Air-Operated Valve Maintenance Guide."

A description and discussion of the operation of main-steam isolation valves (MSIVs) are included in NUREG/CR-6246, "Effects of Aging and Service Wear on Main Steam Isolation Valves and Valve Operators," and power-operated relief valves (PORVs) are described in NUREG/CR-4692, "Operating Experience Review of Failures of Power Operated Relief Valves and Block Valves in Nuclear Power Plants." A detailed discussion of valve and valve trim choices is included in the "ISA Handbook of Control Valves."

Globe, ball, and butterfly valves are commonly used in AOV applications. Air-operated gate valves are less common but some are to be found in nuclear power plants. The two most common AOV actuator types are spring-and-diaphragm and piston. Valve and actuator assemblies can be designed for linear motion or for rotary valve motion using scotch yoke or rack-and-pinion arrangements. AOVs can also be classified as control valves or isolation valves. An AOV "assembly" might be considered to include a dedicated air regulator, positioner, pressure controller, dedicated accumulator(s), accumulator check valves, and dedicated SOVs used as pilots for the valve operator.

Globe AOVs can be cage guided, post guided, stem guided, or skirt guided. The valve plugs can be balanced or unbalanced designs. A variety of packing materials and packing configurations are available, each with its own attributes, advantages and disadvantages.

Air regulators are devices used to provide uniform air pressure. An AOV positioner is a proportional controller that adjusts the air output to the actuator based on control system input and feedback from the valve stem position. Positioners can be pneumatic or electro-pneumatic; they are used to automatically adjust the flow through AOVs in response to changes in the characteristics of the valve or fluid system. Pressure boosters or volume boosters are used to increase the pressure or volume from the common air supply to meet specific needs of an AOV. Pneumatic transducers are used in AOVs to convert current or voltage from the control system into pressure. Accumulators are used as backup pneumatic supplies for safety-related AOVs in order to avoid direct reliance on non-safety-related air systems and/or when multiple repositioning of a valve is required without reliance on common air systems. Accumulator check valves are installed to prevent the air from bleeding out of an accumulator.

SOVs can be used as pilots to control the motion of larger AOVs, or they can be designed to operate equipment, for example dampers, directly. Conceptually, solenoid-operated valves may appear to be simple devices consisting of a coil that, when energized, causes the linear motion of a magnet in order to activate a valve (usually small) to direct a fluid (usually air or nitrogen). In practice, because of the number of SOVs, their use in many systems throughout nuclear power plants, and the number of design, operation and maintenance conditions that must be satisfied in order to ensure their successful operation, establishing and maintaining SOV operability represents one of the more complex and challenging aspects of nuclear plant operation. The considerable complexities regarding SOV operability arise in the details of design, fabrication, installation, maintenance, and testing, all of which involve satisfying numerous environmental conditions and operating requirements in order to ensure successful operation of each of the thousands of SOVs in each plant.

The operation and performance of SOVs have been the subject of a number of studies by the NRC and the nuclear power industry. Detailed technical descriptions of SOVs, including their design, application, and maintenance, are to be found in the "Solenoid Valve Maintenance and Application Guide," NMAC NP-7414, and also in NUREG/CR-4819, Volume 1, "Aging and Service Wear of SOVs in Safety Systems of Nuclear Power Plants." The uses and description of SOVs in nuclear power plant applications are also described in the beginning of NUREG- 1275, Volume 6, "Operating Experience Feedback Report - Solenoid-Operated Valve Problems."

Characteristics of SOVs pertinent to this study, and which raise the likelihood for common-cause AOV failures, include:

- Exceeding the designed allowable maximum operating pressure differential (MOPD), which is the pressure difference between the inlet port and the outlet port, may cause the SOV to spuriously open or not open, depending on the SOV design, and will usually result in future unreliable operation or failure.
- Small orifices make the SOVs subject to interference from small contamination particles or moisture intrusion.
- The design logic of SOV operation and control can be complex and this is a source of design or installation mistakes.
- Many piloted SOVs require a minimum operating pressure differential (Min OPD) to operate properly.

- Many SOVs will not function properly when subject to reverse pressurization or flow in the valve.
- Materials used to construct the valve bodies, internal parts, seals, O-rings, etc. may be subject to binding, creep, corrosion, erosion, adverse material interactions, and/or environmental deterioration.
- Rather small forces, in the range of 10 pounds or less, are produced by a solenoid to operate an SOV; therefore, opposing forces of similar small magnitude can interfere with successful operation.
- SOVs can be damaged if subjected to higher-than-designed pressure in the operating fluid.
- SOVs are subject to damage from the use of thread locking compounds in adjacent pipe or tube connections that tend to migrate into the working parts of the SOVs.
- Incompatible, incorrect, or unapproved elastomers and lubricants used in SOVs have been, and continue to be, a source of many SOV operational problems.

According to section 4.2 of the "Solenoid Valve Maintenance and Application Guide," NMAC NP-7414, nuclear power plants each contain between 1000 and 2500 SOVs, and of these about 20 to 50% are used in safety-related applications.

6. APPLICATIONS OF AOVS IN NUCLEAR POWER PLANTS

All U.S. nuclear power plants use AOVs and some applications appear to be common to most if not all plants. For example, U.S. light water reactors (LWRs) use AOVs for containment isolation functions and for control of main steam. U.S. boiling water reactors (BWRs) use AOVs in conjunction with SOVs to control their scram systems. U.S. pressurized water reactors (PWRs) use AOVs for controlling auxiliary feedwater, main feedwater, and condensate systems.

The majority of AOVs at U.S. LWRs are non-safety-related and are generally associated with the non-nuclear balance-of-plant systems. However, from observations at the plants visited, there are significant numbers of safety-related AOVs as well as important non-safety-related AOVs in U.S. LWRs. The number and general safety categorizations of AOVs in the plants visited as part of this study are described in Table 1 (because of the size of the tables in this report, all of them are presented at the end of the main text).

Applications of AOVs in nuclear power plants include:

- Auxiliary feedwater discharge to steam generator isolation and flow control valve;
- Auxiliary/emergency feedwater turbine trip stop valve;
- Chemical and volume control system (CVCS) letdown isolation valves;
- Containment spray header isolation valve;
- Control rod drive scram discharge volume drain valve;
- Emergency diesel generator air start isolation valve;
- Feedwater containment isolation valve;
- Feedwater regulating valve bypass valve;
- High-pressure core spray emergency diesel generator air start isolation valve;
- Low-pressure core spray containment isolation valve;
- Main feedwater regulating valve;
- Main steam safety/automatic depressurization system (ADS) valve;
- Main steam turbine bypass valve;
- Pressurizer PORV;
- Reactor coolant pressurizer spray valve;
- Reactor core isolation cooling discharge to feedwater check valve;
- Residual heat removal refueling water storage tank suction isolation valve;
- Standby gas treatment system blower isolation valve;
- Standby gas treatment system filter isolation valve;
- Suppression pool/torus vacuum breaker valve; and
- Ventilation dampers for the control room and diesel generator air supply applications.

Applications of SOVs in nuclear plants include:

- Instrument air-pilot valves for AOVs;
- BWR control-rod drive and SCRAM system (over 300 per BWR plant, according to NUREG/CR-4819, Vol. 1);
- High-pressure/temperature (e.g., 2500 psi/600°F) steam/water valves for primary and secondary systems; and

- Low-pressure/temperature (e.g., 160 psi/ 2000°F) process valves for flow control in miscellaneous plant systems.

Overall numbers of valve populations in nuclear power plant applications may include some SOVs as piece-parts of AOVs.

7. CURRENT NRC EQUIPMENT QUALIFICATION, MAINTENANCE, AND TESTING REQUIREMENTS FOR AOVS IN NUCLEAR POWER PLANTS

The NRC's equipment qualification requirements evolved over several decades. As a result, the criteria regarding equipment qualification, particularly environmental qualification of mechanical equipment, may vary among individual nuclear power plants. Current general requirements for environmental design and qualification of nuclear power plant equipment are summarized in the Acceptance Criteria subsection of Section 3.11 of the NRC's Standard Review Plan (SRP), NUREG-0800, as follows:

- The equipment shall be designed to have the capability of performing its design safety functions under all normal, accident, and post-accident environments for the length of time that its function is required;

- The equipment environmental capability shall be demonstrated by appropriate testing and analysis; and

- A quality assurance program meeting the requirements of 10 CFR 50, Appendix B, (Quality Assurance Criteria) shall be established and implemented to provide assurance that all requirements have been met.

Criteria and reference standards for the consideration of "harsh" and "mild" environmental qualification of electrical and mechanical equipment are also included in Section 3.11 of the SRP.

Most AOVs contain organic parts such as diaphragms, O-rings, and seals that are subject to environmental deterioration and aging. Several licensees spoke about qualified life regarding diaphragms and seals during the visits and referred to requirements in their Quality Assurance programs to check such data.

Maintenance requirements for nuclear power plant equipment of interest to the NRC (including AOVs) are driven by the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria," the recent 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (the Maintenance Rule), and licensee commitments toward the operability of systems and equipment described in individual plant FSARs. Preventive and corrective maintenance for safety-related and important non-safety-related AOVs is governed by these requirements. Few licensees (based on the experience with those visited or consulted) have developed predictive maintenance programs for AOVs.

Testing of safety-related AOVs is governed by each licensee's licensing-basis commitments toward specific editions of the valve testing requirements of the "American Society of Mechanical Engineers Boiler and Pressure Vessel Code" (ASME Code). Section XI of the ASME Code requires a stroke-timing test at three month intervals for safety-related power-operated valves. The test interval can be as long as each refueling outage, or even waived entirely, if operational safety considerations dictate. The test consists of stroking the valve in the safety-significant direction (or directions) and measuring the time required to complete the stroke. Licensee actions regarding re-testing or maintenance would depend on comparisons with required stroke time and/or comparisons with trends of previous test results. The performance of any corrective or follow-on surveillance would be dictated by safety and operational considerations. Generally, the valve stroke-timing test is performed with no fluid flow and no fluid pressure (or low fluid pressure) in the pipeline.

No specific periodic testing requirements are imposed by the NRC for non-safety-related AOVs as individual components. System flow testing may have been conducted at startup or immediately after systems (containing AOVs) have undergone maintenance or repair. It is logical to assume that all AOVs would be stroked, if not stroke-time tested or tested against flow or pressure, after repair and prior to being declared ready for service (post-maintenance testing). These tests provide useful information about the performance of AOVs, however, testing may be of limited value if the AOVs are not observed and/or instrumented when they are opened and closed during such tests. These tests will also provide limited assurance of operability under design basis conditions when design basis or full flow or pressure conditions if there is no (or low) fluid pressure or flow in the line (the usual situation) when the test is performed.

8. AOV OPERATING EXPERIENCE AT THE PLANTS VISITED

Information regarding AOV performance was gathered during the visits for this study and is discussed below.

Events or conditions of particular interest were those involving common-cause failures or potential failures, such as design problems or environmental conditions that could adversely affect the performance of a number of AOVs simultaneously or over time. Table 6 provides a list of air-operated valves considered to be risk significant by the licensees in the plants visited.

Each licensee's input to both the LER database and the nuclear industry's databases (NPRDS or EPIX) may vary. Events involving non-safety-related AOVs might or might not be included in either or both databases, depending on circumstances. Interpretations about what is or is not reportable to either or both databases appears to be somewhat variable. Some events or conditions covered by LERs were not reported to the industry's database and some events or conditions reported to the industry's database were not reported as LERs. There were many events (including failures) which were reported in licensees' condition reports but were not reported in LERs or otherwise reported to the NRC. For example, the Palisades events involving common-cause degradation and failure of air regulators were not covered by LERs at the time of the visit for this study. The LER and nuclear industry databases do not capture all of the information about AOV failures and other pneumatic equipment failures received during the site visits. The licensees interviewed indicated that their reporting was in compliance with their interpretations of current NRC requirements.

8.1 AOV Operating Experience at Palo Verde 1, 2, and 3 (PVNGS), Trip No. 1

8.1.1 Atmospheric Dump Valve Failures

For each PVNGS plant, one pneumatic operated atmospheric dump valve (ADV) is installed in each of the four main steam lines to allow cooldown of the steam generators when the MSIVs are closed or when the main condenser is not available as a heat sink. Each valve is designed so as to allow for an unavailable steam generator, concurrent with a loss of AC power and a single failure of one of the remaining ADVs. On March 3, 1989, after a plant trip, all four ADVs on Unit 3 failed to respond when given an open signal from the control room. The only ADV tried from the remote-shutdown panel also failed to respond. Refer to LER 52889005 and NRC Information Notice 89-38. Manual local action was initiated to open one ADV per steam generator, however an actuator for one of the ADVs was damaged in doing so. On March 5, 1989, a test was conducted on the Unit 1 ADVs and one of the ADVs failed to open when given a 50% open demand signal from the control room. Shortly thereafter, all three units were shut down.

The results of the investigation indicated that excessive piston ring leakage, combined with inadequate pilot valve relieving capacity, created high forces in the valve bonnet (also called the balance chamber) that could not be overcome by the actuator. Other problems were also found that compromised the operability of the ADVs. These included:

- Valve oscillations caused by lower than required nitrogen pressure (the regulators exhibited excessive leakage);
- Positioners were not adjusted and/or maintained properly;
- Springs were left on the valve operators that should have been removed prior to startup;
- An actuator piston was fitted with non-qualified Buna-N O-ring rather than a qualified Viton O-ring;
- Air and nitrogen quality was suspect (particulate contamination); and
- Several non-qualified pressure gauges were left installed on the positioners.

Several or these problems were common-cause failure mechanisms which were applicable to either the valve design, valve operator design, or deficiencies in regulators and positioners.

The licensee forwarded a 10 CFR Part 21 report to the NRC. Other plants that have the same or similar ADV AOVs are Catawba 1 and 2, SONGS 2 and 3, and Waterford 3.

8.1.2 Letdown Containment Isolation Valve Leakage

In 1995, PVNGS completed an investigation of the recurring seat leakage, over several years, of three letdown containment isolation valves. LER 52895007 (a previous LER 52894009 also applies) and PVNGS Condition Report 95Q028 of May 11, 1995, described the results of the investigation. The most probable common causes of the seat leakage problems were:

- Undersized pneumatic actuators resulting from not accounting for the high frictional loads of graphite style packing during original sizing of the actuators.
- Not maintaining the specified bench set on the spring-and-diaphragm actuators. "Bench set" is a term used for spring-and-diaphragm AOVs to denote the pressure range (pressure to the actuator diaphragm) through which the valve will stroke its full travel with the actuator uncoupled (packing friction, differential pressure, and seating forces are excluded) from the valve. Basically, the bench set determines the preload on the actuator spring.

There are three letdown isolation valves in each unit at PVNGS. Two of the valves also serve as containment isolation valves. The valves are 2-inch Fisher globe valves. Flow pressure tends to open the valves. The actuators are Fisher pneumatic spring-and-diaphragm actuators. The valves are specified to be able to open and close against a differential pressure of 2485 psig.

Modifications were made to all three AOVs with undersized actuators. Stroke lengths were reduced, the actuators' springs were replaced and the bench sets were increased on all three actuators. Prior to the modification, one actuator had the bench set increased to provide seat load. Later, it was discovered that the higher bench set exceeded the manufacturer's maximum recommended safe spring load and stem load. PVNGS thereafter determined that the higher bench set was acceptable because the valve is normally open and rarely stroked and thus the assumed 1000 cycles used in the calculations was conservative.

8.1.3 Downcomer Feedwater Isolation Valve Leakage

Three downcomer feedwater isolation valves (DCFWIVs) at PVNGS failed to open, following closure after a main steam isolation signal (MSIS) during a Unit 1 reactor trip on November 26, 1995. LER 52895012 covered the reactor trip but not the DCFWIV failures. The source of the information concerning the DCFWIV failures was a comprehensive report provided by the licensee during the site visit for this study. The report (the copy received was untitled) included a root-cause analysis and failure mode evaluation, description of the DCFWIVs, corrective actions description, Maintenance Rule considerations, and observations about the potential for related failures of other valves.

The attempt to reopen the DCFWIVs was made about 14 hours after their closing in response to the MSIS, during which time the valves had cooled considerably. System pressure upstream of the DCFWIVs was bled off, resulting in minimal pressure across the valve disc and a reduction in required actuator force. The three valves still would not open. Finally, an auxiliary feedwater pump was started and the motor-operated valves upstream of the DCFWIVs were opened, providing a slug of water that (it is postulated) finally dislodged the DCFWIV discs and allowed the valves to open successfully using available nitrogen pressure.

There are four DCFWIVs in each unit, two in series to each of the two steam generators. The DCFWIVs are 8-inch flex-wedge gate AOVs. The actuators are Miller Fluid Power single-acting pneumatic cylinders with internal cylinder springs and external spring-and-stanchion set-ups. The actuators are designed to use pressurized nitrogen to open the valves and spring force to close them. The pneumatic control circuits include a 3-way SOV and a 3-way air switch valve. The valves are intended to fail closed on loss of the air signal to the 3-way air switch valve or loss of nitrogen to the operator. The valves are intended to fail open if electric power to the SOVs is lost. The DCFWIVs have three key safety functions: heat removal, trip initiation, and containment isolation. These valves are also Appendix R safe-shutdown components.

The most probable root causes for the multiple valve failures to open were:

- A lack of prudent actuator design margin. The low actuator margin resulted from using a nonconservative valve factor (0.3) in the original sizing. Recent tests on motor-operated gate valves in response to NRC Generic Letter 89-10 indicate that 0.3 is not a conservative valve factor for flex-wedge gate valves.

- Not allowing for the potential effects of thermal binding in the original sizing of the actuator. Pressure locking and thermal binding of power-operated gate valves, including AOVs, is discussed in NRC Generic Letter 95-07. The DCFWIVs were excluded from the licensee's GL 95-07 evaluation because they are normally open valves.

- Not allowing for the potential effects of degradation of the nitrogen supply in the calculation of actuator margin.

Results from static diagnostic testing did not reveal why three of the DCFWIVs failed to open and the remaining AOV opened. A dynamic test program to establish the operability of these AOVs was conducted and valve factors of 0.64 and 0.57 were established.

From the perspective of this study, the significance of these AOV failures was that the actuators were undersized to open the valves. The actuators were not known to be marginally sized to open the valves until the calculation was completed as part of the Root Cause of Failure Investigation. This, combined with thermal binding in the gate valves and marginal nitrogen pressure, all contributed to the common-cause failure of the DCFWIVs. The failures of the vacuum breaker solenoids that caused the reactor trip leading up to these failures is described in section 8.1.4, below.

The licensee did not (and did not have to) report the DCFWIV failures to the NRC. The key safety functions of the DCFWIVs are to close upon actuation of a main-steam isolation signal (MSIS) and to open upon demand to establish a flow path from one of the auxiliary feedwater pumps to the steam generators when that pump is chosen to make up steam generator inventory during periods of hot-standby and plant cooldown. These AOVs successfully closed on demand from the MSIS, thus satisfying their safety function. However, the AOVs failed to open following the cooldown of the unit, and as such, the failures were not classified as safety-related failures. Also, the failures were considered by the licensee as not being

Maintenance Rule functional failures. The failure mode that occurred was not considered by PVNGS to be safety-significant because the open function was not credited in the safety analysis. The failure was not subject to review under the requirements of the Maintenance Rule because the system function was not required in the plant mode when the failure occurred. Further, as noted above, these valves were not within the scope of either NRC Generic Letters 89-10 (these are not MOVs) or 95-07 (DCFWIVs are normally open valves).

8.1.4 Failure of a Vacuum Breaker Solenoid and Subsequent Investigations Involving SOVs

On November 26, 1995, a loss of condenser vacuum condition occurred at Unit 1 of PVNGS and caused a main turbine trip. A reactor trip followed shortly thereafter. Refer to LER 52895012 and to the failure of the DCFWIVs described in paragraph 8.1.3, which also resulted from the trip. The event was initiated when a condenser vacuum breaker inadvertently opened. Two similar SOVs on Unit 2 were found to have air leaks.

The vacuum breakers are non-class, 14-inch Pratt butterfly valves equipped with air-to-open, spring-to-close actuators. The valves are normally closed and require a 3-way ASCO 8321 SOV to be energized to pressurize the actuator and open each valve. The SOVs are normally de-energized and spring force keeps each valve closed.

The vacuum breaker went partially open due to a failure of the solenoid valve which allowed leakage of supply air to the cylinder and out the exhaust port. The exhaust port speed control valve was also throttled which allowed a larger amount of leaking air to pressurize the cylinder. Apparently, the condenser vacuum breaker solenoid failed because the internal gasket material was worn. Other internal parts also showed signs of wear including the seating surfaces on the piston/guide sub-assembly and on the core assembly. The root cause of the SOV failure was attributed to aging. Two other SOV leaks were detected as a result of inspections shortly after the event.

PVNGS evaluated similar SOVs in other plant systems to ascertain their susceptibility to cause a plant trip, transient, or other system operation problem. SOVs that have a high potential to cause such problems were evaluated further to determine if there were current preventative maintenance (PM) programs to replace them. A summary list was prepared of over 1700 SOVs that were determined to need a new PM task generated and/or have the exhaust speed control valve removed.

8.1.5 AOV Margin Calculations

Among the observations made during the visit to PVNGS for this study, it was noted that several safety-related AOVs had low margins, according to the calculations that were furnished for discussion. Specifically:

- Globe valves: The licensee's engineers were using the port area times the differential pressure (d/P) times 1.0 in their thrust estimate calculations. Instrument Society of America guidance is to use the port area times d/P times 1.0. Motor-operated valve testing for other types of globe valves would indicate a valve factor of 0.9 to 1.1. Depending on the particular valve design, the licensee thrust estimates may be as much as 10% low.

  It is important to note that the air-operated globe valves used at Palo Verde have not been dynamically tested in a Generic Letter 89-10 type of program to verify valve factors. Also the licensee has only found physical valve/actuator performance issues with actuators (for globe valves) which were not sized in accordance with the original equipment manufacturer's published recommendations.

- Gate Valves: The licensee's engineers were using the seat area (as provided by the valve manufacturer) times d/P times a valve factor of 0.6 in their thrust estimate calculations. A valve factor of 0.6 is reasonable for cold water systems. INEEL and EPRI test results both support the use of a mean seat area [0.5 X (inside diameter + outside diameter)].

Palo Verde suggested the above wording in their comments on the draft report and provided the following explanation as justification for the gate valve discussion: The Palo Verde engineers had originally used the port area for the thrust estimates but had revised the calculation to use the seat area prior to the site visit. The calculation shown to the authors of this study included an error in that some text explaining the calculation indicated the port area was used rather than the seat area. This text was missed as part of the calculation revision to use seat area and has since been revised. The quantitative part of the calculation reviewed by the authors actually used the seat area even though the text indicated the port area had been used.

Palo Verde initiated several margin improvement modifications based on their AOV capability calculations. These modifications covered 15 Category I and II valves in each unit (45 AOVs total).

The lessons learned in the calculation of valve forces and margins in motor-operated valves are at least partially pertinent for AOVs. The valves used in both AOVs and motor-operated valves are either identical or similar and subject to the same design considerations. Several licensees visited reported that they used the information and techniques developed for motor-operated valves to evaluate inadequate performance of AOVs or investigate events involving AOV failures. It is expected that further investigation of AOV performance would reveal problems involving sufficient margins similar to those found as a result of the studies of operability of motor-operated valves performed in response to NRC Bulletin 85-03 and NRC Generic Letter 89-10.

8.2 AOV Operating Experience at Enrico Fermi 2, Trip No. 2

8.2.1 Failure of Multiple SOVs Controlling Safety-Related AOVs

Inadequate closing force was discovered on four air-operated isolation valves on April 22, 1994. The review in which the deficient condition was found was prompted by two other events in the industry that occurred in 1993. See LER 34194004.

These valves are containment isolation valves that isolate the reactor-coolant pressure boundary. According to data provided during the site visit, they are 3/4-inch Rockwell globe valves with Fisher 667, Size 34 air actuators. The operators are designed to use air to open and springs to close.

The cause of this event was inadequate control of the valve actuator settings. The original specification for these valves was to ensure closure in the control-rod drive hydraulic flow direction at a pressure of 1750 psig. This resulted in a setting by the vendor sufficiently high to meet both the containment isolation as well as the reactor coolant pressure boundary isolation requirements for these valves. However, the nameplate supplied by the vendor incorrectly identified a lower setting. The valves were reset during installation to this lower setting using the nameplate data. The resulting setting was too low to ensure adequate closure of the valves for meeting their reactor-coolant pressure boundary isolation function.

Spring preload settings were adjusted to ensure valve closure and the design-basis documentation was reviewed and revised. LER 34194004 indicates that four similar AOVs were discovered by the licensee to have "a similar problem." It is not known if or how many other similar AOVs at Fermi 2 or in other plants have similar incorrect name plate data that could cause plant personnel to set up the AOVs incorrectly. This was a potential common-cause failure situation.

8.2.2 Failure of Multiple SOVs Controlling Safety-Related AOVs

Failure of three SOVs due to deposits of a mixture of thread locking compound and lubricant on working parts was documented in Fermi 2 DERs 97-1200 and 97-1202. The licensee determined that these events were not reportable. Previous problems of this nature on eight other SOVs were discussed in a Fermi 2 internal report dated September 17, 1997 (enclosed with DER 97-1202). Similar (lubricant) contamination problems going back a number of years are documented in NUREG-1275, Volume 6, "Operating Experience Feedback Report - SOV Problems," subsection 5.2.4. An acronym found in NUREG-1275, Volume 6, "FUSS" (foreign unidentified sticky substance) applies. Similar problems recur throughout the industry. For examples, refer to LER 26095008 for Browns Ferry, LER 28688009 for Indian Point 3, LER 32192003 for Hatch, LER 37495005 for LaSalle (paragraph 8.4.2 below), and LER 44090021 for Perry. Very recently (early September 1999), Clinton reported multiple failures of SOVs used to control AOVs, from a problem that appeared to be similar based on preliminary information.

Fermi 2 DER 97-1202 included a comprehensive "SOV Failure Team Report" that described the failures and a subsequent detailed investigation of SOVs at Fermi 2. A number of actual and potential common-cause failures resulted from the thread locking compound contamination problem. For example, the report indicated that for the normally energized, safety-related ASCO Model 8320 SOVs, there were 15 failures in a population of 66 normally energized valves. The DER additionally reported the failure of three normally energized, safety-related ASCO Model 8316 SOVs. Conclusions in the report were:

- The common cause of the 18 SOV failures was excessive application of Loctite PST-580 thread locking compound.
- There were insufficient material controls and instructions in place to reduce the possibility of (pipe thread sealant and valve lubricant) contamination in the plants pneumatic system.
- Definitions of the categories used by Fermi 2 for restrictions on materials used to maintain the plant were needed.
- NUREG-1275, Volume 6 findings and recommendations were not thoroughly reviewed for applicability at Fermi 2 or compared against their (then) current practices.
- The corrective and preventive maintenance processes did not adequately use actual equipment performance data for SOVs for determining the correct preventive maintenance frequencies. This appeared to have been a problem for other components, such as pumps, valves, heat exchangers, etc., as well.

To emphasize the common-cause failure potential of the above described conditions, the population of SOVs at Fermi 2 includes (as of the publication of DER 97-1202) 1449 QA-1 SOVs, of which 134 are ASCO Model 8320. Of these 134 SOVs, 66 are normally energized. There are also 1040 non-QA-1 SOVs at Fermi 2, of which 258 are ASCO Model 8320. Of these 258 SOVs, approximately 93 are normally energized.

Eliminating the use of thread locking compound and substituting metallic (Grafoil) tape for sealing and locking was suggested to minimize valve contamination problems caused by migration of the thread locking materials or residues. However, the Grafoil product was found by the licensee to be somewhat awkward to apply and difficult to install with a sufficiently tight seal because of the thickness and stiffness of the Grafoil material. Other thread-locking and pipe joint sealing options were being investigated at Fermi 2.

8.3 AOV Operating Experience at Palisades, Trip No. 3

Events at Palisades involving AOVs were generally related to moisture/particulate contamination in the air systems and these events are also discussed briefly in Section 10.1 of this study on air systems operating experience.

8.3.1 Failure of Multiple Air Regulators in the High-Pressure Air System

On March 18, 1997, a safety injection tank test line redundant high pressure injection isolation valve, CV-3018 failed to change position on demand. This situation was originally reporting to NRC/AEOD in April 1997 but was not covered by an LER. Palisades Condition Report C-PAL-97-0404, dated 3/18/97 refers to this event. The stroke test was part of a post-maintenance procedure to verify valve operation after adjustment of packing. CV-3018 is a 4-inch, 1500 lb. Walworth gate valve with a Miller Model DA-63 fail-closed piston air operator. High pressure air is supplied to the valve operator through air regulator PCV-3018 and air to the valve is controlled by SOV SV-3018. Flow rate is controlled by a manual flow control valve between the SOV and the actuator.

During the subsequent investigation, additional attempts were made to stroke the valve and it was discovered that the air regulator was blocked with dirt and rust contamination from the carbon steel piping in the high pressure air system. Based on the condition of the air regulator, it was concluded that the high pressure air system did not meet the cleanliness criteria described in the vendor manual for the air regulator. A walk-down inspection of other air regulators in the system was conducted and 9 of 22 air regulators were found to be contaminated. Significant amounts of material were found in three regulators. The restriction orifice was partially blocked on these regulators. Additional components are considered to have been affected, or potentially affected, from this common-cause failure condition.

A program for regular cleaning of the air restrictors in the air regulators, as well as regular cleaning of filter screens was initiated. The source of the contamination, i.e., the corrosion of the air system piping from moisture in the air, was not addressed in the licensee's actions.

8.3.2 Failure of Shutdown Cooling Heat Exchanger Outlet Isolation Valve

Two events that occurred at Palisades in 1978 and 1981, both involving the same type of failure of valve CV-3025, were discussed with plant personnel during the recent visit as part of this study of AOVs. Refer to LERs 25578003 and 25581030. CV-3025 was found to be inoperable when called upon to open and allow the shutdown cooling heat exchanger to operate. Shutdown cooling flow was lost.

According to section 5.1.1 of NUREG-1275, Volume 2, "(o)n both occasions, water in the IA system filled a valve positioner, causing the control valve to fail closed. The 1978 event lasted for 45 minutes, allowing the primary coolant system to heat up from 130°F to 215°F. The 1981 event lasted over (one and one-half) hours, allowing the primary coolant system to heat up from 123°F to 197°F."

This valve has a design basis function to open during entry into the shutdown cooling mode after a small-break loss-of-coolant accident (LOCA). This valve has a function to be throttled to adjust cooling flow after a small-break LOCA but the licensee considers this to be beyond the design basis. (Refer to Palisades System Level Design Basis Review Calculations EA-AOVSYS-ESS-01.) CV-3025 is a 10-inch, air-operated globe valve. The valve is powered from the non-safety-related IA system.

The event was of interest because failures in the shutdown cooling system are not normally modeled in risk analyses. The potential loss of shutdown cooling merits attention because of the potential for uncovering the core in a relatively short time if boiling or near-boiling conditions occur as a result of the AOV failure.

An attempt was made to gather information about the condition of the plant during and after the events and to assess the current condition of the valve. The old records were retrieved by the licensee's engineers but detailed information about the condition of the reactor coolant system (pressure, temperature) and the plant recovery procedures at the time of the events couldn't be retrieved.

A hand wheel was added to the AOV at some point after the 1981 event to allow the valve to be opened manually. However, the condition of the air system serving the AOV appeared, at the time of the visit, to be similar to the condition in 1981, when the events occurred. Specifically, the potential for contamination of the valve by moisture (or corrosion products from moisture contamination) occurred because the single dryer available for the IA system was, and is, periodically bypassed.

Similar diverse or common-cause failure events can occur, affecting important AOVs associated with the Shutdown Cooling and ECCS systems, because the root cause of these events (i.e., substandard air system design, operation, and maintenance) persists.

8.4 AOV Operating Experience at LaSalle 1 and 2, Trip No. 4

8.4.1 Pneumatic Valves with Less-Than-Designed Effective Diaphragm Areas Result in Inadequate Valve Closing Forces

While developing an AOV preventative maintenance program, inconsistent testing data were obtained for valves with WKM 70-13-1 pneumatic actuators. The inconsistent results appeared to be related to incorrect effective diaphragm areas (EDAs) for the AOV actuators. This occurred in the February-April 1996 time frame. Refer to LER 37396011. Two problems associated with the EDAs of the actuators of the WKM valves were identified. The first was related to the actual versus the manufacturer's published EDAs of the actuator. The second problem was stretching of the diaphragm during valve travel resulting in a reduced EDA. In March 1996, LaSalle Station's AOV Component Engineer contacted Anchor/Darling Valve Company regarding the published versus the actual EDAs and the stretching of the diaphragms. On September 20, 1996, after conducting testing, Anchor/Darling Valve Company acknowledged a reduction of the EDAs for the WKM 70-13-1 pneumatic operators.

Anchor/Darling performed a series of tests to determine the actual effective diaphragm area. These tests indicated that the actual diaphragm areas of the various sizes of the Model 70-13 actuators were approximately 90% of the published values. Further testing by the licensee uncovered a contributing problem that further reduced the effective diaphragm area. The diaphragm case consists of two dome-shape halves bolted together. Generally, one half is deeper than the other. It was discovered that reverse-acting actuators assembled with a deep upper half case caused unintended stretching of the diaphragm within the casing.

The licensee determined that the primary containment isolation valves would have closed at the design-basis-accident containment pressure of 40 pounds per square inch; however, many of the valves may not have been properly set up to close against the normally higher system pressure under some operating conditions. That is, the valves may not have closed under the highest expected differential pressure of the contained system fluid.

There were a total of 36 (18 per unit) WKM AOVs addressed in the LER. Thirteen valves per unit are installed in systems which are part of the Primary Containment Isolation System (PCIS) and five valves per unit are in the Reactor Core Isolation Cooling System (RCIC).

This condition was originally determined by the licensee to be not reportable but the LER was submitted as a voluntary report.

The root cause of this event was the use of incorrect effective diaphragm areas by the

Original Equipment Manufacturer (OEM) for actuator setup. The OEM at the time was WKM.

LaSalle prepared and implemented design change packages (DCPs) to compensate for the reduced EDA and restore the valves to the design specifications. These DCPs were scheduled to be completed prior to Unit 1 and Unit 2 startup from the (then) recent outages. A 10 CFR Part 21 notification on the deficiency of WKM 70-13-1 Pneumatic Actuators currently manufactured by the Anchor/Darling Valve Company was issued on October 4, 1996. NRC Information Notice 96-68, Incorrect Effective Diaphragm Area Values in Vendor Manual Result in Potential Failure of Pneumatic Diaphragm Actuators, was issued on December 19, 1996. The common-cause failure implications of the deficiency involve a whole class of AOVs, widely used in nuclear power plants in both safety-related and important non-safety related applications, the full scope of which has not yet been determined.

8.4.2 Failure of MSIVs Due to Sticking Solenoid Operated Pilot Valves

Unit 2 was shutting down for refueling on February 18, 1995, when a main steam isolation valve (MSIV) failed to close on signal. Refer to LER 37495005 and NRC Information Notice 95-53. The cause of the MISV's failure to close was later determined to be sticking of a solenoid pilot valve. The solenoid pilot valve did not change position, and thus did not allow the pilot air to vent. Failure of pilot air to vent results in air being ported to the under side of the MSIV operating piston, thus holding the MISV in the open position. Shortly thereafter, another MSIV failed to close under similar circumstances. The two Outboard MISVs, which failed to close, were in different Main Steam Lines; however, the Inboard MSIVs in each of these lines operated as designed.

The sticking solenoid pilot valves were disassembled and inspected. Foreign material was observed on several internal parts of the solenoid pilot valve. It was also observed that the interfacing surface of the core assembly and plug nut of one solenoid appeared to have a thin coat of foreign material. When the core assembly and plug nut were pressed together, which is the normal configuration when the solenoid is energized, the film acted as an adhesive. This adhesive material was strong enough that when the plug nut was lifted from the bench, the core assembly adhered to it and was also lifted.

The parts with foreign material were chemically analyzed by the System Material Analysis Department, and the material was determined to be Nyogel 775A. According to the SOV manufacturer, this material is used during assembly as a lubricant on the solenoid cover, under the star washer on the disc holder which is internal to the valve, and at the interface of the solenoid base and housing assembly. Corrective actions included replacement of the SOVs in the MSIV with SOVs from another manufacturer, disassembly and inspection of spare SOVs, and exercising the SOVs in the MSIVs at 30-day intervals in order to minimize the potential for sticking.

These recurring incidents of SOV contamination from lubricants and/or thread locking compounds, described in 1990 in NUREG-1275, Volume 6, and in paragraph 8.2.2 above, continue to be sources of common-cause failures in large numbers of SOVs and their associated AOVs throughout the nuclear industry.

8.4.3 Potential Failure of Safety-Related Dampers Due to Invalid Calculation Assumptions in the Original Builders Calculations

NRC Event No. 33434, dated December 19, 1997, "Inadequate Turbine Building Vent System Exhaust Tunnel/HPCS System Switchgear Room Wall," and a resulting LER 37397046, "Potential Pressurization of the Turbine Building Ventilation Exhaust Tunnel Resulting from a Main Steam HELB due to Calculation Error," refer to a transient analysis performed to predict pressures in the turbine building ventilation exhaust tunnel downstream of a postulated high energy line break. An investigation of the allowable closure time assumptions for these Turbine Building High-energy Line Break

Check Dampers (#1/2VT79YA/B/C) uncovered discrepancies in the original 1977 builder's calculations and assumptions. In the event of a Main Steam High Energy Line Break (HELB), the Turbine Building Ventilation Isolation Dampers, 1(2)VT079YA, B & C, would not close fast enough to prevent the pressure from exceeding the pressure retaining capability of the walls, floors, and ceilings that separate the exhaust tunnel from the safety related High Pressure Core Spray (HPCS) electrical switchgear room. Neither Event Report 33434 nor LER 37397046 specifically described the operating mechanism of the dampers or indicated if the dampers were considered by the licensee to be safety-related.

The design discrepancies led the licensee to reconsider the original builder's assumptions and calculations for the Reactor Building Exhaust Isolation Dampers (1VR05YA and B). LER 37398007 was generated based on the licensee's investigation. It was found that the originally assumed closure time for 1VR05YA and B would not be quick enough to prevent over-pressurization of the Reactor Building exhaust ductwork and downstream masonry plenum. The concern was that there was safety-related Control Room HVAC equipment immediately adjacent to the plenum's masonry walls, which could be damaged by failure of the walls from sudden over-pressurization. Although it is not specifically stated in LER 37398007, it is assumed that dampers 1VR05YA and B and their associated SOVs are safety-related components.

Dampers 1VR05YA and B had malfunctioned and were subsequently rebuilt in 1985 (see LER 37385008 and LER 37385011). Instrument air keeps the dampers open during normal plant operation. A high pressure signal should de-energize the SOVs in the event of a main steam line break, allowing IA to bleed off and the damper to close. In LER 37398007, the licensee indicated that they had revised the calculations and assumed a 0.4 second instrument time delay, and a 0.075 second solenoid valve response time, to correct the original assumption of instantaneous closure.

8.5 AOV Operating Experience at Three Mile Island 1, Trip No. 5

8.5.1 Insufficient Design Margins in Aloyco Air-Operated Gate Valves

Potentially insufficient margins in two AOVs were described by the TMI engineers during the visit for this AOV study. Calculations for five Crane-Aloyco, 2.5 to 6 inch, 1500 and 150 pound class, flex-wedge and split wedge gate AOVs with Miller DA-63-B and A-63-B cylinder actuators were reviewed by one of the TMI engineers. The architect/engineer (AE) requested that the valve manufacturer (Crane-Aloyco) perform thrust calculations on the five AOVs, as part of a limit switch upgrade modification, in order to verify that limit switch installation would not affect valve operability. The resultant thrust calculations, using "present day" methodology, indicated that two of the five valves had negative closing margins for the specified differential pressure (d/P). The AE then requested that the manufacturer re-perform the calculations using the methodology by which the valves were originally sized. The revised Crane-Aloyco calculations, based on the original valve sizing methodology, indicated positive margins in both the opening and closing directions, but using a valve factor of zero. Upon review of the Crane-Aloyco calculations, TMI convened a review group which verified that the two containment isolation valves which had negative margins in the manufacturers' first calculations were, in fact, operable and would be able to perform their designed safety function because the required operating conditions were less severe than the original design basis conditions. This conclusion was based on TMI's calculations using a 0.75 friction factor and d/P of 1600 psi (that required for containment isolation).

A design modification was proposed to increase the closing margin for one of the valves because normal operating differential pressure (2375 psi) was greater than the differential pressure encountered when performing its safety function (1600 psi). The proposed modification included the installation of an accumulator and piping to provide air assist to the spring. Subsequently, TMI performed an analysis of the valve using the EPRI PPM methodology (2375 psi and 0.67 friction factor) and confirmed the operability of the valve. Design problems with a pressure booster for the AOV were also noted. Originally, these conditions were not covered by an LER or other correspondence with the NRC.

During or shortly after the plant visit for this AOV study, the TMI engineers agreed to investigate to determine if there were 10 CFR Part 21 concerns to be addressed, either by TMI or the valve manufacturer (Crane-Aloyco). They determined that the conditions were not reportable under 10 CFR Part 21 for the valves because 10 CFR Part 21 had not been involved in the original purchase documents. However, as described in the next paragraph, there may be generic or common-cause issues with these types of AOVs, related to the design or the OEM calculations, that need to be addressed.

It was learned that Crystal River 3 (a similar B&W plant with the same AE, Gilbert Associates, as TMI-1) had a similar margin problem, with a Crane-Aloyco AOV, as reported in LER 30297015. Crystal River previously encountered design margin problems with Crane motor-operated valves due to degradation caused by normal wear in the valves, as reported by LER 30292004. For the configuration at Crystal River 3, the licensee replaced the AOV's actuator with a larger one.

8.6 AOV Operating Experience at Indian Point 3, Trip No. 6

8.6.1 Failure of Diaphragms in AOVs

Recent events that occurred at Dresden and Quad Cities, involving potential failure of AOV diaphragms on Copes-Vulcan D100 valve operators, were discussed with the maintenance engineers at IP3 during the visit incidental to this study. See Dresden LER 23798003, Morning Report H-98-0045 dated March 6, 1998, and 10 CFR 50.72 Report 33620 of January 28, 1998. The problem was that the elastomer covering the fibers in the diaphragms was too thin and the diaphragms wore out prematurely as the valves were operated. IP3 engineers had not been aware of this problem and, since they have a large number of this type of valve, they started to investigate immediately.

Copes-Vulcan modified the diaphragm design in 1996 to increase the amount of rubber on the side of the diaphragm in contact with the diaphragm plate. Licensees have to determine if diaphragms in the AOVs at their plants have a deficient design such as that described.

A search of the nuclear industry database indicated that there are over 1900 AOVs of this type in U.S. power plants. IP3 has about 90 such AOVs. About 18 failures that involve the diaphragms of these actuators were recognized to be premature failures and reported as such in the industry database. It is not known how many diaphragms having this deficiency were replaced for normal wear and not recognized to have experienced premature failures.

Refer to Sections 9.2 and 9.4 in this study for descriptions of other design problems related to Copes-Vulcan AOVs.

8.6.2 Two Containment Isolation, In-Series AOVs were Inoperable Due to Inadequate Design

On February 15, 1996, two air-operated diaphragm type vapor containment isolation valves in series were found to be inoperable, violating plant Technical Specifications (LER 28696004). The original isolation valve design would close at system pressure when there was a differential pressure or accident design pressure, but not against system pressure with no pressure differential. The cause of this inadequacy in the original design was not determined and the condition was considered by the licensee to have not required any corrective action (although the valve operators and SOVs were subsequently replaced). This was a common-cause event.

These two AOVs (RC-AOV-519 and RC AOV-552) are vapor containment isolation valves on the 3-inch primary water system line to supply spray water to the pressurizer relief tank and make-up water to the reactor coolant pump stand pipes. Both RC-AOV-519 and RC-AOV-552 are 3-inch air operated diaphragm valves manufactured by the ITT Grinnel Company. The valves are normally closed during operation but can be opened when required since they are supposed to automatically close on a containment isolation signal. The engineers determined that both valves behaved radically different, depending on whether or not the system was pressurized or depressurized.

The valve vendor explained that when designing a diaphragm valve/actuator, one must consider whether it closes with a 100% differential pressure across the valve (i.e., valve closed with little or no downstream pressure), a 0% differential pressure (i.e., valve closed with a constant line pressure upstream and downstream) or for both conditions. RC-AOV-519 and RC-AOV-552 are designed to positively seal with a differential pressure and would close against a differential pressure of 150 psi (the primary water system design pressure) but would not close with a line pressure greater than about 120 psig when there is no differential pressure. The original specification for these valves was for a maximum differential pressure of 200 psi, with no reference to a minimum pressure differential or constant line pressure requirement. The valve design had not been modified since the plant was built.

The condition could have existed when closing isolation valve RC-AOV-560, downstream and in series with RC-AOV-519 and 552, in a sequence that maintains a line pressure greater than 12 psi (probably a typographical mistake in LER 28696004 and should be 120 psi). There would then be no differential pressure during stroke testing. Stroke testing under these conditions could have created another problem if the limit switches were adjusted after the valves were to have been shut but did not fully close. The valves could have been assumed to be fully closed during the adjustment so subsequent failures to close would be masked and could have allowed the existence of a spurious control room indication that the valves were fully closed when they were not. There is no direct external indication on the valves to show if they are fully closed.

Tests were conducted that verified the condition described above. Other valves were suspected of having the same problem but subsequent investigation indicated that these two valves were the only ones at IP3 subject to the differential pressure problem described.

The operators and air-supply solenoids for these valves were replaced with re-sized components so that the AOVs meet design requirements.

8.7 AOV Operating Experience at Turkey Point 3 and 4, Trip No. 7

8.7.1 Failure of Auxiliary Feedwater System AOVs from Air System Contamination

Turkey Point Units 3 and 4 experienced recurring common-cause failures in AOVs, pressure transmitters, and pressure controllers in the Auxiliary Feedwater (AFW) systems for both units during surveillance testing in July 1985. The failures were traced directly to IA system moisture and corrosion product contamination and involved simultaneous failure of the AFW flow control valves and steam generator bypass valves.

The event involved the loss of all three trains of AFW (one train for each unit and a swing train). This event was categorized as a precursor event in the NRC's Accident Sequence Precursor (ASP) program. The conditional core damage probability for this event was estimated in the ASP analysis to be approximately 9E-04, making it the fourth highest of the 40 precursor events identified that year.

Other safety-related systems were also supplied with contaminated air during this period (prior to correction) and could have also been adversely affected. Functions, other than AFW, which could have been affected (simultaneously) were steam dump to atmosphere, salt water system flow from the essential heat exchanger, the charging system, and the residual heat removal system.

These conditions, the events leading to them, and the potential consequences were described in detail in Section 5.1.2 of NUREG-1275, Volume 2. LER 25085021, as well as several NRC Inspection Reports (referenced in NUREG-1275, Volume 2), also described these events and conditions.

The licensee installed effective drying equipment and filters, along with continuous dew point monitoring equipment to correct these problems. See Section 10.1 in this report. Although the conditions have since been corrected by dramatic improvement of instrument air quality, they are noted here because of their severity and (then) potential consequences.

8.7.2 Pilot Lockup Valves (POLVs) for Emergency Containment Cooler (ECC) Valves

One failure of a POLV (CV-4-2908) to operate was confirmed in 1996. As-found examination of other POLVs subsequently removed from service indicated that 3 of 12 were stuck. Several contributors to failure including O-ring distortion and grease caking were identified as causing excessive drag. Increased exercise and spring modifications were implemented as fixes. Turkey Point Condition Report 96-0535 dated April 29, 1996, Supplement 1, dated May 16, 1996, Supplement 2, dated September 16, 1996, Supplement 3, dated February 7, 1997, and a summary report, "Investigation of Increased Drag Forces in Pilot Operated Lock-Up Valves," prepared by Altran Corp., dated September 1996, described these conditions; however, an LER could not be found.

The function of the POLVs is to open the emergency core cooling (ECC) outlet valves when air pressure drops to about 45 psig, Unit 4, or 60 psig, Unit 3. The ECC outlet valves are intended to move to the open position on loss of IA, because the actuators require air pressure to function and the IA air system is not safety related. Failure of the POLV to shift on loss of IA pressure could result in insufficient component cooling water flow to support the containment temperature/pressure control design basis safety function. Therefore, the POLVs are considered safety-related. According to the summary report of September 1996, the POLVs (at least those tested) are ITT Hammel Dahl Conoflow Model GVB-12.

The tendency of some of the POLVs to stick was attributed to several factors, including:

- POLVs were originally equipped with Buna-N O-rings. Several POLVs were refitted with Viton O-rings in an attempt to increase their service life; however, it was subsequently discovered that the Viton material is subject to compression set. The detailed discussions in the Condition Report and it supplements included information and conclusions about the various complex formulations, temperature considerations, and interactions of the elastomers with various lubricants. In the end, it was decided to install stiffer springs to overcome any tendency of the valves to stick and reinstall Buna-N O-rings.

- The silicone grease used to lubricate the valves was found to separate and lose lubrication capacity in static applications. This was considered to contribute to the sticking problem.

- The valve's tendency to stick was observed to increase over time. A surveillance interval of 31 days was established, based on the results of the investigation.

8.7.3 Intake Cooling Water Header Inlet Isolation Valve Failed to Fully Close

The Intake Cooling Water (ICW) Header Inlet Isolation Valve (POV-3-4882) failed in approximately 1/4 closed position. Corrosion was found on the lower portion of the yoke and housing below the O-ring seal. The corrosion, caused by lack of maintenance, was identified as the root cause of failure. Minor wear was also found in the housing and cover guides.

POV-3-4882 was described by the licensee as a risk significant, key component and, therefore, Maintenance Rule "a(1)" goals were established based on this event. Failure of this valve to fully close was a functional failure because it prevented the valve from automatically closing on receipt of a safety injection signal and diverting ICW flow from the turbine plant cooling water heat exchangers to the safety-related component cooling water heat exchangers. However, the licensee did not consider the problems to be reportable under 10 CFR 50.72 or 50.73 because no LCO was exceeded. The plant engineers prepared a Condition Report (CR 96-0304) which was reviewed during and shortly after the visit to the site for this AOV study.

A grease fitting was installed to enable relubrication of the yoke-to-housing interface. POV-3-4883, POV-4-4882, and POV-4-4883 were also overhauled.

8.7.4 RCS Letdown Isolation Valve Failure

CV-3-204, a safety-related letdown isolation AOV, failed to close when remotely operated from the control room on September 25, 1996. The active safety-related function of the SOV (SV-3-204) for this AOV is to vent when de-energized to enable CV-3-204 to isolate. The SOV was determined to be defective and was replaced, but no definitive root cause was identified for the SOV failure. A small amount of Teflon tape was found in the body of the SOV. Condition Report 96-1202 refers to this event but no corresponding LER could be found.

Condition Report 96-1202 contains the results of a failure history review for ASCO 8316 SOVs at Turkey Point. A population of 168 SOVs was listed. The licensee concluded that the failure history indicated that these SOVs operated reliably. However, it was noted that 27% of these SOVs had been replaced, for all reasons including both planned maintenance and failure, in the previous 10 years.

8.7.5 Revised Diaphragm Areas Furnished by AOV Vendor were Lower than Designed

Based on the report from LaSalle (see Section 8.4.1) that a valve vendor had furnished AOVs with incorrect effective diaphragm areas, Turkey Point engineers investigated their own (similar) AOVs, as described in their Condition Report 96-1598 of December 17, 1996. The vendor of the actuators at Turkey Point, BS&B, was contacted and it was discovered that 28 AOVs, divided among the two units, had effective diaphragm areas lower than originally designed by as much as 10%.

The engineers at Turkey Point use the effective diaphragm area to establish bench set, lift-off, and supply regulator settings for AOVs. The lower actual diaphragm areas would (potentially) affect the leak tightness and ability to fully stroke the valves. BS&B actuators are used on the emergency diesel generator inlet control valves, the instrument air control valves for the containment header, the bypass control valves for component cooling water from the emergency containment cooler, the main steam to main steam relief control valves, the atmospheric dump valves, and the containment sump pump discharge isolation valves.

The licensee's subsequent operability assessment indicated that there was no safety concern and that all of the AOVs would perform their functions within the existing operational parameters. Although, in this case, the AOVs had sufficient margin to perform their intended function, the actual design margins had been unknown for some time.

These examples of mistakes in vendor calculations are a common-cause failure concern.

9. OTHER PERTINENT AOV EVENTS AND OPERATING EXPERIENCE

The NRC described many deficiencies in AOVs in their Bulletins, Circulars, Generic Letters and Notices. Several of these publications, applicable to the discussion of the capability of AOVs to perform their intended functions, are discussed in this section. Operating experience involving air-operated or solenoid-operated dampers, as well as AOVs or SOVs that serve diesel generators, are also discussed in Section 11. The focus of the following discussions was primarily on the design and qualification problems of AOVs, as opposed to maintenance or air system problems.

NRC Inspection Reports and Immediate-Notification Reports (i.e., reports covered by 10 CFR 50.72) were consulted. In addition, searches were conducted in the NRC's Licensee Event Report (LER) database for pertinent events since 1985 involving AOVs. LERs and events described in other reports that were considered of interest in this study of AOVs are listed in Tables 2, 3, and 4. Table 7 includes some recent (approximately last 5 years) events and conditions involving AOVs and/or air-operated components where the design basis was not met and/or not known. Tables 6 and 7 were included in response to comments on the draft of this report dated April 26, 1999, concerning the number of AOVs that are risk significant and the timeliness and pertinence of events presented in the draft report, respectively. LERs that were cited in Tables 2, 3 and 7 are also listed in numerical order in Table 5 to provide a convenient cross-reference for the reader. Pertinent NRC generic communications regarding AOVs, SOVs, dampers and air systems are listed in Appendix A.

There were very large numbers of pertinent LERs to be reviewed regarding AOVs, dampers, SOVs and air systems. Word searches in the NRC's NUDOCS database, the NRC's Public Document Room (BRS) database, and the Sequence Code Search System (SCSS) Database were conducted. Many key-word searches in NUDOCS and the BRS databases commonly returned hit lists in the high hundreds but the events in these lists were somewhat difficult to retrieve because of limitations of the on-line systems. It is expected that many interesting and pertinent events and failures were not included.

Listings of failures and maintenance data were collected from the nuclear industry's database (Nuclear Plant Reliability Data System [NPRDS] and Equipment Performance and Information Exchange [EPIX]), and a preliminary attempt was made to incorporate the data in a spreadsheet format. However, the effort was abandoned and this data was not used in this study because insufficient time was available to devise a workable sorting scheme, and then sort and evaluate an enormous volume of information in order to determine if and how the data was pertinent.

9.1 Undersized AOV Acuators

In NRC Information Notice 88-94, dated December 2, 1988, the NRC described actuators for Fisher AOVs that were undersized and potentially incapable of providing sufficient thrust to seat, unseat, or properly operate for some design conditions. The problem was traced to a design change where graphite stem packing was substituted for Teflon stem packing. The design calculations did not account for the increased packing drag of the graphite.

Fisher indicated that until the mid-1970s it had supplied sliding stem valves with Fisher's standard single arrangement Teflon packing, and its actuator sizing technology was principally based on Teflon packing. Fisher did not account for the Teflon packing friction forces for sizing actuators with Teflon-packed valves because, based on their experience, the friction forces were calculated to be small compared to the actuator air pressure forces, actuator spring forces, valve differential pressure forces, and valve seating forces. About 1975, Fisher began supplying a number of valves with graphite laminate packing. This change was based on nuclear power industry requests to eliminate Teflon material from radiation environments and because of general personal health and safety concerns resulting in the replacement of asbestos packing material. In April 1976, Fisher began to explicitly account for packing friction forces in sizing all valve actuators after they discovered that graphite packing friction forces and other non-Teflon packing materials contributed significantly to the overall friction forces. The change was fully implemented for all Fisher orders shipped after January 1, 1977.

Fisher also stated that some nuclear power plant licensees may have installed packing different from that provided in the original valve. Such a change would be of concern if the new packing creates higher friction forces than the original packing. For example, data provided by Fisher indicated that increased friction forces can result if Teflon packing is replaced with graphite packing or graphite laminate packing is replaced with graphite ribbon packing. If the size of the installed actuator cannot overcome the increased friction forces associated with the packing change, then the valve may be incapable of performing some of its intended functions.

In addition, Fisher informed the NRC of the potential for undersized actuators on Fisher 9200 series butterfly valves. These particular valves employ rotary shafts rather than sliding stems and are not subject to the same packing friction concerns previously discussed; however, some of these butterfly valves may be equipped with undersized actuators for a different reason. Fisher sized butterfly valve actuators ordered before March 1, 1982, used a method that under some circumstances underestimated the torque required to seat or unseat the butterfly disc in the 9200 series valves. Butterfly valves ordered after March 1, 1982, were not subject to this concern because they were equipped with actuators sized by the current Fisher method that more accurately predicts valve seat/disc frictional forces.

The following AOVs were subject to the design deficiency described above:

- Sliding stem valves supplied by any manufacturer that were repacked using materials or procedures that increased the packing friction forces beyond those accounted for in sizing the actuators.
- Fisher Controls sliding stem valves shipped before January 1, 1977, supplied with graphite and other non-Teflon packing. The actuators for these valves were sized by Fisher without accounting for packing friction forces. Actuators for sliding stem valves supplied by other manufacturers may also be undersized, depending on the actuator sizing methods used by these manufacturers.
- Fisher Controls 9200 series butterfly valves ordered before March 1, 1982. The method used to size the actuators for these valves may have underestimated the torque needed to seat or unseat the butterfly disc.

9.2 Weight and Center-of-Gravity Discrepancies in AOVs

NRC Information Notices 89-28 of March 14, 1989, and 90-17 of March 8, 1990, described discrepancies in the information on weights and center-of-gravity (CG) provided by Copes-Vulcan to a large number of nuclear plants. Numerous types, sizes, and classes of AOVs (as well as other valve types) in various services were involved.

Westinghouse Electric Corporation and Copes-Vulcan prepared an extensive set of revised valve drawings showing the corrected weights and centers of gravity. Westinghouse compiled these drawings into valve lists that represented a complete set of correct weights and centers of gravity as was then currently known to them. However, these valves represented only a portion of the total population of valves that Westinghouse had supplied to various licensees. Tables included in the Information Notices provided condensed information to licensees of the extent of the problem including a brief description of the type and class of the valves most significantly affected by these corrections.

In the information provided by Westinghouse and Copes-Vulcan, approximately 16 models of Copes-Vulcan valves were noted as having weight increases greater than 25%. The largest individual increase was 1710 pounds on a 16-inch, Class 900, air-operated feedwater valve. This represented an increase of 83% in the weight originally given for the valve. Other notable weight increases included a 515-pound weight increase for a 10-inch, Class 600, air-operated modulating valve, and a 107-pound weight increase for a 2-inch, Class 1500, air-operated modulating valve.

The center of gravity discrepancies were far more extensive than the weight discrepancies. Approximately 32 models of Copes-Vulcan valves were noted as having potentially significant discrepancies of this type. In some cases, the center of gravity information was not originally supplied along with the valve. In those instances, piping design analyses were typically performed using conservative estimates. However, in a large number of cases the original information provided with the valves was non-conservative by more than 20%. In the worst case, the center of gravity on a one-inch, Class 600, air-operated isolation valve changed by 15 inches. The staff also concluded that these weight and center of gravity discrepancies were not unique to Westinghouse plants.

Several licensees reanalyzed the associated piping systems using corrected small bore valve weight and center of gravity information. The results of these initial reanalyses disclosed instances where the calculated stress would exceed 100,000 psi during a design-basis earthquake. In all cases the acceptability of the installation was confirmed, but in several cases use of more sophisticated time-history type analyses and NRC-approved increases in allowable stresses were necessary. In addition, some modifications to the seismic support configurations were required in order to bring the installations within the allowable FSAR stress criteria.

This information is included in this current study of AOVs to make the point that the quality of design information provided by vendors has been questionable concerning several important aspects in the design of AOVs.

9.3 Failure of Air Actuators Upon Gradual Loss of Air Pressure

The NRC published Information Notice 82-25 on July 20, 1982, to alert licensees of the potential for certain Hiller actuators to fail in "non-safe" positions if the air supply is gradually decreased.

Mississippi Power and Light Company reported that a large number of isolation valves in the instrument air system at Grand Gulf Nuclear Station (GGNS) failed to pass test requirements. The valves were supplied by the William Powell Company and equipped with actuators supplied by the Ralph A. Hiller Company. Various plant systems were affected, primarily those associated with containment isolation.

During pre-operational testing designed to simulate a slow loss of air in accordance with position C.9 of Regulatory Guide 1.80, a large number of pneumatically operated valves failed to go to their fail-safe condition when the instrument air header was slowly depressurized. Additional testing to simulate an air-line break in accordance with position C.8 was accomplished by depressurizing the instrument header supplying the containment, drywell, and auxiliary building from operating pressure (110 psig) to atmospheric pressure in one minute. Forty-eight valves failed this test.

The specifications for the actuators required the valve to fail to a specified position upon a loss of instrument air but did not specify the rate of depressurization. The Hiller actuators would operate in the specified manner only if the actuator itself was promptly depressurized. The pneumatic actuators include an accumulator with stored air. The stored air is transferred to the actuator cylinder to stroke the valve by means of pneumatic-operated selector valves controlled by (non-safety-related) instrument air. Upon a gradual loss of instrument air pressure, the selector valves will bleed the accumulator air to the atmosphere rather than to the actuator cylinder. This occurs near 20 psig when the selector valve plunger is in an intermediate position.

Mississippi Power and Light Company added safety-related pressure switches to sense air supply pressure to the valves. When the supply air pressure drops to a point slightly above that at which the actuator selector valve would begin to move and bleed off the accumulator, the pressure switch de-energizes the solenoid pilot and immediately causes the valve to go to the fail-safe position.

The instrument air system at Grand Gulf is not seismic Category 1; therefore, a line break causing a rapid loss of instrument air is a realistic concern. Had the condition simulated in the tests (i.e., instrument air-line break) occurred coincident with a postulated loss of coolant accident, then the failure of the pneumatic valves to go to their fail-safe position could have resulted in a loss of containment integrity.

On September 10, 1992, the NRC issued Information Notice 92-67 to alert licensees to problems that might occur as a result of their licensee modifications, made to address potential failure of Hiller actuators upon a gradual loss of air pressure.

Valve assemblies for three Shearon Harris Main Feedwater Preheater Bypass Isolation Valves were specified, procured, and installed for Q Class application. The Anchor Darling Valve Company supplied the valves and the associated Hiller actuators. On January 7, 1992, Carolina Power and Light Company discovered several components associated with the air supply to the actuators of the three main feedwater preheater bypass isolation valves were not qualified for a Q Class application. Specifically, the failure of the air pump in the non-Q Class, non-seismic instrument air supply to the valve actuator accumulator could prevent pressure switches upstream of the air pump from detecting slow leakage in the Q Class, seismic portion of the actuator air lines. The pressure switches were installed to ensure valve closure by sending an automatic close signal if the instrument air system pressure (upstream of the actuator air pump) dropped to 66 psig as discussed in Information Notice 82-25.

The Main Feedwater Preheater Bypass Isolation Valves at Shearon Harris function as containment isolation valves upon receipt of a feedwater isolation signal. The function of the air pump is to raise the normal instrument air supply pressure from between 70 and 100 psig to approximately 150 psig. If accumulator pressure drops from 150 psig to 122 psig, the main feedwater preheater bypass isolation valve may not close within 10 seconds. If pressure drops to a value as low as 20 psig, the air pressure may not be sufficient to close the Main

Feedwater Preheater Bypass Isolation Valves and keep them closed against the maximum differential pressure across the valve seats.

Upon discovery of this condition, Shearon Harris established a surveillance interval for verifying that the actuator components were functioning properly and that the accumulators were fully pressurized. Subsequently, non-Q components were replaced with suitable components and testing was completed satisfactorily.

A postulated air leak in the non-safety-related IA piping could reduce the air inlet pressure to just low enough to affect proper operation of the actuator's 3-way and 4-way pilot valves for the Main Feedwater Preheater Bypass Isolation Valves, and not be detected by the pressure switches in the main header of the Instrument Air System. Refer to Shearon Harris LER 40098001 in Table 2. If this occurred, the pilot valves would shuttle, causing the accumulator pressure to bleed off, which would prevent the valves from closing as required. Operations personnel would have no indication of accumulator low pressure other than local observations made by an auxiliary operator and possibly dual valve indication in the main control room due to the valves cycling slightly. This potential scenario was reported to the NRC via the emergency notification system on January 9, 1998. Refer to LER 40098001.

9.4 Problems with Copes-Vulcan Pressurizer Power-Operated Relief Valves (PORVs)

The NRC issued Information Notice 94-55 to describe design-related problems involving cracking of plug material and severe wear of plugs and cages in Copes-Vulcan PORVs.

On April 7, 1994, at the Salem Generating Station, Unit 1, a reactor trip was followed by two automatic actuations of the safety injection (SI) system. The continued injection of water from the safety injection system filled the pressurizer steam space with subcooled water and, without the normal pressurizer steam space to dampen pressure excursions, resulted in repeated actuation of the plant PORVs to limit reactor coolant system pressure. Salem Unit 1 has two pressurizer PORVs, each of which is actuated through separate automatic controls. During the event, one PORV (1PR-2) cycled at least 200 times, and the other PORV (1PR-1) cycled at least 100 times.

The PORVs are 2-inch, air-operated valves manufactured by Copes-Vulcan that have a plug and cage-type internal trim design. The Copes-Vulcan PORV design used at Salem Unit 1 has a plug which is guided by a cage. Close clearances exist between the outside diameter of the plug and the inside diameter of the cage. When the valve opens, fluid in the system flows from under the plug through several equally spaced ports in the cage, and then to the valve outlet. When the valve closes, the plug seats against a machined surface of the cage. The cage is positioned on a gasket in a close-clearance counterbore in the valve body. The stem transmits the motive force from the air actuator to the plug and is threaded into the plug. A steep taper section on the stem just above the threaded section produces a wedging action at the relatively thin pinning boss. The valve stems were made of Type 316 austenitic stainless steel, and the plug and cage are both made of Type 420 hardened martensitic stainless steel.

The licensee inspected the internal components of the PORVs and discovered three deficiencies: (1) scoring in the plug and cage area (both valves); (2) axial cracking on the pinning boss through which the anti-rotation roll pin passes (both valves); and (3) galling on the stem where it passed through the bonnet (1PR-2 only). The licensee believed the scoring found on the plug and cage of 1PR-2 was the result of out-of-tolerance machining of the inside diameter of the cage and the increased thickness caused by deposition of material as the scoring occurred.

The licensee determined that the cracks in the pinning boss were caused by intergranular stress corrosion cracking (IGSCC). The extent of cracking was similar for both 1PR-1 and 1PR-2. There was no evidence that fatigue contributed to the failure. The licensee found similar cracking, though less prominent than the cracking in 1PR-1 and 1PR-2, on internal components of valves maintained in the warehouse as new spare valves. The licensee performed stress and fracture mechanics analyses to evaluate the stress condition in the valve plug and to assess the potential for additional crack growth. These analyses indicated that differential thermal expansion of the stem and plug materials causes significant stresses in the pinning boss. In addition, because of the steep taper wedging action in the stem-to-plug assembly, high stress concentrations are found in the vicinity of the anti-rotation pin hole of the plug boss. The licensee determined that continued crack growth further into the much heavier plug itself is possible, if the plug is left in service.

To prevent further problems with the internal components made of Type 420 stainless steel, the licensee installed plugs made of Type 316 stainless steel overlaid with Stellite and cages made of 17-4 PH stainless steel in the Unit 1 PORVs. The design of the replacement plug eliminates the pinning boss used for the Type 420 stainless steel plug, thus eliminating the tendency for crack formation in the thin boss section. The stem is now pinned to a thick section of the plug rather than through the relatively thin boss, and the plug height has been increased (to account for the elimination of the boss) to provide the same stroke length as before. The licensee determined that the Unit 2 PORVs did not require modification because their plugs and cages are made of 17-4 PH stainless steel.

The manufacturer determined that Salem was the only nuclear power facility that installed the Type 420 stainless steel internal components in Copes-Vulcan pressurizer PORVs. However, according to the manufacturer, there may be other Copes-Vulcan valves in nuclear service utilizing Type 420 material for the internal components, and these components could experience similar problems.

In summary, NRC Information Notice 94-55 raised questions with respect to: (1) the use of Type 420 stainless steel in this or similar valve applications; (2) valve component misalignment during field assembly; (3) out-of-tolerance machining on Copes-Vulcan PORVs; and (4) high stress concentrations because of the design of the internal components of Copes-Vulcan PORVs.

Additional problems related to air actuators and air supply regulators in Copes-Vulcan PORVs were reported in NRC Information Notice 95-34, dated August 25, 1995. During an inservice testing surveillance at the Haddam Neck nuclear power plant on February 19, 1994, both pressurizer PORVs failed to open fully on demand. The cause was leaks in the air actuator assemblies of both PORVs. Reduced pressure output of the control air regulators compounded the problem. The Haddam Neck PORVs are 2-inch nominal size, air-operated plug valves manufactured by Copes-Vulcan (Model D-100-160).

The licensee determined that the primary cause of the unacceptable valve stroke performance was air leakage from the PORV air actuators caused by improper installation of the diaphragms. Both PORV diaphragms had been replaced by a new style during a 1993 refueling outage. The principal difference in the replacement diaphragms was a change in the material composition. The replacement diaphragm was made of EPDM while the old style diaphragm was made of Buna-N. The licensee switched to the EPDM diaphragms because of a vendor recommendation that EPDM would provide enhanced performance under the temperature and radiation conditions experienced by the PORV. In addition, the EPDM diaphragms have a 24-bolt hole configuration while the old style Buna-N diaphragms only have a 12-bolt hole configuration. The diaphragms also have slightly different shapes.

The licensee apparently had some difficulty installing the EPDM diaphragms because of the bolt hole pattern and shape differences between the EPDM and Buna-N style diaphragms. The licensee believes that the sealing surfaces of the diaphragms were damaged as a consequence of the installation difficulties. Extrusion of the diaphragm from between the base and cover and away from the bolt holes led to small tears at several diaphragm bolt holes locations which ultimately resulted in the air leakage. The licensee chose to use the Buna-N diaphragms to avoid the installation difficulties encountered with the EPDM diaphragms. Copes-Vulcan indicated that they have no reports from other users on installation difficulties of either type of diaphragm.

According to IN 95-34, numerous air-actuator related problems were reported within the nuclear industry regarding this model of Copes-Vulcan valve. The problems can be grouped into three categories: (1) actuator air leaks resulting from in-place diaphragm failures (e.g., holes, rips, and tears); (2) actuator air leaks resulting from loose actuator cover bolts; and (3) valve stroke malfunctions resulting from improper supply of air pressure from the air regulating valves.

The air pressure regulators used at Haddam Neck are ITT Conoflow and contributed to the PORV problems on three separate occasions. In 1993, one of the air pressure regulators failed high, subjecting one of the PORV diaphragms to the full air supply line pressure of 120 psig which is greater than the diaphragm design pressure of 100 psig. Although the PORV actuators are equipped with relief valves to protect the diaphragms from over pressurization, it is believed that the high pressure contributed to or caused premature failure of the diaphragm. In the other two instances, the air supply regulator setpoints had drifted low, resulting in inadequate stroke performance of the valves. The air pressure regulator set points for the Haddam Neck configuration are 85 psig. The PORVs need 65 psig to start opening and 85 psig to open fully. An engineering evaluation by the licensee showed that the valves will come to the full open position with control air pressure reduced to 70 psig and reactor coolant system pressure as low as 840 psig.

Several postulated causes of air pressure regulator set point drift were described in IN 95-34. One was that moisture intrusion from the control air system could cause corrosion of the regulating mechanism; another was that the drift might be configuration related. At Haddam Neck, the air regulating valve is upstream of the (normally closed) solenoid operated valve, meaning that the air regulating valve is constantly subjected to system pressure. The air regulating valve vendor indicated that this configuration may cause set point drift.

9.5 Air-Operated Dampers

Dampers are used in such vital services as control room ventilation and to direct sources of combustion air to emergency diesel generators. Air-operated dampers may be piston-operated with the pistons powered from instrument air or accumulators and SOVs may serve as pilot valves to direct air to the pistons. Alternatively, SOVs may be used to power dampers directly.

Some or all air-operated dampers may not be included in the licensees' AOV programs. For example, Palisades specifically excluded HVAC dampers from the scope of their AOV program. Other licensees may not explicitly state specific exclusions in their program scope; however, their lists of AOVs may not include dampers. Motor-operated dampers are not included in the scope of the Generic Letter 89-10 program.

In the visits to the plants, it was noted that dampers and their operators are usually maintained by the technicians in charge of the particular system in which the dampers are located. This appears to be an approach that could result in nonuniform maintenance or incomplete failure analyses because of the diversity of personnel and concentration.

Several events were noted where air-operated dampers with safety-related and non-safety-related functions may be served by non-safety-related systems such as instrument air without adequate accumulator backups. Interruption of the non-safety-related service would cause interference with the safety functions. A variation of this type of problem is that some safety-related functions of air-operated dampers have been performed by non-safety-related air-operators or SOVs. Some licensees attributed these conditions to design error, inadequate maintenance, or inadequate surveillance testing.

NUREG-1275, Volume 2, includes (page 2-1) descriptions of events at Brunswick, H. B. Robinson, and Cooper, involving potential loss of emergency diesel generators in the event of a loss of offsite power. The loss of offsite power could have resulted in a loss of instrument air which would cause the dampers that serve the emergency diesel generator rooms to close, and subsequently would cause overheating of the diesel generator controls.

By Generic Letter 88-14, dated August 8, 1988, the NRC directed licensees to perform design verifications. Some deficiencies were found as a result but similar conditions were recently discovered. Selected LERs that describe safety-related events or deficient conditions involving air-operated dampers are listed in Table 3.

9.6 AOVs and SOVs Used in Services Related to Emergency Diesel Generators

AOVs and SOVs are used extensively in emergency diesel generator service in nuclear power plants. For example, cooling water for the emergency diesel generators is controlled by AOVs at several plants. SOVs are used as pilots for a number of AOVs mounted directly on the diesel engines.

In general, AOVs and SOVs that are mounted on the diesel generators at the plants visited were maintained by the technicians assigned to the diesel generators. It was observed during the visits that the operability of the valves might be improved if responsibility for maintenance of these valves were assigned to personnel specifically concerned with AOV and SOV maintenance.

NUREG-1275, Volume 2, subsection 5.1.7 and its Appendix B, include detailed descriptions of a number of problems and events related to air service to the AOVs and SOVs that supply diesel generators.

Water in the instrument air caused sticking of an air-operated radiator exhaust damper for Diesel Generator No. 2 at Ft. Calhoun (LER 28587025). Shortly thereafter, it was discovered that Diesel Generator No. 1 had the same degraded condition. If a loss of off site power event had occurred while the DG radiator damper AOVs were inoperable, both DGs would have run to destruction because the high coolant temperature shutoff would have been bypassed during a real demand. This event was analyzed under NRC's ASP program, categorized as a precursor event, and estimated to have a conditional core-damage probability of 6.2E-04, which ranked it as the second highest event of 33 precursor events identified for that year.

Several diesel generator events involving SOVs were described in NUREG-1275, Volume 6, including a simultaneous common-cause emergency diesel generator failure (Section 5.2.1.2 therein) that was traced to deterioration of the elastomer materials used in the SOVs.

A recent example of an SOV design deficiency that was responsible for common-cause inoperability of several diesel generators at several nuclear plants involves SOV springs which were redesigned to solve one problem (air leakage in the SOV) and, as a result, did not meet the minimum requirements for combined voltage and air pressure to actuate the solenoids. Refer to LERs for Point Beach (LER 26698008) and Clinton (LER 46198009) that reference Engine Systems Inc. 10 CFR Part 21 Notification No. 1998120, dated January 26, 1998 (NRC Accession Number 9802020122). The conditions as described in the Part 21 notification were:

- "MKW Power Systems, Inc. (former name of Engine Systems, Inc.) issued Report No. 10CFR21-0055 on June 13, 1990 which provided a 10CFR21 reportable defect with the air start solenoid valve commonly used on EMD diesel generators (EMD #9513134/ Graham-White #712-051). The valves exhibited air leakage when maximum allowable inlet air (200 psig) was applied. This would not prevent the engine from starting; however, the leakage may cause the air compressor(s) to constantly cycle. After consulting with the valve manufacturer (Graham-White), MKW determined that a 275 psig valve was also available. The 275 psig valve was identical to the 200 psig valve except that it had a different spring. MKW made the recommendation to replace the valve's spring with a different spring which was rated for 275 psig. MKW also discontinued sale of the 200 psig valve for nuclear service and recommended the 275 psig valve as a replacement. The 275 psig valve is Graham-White #712-065 (without mounting bracket) or #712-015 (with mounting bracket)."
- "Engine Systems, Inc. (ESI) has recently learned that the 275 psig valve does not

> meet the minimum DC voltage requirement of most nuclear applications when inlet pressures below 200 psig are applied to the valve. The solenoid valve relies on system air pressure to assist the coil in overcoming the force of the spring and thus operating the valve plunger. The reduced system pressure combined with the reduced coil voltage results in the inability of the valve to operate satisfactorily. This problem is not applicable to the original 200 psig valve because it has a weaker spring and therefore less air pressure is needed to overcome the lower spring force."

The SOVs control air flow from the storage tanks to the diesel starting air system. There were 335 SOVs, distributed among about 17 U.S. nuclear plant owners, listed in the Part 21 notice; however, only the two LERs noted above, which refer to these conditions, were found.

9.7 Establishment of and Conformance with the Design Basis for AOVs at Millstone 3

Mistakes in establishing the original plant design basis and/or mistakes in designing equipment in accordance with the plant design basis were reported in several LERs involving AOVs and the air systems that support AOVs at Millstone 3 in 1996. LERs 42396013, 42396028, 42396031, 42396036, and 42396040, as outlined in Table 2, described these events. A NRC Combined Inspection Report, "NRC Combined Inspection Report 50-245/98-206; 50-336/98-206; 50-423/98-206, and Notice of Violation," dated May 26, 1998 (NRC Accession No. 9806030375), summarized the resolution of the issues and the plant modifications resulting from the events and conditions. The events described below are considered to be common-cause failure conditions.

LER 42396013 described an original plant design deficiency where the Residual Heat Removal System (RHS) was outside the design basis of the plant. A loss of control air supplied from the non-safety-related instrument air system could cause the RHS control valves to fail open. If this condition occurred during the initial phase of a plant cool down, the Reactor Plant Component Cooling Water System (CCP) temperatures could exceed 125°F. This is the design temperature used in the system stress analysis. If RHS heat exchanger operation was initiated at 350°F RCS temperature, as assumed, then the RHS heat exchanger CCP outlet temperature could be as high as 250°F if the valves failed open. Under the resultant conditions the CCP piping would not meet the ASME Section III, Appendix F stress criteria. The original plant design did not consider that if the RHS flow control valves failed open on a loss of control air, it could create unacceptably high RHS heat exchanger CCP discharge temperatures.

LER 42396028 described a failure scenario in which a loss of instrument air to temperature control valves in the Charging Pump Cooling (CCE) system serving the charging pump lube oil coolers, coincident with 33°F Service Water (SWP) temperature could result in overcooling of both trains of the charging pump lube oil system and challenge charging pump operability. Failure of the air-operated CCE valves to the full open position due to a loss of the non-safety related IAS system would adversely affect both trains of the charging pumps by allowing excessive cooling of the CCE system, which cools the lube oil system. This condition alone could have prevented the fulfillment of the safety function of the system. The cause of the charging pump inoperability was inadequate original design. This condition would result from overcooling of the lube oil system from a failure of the non-safety related instrument air system coincident with a worst case minimum SWP temperature and maximum flow and heat exchanger cleanliness. Under these conditions, the air-operated CCE valves would fail open and excessive cooling of the lube oil system would occur. This particular combination of conditions was not considered in the initial design.

LER 42396031 described a condition where specific safety related control valves could fail due to exceeding the manufacturer's (ASCO) maximum operating pressure differential pressure (MOPD) rating of the SOVs installed on the control valves in a number of safety-related systems. SOVs could fail to perform safety related functions because of excessive operating pressure differentials. This can result from failures of non-qualified air regulators installed in the instrument air system upstream of the SOVs. The failure of an air regulator would, in turn, result in full IA system pressure being applied to the SOV. The SOV can potentially fail to operate properly since they are not rated for full IA system pressure. A total of 48 SOVs which perform safety-related functions had been originally identified in various systems which would be susceptible to such a failure. The cause of this condition was the failure to consider the potential for pressure regulator failure in the original design and selection of SOVs.

LER 42396036 described a scenario where the High Pressure Safety Injection (SIH) and Low Pressure Safety Injection (SIL) systems would have been subject to degraded performance due to possible mispositioning of normally closed safety related air operated valves. Mispositioning of these 21 valves can be postulated to occur under post-accident harsh environmental conditions due to failure of non-qualified power and control circuits. As a result, the potential diversion of SIH and/or SIL flow under accident conditions could have been more than the margin allowed within the Loss of Coolant Accident analysis. Seventeen additional safety-related air operated and solenoid operated valves were subsequently identified where failures of non-qualified control circuits could degrade performance of a safety system function. These additional valves are located in the following systems: Reactor Plant Component Cooling Water (CCP), Containment Vacuum, Reactor Plant Sampling (SSR); Post Accident Sampling (SSP); and Main Steam to the auxiliary feedwater steam turbine. The cause of the reported conditions was a design mistake. The initial plant design did not adequately consider the potential mispositioning of these valves under harsh environmental conditions or active failure.

LER 42396040 described a scenario where a loss of the non-category 1 Instrument Air System (IAS) would allow CCP valves to reposition to a maximum cooling configuration. Coupled with a low heat load and minimum Service Water (SWP) inlet temperature, the CCP system could reach temperatures lower than values for which they are analyzed, thereby rendering the CCP system, and systems that it serves, potentially inoperable. The CCP system temperature is controlled by three-way valves, each comprised of a single pneumatic operator and a mechanically linked pair of butterfly valves located at the outlet end of each of the three CCP heat exchangers. Each pair of valves, a heat exchanger (through-flow) outlet control valve and a heat exchanger bypass valve, are mechanically linked such that when one valve is fully open, the other is fully closed. This design ensures adequate heat removal under the high heat load conditions assumed in the design basis analysis. However, if the non-safety-related IAS system that operates the three-way CCP temperature control valves were lost, the valves would fail in a configuration that would result in maximum CCP system cooling (i.e., the heat exchanger bypass would fully close and the heat exchanger outlet control valve would fully open). This maximum cooling, if coupled with low CCP heat loads (experienced during an extended plant shutdown and very low SWP temperatures (during winter seasonal conditions), could result in CCP system overcooling. Such an event could cause temperatures to decrease below analyzed limits, thereby rendering the CCP system, and systems that it serves, potentially inoperable. The root causes of this condition were:

- Improper initial design of the CCP system. A review of the CCP design basis and system-related correspondence revealed that the described failure mode described was a design oversight. The design basis analysis for the CCP system focused on high CCP heat load conditions. The designers did not analyze for extremely low CCP heat loads concurrent with very low SWP temperatures.

- Inadequate review of industry and plant operating experience evaluations associated with the CCP system and loss of the

IAS system. Since original design implementation, there have been both industry-wide and plant events dealing with these systems and their interfaces. Many issues dealt specifically with system temperature problems and these should have been investigated.

10. AIR SYSTEMS OPERATING EXPERIENCE

10.1 Air Systems

Air systems in nuclear plants can be a source of common-cause failures in AOVs if the air provided is not clean, dry, oil-free, and furnished at specified pressure. Also, if air supplied to AOVs is once contaminated, the potential for AOV problems is not necessarily eliminated once the quality of the air has been restored. Residues of moisture in the systems can generate AOV problems both directly and over time, from corrosion products that may form and be released into the system.

NUREG-1275, Volume 2, "Operating Experience Feedback Report - Air Systems Problems," was published in December 1987. In the NUREG report, the AEOD staff described air system problems caused by moisture contamination as well as hydrocarbon and desiccant contamination, among other problems. These findings led directly to the NRC issuing

Generic Letter (GL) 88-14, "Instrument Air Supply system Problems Affecting Safety-Related Equipment," in August 1988. In GL 88-14, licensees were asked to verify the quality of instrument air and discuss their programs for maintaining instrument air quality. This verification was to include:

1. Verification by test that actual instrument air quality is consistent with the manufacturers' recommendations for individual components served.

1. Verification that maintenance practices, emergency procedures, and training are adequate to ensure that safety-related equipment will function as intended on loss of instrument air.

2. Verification that the design of the entire instrument air system including air or other pneumatic accumulators is in accordance with its intended function, including verification by test that air-operated safety-related components will perform as expected in accordance with all design-basis events, including a loss of the normal instrument air system. This design verification should include an analysis of current air operated component failure positions to verify that they are correct for assuring required safety functions.

In addition to the above, each licensee/applicant should provide a discussion of their program for maintaining proper instrument air quality." Note that recommendations for periodic verification of air system operability and air quality were not specifically included in GL 88-14.

As noted previously, air systems (including backup nitrogen systems) are supposed to provide clean, dry, oil-free air, at specified pressure, to AOVs. All seven of the licensees visited have, in the past, encountered numerous AOV problems caused by moisture in the air supplied to their AOVs. Six of the seven licensees took aggressive actions to improve the moisture reducing performance of their air systems. Among the actions taken were the installation of additional or more reliable air capacity, installation of redundant dryers, installation of additional air filters for the system and for individual AOVs, installation of automatic drains, and installation of continuous dew point monitoring and alarms.

The history and current condition, as of the time of our visit, of the instrument air system at one of the sites visited demonstrates the importance of supplying dry and clean air to valves and instruments. Two of the events noted in Section 5.1.1 of NUREG-1275, Volume 2, concerned the failure, in 1978 and 1981, of a shutdown cooling system heat-exchanger outlet AOV, CV-3025, to open while the reactor was shut down. LERs 25578003 and 25581030 refer to these events. CV-3025 has a non-safety-related decay heat removal function and is required to be opened after the reactor is depressurized. CV-3025 also has a safety-related functional requirement to open after a small-break loss-of-coolant accident (LOCA) in order to provide for shutdown cooling. This AOV is also intended to be throttled to adjust cooling capacity after a small-break LOCA, but the licensee considers this function to be beyond the design basis for the plant. Note that CV-3025 is powered from the non-safety-related IA system. A handwheel was added after the 1981 valve failure to allow operators to open the valve manually.

Water was present in the IA system at this plant because of improper dryer operation in each of these events. In each event, the water filled a positioner and caused the valve to fail closed. Finally, in each event, the primary coolant temperature approached boiling before operators took alternative actions to remove decay heat. This recurring single failure, in a non-safety-related mode for the valve operation, could have resulted in uncovering the reactor core had the conditions not been mitigated by other operator actions within a few hours.

As of the time of our visit, these events were still significant because the quality of instrument air is compromised by the periodic introduction of moisture into the system. There was, at the time of the visit, only one dryer available for the IA system. As a result, no air drying capability is available when the dryer must be taken off line. This, almost certainly, is causing contamination of the air system from moisture, and subsequently, as a direct result of the presence of moisture in the piping, from corrosion products which can be dislodged from the pipe or valve internals. According to the licensee, air quality directly downstream of the dryer is monitored and moisture content is estimated by direct observation on a daily basis of changes in the color of desiccant crystals. However, such readings are meaningless if the dryer is bypassed. The readings do not alert operators to downstream moisture that would be introduced when the dryer is bypassed.

In addition to the instrument air system, the plant has a high-pressure air system that serves "vital" valves. The air for this system is supplied by three compressors, each with a refrigerant-type dryer that cannot dry the air to a satisfactorily low dew point. The dryers do not operate reliably, i.e., the air compressors for the high-pressure air system must be cycled on and off in order to prevent these refrigerant dryers from freezing up. In April 1997, nine air regulators in the high-pressure air system were found to be contaminated, and it is concluded that this contamination resulted from moisture intrusion because dry air was not provided by this system. This condition was not reported in an LER.

It was also noted that several equipment air filters were, as of the time of the visit, installed downstream of equipment rather than upstream of the components they were designed to protect from particulates. Further, during a more recent NRC Regional inspection, several gallons of water were found at a low point in the air line. The licensee had not installed a low-point drain to remove water from the high-pressure IA system. The licensee committed to do this in response to GL 88-14, but the modification had not been implemented as of the time of the visit for this study. The plant staff prepared plans to improve the air systems but these had not been implemented. They did not make modifications to the air dryers because there were no licensing requirements applicable to them. This is an older plant which was operating before the existence of the ISA 7.3 (1975) instrument air standard.

A somewhat more satisfactory situation regarding air quality was found at Turkey Point during the recent visit for this study. In 1985, Turkey Point 3 and 4 experienced recurring failures in AOVs in the auxiliary feedwater system (LER 25085021) caused by moisture or moisture contamination of the IA system. The events are described in Section 5.1.2 of NUREG-1275, Volume 2. See Section 8.7.1 in this report, as well. Water had been allowed to condense and remain in the instrument air system, subsequently producing corrosion particles which then migrated, along with the moisture, to operating valves and valve control equipment. In addition to the AFW systems for both units, other safety related systems which could have been affected included steam dump to atmosphere, salt water flow to the essential heat exchanger, charging system, and the residual heat removal system.

The Turkey Point units are now equipped with a full-capacity IA system, continuous digital-readout dew point monitors, dew point alarms (although not in the control room), drain traps, and enhanced filtering arrangements. Each unit is equipped with two "double-capacity" compressors (one motor-driven and one diesel-driven), each of which is hard-piped to serve both plants. In addition, the IA dryers are connected to the emergency bus, thereby ensuring a reliable flow of clean, dry air during design basis accidents or events, including loss of offsite power. Recent AOV performance indicates that these measures have been effective in preventing AOV failures from contaminated air.

All of the plants visited took actions to improve the performance of their air systems in response to in-plant events and the recommendations in GL 88-14, although the individual responses and results varied considerably from plant to plant, as indicated above. Modifications included installation of additional air production capacity, additional parallel dryer trains, installation of additional filters, and more attention to preventive or corrective maintenance of the air system.

Palo Verde, Fermi 2, and Palisades do not have continuous dew point monitors in their air systems. They rely heavily on observations of blown-down air to evaluate moisture contamination. TMI-1, Indian Point 3, and Turkey Point have each installed continuous dew point monitors. LaSalle was originally designed with a dew point monitoring system, including alarms. LaSalle, TMI-1, and Indian Point 3 have installed dew point alarms in the control room. Indian Point 3 and Turkey Point 3 and 4 each have installed automatic drain traps on their air receivers. The automatic drain traps have proved to be effective in removing moisture from the air prior to processing by the dryers.

Three of the seven licensees visited do not definitively know, on a day-to-day basis, if the quality of air (or nitrogen) in their plants is acceptable or has degraded with regard to moisture contamination. This could adversely affect the operation of important and safety-related AOVs. LaSalle, TMI, Indian Point 3, and Turkey Point 3 and 4, are equipped with continuous dew point monitoring and alarming of the instrument air systems. The alarms at Turkey Point are local, whereas the alarms at the other three stations are located in the control rooms. Note that these plants also have other air systems containing AOVs of interest that are not equipped with devices to measure dew point. The licensees without continuous monitoring devices check moisture content in their air systems by periodic observation ranging from taking readings at each shift (on rounds) to testing during outages. The extent of particulate contamination at all of the plants visited is estimated from examination of filters and ranges from actual measurement of particle number and size to casual examination of the filters.

Recent operating experience indicates that moisture contamination of air systems in nuclear power plants occurred since GL 88-14 was issued. Some examples are:

- At Beaver Valley 1 in 1990 (LER 33490007), moisture in the IA system caused the failure of feedwater regulating valves and a subsequent reactor trip.
- At Dresden 2 in 1994 (LER 23794005), an air line ruptured because of corrosion in an IA line and this led to a reactor trip.
- During a test of the pressurizer PORVs at Haddam Neck in 1993 (LER 21393007), a failure in the PORV air supply regulating valve was traced to moisture intrusion in the containment control air system due to a faulty air dryer.
- Similarly at Haddam Neck, a letdown AOV failed in 1993 (LER 21393005), because of moisture in the containment control air system.
- In 1992 (LER 32792018) at Sequoyah 1, about 1000 gallons of water was found to have been entrained in the non-essential air system that serves the essential air system. This caused malfunction of the feedwater regulating valves and resulted in a trip.

- In 1997 (LER 32797012) at Sequoyah 1, loss of control air header pressure was attributed to corrosion products in the lines (residue from the moisture intrusion event which had occurred about 5 years earlier).

In several of these events, air was provided to the equipment. However, the air was contaminated with moisture and/or corrosion-products which caused the failures. The interest in air systems in this study is directed toward both the quantity and quality of air supplied to AOVs. If the air supply is not clean and dry, its source cannot be considered reliable regardless of how much air can be provided. IA systems that are not kept clean and dry are sources of common-cause failures of AOVs, both immediate and delayed (See NUREG-1275, Volume 2). As discussed elsewhere in this study, air must be supplied at constant pressure (within specified limits) in order to ensure reliable operation of AOVs and SOVs.

The effects of moisture contamination can be immediate, i.e., the moisture itself causes AOVs or instrumentation to fail. The effects of moisture contamination can also be delayed, i.e., corrosion or contamination products form or build up and cause a number of failures or events. While the long-term hazards of moisture in air lines are not readily predictable, they are certainly detrimental to equipment operability and the hazards may increase or continue to exist long after any one or several particular problems are discovered and corrected. Similar conclusions were noted in NSAC-128, "Pneumatic Systems and Nuclear Plant Safety," in 1988.

NUREG/CR-5472, "A Risk Based Review of Instrument Air Systems at Nuclear Power Plants," included an analysis, from a risk perspective, of the data presented in NUREG-1275, Volume 2, and LERs available between 1980 and 1989. Among the observations in that study, Section 2.4.2 indicated that "(I)nstrument air contamination predominates as a root cause and air-operated valve failure predominates as a direct cause in instrument air common cause events." Other conclusions and data in NUREG/CR-5472 are discussed in Sections 15.1 and 15.4.

A previous NRC report, NUREG/CR-2796, "Compressed Air and Backup Nitrogen Systems in Nuclear Power Plants," focused primarily on loss of instrument air as a source of common-cause failure of AOVs, and concluded that the air systems were reliable in terms of the ability to provide the necessary quantities air. This corresponds to the observations made from the plant visits. Air capacity appears to be plentiful and redundant. As far as the issue of reliably providing air at the necessary pressure is concerned, there have been a number of instances where the failure of a pipe, header, or piece of equipment caused an air leak that lowered pressure locally, or even generally for a short time, but this was not considered of particular concern for the non-safety-related air system, especially when a safety-related event did not occur at the same time.

It must be noted that until NUREG-1275, Volume 2, was published and GL 88-14 was subsequently forwarded to all licensees, the assumptions made by licensees and used in air system studies (including risk studies) concerning adequacy of backups to instrument air (accumulators), as well as intended AOV failure positions, were unverified. Licensees were asked to verify that their plants were capable of responding safely to a loss of (instrument) air, that valves would respond as intended, and that accumulator capabilities were acceptable. Questions still remain (in some of the plants we visited) about the effectiveness of ongoing activities to protect important AOVs against partial or gradual air pressure losses and the effects of contaminated air.

10.2 Accumulators for AOVs

Accumulators are used to provide an air supply to AOVs which are required to function if the normal air supply is lost. This strategy is commonly applied to safety-related AOVs so that credit is not needed for a non-safety-related instrument air system. Essentially, if an accident analysis indicates that air must be supplied to operate a valve, an accumulator is provided.

A number of failures involving accumulators, check valves, and piping are recorded in Table 2 and include the following:

- LER 44595005 at Comanche Peak;
- LER 34688007 at Davis Besse;
- LER 24993005 at Dresden 3;
- LER 33191005 at Duane Arnold;
- LER 28588028 at Ft. Calhoun;
- LER 33689011 at Millstone 2; and
- LER 27093002 at Oconee 2.

In addition to hardware failures, several instances were identified where the capacity of the accumulators was found to be insufficient to meet design basis demands. These include the following:

- LER 25085020 at Turkey Point; and
- LER 29389002 at Pilgrim.

In these last two examples, the design basis demands for the AOVs and/or the margins to meet those design basis demands were not correctly established.

Malfunction (sticking open or closed) of the accumulator check valves associated with particular valve operators can be a significant problem. Note that accumulators, along with the accumulator check valves, are sometimes assumed to be piece parts of a safety-related AOV. First, the loss of air may not be readily apparent and the valve would then fail when operation is attempted. Second, a partial loss of air pressure could cause the valve to fail in an intermediate position or not fail safe if the springs are not properly designed and installed.

Another potential problem involving accumulators was noted during the visits for this study. Most of those encountered have no drains in the bottom and no means of inspection without disassembly of the piping. This condition is typical at many plants. Further, there was no regular inspection program in the plants we visited to determine conditions within the accumulators. Some accumulators have been in service for many years.

Accumulators, by the nature of their design and location in the air systems, are natural drain traps for moisture and/or particulate contamination. The possibility exists for contamination of the valve(s) served by an accumulator if there is moisture or particles (corrosion products for example) in the accumulator. Plants that have had problems with moisture in the air system at one time or another (most plants) could have either moisture in the accumulators, or corrosion products resulting from previous moisture contamination, or both, absent objective evidence (inspection results) to the contrary. The safety-related functions of the AOVs served by the accumulators could be compromised.

Another problem associated with the lack of drains in accumulators is the possibility that the accumulators may trap an appreciable volume of water, thus decreasing the volume of air available to the valve served. If the volume of the accumulators is less than called for in the design bases, the AOVs may not be able to stroke the number of times required (as assumed in the plant safety analyses).

11. SOLENOID OPERATED VALVE (SOV) OPERATING EXPERIENCE

As noted in Section 5, characteristics of SOVs pertinent to this study include:

- Exceeding the maximum operating pressure differential (MOPD), which is the pressure difference between the inlet port and the outlet port, may cause the SOV to spuriously open or not open, depending on the SOV's design.

- Small orifices make the SOVs subject to interference from small contamination particles or moisture intrusion.

- The design logic of SOV operation and control can be complex and this is a source of design or installation mistakes.

- Many piloted SOVs require a minimum operating pressure differential (Min OPD) to operate properly.

- Many SOVs will not function properly when subject to reverse pressurization or flow in the valve.

- Materials used to construct the valve bodies, internal parts, seals, O-rings, etc., may be subject to binding, creep, corrosion, erosion, and/or environmental deterioration.

- Rather small forces, in the range of 10 pounds or less, are produced by a solenoid to operate an SOV; therefore, opposing forces of similar small magnitude can interfere with successful operation.

- SOVs can be damaged if subjected to higher-than-designed pressure in the operating fluid.

- SOVs are subject to damage from the use of thread locking compounds in adjacent pipe or tube connections that tend to migrate into the working parts of the SOVs.

- Elastomers and lubricants used in SOVs have been, and continue to be, a source of many SOV operational problems.

One of the pertinent observations reported in NUREG/CR-4819, Volume 1 (page 25 therein), was that "SOVs used in nuclear power plants were originally designed for industrial applications where an energize-to-operate philosophy is prevalent. In the industrial applications, the SOV is normally maintained in the de-energized state, resulting in the coil and elastomers being maintained at room-ambient or process fluid temperature. The nuclear safety philosophy of returning to a fail safe condition on loss of air or electrical power can result in a valve being maintained in an energized state throughout most of its installed life. The consequence of the continuously energized SOV is more rapid degradation of the elastomers and solenoid coils because of the long periods at elevated temperatures. Although SOV manufacturers have considered the (continuously energized state) in design changes and material changes (to) their products, the continuously energized SOV will generally undergo more rapid degradation than an SOV used mostly in a de-energized state."

Pertinent reports that described various aspects of the design, operational, or testing problems involved in the use of SOVs include:

- NUREG-1275, Volume 6, "Operating Experience Feedback Report - Solenoid-Operated Valve Problems;"

- NUREG/CR-3424, "Equipment Qualification Research Test Program and Failure Analysis of Class 1E Solenoid Valves;"

- NUREG/CR-4819, Volume 1, "Aging and Service Wear of Solenoid-Operated Valves Used in Safety Systems of Nuclear Power Plants, Operating Experience and Failure Identification;"

- NUREG/CR-4819, Volume 2, "Aging and Service Wear of Solenoid-Operated Valves Used in Safety Systems of Nuclear

Power Plants, Evaluation of Monitoring Methods;"

- NUREG/CR-5008, "Development of a Testing and Analysis Methodology to Determine the Functional Condition of Solenoid Operated Valves;"
- NUREG/CR-5141, "Aging and Qualification Research on Solenoid Operated Valves;"
- NUREG/CR-5292, "Closeout of IE Bulletin 80-23, Failures of Solenoid Valves Manufactured by Valcor Engineering Corporation;" and
- NUREG/CR-6246, "Effects of Aging and Service Wear on Main Steam Isolation Valves and Valve Operators."

SOV failure causes and mechanisms discussed in NUREG-1275, Volume 6 include:

- High ambient temperatures;
- Heatup of the SOV from being continuously energized;
- Subjecting SOV to pressure or flow in a direction not designed for;
- Misapplication of materials (metallic or organic) in design or replacement;
- Over pressurization;
- Incorrect wiring;
- Incorrect electric current or voltage;
- Inadequate preventive maintenance;
- Mistakes made in rebuilding SOVs;
- Misapplication of lubricants;
- Misapplication of thread sealants;
- Contamination (moisture and/or particulate);
- Inadequate surveillance testing;
- Not recognizing and treating SOVs as (internal) piece parts of AOVs or other equipment; and
- Use of unqualified SOVs.

SOVs are sources of common-cause failures in AOVs. SOV-related root causes of such common-cause AOV failures were described in NUREG-1275, Volume 6 and include:

- Incorrect specification of operating parameters such as maximum operating pressure differential (MOPD) or Min OPD;
- Incorrect specification of valve orientation with respect to fluid flow;
- Incorrect material selection for the operating environment;
- Incorrect specification of environmental (temperature, radiation, moisture) conditions;
- Incorrect estimate of expected service life when coil is continuously energized;
- Failure to replace worn parts or parts with limited service life in time;
- Rebuilding or re-assembling SOVs incorrectly;
- Failure to maintain the supply of clean, dry, oil-free, instrument air at required pressure;
- Excessive lubrication of SOV internals;
- Installing SOVs in an incorrect orientation;
- Failure to provide correct electric current or voltage;
- Inadequate or incorrect electrical connections; and

- Manufacturing defects such as lubrication mistakes, defective materials, and assembly mistakes.

A search of the LER database returned 739 LERs since 1985 that involved SOVs. A widely used industry SOV application guide indicates that there were 2350 SOV failures cataloged in the nuclear industry database between 1985 and 1990. Several pertinent LERs involving failures of SOVs, which were reviewed as part of this study, are listed in Table 2. Refer to Section 9.6 for a description of a recent common-cause failure condition in diesel generator air-start SOVs.

One recent Event Report (#34262, dated 5/19/98, see Table 4) from Susquehanna described a 10 CFR Part 21 Notice by Valcor Corporation regarding their Model V70900-65-11 AC powered solenoid-operated air pilot valves. Also see Susquehanna Condition Report No. 98-2296. According to the reports, there were 40 of these SOVs in service, all at Susquehanna. The design of the SOV was such that residual magnetism in the plunger caused the solenoid to stick against the stop if the air gap was too small. Compression set in the O-ring seat caused the air gap to decrease to where the solenoid would stick after about 4 to 18 months in service. This was a common-cause failure that resulted in multiple failed and degraded primary containment isolation valves and at least 5 degraded primary containment penetrations. The failures included one stuck open PCIV for over 4 hours, 2 delayed closing PCIVs, 7 imminent solenoid failures to the non-fail-safe position, and a 59% expectation of near term failure to the non-fail-safe position for all 40 SOVs.

Observations made during the visits to the seven nuclear plants for this study of AOVs, as well as an examination of pertinent LERs and other reports, indicated that SOV failures similar to those described above are currently common in the nuclear industry. However, the Maintenance Rule is driving efforts by licensees to implement many of the independent recommendations made in 1991 in Section 9 of NUREG-1275, Volume 6. Those recommendations were:

- Take corrective actions to address the root-causes of SOV failures based on risk significance and determined from a plant-specific prioritization scheme;
- Review SOV design specifications, calculations, and operating conditions including temperature and pressure limitations;
- Conduct reviews to identify SOVs that may have been overlooked and to verify the orientation of SOVs;
- Replace or refurbish SOVs on a timely basis; and
- Ensure that the air systems that serve SOVs are a reliable source of clean, dry, oil-free air at proper pressure.

12. AOV PROGRAM PLANS AT THE SITES VISITED

Six of the seven plants visited all had AOV action plans in varying stages of completion to address design basis determination or verification, calculation of margins, diagnostic testing to confirm margins, and maintenance. One plant was focusing on maintenance practices and assumed that the design basis operability, as defined in the original design is acceptable.

The AOV action plans were comprehensive and indicate an awareness of the problems associated with AOVs, diagnostic tools available in the industry, and the work of various organizations and utilities regarding AOVs.

The licensees visited were aware of the work of the AOV Users Group, EPRI, NMAC, and ASME, among others. The scope and schedule for implementing program plans on AOVs, varied widely among the plants.

Recent licensee efforts are being stimulated by their implementation of the Maintenance Rule, 10 CFR 50.65, recent industry-wide correspondence, a number of events involving AOVs, results from plant IPE and PRA investigations, and/or experience gained from the motor-operated valve programs established to meet the recommendations of Generic Letter 89-10. At the time of the plant visits, some licensees had not started these efforts while others were well on their way. A similar situation existed with regard to licensees' determination of the margins available for their AOVs to meet the design basis demands. It appears that improvements might include further aging assessments and consideration of additional failure mechanisms that the plants may not be considering now, in their assessments of the available margins in AOVs.

12.1 Palo Verde Nuclear Generating Station (PVNGS) AOV Program

In March 1989, PVNGS Unit 3 suffered a loss of offsite power which resulted in the loss of the air system. A comprehensive review of the air system design was subsequently performed and they concluded that a number of improvements with respect to the air system and associated equipment, including several types of AOVs, were needed to ensure the as-designed operability of the air system and the equipment served by it. A comprehensive "Compressed Gas System Analysis Report" was prepared (13-MS-A20, Revision 2, dated June 15, 1989, Revision 3 undated), and still serves as an active corrective action guidance document for the plant engineering staff concerned with the PVNGS pneumatic systems and equipment.

The PVNGS action plan regarding AOVs was established based on their discovered need to improve AOV performance and included the following:

- Establish and maintain lists of Category I (safety-related with "active" safety functions) and II (safety-related without "active" safety functions, or ASME Section XI tested, or 10 CFR 50, Appendix J local leak-rate tested) AOVs.
- Evaluate and improve actuator margin for Category I and II AOVs, including dynamic testing of DCFWIVs and steam-generator blowdown isolation valves in open direction. Note that the tests of DCFWIVs were completed and valve factors of 0.64 and 0.57 were established in these tests. Refer to Section 8.1.3 in this study.
- Evaluate the need to develop procedures such as for diagnostic testing, trending, and revision/expansion of maintenance procedures.
- Determine Category I and II AOVs.
- Develop a method to determine which Category II AOVs should be subjected to a sizing review, set point verification, and PM review.

- Develop a method to determine which non-safety-related (NQR) AOVs should be subjected to a sizing review, set point verification, and PM review, and coordinate with the Maintenance Rule system AOVs.
- Perform design sizing evaluations of Category I AOVs.
- Revise the AOV Controlled Set Point Database.
- Evaluate uncertainties regarding actuator sizing for Category I and II AOVs.
- Evaluate possible permanent installation of test connections and transducer mountings on Category I AOVs.
- Determine a replacement for the (then) current AOV diagnostic system.
- Evaluate Category I AOVs to determine if current diagnostic equipment can be used.
- Determine scope of baseline and periodic test program for Category I AOVs. (No testing was proposed for Category II AOVs.)
- Review the maintenance program for Category I AOVs.
- Determine the post-maintenance requirements for AOVs.
- Develop or revise database for AOV trending.
- Evaluate the quality class of IA components used on Category I AOVs.
- Review the AOV training program.
- Review the interface between the AOV program and the valve packing program.
- Verify the Safety Issue Management Systems (SIMS) database for AOVs.
- Evaluate live-loading of the packing for throttling valves.
- Perform pressure-locking and thermal-binding evaluations for air-operated gate valves.
- Determine the necessary tasks for inclusion of hydraulic-operated valves in the AOV program.

This program was being implemented and work on individual items was either completed or ongoing as of the time of the visit for this study.

12.2 Fermi 2 AOV Program

Fermi 2 is the lead BWR plant in an EPRI program to improve the operability and performance of AOVs in nuclear power plant applications. The following quoted pilot program description was extracted from a Detroit Edison (owner of Fermi 2) Statement of Work (WO-5436-01):

"Detroit Edison (DECO) is involved in an effort sponsored by the Electric Power Research Institute (EPRI) to develop an overall Air Operated Valve Program document and, following the methods defined in the document, to conduct design basis system level (differential pressure, flow, temperature) and component level (required thrust/torque and actuator output capability/margin) evaluations for their category 1 AOVs. Where applicable and with DECO concurrence, it is desired to apply the EPRI MOV Performance Prediction Methodology (PPM) to evaluate required thrust/torque and to utilize methods defined in EPRI Report TR-107321, 'Application Guide for Evaluation of Actuator Output Capability for Air Operated Valves in Nuclear Power Plants' to evaluate actuator output capability."[a]

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a. The Electric Power Research Institute (EPRI) MOV Performance Prediction Methodology (PPM), EPRI TR-103237, Revision 2, "MOV Performance Prediction Program Topical Report" was referenced in EPRI report TR-107321 for application to AOVs.

The Detroit Edison Statement of Work described the objectives of the program as follows:

- "Develop an AOV Program Document which defines the overall approach and specific methods to be used in conducting system and component level design basis reviews of air operated valves."

- "Determine the design-basis system parameters for input to AOV thrust/torque evaluations. These include stroke direction, differential pressure, line pressures, flow, fluid media and temperature."

- "Determine thrust/torque requirements for 41 AOVs selected by the Fermi 2 AOV Program staff under design basis flow and differential pressure conditions."

- "Determine the air actuator thrust/torque capabilities and margin available for AOVs selected ... above. Provide recommendations for corrective action cases where margin improvement is desired."

DECO, Fermi 2, and their contractors were in the process of implementing the program plan at the time of the visit for this study. The categorization process was completed (see Table 1) and calculation efforts were underway. The original schedule called for the evaluations to be completed by the end of 1997.

The Fermi 2 AOV categorization process is closely tied to their efforts to satisfy the requirements of the Maintenance Rule, 10 CFR 50.65. They are using risk-based techniques and analysis tools developed to implement the Maintenance Rule in their efforts to categorize AOVs in terms of risk and safety significance, as well as economic importance. Several AOVs were found to be much more important after such techniques and analyses were applied. As with all of the plants visited, the category definitions vary (see Table 1) between licensees.

DECO is a member of a group of utilities known as the Utilities Service Alliance (USA Group) that pool their resources to devise integrated solutions to problems and concerns. The plants in the USA Group were, at the time of the visit, Fermi 2, WNP 2, Palisades, Fort Calhoun, Clinton, Cooper, and Wolf Creek. AOVs are being studied at Fermi 2 as one of the USA Group efforts.

12.3 Palisades AOV Program

Palisades was the lead PWR in the EPRI program to improve the operability and performance of AOVs in nuclear power plant applications and also was a member of the USA Group of utilities at the time of the visit. The Palisades program was described in their Procedure No. EM-28-03, dated September 26, 1997, generally as follows:

- "The purpose of this procedure is to provide a systematic approach for addressing and resolving issues associated with all aspects of air operated valve (AOV) performance and to provide an auditable program plan for plant compliance and utilization of industry recommendations affecting AOVs.... This procedure also incorporates the reciprocal agreement in program methodology being shared with other USA member plants."

- The Background section of the Palisades AOV program plan referred to ASME OM-19 "Preservice and Periodic Performance Testing of Pneumatically and Hydraulically Operated Valve Assemblies in LWR Power Plants." Palisades intended to use the information in this (soon to be published as final) draft guide in developing their testing methods.

- The general description of the Palisades AOV program included the following:

  - "This program will be similar in content to the existing Palisades Motor Operated Valve Program which was implemented as part of NRC Generic Letter 89-10. The program is designed to capture the design basis of the system and establish a design database of vendor references, design calculations, and field settings for valves. The objectives of this plan are to improve the design basis knowledge of the AOVs, to support Operations and Maintenance in their various activities, and to improve the performance of the valves through maintenance or modifications based on design basis evaluations. Where appropriate, lessons learned from the MOV Program will be applied to the AOV Program. Information and data from the MOV program, where applicable, will be utilized as necessary to facilitate AOV evaluations."

- "GL 89-10 for MOVs identified the following basic problems in design and maintenance practices that are generally applicable to AOVs.

- Disc Factor for Gate Valves—It is now generally accepted that past sizing practices for actuators on gate valves may have resulted in underestimating thrust by a factor of 1.5 to 2.0. Recent industry gate valve testing supports this conclusion.

- Design Basis—The design bases for MOVs (were) found to contain incomplete information and/or nonconservative assumptions in a number of cases. Since AOVs and MOVs appear in the same systems, it is likely that this is a problem for AOV design bases as well.

- Control of Field Adjustments—MOV switch settings were not procedurally controlled prior to the GL 89-10 program. As a result, a number of switch settings were found to be outside acceptable min/max ranges. AOV regulator and bench settings may also need to be procedurally controlled and packing forces verified to ensure they are within design assumptions. The program will define the necessary testing requirements to ensure the valves can perform their design function."

- "The goal of this program is to economically verify that AOVs providing active safety-related functions and balance-of-plant AOVs which are critical in terms of cost, radiation exposure, and reliable power generation, perform as required. All valves in this program will be categorized based on safety significance, PSA, an importance to plant availability and maintenance history. Design basis review, testing and corrective action (are) then performed on valves prioritized above a plant defined action threshold."

- The scope section indicated that HVAC dampers were excluded from the AOV program. AOVs were then categorized as 1, 2 or 3 (see Table 1).

- The Palisades AOV program scope that was obtained during the site visit divided AOVs into 3 categories as follows:

    - "Category 1: Valves in this category are safety-related with active safety functions, are important to safety based on their PSA risk significance, or are included based on experience from Expert Panel meetings. Category 1 AOVs require a documented design basis review and setpoint verification periodically confirmed by diagnostic testing."

    - "Category 2: Valves in this category may be safety-related with low risk significance for PSA or nonsafety-related in critical applications that could affect plant availability, capacity factor, heat rate, or have high maintenance costs. Category 2 AOVs may receive design reviews and/or diagnostic testing on an as-need basis at the discretion of the AOV Program Engineer. Any AOV with a Maintenance Preventable Functional Failure on record in the Maintenance Rule Program shall be considered for inclusion in the AOV Program as a Category 2 AOV by the AOV Program Engineer."

- "Category 3: Valves in this category are the remaining air operated valves not in Categories 1 or 2. Design basis reviews or diagnostic testing are not performed for Category 3 AOVs."

- The design basis review was described and consists of a system level review and a component level review.

- The system level review identifies the design basis conditions under which the AOV must operate and includes:

  - Line pressure (both upstream and downstream)
  - Fluid media
  - Fluid flow (only if required)
  - Fluid temperature.

- A component level review section described requirements for AOV actuators in order to meet worst-case system demands and includes a list of elements to be considered, as follows:

  - Minimum air pressure required
  - Maximum air pressure allowed
  - Diaphragm air
  - Required bench set
  - Spring rate
  - Seat load
  - Valve factor (for gate valves if PPM is not used)
  - Minimum required thrust/torque
  - Stroke length
  - Seat diameter
  - Valve trim (includes valve dimensions and balanced/unbalanced areas)
  - SOV characteristics/ratings
  - Packing load assumptions
  - Flow direction (for globe valves either under or over seat)
  - Shaft orientation (upstream or downstream for butterfly valves)
  - Maximum spring allowable force
  - Butterfly valve bearing material and coefficient of friction assumptions
  - Piston breakaway force.

- The EPRI Motor Operated Valve Performance Prediction Methodology (PPM) was referenced for predicting torque/thrust requirements without the need for dynamic testing. The PPM methods are considered to be directly applicable to gate and butterfly AOVs; however, the PPM methods may be of limited applicability for many globe/plug valves because of the wide variety in use in the nuclear industry compared to the types and small population tested to establish the PPM methodology.

- Diagnostic testing methods are to be used to establish baselines for periodic monitoring and post-maintenance testing.

- Corrective and preventive maintenance is to be coordinated with the requirements of the Maintenance Rule.

- Criteria for AOV modifications are discussed, as well as pressure locking and thermal binding, SOV evaluations, and air regulator setting guidelines.

12.4 LaSalle AOV Program

The AOV program at LaSalle was described in a draft administrative procedure dated December 1997 just prior to our visit on December 17–18, 1997. The stated purpose is to define the AOV program, identify the interfaces and provide a comprehensive approach to verify that the valves covered in the scope of the procedure are capable of performing their intended function under normal, abnormal, or emergency operating design basis conditions.

The procedure included the following provisions:

- Systematically review AOV requirements

- Categorize AOVs based on their safety significance and functional requirements

- Prioritize AOVs based on past performance and industry data

- Establish AOV design bases by reviewing design data

- Determine as-built configurations by performing walkdowns

- Track and trend data.

AOVs are to be categorized as 1, 2, 3 or 4 (see Table 1) and the categorization process is coordinated with the requirements of the Maintenance Rule.

We were informed that LaSalle intended (as of December 1997) to rely on diagnostic testing to ensure the operational readiness of AOVs. They had done many static diagnostic tests but had not tested any AOVs under dynamic conditions. The method(s) by which they intended to establish the available margins with respect to design basis loadings of AOVs was unclear as of the time of the site visit.

12.5 Three Mile Island I AOV Program

The scope of the draft AOV program at TMI-1 included the following:

- Identify and categorize the population of AOVs into specific groups. The valve category will define the requirements for each valve in the group based on risk ranking. The techniques used to implement the requirements of the Maintenance Rule are being used in this process. AOVs are categorized as 1, 2 or 3, depending on their safety and economic significance. See Table 1.

- Determine the limiting fluid system operating conditions under which each valve in Categories 1 and 2 must operate, including design basis accidents, safe plant shutdown, and normal/abnormal operating conditions (system level design basis review).

- Develop and implement a methodology to determine valve thrust/torque requirements and air actuator thrust capabilities (component level design basis review). It appeared from the program text that this methodology will be applied to Category 1 and 2 AOVs.

- Evaluate the design capability against the valve requirements and initiate any necessary valve or actuator modifications or adjustments.

- Adopt a validated test methodology with known accuracy characteristics that is compatible with the different valve design characteristics.

- Develop procedures and provide training for using the adopted test methodology.

- Upgrade and improve the periodic maintenance and test program to ensure long-term AOV operability. This includes the provision for a qualified review of the test data for valve operability.

- Continue to trend and evaluate AOV failures, preventive maintenance results and diagnostic test data. When appropriate, preventive maintenance task frequencies will be revised to address the findings.

The AOV program is to be coordinated with the requirements of the Inservice Testing Program mandated in 10 CFR 50.55a. The draft AOV program description (GPUN Topical Report 118) included an appendix that described the system level design basis reviews and an attachment that provided samples of the design basis reviews for individual AOVs.

12.6 Indian Point 3 (IP3) AOV Program

The IP3 AOV program was initiated primarily because the plant AOV failure rates were averaging about twice the industry AOV failure rate over the previous few years. The purpose of the program was described as "to improve plant safety, reliability, and efficiency."

An action plan was developed, the key elements of which included:

- Identify the key elements of a successful AOV program

- Categorize AOVs

- Perform design basis reviews (both system level and component level)

- Validate actuator sizing

- Make program recommendations based on these reviews.

The program was divided into two phases, as follows:

- Phase 1, Concentrate on AOVs to be maintained during Refueling Outage No. 9

- Phase 2, Address remaining AOVs.

Phase 1 included the following:

- Twenty AOVs were selected based on safety-related or plant efficiency/thermal performance considerations.

- The scope of work included overhauls, engineering reviews, and diagnostic testing.

- A review of the Phase 1 results was conducted to verify the program and provide insights for necessary modifications.

About half of the 20 AOVs tested were found to have problems, some of them long-standing, that needed correction. Corrections made, based on diagnostic testing methods, were considered successful. The majority of problems involved packing, seating/unseating, and positioner calibration.

Phase 2 implementation included:

- Conduct regular meetings with the AOV peer group

- Complete and verify a list of all safety-related AOVs to ensure that none are excluded from the evaluation

- Categorize all AOVs for the program based on PRA significance, active safety function, effect on plant availability, effect on plant thermal efficiency, maintenance history, and operational problems

- Perform system and component level reviews on Phase 2 AOVs, validate actuator sizing requirements, make recommendations regarding scope and frequency of testing/maintenance, and evaluate training procedures.

The stated goal was to complete the program by January 1999 and implement Phase 2 during refueling outage 10 on the full scope of AOVs.

12.7 Turkey Point 3 and 4 AOV Program

The AOV program at Turkey Point was initiated in 1991 as a maintenance initiative and focused on procedures, diagnostics, and preventive maintenance. Since that time, and based to some extent on compliance with the Maintenance Rule, AOVs at Turkey Point have been categorized in accordance with their risk and safety significance (see Table 1). Turkey Point now employs diagnostic testing, troubleshooting, and AOV history as tools to guide preventive and corrective maintenance. An extensive training effort is used to ensure that technicians can diagnose and repair AOV problems.

At the time of the visit, Turkey Point had no plans to further investigate the design bases vs. available margins for AOVs. They planned to rely on the information originally furnished by the vendors and considered that their primary focus should be on maintenance of their equipment.

13. INDUSTRY INITIATIVES REGARDING AOVS

The nuclear power industry is aware of the significance of AOV operability as a logical extension of their work to ensure the design basis operability of motor-operated valves. They are also aware of the significance of AOVs as a result of the work done to prioritize the importance of equipment in nuclear power plants, in order to comply with the requirements of the Maintenance Rule. A number of industry organizations and individuals are working on these issues.

Key points of interest in the nuclear power industry regarding assessing margins for AOVs include:

- The Electric Power Research Institute (EPRI) MOV Performance Prediction Methodology (PPM), EPRI TR-103237, Revision 2, "MOV Performance Prediction Program Topical Report" was referenced in EPRI Report TR-107321, "Application Guide for Evaluation of Actuator Output Capability for Air Operated Valves in Nuclear Power Plants." The PPM methodology (originally developed as part of industry efforts regarding MOVs) was endorsed by EPRI as providing guidance for predicting the forces needed to operate globe, gate, and butterfly valves for at least some AOVs. The PPM methods are considered to be directly applicable to gate and butterfly AOVs; however, they may not be appropriate for many globe/plug valves because of the wide variety in use in the nuclear industry compared to the types and small population tested to establish the PPM methodology.

- NRC Information Notice 96-48 included a summary of important contributions and findings resulting from the EPRI MOV PPM. Important findings (or confirmatory information) from the EPRI MOV Program included the following:

  a. The traditional methods for predicting gate valve performance might not be conservative for many applications because of incomplete equations, design features, manufacturing controls, and wide-ranging friction coefficients.

  b. The edge radii on disk seats and guide slots are critical to gate valve performance and predictability.

  c. Stellite friction coefficients increase with differential-pressure valve strokes in cold water to a plateau level, stabilize quickly in hot water, and decrease as differential pressure increases.

  d. Gate valves with carbon steel guides and disk guide slots with tight clearances might fail to close under blowdown conditions.

  e. Many existing gate valve manufacturing and design processes and controls, and plant maintenance practices, might contribute to poor valve performance.

  f. Traditional methods for predicting globe valve performance for incompressible flow conditions are not conservative for globe valves in which differential pressure acts across the plug guide.

  g. Globe valve thrust requirements for some designs can be excessive under compressible flow and blowdown conditions because of the potential for plug-side loading.

  h. Rate-of-loading effects (load-sensitive behavior) can reduce the static thrust output by up to 30 percent under dynamic conditions.

  i. Hydrodynamic torque coefficients used by some butterfly valve manufacturers might not be conservative for certain applications, with valves located near piping elbows especially vulnerable.

j. Butterfly valve seats should be periodically replaced to avoid hardening or degradation.

In addition to these reported important findings, EPRI confirmed that thrust requirements to unwedge a gate valve can be higher under dynamic conditions than under static conditions."

- With the exception of item h above, which refers to a characteristic that may be unique to motor operators, the findings quoted from IN 96-48 are applicable to AOVs using the same types of valves. However, note that MOVs include large populations of gate valves while AOVs include large populations of globe valves.

- Thrust requirements for some AOVs may be greater than originally assumed in vendor calculations and some plants have yet to address this issue.

- Valve packing friction force estimates may not be conservative for graphite packing or for packing modifications made to reduce leakage.

- Calculated effective diaphragm areas used by some vendors were found to be nonconservative.

EPRI contracted with Detroit Edison (Fermi 2) and Consumers Energy (Palisades) as part of the development of an overall AOV program document. Based on the (draft) EPRI AOV Program document, a pilot program was being prepared which included design basis system level tests (differential pressure, flow, and temperature) and component level tests (required thrust or torque and actuator output capability or margin) for Category 1 (safety-related and/or highly risk significant) AOVs. The PPM, originally prepared in response to MOV issues, and TR-107321 are to be used in the evaluations of AOVs.

The Air-Operated Valve Users Group (AUG) is an organization dedicated to improved performance of AOVs. They have held 17 semi-annual meetings so far. The 14th and 16th semi-annual meeting, in December 1997 and December 1998 respectively, were held jointly with the Motor-Operated Valve Users Group. The focus areas of the AUG were:

- Provide a forum for exchange of technical information for the industry on AOV issues.

- Support the member's regulatory interest by providing reliable guidance on regulatory issues, as well as striving to achieve an equitable balance between safety and efficient plant operation.

- Provide technical expertise for problem identification and resolution, implementing advanced technologies, and technical direction.

- Develop products and services that allow members to operate in the most safe, reliable, and cost-effective manner.

A Joint Owners Group comprised of the B&W Owners Group, CE Owners Group, BWR Owners Group, Westinghouse Owners Group, and 104 operating U.S. nuclear power plants was formed to address issues involving AOVs (and is referred to herein as the JOG AOV). In November of 1997 the JOG AOV agreed to:

- Develop an AOV program document which encompasses all aspects necessary to ensure the required function of AOVs;

- Utilize existing, ongoing, and planned industry efforts to the extent possible in the development of the program; and

- Produce a document which would be suitable as guidance for all plants to utilize in developing and maintaining an acceptable AOV program.

The JOG AOV described their efforts to the NRC in a presentation on June 3, 1999, at NRC headquarters. NEI forwarded a copy of the JOG AOV Program (Revision 0, dated March 9, 1999) to the NRC in a letter from D. Modeen to E. Imbro dated July 19, 1999. In the letter NEI stated that "(w)ithin the industry there is broad recognition that AOV design configuration, operation, testing, and maintenance are important factors in safe, reliable, and efficient plant operation." NEI concluded:

> "Industry experience and various published reports do not indicate safety-significant AOV concerns that warrant industry action. The industry actions described above provide ample evidence that industry is addressing any AOV performance issues. Given this perspective, these industry activities are not a topic that the industry desires credit for in the context of SECY 99-063, *The Use by Industry of Voluntary Industry Initiatives in the Regulatory Process*. Consequently, we are not requesting NRC review or endorsement of the enclosed document."

Observations and comments regarding the JOG AOV Program (Revision 0, dated March 9, 1999) are as follows:

- "CURRENT DOCUMENT STATUS," page ii[b]

  No formal implementation action is being recommended with issuance of the JOG AOV Program Document, Revision 0. The text indicated that "...individual utility executive implementation directions are provided at some future time." No schedule recommendations to licensees were included to implement the JOG AOV Program.

- Section 1.3, "AOV Program Elements," page 2

  "Nine key elements for an AOV program are identified (in the JOG AOV Program) as follows:

  Scoping and Categorization
  Setpoint Control
  Design Basis Reviews
  Testing
  Preventive Maintenance
  Training
  Feedback
  Documentation/Data Management
  Tracking and Trending of AOV Performance"

  The elements appear to be comprehensive and include the necessary scope.

- Section 1.5, "Instrument Air Systems," page 2

  It is critically important that licensees provide reliable supplies of clean, dry, oil-free air at the proper pressure, to ensure the successful operation of AOVs.

- Section 2, "DEFINITIONS," page 3

  The stated definition of an "active valve: a valve that must perform a mechanical motion during the course of accomplishing a system safety-significant function" appears to preclude consideration of mispositioning as a consideration in the JOG AOV program. Refer to the remarks about Section 4.1.4 of the JOG AOV program below.

  The stated definition of "high safety significance: designation referring to the importance to plant safety by a blending process of risk ranking and expert panel evaluations," includes risk ranking and expert panel opinion as the only criteria, without reference to the contents of the existing plant design basis.

  The term "system safety significant function" is not defined but is used in several of the other definitions.

b. Section numbers and page numbers refer to those found in the JOG AOV Program plan.

- Section 3, "PROGRAM REQUIREMENTS," page 4 and Table 3-1, page 5

  Design basis reviews, baseline testing, periodic testing and post-maintenance testing are not required in the JOG AOV Program for Category 2 AOVs even though these valves are either safety-related or have high safety-significance. This appears to be a relaxed standard for such important equipment.

- Section 4.1.2, "Scope," page 6

  Dampers are specifically excluded from the program, based on similar treatment by the industry for motor-operated dampers. Air-operated dampers provide several critical functions such as control room ventilation, containment protection, diesel generator protection, and diesel generator air supply.

- Section 4.1.3, "Categorization Process," page 6

  AOVs within the scope of the JOG AOV Program are classified into two categories as follows:

  Category 1: AOVs that are safety-related, active, and have high safety significance.

  Category 2: AOVs that are safety-related, active, and do not have high safety significance, or AOVs that are non-safety-related, have high safety significance, and are active.

  The JOG AOV Program indicates that "AOVs not in Categories 1 or 2 are considered outside the scope of this (the JOG AOV) program, as they are deemed not to be critical to plant safety." Any valve could be excluded from Category 1 based on expert panel opinion or findings of low risk ranking (see next comment). Further, only active valves, as defined in accordance with the terminology of the program document, are included in either Category 1 or 2 and thus subject to any requirements under the JOG AOV program.

- Section 4.1.3, "Categorization Process," page 6

  Common-cause failures are not taken into consideration in the JOG AOV Program when ranking the risk-based importance of AOVs. Individual AOVs may be found to be of low risk significance, but the common-cause failure (CCF) of two or more AOVs performing the same function may have considerable risk significance. (Refer to the last paragraph in Section 15.4 of this study for specific examples.) The JOG AOV Program does not refer to this issue

- Section 4.1.3.2, "Determination of Safety Significance," page 7 and 8

  Five specific references are included in the paragraph, "any one" providing an acceptable method for ranking safety significant AOVs. A conservative approach would be for each licensee to consult at least two or three of the five, rather than relying on one as the paragraph suggests, especially considering the sentence at the top of page 8.

- Section 4.1.4, "Mispositioning," page 8

  Mispositioning or inadvertent operation of an AOV is specifically excluded from the JOG AOV Program. It is yet to be demonstrated, on the basis of safety significance, that mispositioning of AOVs is a consideration that can be ignored.

- Section 4.3, "Design Basis Reviews," pages 10–15

  The JOG AOV Program document includes detailed instructions for conducting design basis reviews. Several specific references to methods developed in response to Generic Letter 89-10 programs for MOVs are included. In general, design basis reviews are only required for Category 1 AOVs.

- Section 4.3.3.2, "Actuator Output Capability, page 14

- This section refers to EPRI TR-107321, "Application Guide for Evaluation of Actuator Output Capability for Air Operated Valves in Nuclear Power Plants," as providing acceptable first principle equations for evaluation of actuator output capability for AOVs. EPRI TR-107321 references the EPRI MOV Performance Prediction Methodology (PPM), EPRI TR-103237, for application to AOVs. Extrapolation of the methodology, originally developed in response to issues involving MOVs, may be acceptable; however, the range and conditions of applicability remain to be examined and verified.

- Section 4.4, "Testing," pages 15-19

  Baseline, periodic, and post-maintenance testing methods are described in some detail but testing, including post-maintenance testings, is restricted to Category 1 AOVS.

- Section 4.9, "Tracking and Trending," page 21 and Section 5, "FULL PROGRAM IMLEMENTATION," page 22

  Tracking and trending did not include establishing and maintaining a history of the program, including changes and updates.

Previous activities of a JOG for motor-operated valves involved production of an industry topical report for periodic verification of design-basis motor-operated valves. This report was assembled to meet the recommended actions in NRC Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves."

The ASME Committee on Operations and Maintenance Activities developed a guideline (to have been published as a standard) for AOVs, ASME OM Part 19, "Preservice and Periodic Performance Testing of Pneumatically and Hydraulically Operated Valve Assemblies in LWR Power Plants." OM-19 is a set of voluntary applied guidelines which apply to active safety-related AOVs and can be applied to other AOVs at the user's option. In the draft that was reviewed for this study (October 18, 1996), preservice testing is recommended including consideration of dynamic loads. Several testing options are offered including dynamic testing at service conditions. Periodic stroke/timing testing and other tests include measurement of parameters such as bench set, maximum available pneumatic pressure, seat load, spring rate, actual travel, pneumatic pressure required to accomplish safety function, and friction forces. Provisions for analysis and evaluation of data as well as corrective action criteria are included.

13.1 The Maintenance Rule

The requirements for "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," were invoked in 10 CFR 50.65. The implementation of these requirements is being accomplished in accordance with NRC Regulatory Guide 1.160, which endorsed an industry document, NUMARC 93-01. The Maintenance Rule does not deal with specific maintenance requirements, but rather with plant and system performance which demonstrates that nuclear power plant maintenance activities are ensuring acceptable performance of structures, systems, and components.

Licensees are addressing how effective their maintenance practices are with respect to the operability of safety-related and important non-safety-related systems and equipment in their plants using a variety of risk related tools and techniques. The licensees are ranking systems and equipment in terms of risk significance in order to get improved effectiveness out of their maintenance activities as well as improved safety. The system of establishing goals for systems and equipment found to have operational problems appears to be effective (at least as far as AOVs are concerned) in the plants visited. Its effect (and perceived effectiveness) was noted in all of the plants visited.

Licensees at the plants visited are using techniques developed in response to the Maintenance Rule to assess the performance of equipment outside, and in addition to, the scope of NRC's interest in order to increase the reliability and efficiency of their plants. As a result of their Maintenance Rule activities, licensees became increasingly aware of the importance of AOVs, in terms of plant safety, availability, and heat rate.

14. DIAGNOSTIC SYSTEMS, MAINTENANCE, AND TESTING OF AOVS

The rapid evolution of computerized diagnostic systems for evaluating the condition and performance of valves in nuclear plants provided an essential tool to the nuclear power industry. All nuclear plants have used and are using such diagnostic equipment to comply with the recommendations of NRC Generic Letter 89-10 and its supplements. Diagnostic equipment for evaluating the condition and performance of AOVs has been available since the mid-1980s and evolved in parallel with computer software, computer hardware, testing equipment, and vendor experience. Recently, diagnostic system vendors who were primarily focused on providing diagnostic systems for motor-operated valves turned their attention to AOVs and released state-of-the-art products specifically designed for AOVs. At least one diagnostic system on the market was specifically designed for SOVs.

The AOV diagnostic systems are used to monitor, record, and provide numerical and graphical output for a number of measured AOV parameters. These parameters include total travel, signal times incidental to travel, friction, bench set (the high and low pressure values applied to a single-acting actuator to produce the nominal valve travel with no external forces on the actuator stem), spring rate, seat load, signal pressures at various points of travel, positioner parameters, I/P (current-to-pressure) or E/P (voltage-to-pressure) transducer signal parameters, and system air pressure variations.

AOV diagnostic systems are extremely useful and powerful tools for preventive or predictive maintenance. In all of the plants visited, the engineers were aware of what is available and several were actively seeking to upgrade their existing diagnostic system capabilities. The diagnostic system vendors and users have become adept at diagnosing a wide variety of valve and operator defects and deteriorated conditions.

The problem of predicting, with confidence, design basis behavior of a valve from information gathered from no-load testing (no pressure and/or fluid flow in the line) remains. The EPRI valve performance prediction methodology (PPM) and the included data are offered by EPRI for predicting the forces needed to operate globe, gate, and butterfly AOVs at design-basis conditions. Refer to EPRI TR-103237 as endorsed in EPRI report TR-107321. Extrapolation of the methodology, originally developed in response to issues involving MOVs, may be acceptable; however, the range and conditions of applicability remain to be examined and verified. Data from recent or planned flow or pressure testing (EPRI or other sources and individual plant tests) could be useful in verifying the assumptions used in the PPM.

As was the case originally with motor-operated valves, the claims of the AOV and SOV diagnostic system vendors regarding accuracy and abilities of their systems remain to be verified. Calculated estimates of packing loads, side loads, and other parameters should be verified to be within realistic ranges. The relationship between diaphragm pressure and stem thrust delivered to the valve disc or plug should be confirmed to be as assumed.

The currently required test for most safety-related AOVs in nuclear power plants consists of a periodic stroke-timing test in accordance with ASME Code Section XI. The test consists of operating the valve with no required pressure or flow in the lines and measuring the time required to stroke in the safety-related direction(s). Records are kept for trending purposes and to satisfy NRC documentation requirements. The stroke-timing test by itself does not provide much in the way of assurance of future operability. AOVs can and have been successfully stroke-time tested and then failed on the next test or the next service demand. The use of diagnostic systems to identify AOV part failures or misadjustments can provide for enhanced reliability. However, the problem of predicting design basis operability of safety-related AOVs from a no-load test remains, if not supplemented by knowledge of the design basis capability and demand.

The plants visited have a number of AOVs (including dampers) associated with safety-related, important non-safety-related, or risk significant systems, such as the diesels or HVAC systems, that are not maintained under the control of the AOV engineer(s). These AOVs are serviced by the technicians and engineers that service the particular system. These AOVs may not receive the same level of attention relative to engineering, maintenance, and testing, or the benefit of safety-related performance information available to personnel more closely involved with the design, maintenance, and performance evaluation of AOVs within each plant's AOV program.

15. RISK ANALYSIS INVOLVING AOVS

15.1 Previous Risk Analyses Results

Several risk analyses have been conducted concerning air systems and AOVs. NUREG-1275, Volume 2, Section 6.4 refers to three risk studies regarding instrument air systems' relation to nuclear plant safety. NUREG-1275, Volume 6, Section 8.1 includes a discussion of the risks associated with common-cause failures of SOVs.

A study produced for the industry by Pickard, Lowe and Garrick, Inc., NSAC-128, "Pneumatic Systems and Nuclear Plant Safety," October 1988 offered the following findings:

- "The safety significance of support systems (especially pneumatic systems) is not always obvious. Most plant analysis and documentation (FSAR, technical specifications, emergency procedures) is produced by reactor vendors and focuses on the front line systems they design. In PRA analysis and maintenance records, support system failures are often categorized as failures in the affected front line system."

- "Contamination has far more impact on pneumatic system performance than does loss of pressure. System designers provide highly reliable, redundant air supply systems, often backed up by nitrogen bottles, and then add another layer of protection by installing valves that fail-safe on loss of air. On the other hand, they generally provide neither redundancy for preventing contamination nor warning against its occurrence or effects. The effects of contamination are unpredictable: sometimes failures occur immediately; in other cases, contamination causes multiple failures in air operated equipment at random times long after the contamination has been released. Contamination has the potential to defeat all redundancy in supply."

- "There is no way to predict exactly what sequence of events will occur during gradual degradation of air pressure and no assurance that, if it happens twice at the same plant, the sequences will be the same."

- "Air system capabilities and requirements, as well as the extent and significance of air system problems, are not always well understood by plant staffs. System capacity, dryer lifetime, and the effects of degraded desiccant are sometimes not known. Symptoms, rather than root causes, are often corrected."

- "The extent of reliance on pneumatic systems for safety and operational function is plant specific."

- "Plant availability is more likely to be affected by air system problems than is plant safety."

- "More extensive lists of findings and recommendations found elsewhere (references in quoted text are omitted) tend to obscure the significance of contamination stressed (in NSAC-128)."

In a 1989 study, NUREG/CR-5472, "A Risk-Based Review of Instrument Air Systems at Nuclear Power Plants," the researchers studied the IA systems in nuclear power plants in terms of their contribution to risk. The overall Conclusions and Recommendations section (5) of that report included the following paragraphs:

"A systematic review of IA-related events, system designs, and risk impacts was performed. Although many events related to the IA system have been reported, there is neither such a plurality of events nor do the events place a typical plant in such danger of core damage or significant release of radioactivity that treatment of the IA system should be significantly revised. This study yielded three general conclusions:

1. The IA system contribution to total core melt frequency is generally much lower than that of frontline systems, and is significantly lower at BWRs than at PWRs.

2. The risk contribution of the IA system cannot be significantly reduced by modifications or reliability improvements within the IA system.

3. Most plants which had notable IA-related risk sequences needed modifications outside the IA system (e.g., the condensate system at Oconee, and the HPI system at Haddam Neck)."

"However, risk and reliability analyses that have systematically considered the IA system, its interactions with frontline systems, and the effect of loss of IA on the plant have uncovered plant-specific operating and design weaknesses that impact risk. The following conditions increase the risk impacts of the IA system:

- The possibility of common cause failures of air-operated equipment (e.g., Calvert Cliffs);

- Unique designs in fail-safe valve positions (e.g., Oconee);

- Contamination of the air system such that the common-cause failure probabilities of air-operated components are significantly increased (e.g., Turkey Point)

- Accumulator and associated check valve reliabilities especially those not adequately or frequently tested; (and)

- EDG dependencies on IA during an actual (loss of offsite Power)."

"The following actions can ensure that IA system contributions to plant risk remain low:

1. Ensuring that appropriate standards of design quality (moisture, particulate size), design intent (compressor capacity, backup sources of air), and operational performance (minimization of maintenance-related and other human errors) are maintained.

2. Including the IA system in risk-based review of plant systems (e.g., PRAs) and, when risk sequences are quantified using an estimate of the frequency of loss of IA that reflects the generic frequency and nature of problems in the system.

3. Locating and correcting any EDG/IA interactions in which non-safety grade portions of IA can cause EDGs to fail during a LOOP (loss of offsite power). Such a review would include identification and elimination of diesel room dependence for cooling on systems that are off-line during a LOOP.

4. Ensuring that the design and functionality of accumulators is consistent with safety analyses.

5. Prior to making changes to solenoid valves and/or air pressure regulators(,) perform (an) analysis that takes into consideration potential overpressurization problems. Follow changes by a test to (ensure) that the fail-safe conditions of AOVs are not being compromised."

The situation at LaSalle, a BWR, where loss of instrument air is the largest initiating event, contributing almost one-third to core-damage probability (refer to the trip report in Appendix C), is at odds, for at least one plant, with the in NUREG/CR-5472 that IA system contribution to core melt is generally much lower than for frontline systems. Observations from the site visits, particularly Palisades and Three Mile Island, are not in agreement with the conclusion in NUREG/CR-5472 that the risk contributions of the IA system cannot be reduced by improvements to the system. The other conclusions and recommendations are generally in agreement with this current study.

Many of the details identified as sources of concern in this current study of AOVs were described in NUREG/CR-5472 (sections therein noted in parenthesis), such as:

- A significant reduction in risk due to common-cause events could be achieved by reducing instrument air contamination. (2.4.4)

- An important safety concern is the ability of AOVs to achieve the fail-safe position. Moisture contamination of the IA system is an important common-cause event from a risk perspective. (3.2)

- The IA system affects PRA models in somewhat the same way as does the electric power system. The IA system supports frontline components, both in normally operating and standby systems, and a loss of IA causes a reactor trip. IA uses accumulators as a back-up source of motive force to important components. These accumulators function in a manner analogous to that in which emergency AC power is applied to important loads upon loss of normal AC power. (4.)

- In all PRAs reviewed, IA was discussed qualitatively both as a potential initiating, and as a support system. However, most PRAs did not treat loss of IA as a unique initiating event because it is not regarded as being significantly different from the other events that cause a loss of the PCS and a reactor trip. (4.)

- The significance of relying on an air-pressure regulator that may be non-safety-related to prevent the over-pressurization and failure of a safety-related (solenoid-operated valve) appears not to have received the emphasis it deserves in the original design basis. Even though safety-related components that depend on the air system are designed to assume a fail-safe condition on loss of air, the converse condition, of air over-pressurization, may not have been consistently considered. (4.4.1)

15.2 Accident Sequence Precursor Analysis

As part of this study, an analysis was made of previous Accident Sequence Precursor (ASP) analyses. The results of that review are included in Appendix B. Several relatively significant events involving failures of AOVs and SOVs were identified. These events involved failures of the valves caused by design, installation, and maintenance activities, as well as failures in the air systems. Conclusions from the review of the ASP analyses include:

- AOV-related precursors have been important contributors in terms of conditional core-damage probability. There were 26 AOV-related precursors, of the 288 total precursors (i.e., events with conditional core damage probability (CCDP) contribution greater than or equal to 1E-6) from 1984 through 1995. The 1985 Turkey Point AFW event ranked highest among AOV precursors, with a CCDP contribution of about 9E-4.

- Twelve AOV-related precursors had CCDP contributions greater than or equal to 1E-4 from 1984 to 1995.

- Nineteen (12 operational, 5 testing, 2 maintenance) of the 26 AOV-related precursors occurred while the plants were at power.

- Eleven of the 26 AOV-related precursors involved common-cause considerations. Four of those eleven events had CCDP contributions greater than 1E-4.

- A variety of safety-related and non-safety-related systems were impacted by the precursors.

15.3 Recent AOV risk Studies

15.3.1 Risk Studies in Several of the Plants Visited

Information on risk significance of air systems, particularly contributions of failure of the air system as a percentage of core damage frequency (CDF), was gathered during the plant visits. Relative risk significance of particular AOVs and SOVs were also studied and information was offered at most of the plants visited because the licensees were engaged in PRA-related activities. These activities were prompted by their efforts to comply with the Maintenance Rule (10 CFR 50.65), and their ongoing efforts regarding probabilistic and risk-informed studies, including Individual Plant Examinations (IPEs) in the form of Probabilistic Safety Assessments (PSAs) produced in response to NRC Generic Letter 88-20 and its supplements. The methods endorsed in NRC Regulatory Guide 1.160 and described in NUMARC 93-01 rely heavily on PRA tools such as risk-reduction worth (RRW), CDF, and risk-achievement worth (RAW) to rank the importance of systems, and thereby, the valves of interest here.

The engineers at Fermi 2 used PRA techniques and results from their risk analyses to classify AOVs in terms of risk significance. Lists of risk significant and "critical" AOVs were generated. Among the AOVs that ranked high in terms of Fussell-Vesely (FV) importance were the hardened vent isolation valves and those associated with the RHR heat exchangers. Thirty-three AOVs (22 safety-related and 11 non-safety related that perform a risk significant function) and 370 control-rod-drive scram inlet and outlet valves were considered safety-significant.

Similar techniques were used at Palisades to identify "important" AOVs. A list of 11 risk significant AOVs (out of 84 modeled as active in the PSA, 9 of which are not categorized as active in the plants' IST program) was provided during the site visit.

According to the LaSalle Summary PRA, "(t)ransients with loss of instrument air, (T11), are the largest initiating event category, contributing 32% of the CDF. These transients are significant because venting containment cannot be performed without instrument air. Failure to vent results in the loss of the ADS function (and subsequent loss of the low pressure injection systems) and eventual containment failure, causing potential loss of injection systems in the reactor building due to severe environments."

The LaSalle Summary PRA indicated that "Loss of offsite power, LOSP, events are the second highest contributor to CDF. Single unit LOSP events contribute 6.5% of CDF and dual unit LOSP events contribute 22.9%. The core damage contribution of the SBO sequences (subset of LOSP) is 17.2%." A loss of offsite power to Unit 1 or to both units at LaSalle leads directly to a loss of instrument air, and the CDF resulting from such an event would be expected to be at least as significant as a loss of instrument air alone. The "Accident Sequence Event Descriptions" listed in the tables in the LaSalle Summary PRA include the event described as "LOSS OF INSTRUMENT AIR 1E OR LOSP AT UNIT 1;" however, the Summary PRA does not provide a discussion of the relationship or dependencies between the two events.

Risk studies were also conducted at TMI-1 to rank valves of all types in terms of relative risk significance. Several air-operated containment isolation valves ranked high in importance. It was also noted that loss-of-instrument-air was ranked sixth among the top 10 PRA core damage sequences, accounting for 5.3 percent of the total (small LOCAs contributed 18.8 percent).

Tables of AOVs indicating risk significance and tables of events showing importance rankings were furnished by the engineers at Indian Point 3 during the site visit. By hand count, 21 of 176 AOVs (with safety-function listed as "active" or "unknown") were considered by the engineers to have a "High PRA." Over 60 AOVs in the plant were classified as risk significant, based on a combination of evaluation criteria. The individual failures of MSIVs and EDG Flow Control Valves were found to be not particularly risk significant but the CCF of two or more of each type of valve was found to have considerable risk significance. (RAW was computed by the licensee to be 1.614 for failure of an MSIV to close on demand. RAW was computed to be 50.73 for CCF of two or more MSIVs. RAW was not computed or not significant for failure of an individual EDG flow control valve. RAW was computed to be 46.97 for CCF of both EDG flow control valves.)

The Reliability and Risk Assessment Group (RRAG) from Florida Power and Light (FPL) undertook a preliminary AOV risk ranking analysis during and shortly after the visit for this AOV study. Their analysis was based on the 44 AOVs that are modeled in the Level 1 PSA for Turkey Point 3 as well as some shared systems between Units 3 and 4 (e.g., AFW). Of the 44 AOVs, 17 met the Maintenance Rule risk significance criteria for further evaluation by an expert panel. Three AOVs of the 17 that met the Maintenance Rule risk significance criteria were reported to have a risk achievement worth (RAW) of about 2.4; however, they were considered to be not risk significant by the expert panel because of mitigating factors such as isolating leaks or breaks using in-line manual (hand-operated) valves that were not credited in the PSA. According to FPL, one of the 17 AOVs had a RAW of about 1.3, and one had a RAW of 1.2; the other 12 AOVs had a RAW of 1.

The RRAG/FPL, incidental to discussions during the visit for this study, increased the assumed failure rate for AOVs by a factor of 10. The CDF increased from 5E-5 to 1.1E-4. They also described a study that they did for the common-cause failure sensitivity of two AOVs, 856A and B, which had been replaced by MOVs in 1987 as part of Bulletin 86-03 compliance. See NUREG/CR-5295, "Closeout of IE Compliance Bulletin 86-03: Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line." The sensitivity study, prior to modifying the plant design, indicated a risk achievement worth (RAW) of 21.3 for the common-cause failure of the two (originally air-operated) valves due to loss of IA.

Table 6 lists air-operated valves considered by the licensees in the plants visited to be risk significant.

15.3.2 Generic Issue 158 Draft Study and Subsequent Sensitivity Analysis

In 1996, a sensitivity analysis draft report, INEL-95/0550, entitled "GI-158: Performance of Safety Related Power Operated Valves Under Operating Conditions," was prepared for the NRC as part of their process for resolving Generic Issue (GI) 158, "Performance of Safety-Related Power-Operated Valves Under Design-Basis Conditions." That draft study included a sensitivity analysis and has since been designated as NUREG/CR-6644 and published in September 1999.

As was noted in the Executive Summary of NUREG/CR-6644, "(t)his study reveals that changes in the failure probabilities of some POVs will result in proportional changes in the system unreliability and indicates that POVs have an important role in system reliability."

Another sensitivity analysis was prepared (see reference 57), as part of this current study of air-operated valves (AOVs) in nuclear power plants, to assess the impact on core damage frequency (CDF) resulting from the operating performance of AOVs. Additional sensitivity analyses, focused on AOVs and SOVs, were performed in an effort to assess core damage frequency (CDF) for failures of power-operated valves. Some of the insights and results of these analyses were used in the preparation of NUREG/CR-6644.

Failure probabilities used in PRAs of 44 plants were examined in NUREG/CR-6644. Bounding failure probabilities for AOVs and SOVs of three per hundred demands were identified. Bounding failure probabilities for HOVs of five per hundred demands were also identified. These values were used in the CDF sensitivity analyses conducted as part of the study. It was noted in NUREG/CR-6644 (Section 2.1 after Section 2.1.3) that "(s)ome plants with several years of operating experience are using very low generic probabilities in PRA calculations." Common-cause beta factors of 0.022, 0.062, and 0.115 were calculated for AOVs, HOVs, and SOVs respectively. The SOV beta factor was calculated as 0.015 if "...the anomalous performance of ... one site is excluded from the calculation."

Sensitivity analyses for a General Electric BWR, two Westinghouse 4-loop PWRs, a Westinghouse 3-loop PWR, a B&W PWR, and two Combustion Engineering PWRs (7 plants) were conducted as part of the GI-158 study.

The analysts who conducted the GI-158 study used the models and data that were available to them and employed standard techniques for performing the risk studies. However, the failure scenarios and failure data may not have reflected higher risk significance of AOV and SOV failures and degraded conditions because:

- The risk models for the plants considered were not set up to account for common-cause AOV or SOV failure contributions resulting from air contamination (over long periods), gradual air pressure degradation, or over-pressurization.

- Failures of AOVs or SOVs in the BWR reactor protection system were not included in the BWR risk model. Scram pilot SOVs contribute about one third to the total unavailability of scram systems (NUREG/CR-5500, Vol. 3).

- A number of pertinent AOV and SOV-related events and conditions, including common-cause failure events and conditions were not included in LERs and other reports and thus, were not available for consideration in these (or other) risk studies.

- Core damage was the only "figure-of-merit" included in the studies. Large early release is a pertinent figure-of-merit for AOVs, but was not considered.

15.4 Common-Cause Failures

NUREG/CR-4780, "Procedures for Treating Common Cause Failures in Safety and Reliability Studies" was reviewed during the preparation of this study of AOVs. That report is devoted to an exploration of common-cause failures and events in their relation to risk analyses, as well as the difficulties involved in arriving at consistent characterizations of common-cause failures and events. Section 1.4 of NUREG/CR-4780 includes the following definition, which is considered useful for the discussion here. "In a PRA model, a common cause event is defined as the failure or unavailable state of more than one component at the same time and due to the same shared cause. Common cause events require the existence of some cause-effect relationship that links the failures of a set of components to a single shared root cause."

Recently, NUREG/CR-6268, "Common-Cause Failure Database and Analysis System:..." (Four Volumes) offered, in Volume 3, Section 1.3, the following criteria to define a common-cause failure event:

1. "Two or more components fail or are degraded at the same plant. Failures are discovered during equipment challenges to operate, surveillance testing, or design deficiencies that are detected prior to operating the equipment. In the case of a failure resulting from a design deficiency, a potential failure is considered to have the same severity as a failure that results from a challenge to the equipment, provided the design deficiency would have caused a component to fail on demand. For example, a wiring discrepancy that would prevent a pump start is considered to be a complete failure, even if no start was attempted.

2. Component failures occur within a selected period of time.

3. The component failures result from a single shared cause and are linked by a coupling mechanism such that other

components in the group are susceptible to the same cause and failure mode.

4. The equipment failures are not caused by the failure of equipment outside the established component boundary."

Section 2.4 of NUREG/CR-5472, "A Risk-Based Review of Instrument Air Systems at Nuclear Power Plants," included a review of instrument air events based on the events and findings in NUREG-1275, Volume 2. A "Summary of Instrument Air Common Cause Events" prior to 1986 was included in Table I-5 of NUREG/CR-5472. Some of the events described in NUREG-1275, Volume 2, as common-cause failures were characterized in NUREG/CR-5472 as not really fitting the common-cause concept because the time interval between events or the discovery of failures was considered to be too long and corrective or defensive measures could have been taken to mitigate the effects of the failures or events. However, many of these events meet the definition of common-cause failure described in NUREG/CR-6268.

The NRC/AEOD published a report, AEOD/E92-02, "Insights From Common-Mode Failure Events," dated June 1992 and Supplement 1, dated February 1993. These reports were referred to in NRC Information Notice 93-35, "Insights From Common-Cause Failure Events," dated May 12, 1993. The significance of common-cause failures in the plants was reviewed in the reports and the reviews indicated that many common-cause failures are associated with errors in design and equipment qualification. The reports emphasized that common-cause failures defeat the single-failure criteria which is basic to the satisfactory safety performance of nuclear plants. The reports also referred to several events that occurred over extended periods of time.

Common-cause failures of AOVs are considered to include failures that occurred or could occur from the same event or root cause, such as loss of instrument air, contamination of instrument air, SOVs that failed or were damaged from the same source of particulate contamination, SOVs improperly installed in a harsh environment, SOVs designed with a common defect, or a group of the same AOVs installed with inadequate thrust capability to meet design basis conditions. The point is that the common-cause failure mechanism did or could cause the same failure of two or more AOVs at the same time or over time, and if not detected and corrected, compromised or could compromise safety because these multiple failures occurred or could occur at some, not necessarily simultaneous, minimum demand conditions.

Functional failures in one AOV that remain undetected until another of the same type AOV fails in the same or a different system from the same cause, and then the failure in the first AOV is subsequently discovered, are of particular concern because the vulnerabilities may exist over long periods.

Common-cause failures should be taken into consideration when ranking the risk-based importance of AOVs. Individual AOVs may be found to be of low risk significance but the common-cause failure (CCF) of two or more AOVs performing the same function may have considerable risk significance. For example, at Indian Point 3, the individual failures of MSIVs and EDG Flow Control Valves were found to be not particularly risk significant but the CCF of two or more of each type of valve was found to have considerable risk significance. (RAW was computed by the licensee to be 1.614 for failure of an MSIV to close on demand. RAW was computed to be 50.73 for CCF of two or more MSIVs. RAW was not computed or not significant for failure of an individual EDG flow control valve. RAW was computed to be 46.97 for CCF of both EDG flow control valves.)

15.5 AOV Failure Rates

NUREG/CR-1363, "Data Summaries of LERs of Valves at U.S. Commercial Nuclear Power Plants, January 1, 1975 - December 31, 1980," reported an estimate of two failures per thousand demands for AOVs. This number appears to be a number commonly used in risk analyses.

NUREG/CR-1363 also reported an estimate of six failures per thousand demands for motor-operated valves, a commonly used figure at that time. NUREG/CR-5140, "Value Impact Analysis For Extension of NRC Bulletin 85-03 To Cover All Safety-Related MOVs," reported an estimated failure rate of over eight per hundred demands, based on specific licensee reports of MOV failures for loaded demands, prior to the publication of Generic Letter 89-10. The discrepancy between the estimates in the two reports may have been in the "demand" denominator. No-load demands for stroke-timing tests and tests after repairs were included as demands in NUREG/CR-1363 data, along with the loaded demands in which the motor-operated valves were subjected to pressure or flow.

The failure rate for AOVs is expected to be higher than two failures per thousand loaded demands, although it may not be as high as it was found to be for MOVs because of the experience gained by the licensees over the past 10 years. However, much of the experience gained by licensees from motor-operated valves has yet to be incorporated into the design and setup of AOVs. Also, as was the case with MOVs, many important AOVs have not been tested at design basis (including accident) conditions. Performance predictions for AOVs that have been extrapolated from no-load and low-load tests or service conditions might not be valid at design basis conditions.

15.6 Observations Pertinent to AOVs from Examination of Risk-Related Analyses and Probabilistic Safety Assessments (PSAs) for Motor-Operated Valves (MOVs)

The NRC and the nuclear power industry have used risk analysis tools in several studies involving the implementation of Generic Letters 89-10 and 96-05 to improve the reliability and performance of safety-related MOVs. Many of the methods and observations in these studies of MOVs provided insights that are considered pertinent to the present study of AOVs and licensees' prospective programs for the improvement of AOV performance. The studies of MOVs that were reviewed included:

- NUREG/CR-5140 (BNL-NUREG-52145), "Value-Impact Analysis For Extension Of NRC Bulletin 85-03 To Cover All Safety-Related MOVs," July 1988.
- BNL TECHNICAL REPORT A-3869-T52-10-91, "Safety Significance Of Inadvertent Operation Of Motor Operated Valves In Safety-Related Piping Systems In Boiling Water Reactors," October 1991.
- BNL TECHNICAL LETTER REPORT E-2071-T1-12-93, "Safety Significance Of Inadvertent Operation Of Motor Operated Valves In Safety-Related Piping Systems In Pressurized Water Reactors," Revision 1, March 1995.
- BWR Owners Group Report NEDC-32264A, "Application Of Probabilistic Safety Assessment To Generic Letter 89-10 Implementation," October 1, 1996.
- Westinghouse Engineering Report V-EC-1658-A, "Risk Ranking Approach For Motor-Operated Valves In Response To Generic Letter 96-05," Revision 2, July 1998.

BWR Owners Group Report NEDC-32264A and Westinghouse Owners Group Report V-EC-1658-A were referenced in the Joint Owners Group AOV Program (references 6.10 and 6.11 respectively), Revision 0, dated March 9, 1999. These reports were endorsed to provide acceptable methods, among others (see Section 4.1.3.2 of the JOG AOV Program) for ranking safety significance and conducting an expert panel review of AOVs.

The value/impact analysis in NUREG/CR-5140 included a risk-based analysis of the effects of the higher MOV failure estimates reported by licensees in response to NRC Bulletin 85-03 in relation to the previously estimated failure rates. We have not been able to accurately quantify AOV failure rates for our current study of AOVs in a manner comparable to those documented for MOVs in NUREG/CR-5140. However, based on our previous observations, we believe that AOV failure rates for demands at pressure or flow conditions are substantially higher that the estimate of approximately two per thousand demands commonly used in risk studies. If a sufficiently reliable failure rate for AOVs can be documented, the methods used in NUREG/CR-5140 would be appropriate for a value/impact study of AOVs.

BNL reports A-3869-T52-10-91 and E-2071-T1-12-93 were prepared in response to BWR Owners Group and Westinghouse Owners Group questions regarding the issue of the risk significance of MOV mispositioning. The NRC's concern was related to the ability of "position changeable" MOVs in the event of their inadvertent operation from the control room. In the Davis-Besse event in 1985 (that subsequently led to the publication of NRC Bulletin 85-03 and Generic Letter 89-10), several MOVs were inadvertently closed from the control room and could not be reopened because their torque switches were improperly set. The reports indicated that plant risk increased for BWRs and PWRs by about one order of magnitude for mispositioning of MOVs and their subsequent failure.

Failure of AOVs in previously unanticipated positions has been documented a number of times. Drift of AOVs is also possible as a result of partial and/or gradual loss of pneumatic pressure. Such losses have occurred by failures in SOVs or other pneumatically controlled devices. Although the hardware and failure mechanisms may be somewhat different for the MOV and AOV operators, the concern is the same, i.e., the ability of the valve to perform the designed function. In addition, the failure mechanisms for AOVs include common-cause failure mechanisms based on failure of the air supply or contamination therefrom, and other failure mechanisms such as material-related failures in control solenoids. The methods used in these two BNL technical reports provide useful guidance for work to estimate risks associated with mispositioning of AOVs.

15.6.1 BWR Owners Group Report NEDC-32264-A

BWR Owners Group Report NEDC-32264-A described a method for categorizing MOVs for initial testing and subsequent periodic verification to meet the recommendations in GL 89-10. The methods discussed in NEDC-32264-A described a risk-based categorization and ranking process for MOVs that may be of interest in the study of AOVs. This BWROG topical report described a seven step process for ranking MOVs in order of risk significance, as follows:

- REVIEW PLANT PSA TO DETERMINE HOW WELL THE PSA REPRESENTS THE MOVS PERTINENT TO THE GL 89-10 PROGRAM.

  The PSA for a particular plant is to be reviewed to develop an understanding of MOV performance. Level 1 and 2 PSAs are to be reviewed in order to consider MOVs that affect core damage and containment integrity. Failure of MOVs to change position is to be emphasized. The PSA may "mask" the importance of some MOVs depending on how it is constructed. High-energy line break scenarios are to be considered. Finally, the role of MOVs in initiating events is to be investigated.

- REVIEW GL 89-10 MOVS NOT INCLUDED IN THE PSA.

  The reasons why particular MOVs are not modeled in the PSA are to be considered and documented.

- REVIEW IMPORTANCE MEASURES (FIGURES-OF-MERIT) USED IN THE PSA.

  Completion of this task results in a listing of the MOVs and their associated numerical importance measures. Various ranking processes and their attributes are described in the report.

- QUANTIFY THE IMPORTANCE OF MOV IMPLICITLY MODELED IN THE PSA.

  Because PSAs generally treat some potential failures implicitly, an expanded review of MOV importance is required. Implicit modeling refers to actions where the successful function of an MOV is required but may not be explicitly stated, for example "operator aligns containment venting." The role of MOVs in such events is to be considered.

- PERFORM SENSITIVITY ANALYSES TO ENSURE THAT MOVS ARE EVALUATED PROPERLY.

  Three issues are to be addressed:

  - Truncation (base events dropped) in the generation of cutsets from fault trees due to low failure probabilities;
  - Truncation due to calculation cut-off values (set to minimize program run-times); and
  - Common-cause failures of MOVs due to potential widespread performance deficiencies. These issues are discussed in the report and are to be considered in the analyses. Testing and verification of performance are suggested, along with expert-panel opinions, as are methods to minimize the impact of these issues, particularly common-cause failures.

  MOVs determined to be important due to intersystem common-cause failure potential would, at least initially, be subjected to increased verification testing. Methods for considering sensitivity analyses involving intersystem common-cause failures, which are not generally modeled in PSAs, are discussed.

- CATEGORIZE MOVS AND APPLY TESTING CRITERIA.

  A tabular form is offered for MOV risk prioritization results. Various risk ranking methods are discussed and review of the results by an expert panel is recommended.

- MOVS ARE CATEGORIZED AS HIGH, MEDIUM, OR LOW, IN ACCORDANCE WITH THEIR IMPORTANCE TO CORE DAMAGE OR LARGE RELEASE OF CONTAMINATION.

Table 2 of NEDC-32264-A provides the proposed ranking criteria for MOVs as follows:

TABLE 2 (OF NEDC-32264-A)RANKING CRITERIA FOR MOVSFOR GENERIC LETTER 89-10

RANK	CRITERIA[a][b][c]	NOTES
High	>1% CDF GL 89-10 MOVs	Additional MOVs can be added based on judgement, sensitivity analyses.
Medium	≥1% CDF ≥ 0.1%GL 89-10 MOVs	
Low	Remaining GL 89-10 MOVs < 0.1% CDF	Adequate justification for valves in this category should exist.

[a] These importance criteria establish the baseline for valve inclusion. However, as noted in Task 4, qualitative assessments further evaluate the inclusion of other MOVs.

[b] Similar criteria for Level 2/RRF should be utilized.

[c] See ADDENDUM 1 for correlation of % CDF and F-V.

Failure rates for the MOVs were modeled in NEDC-32264-A primarily as 3 per 1000 demands and 8.7 per 100 demands for the BWRs studied. A failure rate of around 3 per 1000 demands had been used for MOVs in previous risk studies such as WASH-1400 (as reported in NUREG/CR-1363). NUREG/CR-5140 documented a failure rate of about 8.7 per 100 demands based on the licensee responses to NRC Bulletin 85-03.

Section 2.3.1 and 2.3.2 of NEDC-32264-A included discussions of "multi-component" issues where an MOV might be more important because of the simultaneous failure of another MOV, along with related discussions of common-cause failure modeling. It appears that conditions related to common-cause failure of multiple MOVs caused by common adjustment to design defects (e.g., torque switch setting deficiencies or insufficient operator margin) are minimized in the discussions. The common-cause failure mechanisms for AOVs and MOVs may not be directly comparable with respect to the deficiencies in the powering mechanisms and valve operators, and thus the assumptions used to estimate risks due to common-cause MOV failures might not be appropriate if extrapolated to discussions of common-cause failures of AOVs.

A discussion of the results of application of the above outlined methods to five BWR plants follows in the NEDC-32264-A report. The MOV prioritization results are summarized in Table 4 of NEDC-32264-A. Details of the studies for each of the five BWRs are included in Appendices to the NEDC-32264-A report. The ranking process and use of various risk importance measures were studied further in a section labeled Addendum 1 which was included at the end of the report.

Definitions of importance measures were included in Section 3.1 of the Addendum in the NEDC report. These are considered to be pertinent to discussions of risk and are therefore reproduced below for information.

SYMBOLS:

T = Base core damage frequency for all basic events

U = Failure probability (or unavailability) of individual basic event

T(0) = CDF with basic event assumed to never occur (i.e., probability set equal to 0)

T(1) = CDF with basic event assumed to occur (i.e., probability set equal to 1)

From the above basic PSA inputs the following importance measures can be calculated for each individual basic event:

a. Risk increase where basic event is assumed to occur (i.e., basic event probability set equal to 1).

   Risk Increase = T(1) - T

b. Risk reduction where basic event is assumed never to occur (i.e., basic event probability set equal to 0).

   Risk Decrease =b T - T(0)

c. Fussell-Vesely (FV) importance is the fraction of the CDF which involves the basic event divided by the base CDF. In some PSAs this represents the sum of the CDF for the minimum cutsets containing the basic event divided by base CDF. A minimum cutset is defined as the smallest combination of failures (or basic events) which, if they all occur, will cause the top event (core damage) to occur.

   FV = {bT - T(0)}/T

d. Criticality (CRIT) importance is as follows:

   CRIT = {[T(1b) - T(0)] x U}/T = FV

e. Risk Reduction Worth (RRW) is the base CDF divided by CDF with U = 0.

   RRW = T/T(0)

f. Birnbaum (BIRN) importance is as follows:

   BIRN = T(1) - T(0) = CRIT x (T/U)

g. Risk Achievement Worth (RAW) is the CDF with U = 1 divided by base CDF.

   RAW = T(1)/T

h. Cumulative % Risk Contribution is calculated by first ranking (sorting) the basic events by decreasing "Risk Decrease" or decreasing "F-V." The % risk reduction is the risk decrease divided by the sum of the risk decreases of all basic events. The cumulative % risk reduction is then the sum of the individual % risk reduction in order of their size. The number of SSCs included depends upon the total cumulative % of interest.

$$\%\ \text{Risk Reduction} = \frac{[T - T_j(0)]}{\sum_{j=1}^{m}[T - T_j(0)]} \times 100$$

$$\text{Cumulative \% Risk Contribution} = \sum_{i=1}^{n} (\%\ \text{Risk Reduction})$$

where "n" is the number of SSCs required to obtain the cumulative % risk of interest.

A NRC Preliminary Safety Evaluation Report (SER) on the first revision of NEDC-32264-A, comments from the BWR Owners Group, and a final SER were included at the end of the report. The NRC staff determined that the methodology described in the report was acceptable, subject to a number of comments in the SER. The use of risk-related rankings to establish test intervals was accepted and it was noted that the methodology was considered to be a pilot in the NRC's PRA Implementation Plan. The nuclear power industry proposed to apply risk analysis techniques in the inclusion of MOVs for the licensees' Generic Letter 89-10 programs. They planned to exclude certain "low risk" safety-related MOVs and perhaps include certain "high risk" but non-safety-related MOVs. It appeared that their proposed commitments toward surveillance of high risk but non-safety-related MOVs were ambiguous. However, the NRC's SER indicated that licensee commitments to ASME Code testing and to the recommendations in Generic Letter 89-10 to include all safety-related MOVs were to remain in place.

15.6.2 Westinghouse Engineering Report V-EC-1658-A

The purpose of the Westinghouse Report is to provide a guideline on how Westinghouse NSSS Owners can rank MOVs according to their risk importance. The report describes a suggested process for MOV risk ranking and is based on previous risk ranking programs including NEDC-32264-A, described above. This is part of a Joint Owners Group (JOG) effort to ensure MOV operability. The Westinghouse guideline is oriented toward implementation of NRC Generic Letter 96-05 regarding periodic verification of MOV operability at design basis conditions.

The Westinghouse report includes a separate (from NEDC-32264-A) risk ranking guideline for MOVs based on a six step process, as follows:

- IDENTIFY MOVS TO BE CONSIDERED.

  A separate JOG report (MPR-1807) is used to identify MOVs within the scope of the program.

- CALCULATE MOV AT-POWER RISK IMPORTANCES.

  Plant PSAs are used to determine at-power risk importances for MOVs within the scope of the program. The failure modes considered were failure to change position on demand. Importance measures similar to those described above in NEDC-32264-A are used in the risk ranking process. In addition, the Westinghouse Report notes (Section 3.2.1) the relationship between FV and RRW as RRW = 1/(1 - FV) and quotes the importance measure used by the BWROG for risk ranking MOVs for implementation of GL 89-10 as "Importance = (Sum of all accident sequence frequencies related to a specific MOV) / (Total CDF)."

A three level approach to risk ranking MOVs is recommended in the Westinghouse Report based on the following criteria:

Risk Category	Ranking Criteria
High	FV > 0.01 or RAW >10
Medium	0.01 > FV > 0.001 and RAW < 10 or 10 > RAW > 2 and FV < 0.01
Low	FV < 0.001 and RAW < 2

- ASSESS PSA COMPLETENESS ISSUES.

  Several issues are discussed in Section 3.3 of the Westinghouse report that are directly pertinent to the study of AOVs as well as MOVs. The issues of the completeness of failure data and the accuracy of failure rates are discussed in the Westinghouse report. The report indicates that failure rates ranging from 4 in 10000 demands to 1 in 100 demands to open or close were used. It was noted that the NRC took issue with these rates in their SER (enclosed as part of the report) and noted the failure rate of 8.7 per 100 demands documented in NUREG/CR-5140.

  Accident sequence and cutset truncation limits that account for at least 90% of the PSA CDF are recommended. Also, common-cause evaluations for at-power risk analyses are to be considered in the normal manner for PSA analyses.

  The Westinghouse Report endorses PSA Level II analyses for evaluating the importance of MOV failures on release of contamination due to containment failures and indicates that RAW and FV importances should be evaluated.

  The Westinghouse Report notes that CDFs for some plant initiating events are based on plant-specific design and component reliability. The initiators listed are loss of normal service water, loss of component cooling, loss of chemical and volume control system, and total or partial loss of main feedwater. These issues, along with loss of instrument air or air system or component overpressurization, as noted previously, are pertinent to evaluation of the risk impact of AOVs. Similarly, it is noted in the Westinghouse Report that not all MOV failures are modeled as basic events in the PSA model and that they may be subsumed or combined with other events. Engineers should be aware of these factors and consider them in the risk analyses. This is also the case for AOVs.

- EVALUATE OTHER CONSIDERATIONS.

  Section 3.4 of the Westinghouse Report indicates that in addition to the PSA importance measures, the following considerations are pertinent and should be considered by a designated panel of experts:

  - Components not modeled in the MOV program or PSA model;
  - Shutdown risk;
  - External events;
  - Component operating history;
  - Assessment of interchanging functions of MOVs;
  - Plant-specific configurations (equipment out of service for maintenance and trains or equipment out of service for long periods of time, as referred to in 10 CFR 50.65, The Maintenance Rule);
  - MOV importance in the design basis analysis performed to satisfy the FSAR and Technical Specification requirements; and
  - Failures of MOVs that could lead to other accident scenarios.

- DEVELOP COMPONENT RANKING WORKSHEETS.

  Worksheets are included in the Westinghouse Report that are to contain the pertinent information gathered during the ranking and risk evaluations. These are also used to assist the expert panel in making decisions about ranking specific MOVs.

- CONDUCT EXPERT PANEL SESSION FOR MOV RANKING.

  The expert panel is to evaluate the importance of each MOV and categorize them in priority ranking.

A NRC SER was published in the final Westinghouse report and use of the report was determined to be acceptable subject to several conditions and limitations summarized at the end of the SER. Among the conditions were:

- Licensees must document their reasons for categorizing MOVs as having low safety significance.

- Accuracy of failure rate data and operating conditions under which the data were collected are to be verified.

- Diagnostic test intervals of up to 10 years are approved provided the potential for common-cause failures caused by maintenance or test activities are considered by the expert panel.

- Both risk and deterministic criteria for test frequencies to ensure design basis capability are satisfied.

Intersystem common-cause failure rates were not considered to be a pertinent issue for MOVs in the Westinghouse Report (as well as in the BWROG report described previously) because the implemented recommendations of GL 89-10 and GL 96-05 precluded such occurrences. Further, implementation of the recommendations of GL 89-10 and GL 96-05 was relied upon within the industry to ensure that MOV failure rates were actually around to 3 in 1000 demands so that the assumptions in the various risk analyses (PSAs, etc.) could be considered accurate. It is noted that no such program exists at the current time for AOVs; therefore, the issue of accuracy of failure rate estimates for AOVs is significant in assessing the conservativism of risk analyses.

16. CONCLUSIONS

16.1 AOV Population

There are large populations of safety-related and important non-safety-related AOVs including risk significant AOVs, in all of the plants visited. AOVs generally number in the hundreds and SOVs number in the thousands in each plant. AOVs may be found in a large number of systems including engineered safety feature systems at many plants. There are large varieties of AOV designs and manufacturers. Combined choices of valve, operator, and trim configurations are almost unlimited. One licensee has installed AOVs from 38 different manufacturers. Licensees may not be fully aware of the presence of SOVs that are internal to equipment. The population of AOVs and the variety of designs present a serious challenge to the operators to ensure that all of their important AOVs have been properly selected for the intended application, designed, installed, set up properly, and thereafter maintained in operating condition.

16.2 AOV Design Basis Demands and Margins

One of the first tasks planned for licensees to perform in the proposed model AOV programs (Fermi 2 and Palisades) was to review and confirm the design bases for AOVs. Environmental conditions as well as load demands are to be reviewed. This involves a considerable amount of effort on the part of licensees because the information may be scattered, fragmented, or missing. Review or establishment of design basis demands was also one of the first tasks performed for motor-operated valves under the guidance in Generic Letter 89-10. Licensees may find, as was the case with motor-operated valves, that the design bases for AOVs include incomplete information and/or nonconservative assumptions (refer to Section 12.3). Design basis demands need to be reviewed, or established, as a first step in order to compare them with the margins available in AOVs.

Licensees may not know that the design basis loads or environmental conditions can be met with acceptable margins for the important AOVs in their plants. Licensees that were visited are relying on architect/engineers or manufacturers' information and calculations or, more recently, are conducting design reviews to determine the design basis margins for their AOVs. Several manufacturers' AOV calculations or valve descriptions were found that included mistakes or inaccurate information. Some licensees have conducted analyses and/or tests to verify or determine the design bases and margins for selected AOVs.

Licensees are at various stages regarding their determination or confirmation of the design basis demands on safety-related and important non-safety-related AOVs. Palisades and Fermi are lead plants in an EPRI–sponsored analysis project to determine design basis demands and thereafter calculate the ability of AOVs to meet those demands. The effort is restricted to Category 1 and 2 AOVs as described in Section 12.3. The categorization of AOVs, in terms of safety significance (and thus, subsequent attention by the licensees) varies from plant to plant. There is no NRC requirement for a standardized categorization.

Plant licensing documents imply that licensees have confirmed that positive margins are available for safety-related AOVs to meet their design basis demands. The plant licensing documents also imply that failures of AOVs in important non-safety-related systems will not interfere with safety-related functions (shutdown, remain shutdown, and mitigate offsite dose). A similar situation existed regarding the operability of motor-operated valves before Generic Letter 89-10 was issued. Licensees discovered a number of interesting details about their design bases and the abilities of motor-operated valves to meet their commitments. Similar efforts on the part of licensees regarding AOVs can be expected to have similar results.

16.3 AOV Failures

A number of examples were found where AOVs, both safety-related and non-safety-related, did not have sufficient designed margin to meet their design basis or operational demands. Problems associated with the AOVs and their operators included improperly defined diaphragm areas (LaSalle), inadequate diaphragm materials (Dresden, Quad Cities, Indian Point, et. al.), improper estimates of packing loads, and underestimated valve factors (Palo Verde). The issue of the ability of safety-related and important non-safety-related AOVs to perform their functions is considered to be safety significant.

In addition to the operability problems caused by insufficient margin or by the air systems, AOVs are subject to all of the environmental conditions, maintenance problems, and aging mechanisms of other valves in nuclear power plants. NRC generic communications are included in Appendix A that describe problems encountered with AOVs/SOVs or air systems in the nuclear power industry. The events and failures included in Tables 2, 3, 4, and 7 in this study represent only a sample of the total number of events and failures involving AOVs.

At almost all plants visited, a number of pertinent AOV and SOV-related events and conditions, including common-cause failure events and conditions, were not included in Licensee Event Reports or reports to the industry databases that the authors had access to. Licensees determined that they did not have to report these events or conditions to the NRC if a safety function of safety-related equipment was not compromised.

16.4 Air Systems

High quality air is critical to ensure the operability of AOVs. A reliable supply of clean, dry, oil-free air at specified pressure is a necessary, but not a sufficient condition for operability of AOVs. Failures in air systems that serve AOVs are a source of common-cause failures in AOVs and this consideration is safety significant.

Every plant visited had experienced major problems with air quality and had taken actions, some quite extensive, and mostly in response to Generic Letter 88-14, to address their air quality problems. Three of the seven licensees visited do not definitively know, on a day-to-day basis, if the quality of air (or nitrogen) in their plants has degraded with regard to moisture contamination. LaSalle, TMI, Indian Point 3, and Turkey Point 3 and 4 are equipped with continuous dew point monitoring and alarming (Turkey Point does not have alarms in the control room) of the instrument air systems. The other licensees check moisture content in their air systems by observation at varying periods ranging from taking readings at each shift (on rounds) to testing semi-annually. Note that these plants also have other air systems containing AOVs of interest that are not equipped with devices to measure dew point.

AOVs, positioners, and regulators can be, and many times have been, directly compromised by moisture in the air that is used to operate the valve. These components can also be adversely affected by particulate contamination generated by moisture-induced corrosion of air piping and valve parts, or can be damaged by hydrocarbon attack on some elastomers. AOV positioners can malfunction because of variations of air pressure in the air system.

High air pressure can damage some valves, particularly solenoids, and can do so simultaneously, depending on the circumstances. Low air pressure can directly affect the performance of an AOV (or several AOVs) or cause it (them) to move to unexpected and/or unanalyzed positions. Particulate contamination from external sources is normally removed by filtration. Corrosion products can be released into the air system downstream of filters, as the result of shock, vibration, or seismic events, depending on the material and condition of the piping.

Over a period of time, contaminated air from moisture and/or particles may be a more serious problem than a sudden failure of the air system to provide air (the usually assumed failure in risk assessments and other system failure analyses) in that the contaminants provide a mechanism, suddenly or over time, to cause multiple AOV failures or degradation leading to failure. This situation calls into question the argument that AOVs connected to high pressure air systems, accumulators, or instrument air systems will "fail-safe," i.e., move to the safe positions that are assumed from a loss of air. Air quality must be verified (continuously or very frequently) to be satisfactory in order to substantiate the "fail-safe" assumption. The provisions of Generic Letter 88-14 to ensure air quality over the long term are somewhat vague on this issue and some events indicated examples of deteriorated air quality over time (refer to the discussion in Section 10.1 of the provisions of GL 88-14 and operating experience since its publication).

16.5 Accumulators

The quality of air or nitrogen supplied by accumulators serving individual AOVs is generally not monitored or verified on a regular basis to ensure operational reliability. Licensees may get verification and certification of nitrogen bottles from outside vendors. Many accumulators do not have drains and/or convenient means to check them to ensure that they are clean and dry. Further, if an accumulator has water in it, the lost volume of air or nitrogen could be significant if repeated strokes are required by the design basis for a particular AOV. There is also the potential for contamination of the air or direct interference with the performance of the operator from moisture in the air. Since accumulators inherently function as drain traps in the air system, the potential for having moisture and/or particulate contamination in them is likely, unless the air system is "extremely dry" at all times and the accumulators are monitored in some way.

Accumulator check valve failures can cause partial loss of air pressure. A partial loss of air pressure could cause the valve to fail in an intermediate position or not fail safe. Also, the loss of air may not be readily apparent and the AOV could then fail when operation is attempted.

Safety-related AOVs may not be supplied directly by non-safety-related air systems but the air supplied to safety-related accumulators is usually, but not always, supplied by a non-safety-related instrument air system. The potential for moisture or particulate contamination, partial loss of pressure due to check valve malfunction, or loss of accumulator volume through water intrusion indicates a direct safety concern to ensure the operational reliability of both the air system and the accumulators that serve safety-related and important non-safety-related equipment.

16.6 Solenoid-Operated Valves

SOVs are a source of common-cause failures in both safety-related and important non-safety-related AOVs because of their numbers, their dependence on the air system, and their similar characteristics and common-cause failure mechanisms (see Section 11). The SOV failures and failure mechanisms described in NUREG-1275, Volume 6, in 1991 are currently commonplace in the nuclear industry. The safety significance of particular SOV failures depends on the application, however the number of safety-related and important non-safety-related applications is large enough that attention should be focused on these components.

An NRC study, NUREG/CR-5500, Volume 3, "General Electric Reactor Protection System Unavailability, 1984-1995 (draft 2)," indicated that about one third of BWR RPS unavailability was due to common-cause failures of the hydraulic control unit and backup scram SOVs in that system.

The Maintenance Rule is driving efforts by licensees to implement many of the independent recommendations regarding SOVs made in 1991 in Section 9 of NUREG-1275, Volume 6. Those recommendations were:

- Take corrective actions to address the root causes of SOV failures based on risk significance and determined from a plant-specific prioritization scheme.
- Review SOV design specifications, calculations, and operating conditions including temperature and pressure limitations.
- Conduct reviews to identify SOVs that may have been overlooked and to verify the orientation of SOVs.
- Replace or refurbish SOVs on a timely basis.
- Ensure that the air systems that serve SOVs are a reliable source of clean, dry, oil-free air at proper pressure.

16.7 AOV Program Plans

All of the seven plants visited had AOV action plans in varying stages of completion at the time of the visits. One plant was focusing on maintenance practices, assuming that the design basis operability, as defined in the original design, is acceptable. The AOV action plans appeared to be comprehensive and indicated awareness of the problems associated with AOVs, the diagnostic tools available in the industry, and the work of various organizations and utilities regarding AOVs.

Recent licensee efforts are being driven by the Maintenance Rule, recent industry-wide correspondence, a number of events involving AOVs, results from their IPE and PRA investigations, and/or experience gained from the MOV programs established to meet the recommendations of Generic Letter 89-10. Some licensees have not started these efforts while others are well on their way, but no licensees visited had completed their investigations. A similar situation exists with regard to licensees' determination of the margins that exist within their AOVs to meet the design basis demands. Concerns include the quality of aging assessments and various failure mechanisms that the plants may not be considering in their assessments of the available margins in AOVs. There is no uniform schedule to implement AOV program plans in nuclear plants.

16.8 Industry Initiatives Regarding AOVs

The nuclear power industry is aware of the significance of AOV operability as a logical extension of their work to ensure the design basis operability of motor-operated valves, as well as a result of the work done to prioritize the importance of equipment in nuclear power plants, in order to comply with the requirements of the Maintenance Rule. A number of industry organizations including the Electric Power Research Institute, the Air-Operated Valve Users Group, and the Air-Operated Valve Joint Owners Group (JOG AOV) recently produced and distributed an industry-wide program plan to "provide assurance that AOVs are capable of performing their intended safety-significant, i.e., risk significant functions."

16.9 The Maintenance Rule

Licensees are addressing maintenance practices to meet the requirements of the Maintenance Rule with respect to the operability of safety-related and important non-safety-related systems and equipment in their plants using a variety of risk related tools and techniques. The system of establishing goals for systems and equipment found to have operational problems appeared to be effective (at least as far as AOVs are concerned) in the plants visited. In general, licensees at the plants visited are using techniques developed in response to the Maintenance Rule to cover equipment outside, and in addition to, the scope of NRC's interest in order to increase the reliability and efficiency of their plants. As a result of their Maintenance Rule activities, licensees are increasingly aware of the importance of AOVs, in terms of plant safety, availability, and heat rate.

16.10 Diagnostic Systems, Maintenance, and Testing of AOVs

Diagnostic testing (computerized measurement and software) tools are used by all licensees for motor-operated valves and appear to be valuable for determining maintenance or aging-related valve or actuator problems. Modern AOV diagnostic tools are now becoming available. Extrapolating no-load test data to design basis conditions remains to be verified as acceptable, just as with motor-operated valves.

Claims of the diagnostic system vendors regarding accuracy and abilities of their systems have not been independently verified, along with assumptions about calculated forces such as packing loads and side loads.

The ASME Section XI stroke-timing test alone does not provide assurance of future operability of AOVs.

16.11 Risk Analysis Involving AOVs

AOVs have been examined in several risk studies. The findings in the study by Pickard, Lowe and Garrick, NSAC-128 (see Section 15.1 herein) described the importance of the air system and emphasized the importance of providing clean, dry, oil-free air at specified pressure to AOVs. Recently, NUREG/CR-6644 (see Section 15.3.2 herein) indicated that at some plants AOV reliability can have important effects on system reliability and plant risk.

In NUREG/CR-5472, "A Risk-Based Review of Instrument Air Systems at Nuclear Power Plants," the researchers studied the IA systems in nuclear power plants in terms of their contribution to risk. In general, the conclusions and recommendations in NUREG/CR-5472, support the findings in this current study. See Section 15.1 for a discussion of those conclusions and recommendations.

An Accident Sequence Precursor analysis (Section 15.2 and Appendix B) indicates the relative risk significance of a number of events involving AOVs. There were 26 precursor events involving AOVs, 12 of which had CCDPs $\geq$ 1E-4. For several years, AOV precursor events were in the top quarter of all precursor events. The highest CCDP for an AOV event was 9E-4, that being the common-cause failure of all trains of AFW at both Turkey Point units due to water contamination of the instrument air system. See Section 8.7.1 herein and Section 5.1.2 in NUREG-1275, Volume 2.

The operating experience gathered as part of this study, and observed in the plants during the recent visits, indicated that the detailed reviews conducted by each plant (see Section 15.3.1) disclosed AOVs which are important and risk significant. Many of the AOV events, failures, and deficiencies were not reported to the NRC and, as a result, were not reflected in the data used for risk analyses. A number of examples are included in Tables 2, 3, and 4, of this study. Also, recent risk studies (see Section 15.3.2) appear to have underestimated AOV risk. They were prepared using models which did not account for air system contamination, air system overpressurization, or other conditions which could cause AOV common-cause failures.

Many of the licensees' individualized plant reviews for Maintenance Rule evaluations and for the plants' AOV programs, included reviews of PRA findings by expert panels. The licensees found that by using their plants' operating experience and plant-specific probabilistic risk assessments, certain AOVs were found to have high risk importance, high risk achievement worth, or were important to preventing large early releases.

Generally, current risk models and databases either do not model the air systems, or do not include consideration of malfunctions of the air systems (i.e., air systems that do not consistently provide clean and dry air at proper pressure).

These models may also not include consideration of shutdown modes, not include (or model accurately) a number of common-cause AOV failure scenarios, or they may include overly optimistic nonconservative AOV failure rates (Section 15.5).

Some of the PRAs lump the AOVs with systems or consider skid mounted equipment in one single failure scenario, and the dependency on AOVs may not be apparent in such cases. Non-safety-related air systems are assumed to fail or shut down upon loss of offsite power in most cases and the AOVs that provide a safety function are assumed to move to, or remain in a "fail-safe" position if that occurs. Alternatively accumulators are provided to allow the AOVs to be manipulated if the air systems are unavailable and these accumulators are assumed to provide the necessary quantities of clean, dry air. These assumptions are predicated on the performance of equipment which is designed, installed, qualified, maintained, and tested to meet the required demands.

17. SAFETY SIGNIFICANCE

The following findings and conclusions of this study are considered to be safety significant:

- There have been a number of events involving AOVs and reported conditions of AOVs that indicate that AOVs have reduced operating margins caused by such factors as aging, load mechanisms not understood or considered in the original design, or previously contaminated air. Licensees may not know if the design basis loads or environmental conditions can be met with acceptable margins for the AOVs in their plants. Licensees are either relying on manufacturers' information and calculations or, more recently, design reviews of the AOV design bases. Several manufacturers' AOV calculations or valve descriptions were found that included mistakes or inaccurate information. The knowledge (or lack thereof) of the design bases and margins in AOVs to meet the design basis demands can be a matter of safety significance.

- Air systems are potential sources of common-cause failures of AOVs. High quality air, i.e., clean, dry, oil-free air, at specified pressure, is essential to acceptable performance of AOVs and SOVs. A reliable supply of high quality air is required at all times in order to ensure that AOVs will be able to move to or remain in their fail-safe positions.

- Accumulators in the air system that supply air to safety-related and important non-safety related AOVs can be sources of contamination, reduced capacity, and subsequent AOV inoperability unless it is verified that the accumulators are free of contamination, do not contain trapped water, and the accumulator check valves are functioning. Licensees need to know that the accumulators are functioning properly in order to justify the assumptions that AOVs will move or remain in their fail-safe positions.

- Air-operated dampers are potential sources of failure of emergency diesel generators, control room ventilation systems and other safety-related systems that they may serve.

- SOVs are potential sources of common-cause failure of AOVs. Root causes of failure include potential contamination from the air system, design, qualification, and maintenance. These causes have been reported previously and need to be systematically addressed by licensees based on the safety status and risk significance of the SOVs.

- The operating experience gathered for this study and observed in the plants during the recent visits indicates that many AOVs are risk significant. During the plant visits, we learned of many common-cause failure events and deficiencies that had been reported in plant condition reports or plant deficiency reports but were not reported in LERs or in other reports to the NRC. As a result those common-cause failures were not included in common-cause failure databases. Therefore, factors used in risk analyses for estimating the effects of common-cause failures of AOVs were nonconservatively affected by such under reporting. Incomplete modeling and quantification of common-cause air system failures and degradations appear to have resulted in under-estimates of the risk significance of AOVs. Individual AOVs may be found to be of low risk significance but the common-cause failure of two or more AOVs performing the same function may have considerable risk significance.

- Accident Sequence Precursor analyses indicate that there have been a number of risk significant events involving AOVs. There were 26 precursor events involving AOVs, 12 of which had CCDPs $\geq$ 1E-4. For several years, AOV precursor events were in the top quarter of all precursor events. The highest CCDP for an AOV event was 9E-4, that being the common-cause failure of all trains of AFW at both Turkey Point units due to water contamination of the instrument air system. This event is noted, although it occurred over 14 years ago and before the publication of Generic Letter 88-14, because it demonstrates that AOVs can be important components that can significantly affect risk.

- The number and scope of NRC generic communications and studies of AOV and air systems provide an important indicator of the overall safety significance of these components and systems.

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Tables

Table 1. Populations of air-operated valves in plants visited.

Plant Name	Safety-Related AOVs	Category 1 AOVs	Category 2 AOVs	Category 3 AOVs	GL 89-10 MOVs
Palo Verde 1, 2, and 3	41 + 131 = 172 AOVs per plant are classified by the licensee as safety-related. See Category 1 and 2.	41 AOVs per plant are classified by the licensee as Category 1. The licensee refers to active safety-related AOVs as Category 1.	131 AOVs per plant are classified by the licensee as Category 2. The licensee refers to non-active safety-related AOVs as Category 2.	Approx. 2628 AOVs per plant are classified by the licensee as Category 3. The licensee refers to non-safety-related AOVs as Category 3.	There are 831 MOVs on site (3 plants) of which 336 are in the GL 89-10 program.
Fermi 2	29 AOVs in Category 1 and 34 AOVs in Category 2 (are safety-related, as well as 370 AOVs for Scram inlet and outlet valves.(There are also 2482 SOVs of which 1442 are classified by the licensee as QA1.)	410 AOVs are classified by the licensee as Category 1. The licensee refers to AOVs having "high safety-significance" as Category 1. Included are 370 SCRAM inlet and outlet valves, 29 other safety-related valves, and 11 AOVs that support a non-safety-related, risk significant function.	84 AOVs are classified by the licensee as Category 2, including 34 safety-related AOVs. The licensee designates as Category 2 those less safety significant AOVs that support safety-related functions or have relatively high economic consequences if they should fail.	Category 3 AOVs are those "having little or no safety significance or economic consequences." (Note: The original 1995 rough outline for development of the Fermi 2 AOV program lists a total of 2058 AOVs, of which 598 were considered safety-related valves or dampers and 1460 were considered non-safety-related valves or dampers.)	147 MOVs are in the GL 89-10 program.
Palisades	191 AOVs.	111 AOVs. Valves in this category are safety-related with active safety functions, important-to-safety based on their PSA risk significance, or included based on Expert Panel determinations.	42 AOVs are classified by the licensee as Category 2. These AOVs are safety-related but of low risk significance or non-safety-related but used in "critical" applications.	Approximately 561 AOVs, which are not Category 1 or 2 are classified by the licensee as Category 3 AOVs.	There are 54 MOVs in the plant of which 30 are covered by GL 89-10.
LaSalle 1 and 2	84 for both units. In addition, 370 CRD valves in each unit are classified by the licensee as safety-related.	AOVs having high safety significance. Number not provided.	AOVs having low safety significance. Number not provided.	AOVs having high economic significance. Number not provided. (LaSalle categorizes AOVs with no or limited safety/economic significance as Category 4.) (There are 1575 non-safety-related AOVs for both units.)	There are 200 MOVs in the GL 89-10 program, for both units.

Table 1. (continued).

Plant Name	Safety-Related AOVs	Category 1 AOVs	Category 2 AOVs	Category 3 AOVs	GL 89-10 MOVs
TMI 1	98 AOVs are classified as safety-related (designated "Q-class" or "Class 1") by the licensee.	98 AOVs are categorized as Class 1 by the licensee. These are AOVs with an active safety function.	328 AOVs are categorized as Class 2 by the licensee. These are AOVs with an EOP function or operational economic significance.	484 AOVs are categorized as Class 3 by the licensee. These are AOVs not categorized 1 or 2.There are a total of 910 AOVs at TMI-1.	There are 81 MOVs in the GL 89-10 program for this plant.
Indian Point 3	263 AOVs are classified as safety-related by the licensee.	The licensee did not classify AOVs as Category 1, 2, or 3. (215 AOVs were classified by the licensee as being within the scope of the Maintenance Rule, 10 CFR 50.65.)	The licensee did not classify AOVs as Category 1, 2, or 3.	The licensee did not classify AOVs as Category 1, 2, or 3. (There are 578 AOVs in the plant, therefore, 578 - 263 = 315 AOVs are non-safety-related.)	89 MOVs are within the scope of GL 89-10.
Turkey Point 3 and 4	The licensee classified 191 AOVs (total for both units) as safety-related.	174 AOVs (98 active, 76 passive, total for both units) are classified by the licensee as Category 1.	53 (34 active, 19 passive, total for both units) are classified by the licensee as Category 2.	There are 836 AOVs in both units. It is not known if the licensee specifically designated some AOVs as Category 3.	111 MOVs (total for both units) are within the scope of GL 89-10.

Notes:

1. The category designations in the table vary from plant to plant. The use of the categories for each plant is explained with the entry.
2. There may be SOVs in the plants that are classified as part of the AOV. Figures for SOVs were included if separate data was provided.

Table 2. Selected events documented by NRC Licensee Event Reports (LERs).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Beaver Valley 1	33490007	900330	Operation at 100% reactor power, operators noted closure of the "C" Main Feedwater Regulating Valve, FCV-FW-498. The operator tried to manually open FCV-FW-498; however, the valve would not respond. A reactor trip on "SG 'C' Low Level & Feedwater Flow Low" occurred. The cause for the reactor trip was the closure of FCV-FW-498, in response to insufficient instrument air pressure supplying the valve positioner. Moisture in the instrument air supply had plugged the filter in the air regulator for the current-to-pneumatic valve positioner for FCV-FW-498, causing the valve to close. This moisture was present in the instrument air system as a result of the instrument air dryer being out of service. While the air dryer was out of service, the air compressors' discharge was directed through the instrument air bypass filters which did not remove the moisture. This is considered to be a common-cause failure condition.	a, c	Main Feedwater Instrument Air
Calvert Cliffs 1	31788009	880824	The unit tripped on Loss of Load when the Main Turbine tripped on high Steam Generator level. The Main Turbine tripped when the air line on #12 Main Feed Regulating Valve failed and the valve failed to the full open position causing a high level in #12 steam generator. The air line to the feed regulating valve failed due to vibration and stress. The 1/4 inch instrument air line nipple at the junction of #12 MFRV's valve positioner sheared due to cyclic stress and fatigue. The improper location of a pressure switch in the instrument air supply line contributed to the total stress on the nipple. The weight of the pressure switch, along with feed header-induced vibrational stress, eventually caused the nipple to fail. Air pressure on the top and bottom of the valve diaphragm bled off simultaneously, and pressure beneath the valve plug from main feedwater forced it up, thus fully opening the valve. Normally, the valve will fail in the "as is" position if air pressure decreases to less than 70 psig as seen at the pressure switch. In this event the loss of air pressure was downstream of the pressure switch and thus the valve moved, unexpectedly, to the full open position.	c	Main Feedwater

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Calvert Cliffs 1	31789005	890314	With the Unit operating at 100% power, a partial loss of Instrument Air (IA) occurred due to a failed check valve. The low IA event was ended when the Air Compressors were started and restored IA pressure. The event was caused by the wrong check valve type installed in the IA system causing the valve internals (flapper and swing hinge) to fail due to improper wear pattern. The repair and leak testing of the similar replacement check valve showed that the valve type was inappropriately chosen for its intended application. The existing check valve was primarily designed to function with a higher header pressure (i.e., about 250 psig instead of an actual operating pressure of 100 psig) and uses a hard seat and disk. The check valve needs the higher differential pressure and rapid flow reversal to ensure an airtight closure. This valve was to have been replaced with a different check valve better suited for its intended use. Therefore, the root cause of this event was the valve type being inappropriately chosen for its intended application. This was an ASP Program Event. See Appendix B.	c	Instrument Air
Calvert Cliffs 1	31789018	891106	A condition was discovered that could have prevented the fulfillment of certain systems to remove residual heat and control the release of radioactive material after a Loss of Coolant Accident (LOCA). During the performance of a test to satisfy specific requirements in Generic Letter 88-14, it was discovered that many air-operated control valves and piston-operated ventilation dampers which utilize safety-related air accumulators would not have performed as expected after a loss of normal non-safety-related instrument air. The root cause of the event was identified as a lack of an adequately documented design basis combined with inadequacies in the testing and preventative maintenance program for the Instrument Air System. This is considered to be a common-cause failure condition.	c	6 Different Systems Instrument Air

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Catawba 1	41397002	970506	Feedwater Containment Isolation valve 1CF-51 was inoperable due to actuator low nitrogen pressure. Each Steam Generator Feedwater Containment Isolation Valve is a pneumatic-hydraulic controlled gate valve whose safety related function is to terminate flow in either direction. The valve actuator is a Borg Warner #37981 that uses nitrogen pressure to close the valve. Per the Borg Warner instruction manual, the valve is inoperable when nitrogen pressure is below 2050 psig decreasing signal. The alarm pressure switch setpoint is calibrated at 2100 psig (+ or - 50 psig) and provides alarm indication through the Operator Aid Computer (OAC). On April 3, 1997, between the hours of 0608 and 2319, valve 1CF-51 was inoperable due to low nitrogen pressure. This inoperability time exceeded the 10-hour time limit allowed by Technical Specifications. The root cause of this event was inadequate information in the OAC Alarm Response. The Alarm Response did not indicate the need for immediate action and did not include setpoint information that would have led the operators to recognize the close proximity of the alarm setpoint to the minimum allowed nitrogen pressure for valve operability.	c	Feedwater Containment Isolation Valve
Clinton	46190004	900703	Investigation of information notice 88-24 revealed that the maximum operating pressure differential (MOPD) of 73 solenoid-operated valves (SOVs) supplying air to active safety-related air operated valves (AOVs) and dampers was less than the maximum instrument air (IA) system pressure (max IAP). The SOVs would be subjected to max IAP in the event of air regulator (AR) failure. Loss of air had been previously considered as a failure mode; over pressurization had not. Further investigation identified that AOV components and other safety-related end-use devices also have the potential for over pressurization if their upstream ARs fail open. The over pressurizations (AOV components, SOVs, and other end-use devices), may result in failure of safety-related devices to reposition to their safety positions. The potential for over pressurization was the result of errors by vendors, the nuclear station engineering department, and the architect engineer (AE), and miscommunication with the AE. Corrective actions included installing rupture disks and/or high MOPD SOVs, and replacing ARs with safety-related, seismically qualified, ARs. The potential over pressurization of the AOVs and two end-use devices was determined to be reportable in accordance with the provisions of 10 CFR Part 21 This is considered to be a common-cause failure condition.	b, c	Miscellaneous Active Safety-Related Systems

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Clinton	46198009	980203	Engineering personnel determined during their review of a 10CFR21 notification by Engine Systems, Incorporated, that the air start solenoid pilot valves for the emergency diesel generators (EDG) would not operate as required by the design basis. The solenoid valves, which are used to valve air to the air operated main valves that supply starting air to the start motors on all three EDGs, would not operate reliably at the low end of the design basis air start system pressures and DC voltages. The apparent cause of the inadequate design of the solenoid valve was a failure of the design engineer to consider the full range of design basis design pressures and DC voltages when changing the spring size of the solenoid pilot valve. Corrective action for this event included: changing the spring size in the solenoid valve for the Division III EDG air start System; replacing the Division I and II EDG air start system solenoid valves, or otherwise modifying the system to meet the design basis requirements; and revising the annunciator procedures for the EDG air start system receiver low pressure alarm to require that the EDGs be declared inoperable when supply air pressure drops below 200 psig. There are 335 SOVs, distributed among about 17 nuclear plant owners, listed in the Part 21 notice. Point Beach LER 26698008 also refers to Engine Systems Part 21 Notification #1998120 dated January 26, 1998. This is considered to be a common-cause failure condition.	b, c	EMDs
Comanche Peak 1	44595005	950831	Engineering personnel identified nonconservatism in the calculation that determined (1) leakage rates for accumulator check valves associated with various air operated valves and the nitrogen accumulators for the pressurizer Power Operated Relief Valves (PORVs), and (2) the pressure switch alarm set points for these accumulators. Engineering personnel performed evaluations which revealed that, with the exception of the PORVs, the valves associated with these accumulators were operable. The PORV accumulator low pressure alarm set points would still ensure operability during Modes 1, 2 and 3. However, for Modes 4, 5 and 6, the set points would not ensure operability for all conditions. TU Electric believed that the cause of the event was nonconservative design in the original calculation used to determine the low pressure alarm set point for the PORV accumulators and the accumulator check valve leakage rates. The PORV nitrogen accumulator low pressure alarm set points were to be raised to 90 psig. A review was to be performed for all safety-related accumulators used for valve actuation to verify that the appropriate set points are being used. This is considered to be a common-cause failure condition.	c	Pressurizer Power Operated Relief Valves

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Comanche Peak 1	44598001	980110	During performance of a partial stroke test of feedwater isolation valves (FWIVs), the Plant Equipment Operators (utility, non-licensed) discovered that Feedwater Isolation Valves 1-HV-2134 and 2-HV-2137 had low nitrogen pressure. The Shift Manager declared that both FWIVs 1-HV-2134 and 1-HV-2137 were inoperable. Maintenance personnel proceeded to charge the accumulators with nitrogen. The low nitrogen pressure for FWIV 1-HV-2134 was caused by a leak in a nitrogen solenoid valve and on the nitrogen cylinder head. The low nitrogen pressure for FWIV 1-HV-2137 was caused by a leak in a nitrogen solenoid valve. Immediate corrective action was to restore the nitrogen pressure to declare the FWIVs operable. The leak rate on 1-HV-2134 was reduced and work planned to replace elastomer parts for this valve. The solenoid valve on 1-HV-2137 has been replaced to prevent recurrence. The elastomer seal on FWIV 1-HV-2134 was degraded and caused the nitrogen to leak. The LER does not describe the root cause for the SOV failure or the mechanism for the elastomer seal degradation.	c	Feedwater
Cooper	29894013	940712	A Reactor Scram and Group 2 (Shutdown Cooling), 3 (Reactor Water Cleanup), and 6 (Reactor Building Ventilation) Isolations occurred and Standby Gas Treatment started as a result of a spurious decrease in the indicated Reactor vessel water level on the "B" channel level instruments. The cause of the spurious actuation was leakage through a solenoid valve due to wear. This solenoid valve provides injection of Core Spray to backfill the "B" channel reference leg of the Reactor vessel water level indication system. The wear was due to pressure transients from valve surveillance testing in the Core Spray system, which had not been anticipated. The SOV (NBI-SOV-SSV739) is rarely operated, and the wear found would not be expected from normal operation. Root cause investigation indicated that the condition of the valve was due to pressure transients from the surveillance testing of CS-MOV-MO26B with the Reactor depressurized. The solenoid valve is designed to use process system pressure to assist in obtaining a tight shutoff, a condition which is not attainable with the Reactor depressurized. The selection of this valve design for the system did not appropriately anticipate the spectrum of operational conditions to which it could be exposed when the equipment was installed in 1989.	c	RCS, Core Spray

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Crystal River 3	30297015	970612	During a review of the differential pressure calculation for the Letdown Line [CB] Inboard Containment Isolation Valves, FPC discovered that the evaluation of the maximum differential pressure (d/P) that these valves could be subject to was in error. These valves are rated to close against a maximum d/P of 1800 psi, but could be subjected to a d/P in excess of 2000 psi in the event of a letdown line rupture downstream of outboard containment isolation valve (MUV-49), concurrent with a failure of MUV-49 to close and operator action in accordance with Emergency Operating Procedure (EOP) 3. Outboard containment isolation valve MUV-49 would not be capable of closing if subjected to 2000 psi d/P. Isolation at Penetration 333 requires either MUV-49 or MUV-40/41/ 505 to close. The cause of this event was the use of inappropriate assumptions in the calculation for the determination of maximum valve d/P. FPC planned to install a new inboard containment isolation valve prior to restart of CR-3. In addition, the air operator on MUV-49 was to be modified to allow closure against the predicted d/P. The closure of these valves is required to mitigate the effects of a Makeup System Letdown Line Failure Accident and in response to a Reactor Building Isolation signal. Consequently, the valves were outside of their design basis and their failure to close in the described scenario during previous operating periods, could have created an non-isolatable loss of reactor coolant to the Auxiliary Building. This is considered to be a common-cause failure condition.	c	Containment
Davis Besse	34687015	871207	A low pressure instrument air pressure alarm was received in the control room. The low instrument air pressure caused several valves in the steam and feedwater systems to open and dump steam and condensate to the condenser. The Integrated Control System responded by increasing feedwater flow and pulling control rods out. Reactor power increased and reached the high flux trip on the Reactor Protection System. The loss of instrument air pressure was caused by direct venting of the Instrument Air header to atmosphere when a solenoid valve failed on Instrument Air Dryers 1-1 and 1-2.The air dryers were isolated and bypassed in order to re-establish normal instrument air header pressure. No information was provided in the LER concerning how long the dryers were out of service or what the short or long-term effects on air quality might have been. According to the LER, the dryers were subsequently replaced. This is considered to be a common-cause failure condition.	b, c	Main Steam Main Feedwater Instrument Air

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Davis Besse	34688007	880304	Several AOVs were declared inoperable due to the potential for moving from their fail-safe position upon Loss of Instrument Air following a Safety Features Actuation System Initiation. Service Water Valve SW 1434 and four other valves of similar design require air pressure to hold them in their "fail-safe" position. These valves are SW 1424, SW 1429, CC 1467, and CC 1469 which are the Temperature Control Valves for two of the Component Cooling Water Heat Exchangers and Decay Heat Removal Heat Exchangers respectively. Each valve operator is similar and is provided with two safety-grade accumulators. One accumulator is piped to the bottom of the valve actuator piston and provides the motive force to open the valve when air is vented from the top of the actuator piston. The second accumulator is piped to the top side of the actuator piston through two solenoid valves and provides the motive force to close the valve. The cause of the valve drifting from its open position was air leakage from the valve's accumulator system. The cause of the accumulator leakage was not described in the LER.A mechanical locking device was to have been added which, on the loss of air, will automatically lock its associated valve in the fail-safe position subsequent to the valve's receiving an SFAS initiation signal. This is considered to be a common-cause failure condition. This was an ASP Program Event. See Appendix B.	c	Decay Heat Removal Instrument Air
D. C. Cook	31597026	970925	Due to a lack of overpressure protection on the 85, 50, or 20 psig control air headers, if a non-safety-related air regulator failed open it would result in an overpressurization of a control air header. This would result in the potential for common mode failure of both trains of safety related equipment. The lack of overpressure protection on the control air headers due to a regulator failing open had not been identified as a mechanism that could overpressurize the low pressure headers. As a result, single failure of a non-safety related component affecting both trains of safety related equipment was not identified. There would have been no significant effects for the 85 and 50 psig headers, however, overpressurization of the 20 psig header could have resulted in the degradation of the RHR system and the partial opening of the Unit 2 Steam Generator (SG) Power Operated Relief Valves (PORVs) for the duration of the overpressure event. Due to a single failure being identified that could have potentially resulted in the degradation of both trains of RHR this event could have been significant. (continued next page)	c	RHR HX Outlet Valves Steam Generator PORVs

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
D. C. Cook	31597026 (continued)	970925	The cause of the lack of overpressure protection on the Control Air System was the fact that a regulator (non-safety-related) failing open was not identified as a mechanism that could overpressurize the low pressure headers. As a result, single failure of a non-safety related component affecting both trains of a safety related system was not identified.		RHR HX Outlet Valves Steam Generator PORVs
Dresden 2	23787023	870717	Four component failures involving Copes-Vulcan P-200-12 and B&W-B-209-12 feedwater regulating valves, as well as booster valve and positioner failures led to a scram. Subsequent investigation revealed a number of problems, including a broken handwheel stop, air restrictions in an SOV due to wear, a positioner out of adjustment, and improper booster relay valve diaphragm seating caused by either age or wear. The Component Failure Data section of the LER indicated that an industry-wide search revealed 81 failures in one year of Copes-Vulcan Model P-200 AOVs due to positioner operating abnormalities. The B&W operators had 6 failures, 5 of which were SOV related. This is considered to be a common-cause failure condition.	c	Main Feedwater
Dresden 2	23788012	880517	All eight Main Steam Isolation Valves (MSIVs) were declared inoperable due to their failure to fully close on a loss of pneumatic supply. The root cause of the MSIVs failure to fully close has been attributed to high drag forces exerted on the valve stem by the valve packing. This is considered to be a common-cause failure condition.	c	Main Steam Isolation Valves

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Dresden 2	23798003 (Event No. 33620, dated 1/28/98 followed up by Morning Report H-98-0045, dated 3/6/98 referred to this condition.)	980128	Failures in diaphragms of D-100 Copes-Vulcan AOVs were caused by wear attributed to inadequate rubber covering over the fiber in the diaphragms. The root cause of this event was manufacturer's design deficiency of the valve operator's diaphragm (NRC Cause Code B). The diaphragm which failed has the embedded reinforcing fiber layer too close to the surface on the lower side of the diaphragm. This significantly increases the susceptibility of the diaphragm to a mechanical abrasion failure because of rubbing between the diaphragm and the actuator plate. Essentially, the diaphragm wears out prematurely. This failure mechanism was also documented by the ComEd System Materials Analysis Department (SMAD) for three other failed diaphragms of the same make and model from the Quad Cites Nuclear power plant during detailed laboratory examinations. In 1996, the valve manufacturer (Copes-Vulcan) upgraded the diaphragm design by increasing the amount of rubber on the side of the diaphragm in contact with the diaphragm plate. This same valve (AO3-2301-64, HPCI Turbine Stop Valve above seat drain valve) failed at Dresden 2 in March 1995 for what appears to be the same reason (LER 24995004). LER 24997003 described another failure of AOV AO3-2301-64 due to misadjustment. There are over 1900 Copes-Vulcan D100 valves in service in U.S. nuclear power plants. This is considered to be a common-cause failure condition.	c	HPCI
Dresden 3	24993004	930116	At 83% power, an Instrument Air Header Pressure low alarm was received in the Control Room. Instrument Air Header pressure was observed to be decreasing rapidly. Loss of Instrument Air was attributed to mechanical failure of the 3A Instrument Air Compressor dryer inlet valve to close during the dryer purge cycle with concurrent failure of the backup Service Air to Instrument Air Cross-tie valve to promptly open on a low instrument air pressure signal. Based on bench testing it was determined that inlet valve binding in the open position was the root cause for dryer blowdown. Binding was attributed to excessive friction in the valve ball. The root cause for the cross-tie valve slow response time was attributed to malfunction of its air pressure regulator, due to binding of internal parts. It was also noted in the LER that a solenoid valve needed to be relocated. Prior to returning the 3A Instrument Air Compressor to service the following corrective actions were taken: – replacement of the 3A dryer inlet and exhaust valves, – replacement of the inlet and exhaust solenoid valves, and – replacement of the cross-tie valve air pressure regulator and solenoid relief valve.	b, c	Scram Header Instrument Air

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Dresden 3	24993005	930126	An air leak occurred on the fail-safe accumulator system for Drywell Vent Valve 3-1601-63. Air was identified to be leaking out of a weep hole on the two way mechanical Versa Valve. The apparent cause for the leakage out of the weep hole in the two way Versa Valve was due to degradation of the O-ring seals caused by aging and fatigue of the O-ring material.	c	Drywell Environmental System
Duane Arnold	33191005	910622	With the reactor at approximately 100% power, a single outboard Main Steam Isolation Valve (MSIV) closed, resulting in a high flux automatic reactor scram. The cause of the MSIV closure was determined to be a non-safety related pipe joint failure. The two-inch nitrogen supply pipe that supplies the outboard MSIVs' control accumulators separated sufficiently at a soldered coupling to reduce supply pressure. Although check valves are installed to maintain control pack accumulator pressure, two smaller fitting leaks on the control pack for the 'B' outboard MSIV slowly reduced the pressure to the nitrogen-operated MSIV position control valve. This caused the control valve to slowly change position, porting actuating nitrogen to close the MSIV.	c	Main Steam Isolation Valves
Fermi 2	34194004	940822	Four air-operated, spring closing isolation valves may not have been capable of performing their Reactor Coolant Pressure Boundary (RCPB) function of isolating against Reactor Coolant System (RCS) pressure at their as-found actuator spring preload settings. The cause of this event was inadequate control of the valve actuator settings which resulted in insufficient preload being maintained on the actuator closing springs. The spring preload settings were to have been increased to a value sufficient to ensure adequate valve closure prior to plant startup. Two industry events during 1993 involving the failure of air-operated, spring closing valves to adequately isolate against full RCS pressure prompted a review of similar valves at Fermi 2. Four, 3/4-inch Rockwell-Edwards globe valves with Fisher air actuators were Identified with a similar problem. These are normally open, spring to close valves which provide automatic containment isolation for reactor recirculation pump seal purge flow from the Control Rod Drive Hydraulic (CRDH) system. Isolation occurs on either a High Drywell Pressure or a Reactor Vessel Low Level 2 signal. The valves are oriented with normal system flow from the CRDH system over the seat, to the reactor recirculation pump seals which are at normal reactor pressure. (continued next page)	c	Control Rod Drive Hydraulic System Reactor Coolant System

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Fermi 2	34194004 (continued)	940822	In the event of an accident with a concurrent loss of CRDH system pressure, the valves could be subjected to a back pressure (under seat) differential of up to about 1200 psid, which may either prevent the valves from fully closing or allow them to lift off of their seats after they have closed. The original specification for these valves was to ensure closure in the CRDH flow direction at a pressure of 1750 psig. This resulted in a setting by the vendor sufficiently high to meet both the containment isolation as well as the RCPB isolation requirements for these valves. However, the nameplate supplied by the vendor incorrectly identified a lower setting. The valves were reset during installation to this lower setting using the nameplate data. The resulting setting was too low to ensure adequate closure of the valves for meeting their RCPB isolation function. This condition had existed since initial operation of the plant. This is considered to be a common-cause failure condition.	c	Control Rod Drive Hydraulic System Reactor Coolant System
Fort Calhoun	28587018	870504	The valve operator system for HCV-385 and HCV-386 had design deficiencies which could have prevented them from performing their safety function. These are the isolation valves for the containment spray (CS) and High and Low Pressure Safety Injection (HPSI/LPSI) pump minimum recirculation line. They were designed such that a single failure in conjunction with a Loss of Coolant Accident (LOCA) and loss of the non-safety grade instrument air system would have prevented these valves from going shut. This failure would have allowed primary coolant to be pumped to the Safety Injection and Refueling Water Tank (SIRWT) which vents to the Auxiliary Building; thus, violating containment integrity. HCV-385 and HCV-386 rely on a safety grade accumulator to provide the motive force to close the valves. Two design deficiencies were identified. First, both valves shared a single accumulator. Thus, a single failure could disable the ability to close the valves. Second, the valves' required time to close is 45 seconds to ensure that the coolant from the containment sump does not reach the valves. If highly contaminated coolant reaches the two valves they could be inaccessible for up to 1000 hours, and the air accumulators would have been required to hold the valves shut for this full time period. If the valves close in less than 45 seconds, they must hold only long enough for the operators to locally isolate them by handwheel. Test data indicated that closure time had been at or near 45 seconds. This valve operator/air accumulator system deficiency was an original design deficiency present since the plant started operation in 1973. This is a common-cause failure condition.	c	Containment Spray High Pressure Safety Injection Low Pressure Safety Injection

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Fort Calhoun	28587033	870706	Clarified water entered the Instrument Air System during a surveillance test of the diesel generator room dry pipe sprinkler system. The interfacing check valves between the Instrument Air and Fire Protection systems were prevented from completely closing by foreign material. Additionally, inadequate procedures and inadequate operator training on this unique dry pipe valve contributed to this event. This event occurred because: 1. Instrument air check valves IA-575 and IA-576 were prevented from closing by foreign material. 2. The operator performing the test failed to properly reset the dry pipe valve as a result of inadequate procedures and inadequate training on the unique dry pipe valve. The air maintenance device was bypassed thus removing another check valve and orifice that could have prevented and/or restricted flow of water into the air system. During the reset process, as performed, a flow path existed to the Fire Protection System. The Fire Protection System pressure is approximately 30 psi greater than Instrument Air pressure. Thus, water flowed into the Instrument Air System. This event was the cause of failure of the DG-2 exhaust damper and shutdown of DG-2 on September 23, 1987, which is addressed in LER 28587025 in Table 3. This is considered to be a common-cause failure event.	a, c	Instrument Air Fire Protection System
Fort Calhoun	28588002	880125	The High Pressure Safety Injection alternate discharge header isolation valve, HCV-2987, was declared inoperable due to failure to meet the design criteria for operability. The purpose of HCV-2987 is to ensure long term core cooling through hot leg injection. It is a normally open valve that fails as is on a loss of instrument air. HCV-2987 uses an air intensifier to convert instrument air received at about 40 psig to supply the valve receiver reservoir air pressure at 320 psig. A check valve prevents depressurization of this reservoir on a loss of instrument air. The valve then can be cycled closed one time without the use of instrument air. This is the valve's required position for simultaneous hot and cold leg injection for long term core cooling if the "B" HPSI pump is operating. The equivalent of closing HCV-2987 could be obtained by (continued on next page)	c	High Pressure Safety Injection

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Fort Calhoun	28588002 (continued)	880125	closing the four HPSI isolation valves located downstream. HCV-2987 must close to meet the single failure criteria for hot leg injection. It was determined that the air intensifier had failed (for reasons not included in the LER). A temporary mechanical jumper was installed to bypass the valve's air intensifier. A nitrogen gas bottle supplying 320 psig pressure to the valve's receiver reservoir was used to supply sufficient pressure for valve operations.	c	High Pressure Safety Injection
Fort Calhoun	28588004	880311	During a LOCA, a potential leakage path through isolation valve, PCV-1849, for the Instrument Air (IA) System Containment Penetration M could exist if the IA system pressure is not maintained above the containment pressure. PCV-1849 is designed to close on receipt of a low IA system pressure in conjunction with a Containment Isolation Actuation Signal (CIAS). However, PCV-1849 is an air operated valve that has an actuator which would allow the valve to open on loss of air. During a Loss of Coolant Accident (LOCA) with a concurrent Loss of Instrument Air System pressure, PCV-1849 may not have been capable of being closed, thus containment integrity could not be assured. This resulted in the determination that the actual design was outside the USAR. This is not consistent with the USAR Appendix G Criterion 53 and therefore, containment integrity could not be ensured during a LOCA with concurrent Loss of Offsite Power. Although the LER does not say it, this appears to be an original design defect.	c	Instrument Air
Fort Calhoun	28588009	880406	During a self-conducted Safety System Functional Inspection (SSFI) of valve operators that have installed accumulators, the capability of certain valves to perform their design function during a design basis accident with a concurrent loss of instrument air was questioned. The valves of concern are LCV-383-1&2, Safety Injection and Refueling Water Tank (SIRWT) isolation valves; HCV-438B & D, Component Cooling Water (CCW) to Reactor Coolant Pump (RCP) seal cooler isolation valves; HCV-238 & 239, charging pump header to Reactor Coolant System (RCS) isolation valves; and HCV-240 auxiliary pressurizer spray isolation valves. In general, the concern was that the accumulators for these AOVs, as originally designed and configured, did not have sufficient capacity to ensure that air pressure would be available for the time needed to complete the design basis functions. This is considered to be a common-cause failure condition.	c	High Pressure Safety Injection Component Cooling Water

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Fort Calhoun	28588028	881020	A check valve on the air accumulator providing back up air supply to steam isolation valve YCV-1045A was discovered inoperable during function testing. Immediate corrective action was required to repair the check valve. Upon disassembly, it was discovered that a metal particle, which had lodged in the valve seat, rendered the accumulator inoperable.	a	Main Steam Isolation Valves
Fort Calhoun	28590025	900929	Conditions were identified involving the Component Cooling Water (CCW), Raw Water (RW), and Containment Spray (CS) system which placed the unit outside its design basis for post-accident containment cooling. The CCW conditions involved the potential for degradation of containment air cooler performance and/or loss of CCW system operability following loss of Instrument Air. The CCW/RW interface valves would fail open on a loss of instrument air (IA) assumed as part of the post-LOCA accident scenario. The IA system is not a safety-related system; therefore its availability cannot be credited in a post-accident situation. Although these valves are equipped with backup air accumulators, the accumulators are not qualified as safety-grade items and also cannot be credited to operate. In addition, valves to many essential and nonessential heat exchangers served by CCW fail open upon a loss of instrument air. The resultant system flow distribution would deprive the containment air cooling coils of their design CCW flow by allowing CCW flow to nonessential equipment. This would result in heat removal performance by the containment air coolers below that assumed in the design basis. The primary cause is attributed to deficiencies in the original system design. This is considered to be a common-cause failure condition. This was an ASP Program Event. See Appendix B.	c	Component Cooling WaterRaw WaterContain-ment Spray System
Haddam Neck	21393005	930518	A containment isolation valve for the reactor coolant letdown line (LD-TV-230) could not be closed from the control room using the control switch. LD-TV-230 is a normally open air operated valve located inside the containment. The problem was traced to a failure of the air pilot solenoid valve. The cause was corrosion inside the valve as the result of water intrusion from the containment control air system due to a malfunction of the containment control air dryer. Corrective actions included replacement of the failed solenoid valve, a blowdown of the air system to remove any remaining water, and repair of the dryer. It was unknown how long this valve was inoperable prior to discovery, but is believed to have been longer than the action time allowed by the plant Technical Specifications. The failed SOV was an ASCO Model (continued on next page)	a, b, c	Instrument Air Reactor Coolant Letdown

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Haddam Neck	21393005 (continued)	930518	NP 8320A182V. The cause of the water in the air system was a malfunction of the containment control air dryer. The dryer is a Model 25HA4 manufactured by Pall Pneumatic Products Corp. The malfunction involved a failure of the desiccant towers to switch from drying to regeneration mode properly. It was not known how long this condition may have existed. This is considered to be a common-cause failure condition. See LER 21393007.	a, b, c	Instrument Air Reactor Coolant Letdown
Haddam Neck	21393007	930525	While performing a pressure decay test of the pressurizer Pilot-Operated Relief Valves (PORVs) emergency air supply system, it was determined that the pressure decay exceeded the Technical Specification acceptance criterion of 0.3 psi/hr. The problem was traced to a leak in the diaphragm assembly of one of the PORVs (PR-AOV-568). This leak was caused by both the inadequate sealing of the PORV diaphragm assembly and the failure of the PORV air supply pressure regulating valve (CA-PRV-836A).The failure of the air pressure regulating valve was caused by corrosion inside the valve due to water intrusion from the containment control air system. The moisture in the containment control air system was due to a malfunction of its air dryer. The diaphragm of the failed brass regulating valve was covered with a powdery blue-green corrosion product when it was inspected. This corrosion caused a leak in the diaphragm allowing air to vent through the regulator eventually causing the pressure to equalize on both sides. This is considered to be a common-cause failure condition. See LER 21393005.This was an ASP Program Event. See Appendix B.	a, c	Instrument Air Pressurizer Pilot-Operated Relief Valves
Haddam Neck	21394005	940219	With the plant in Mode 5 (Cold Shutdown) while performing a stroke test of the pressurizer Pilot-Operated Relief Valves (PORVs), it was determined that the valves would not fully open. The problem was traced to a leak in the diaphragm assembly of the PORVs (PR-AOV-568 & 570). The leaking diaphragms were caused by loose diaphragm cover bolts. Both PORV diaphragms were replaced during the 1993 refueling outage with a new style. The principle change was the substitution of a longer lasting material (EPDM) for the old Buna-N material. The manufacturer also changed the shape of the diaphragm somewhat although this was never communicated to the licensee. This change resulted in some difficulty installing the diaphragm. To overcome this, a commonly used lubricant (Moly 55) was applied to aid installation. The PORVs were subsequently retested satisfactorily. After the February 1994 failures, an in-depth discussion with the manufacturer on the possible causes for failure revealed several aids to overcome installation problems. (continued on next page)	c	Pressurizer Pilot-Operated Relief Valves

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Haddam Neck	21394005 (continued)	940219	The most significant was the use of a sealant around the diaphragm's bolt circle. It is believed that the presence of lubricant instead of the sealant allowed some extrusion of the diaphragm from between the base and cover and away from the bolt holes. This extrusion also led to small tears at several diaphragm bolt holes, allowing the bolts to loosen over time. Also, air regulators supplying both PORVs were set too low (77.3 PSIG and 75.1PSIG versus the required 85 PSIG). Program Event. See Appendix B. This event was reported as a common-mode failure. This was an ASP event.	c	Pressurizer Pilot-Operated Relief Valves
Haddam Neck	21396012	960611	An engineering analysis determined that the main feedwater regulating valves (FRV) would not fully isolate feedwater flow as required for a main steam line break accident in containment. It was determined that the differential pressure across the valves would overcome the valve spring's closing forces. In a design basis steam line break analysis, the feedwater motor operated valves (MOV) are required to isolate; however, in the event of a single failure of the MOV the FRV is credited with isolation. The failure to isolate feedwater for a steam line break in containment could result in exceeding maximum containment design conditions. The cause of this condition was an erroneous assumption in the accident analysis that the FRVs would isolate against a high differential pressure. The differential pressure across the valve would overcome the valve spring's closing force. This is considered to be a common-cause failure condition. See LER 21396018. (Event No. 30619 dated 6/11/96 applies.)	c	Main Feedwater

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Haddam Neck	21396018	960822	An engineering analysis revealed that the feedwater regulating bypass valves would not fully isolate feedwater flow as required for a main steam line break inside containment. This condition was discovered during a follow-up to a similar problem with the main feedwater regulating valves (LER 21396012). The failure to isolate feedwater for a steam line break inside containment could result in exceeding maximum containment design conditions. This event did not involve any actual equipment failures. The cause of this condition was an erroneous assumption that the feedwater bypass valves would close and isolate against the differential pressure experienced between the steam generator feed pump and a faulted steam generator. Additionally, credit was taken for isolation of the bypass valves from the control room ten minutes after an accident, however the control air system which operates the valves is not a credited system. The bypass valves (FW-HICV-1301-1,2,3,4) are 1-1/2 inch, air to close, spring to open, manufactured by Masoneilan. They are normally closed during full power operation. On an auxiliary feedwater actuation signal they go full open until manual operator action is taken at the main control board to throttle flow. This is considered to be a common-cause failure condition. See LER 21396012.	c	Main Feedwater
Hatch 1	32192003	920102	Valves 1P41-F039A and B, air operated cooling water supply valves to Emergency Equipment Room coolers 1T41-B002A and B, failed to open automatically as required during routine operability testing. These valves are designed to open automatically to provide cooling water to the room coolers to maintain the temperature below 148°F when the Core Spray and/or Residual Heat Removal pumps are in operation. With both the normal and standby coolers for this room inoperable, Core Spray pump 1E21-C001A and RHR pumps 1E11-C002A and C were declared inoperable. An LCO was invoked and a temporary modification was implemented to place valves 1P41-F039A and B in the open position to ensure a supply of cooling water to the Emergency Equipment Room coolers. An unanticipated breakdown of a material used in the manufacturing process may have been a root cause. The solenoid operated valves (SOVs) in the air supply lines to valves 1P41-F039A and B failed to reposition as required. Consequently, valves 1P41-F039A and B could not open. It appeared that the SOVs failed because their solenoid cores stuck to the top of the core housings, perhaps as a result of the gelling of a lubricant (Dow Corning 550) used in the assembly process. Corrective actions include replacing the SOVs, increasing the cycling of SOVs of this type, and changing these valves to another type of SOV. This is considered to be a common-cause failure condition.	b, c	RHR Core Spray

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Hope Creek	35486063	860828	Following an extensive investigation the determination was made that twelve air operated valves had solenoid valves installed which had an operating air supply pressure rating less than the maximum expected air supply header pressure. Ten of the valves in question had non-Q pressure regulators installed to limit the pressure to within the design value. Exceeding the pressure rating may result in failure of the solenoid valve to open. The root cause of this event was design inadequacy in that solenoid valves of incorrect pressure rating were specified for use. The subject valves were installed in the Containment Atmosphere Control (CAC) System and the Safety Auxiliaries Cooling System (SACS). This is considered to be a common-cause failure condition.	b, c	10 Different Systems Instrument Air
Hope Creek	35494017	941110	Two room cooler isolation valves failed to open during a quarterly test of the Safety Auxiliary Cooling System. The valves are associated with the room coolers for two emergency diesel generators. These are Anchor-Darling flex-wedge gate AOVs with Hiller actuators. The most probable cause of failure was identified as binding associated with the stem packing and a nonconservative valve friction coefficient used in calculating the actuator size. (Lessons learned from the testing of motor-operated valves in accordance with the guidance in Generic Letter 89-10 led to this conclusion.) The licensee determined that this was a common-mode failure attributable to a design deficiency. A similar condition (referenced in the LER) was reported in Hope Creek LER 35493006 in October 1993 for these AOVs. In that event, excessive air pressure caused the gates to bind in the seat. This condition resulted when the packing style was changed which reduced the packing drag on the valve stem. The design change package did not account for the reduced packing drag and did not lower the air supply pressure to the actuator. The additional seating force and gate travel reduced the ability of the spring in the actuator to drive the valve open.	c	Diesel Generator Room Cooling
Hope Creek	35497020	970828	Design deficiency (failure of instrument air) could have resulted in the loss of Emergency Diesel Generators and Emergency Core Cooling System. See Hope Creek Event #32836 in Table 4.	c	Safety Auxiliary Cooling System (SACS)

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Indian Point 2	24793010	930818	It was determined by an engineering analysis that regulating valves installed in the nitrogen backup to the Instrument Air System for the Main Steam Power Operated Relief Valves and the Auxiliary Feedwater System (AFWS), were not fully capable of performing their function in the manner set forth in the Final Safety Analysis Report (FSAR). Specifically, the nitrogen backup regulator is described in the FSAR as capable of automatically supplying nitrogen in the event of loss of instrument air. The nitrogen regulating valves were designed and installed with an extraneous equalizing line between the upper spring chamber vent and the downstream air line. This equalizing line compromised the ability of the nitrogen regulating valves to automatically modulate (or regulate) the nitrogen pressure and flow. The nitrogen regulating valves would therefore only pass nitrogen to the extent that it leaked past the valve seat or the valve was preset to a given opening. The nitrogen regulating valves could still have been operated manually. The valves are CASHCO, model HP 1-32-45-S36, 0.5 inch size. Thus, the cause of the failure of the nitrogen backup regulator was an incorrect design application of the regulator by the licensee that was not caught by testing. This is considered to be a common-cause failure condition.	c	Main Steam Power Operated Relief Valves Auxiliary Feedwater System
Indian Point 3	28688009	881025	Automatic Switch Company (ASCO) indicating a potential problem existed with ASCO NP8314 series solenoid valves (SOVs) failing to shift to a de-energized position following extended periods of being energized. ASCO identified the failure mechanism as the solidification of a lubricant used in the initial manufacturing processes that had migrated into critical surfaces of the valve's subassembly and which subsequently caused the SOVs to fail to shift when called upon. The plant identified six normally energized NP8314 series SOVs in use. Two of the normally energized NP8314 series SOVs were replaced with improved series NP8314 SOVs from ASCO. The two that were removed were disassembled and tested with no lubricant solidification problems noted. Refer to NUREG-1275, Volume 6. ASCO employees used the lubricant, without authorization, to make assembly easier. This is considered to be a common-cause failure condition.	b, c	Reactor Coolant System Liquid Waste

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Indian Point 3	28693035	930916	An existing design flaw could allow an automatic Safety Injection actuation signal combined with a loss of offsite power to result in the loss of the Weld Channel and Containment Penetration Pressurization System (WCCPPS). The signal would cause the system's pressure control valves to fail in a closed position which cuts off the air supply. The design flaw is outside the design basis of the plant and had existed since initial plant startup (August 1976). The IA system supplies compressed air at approximately 100 psig through a check valve to the WCCPPS header which supplies the four WCCPPS zones. Each zone has an air receiver upstream of a pressure control valve (PCV) that is sized to maintain the air supply for at least four hours in the event the IA and SA are lost. The PCVs are designed as air-to-open valves, meaning that air pressure must be supplied to the dome of the PCV in order for the valve to open and maintain the downstream air pressure at about 46 psig. The PCVs rely upon the pressure in the IA system to remain open since there are check valves between the WCCPPS air receivers and the connections to IA. An automatic SI System actuation signal with a loss of offsite could have caused a loss of IA because the safety related electrical busses are stripped and the IA and SA compressors (CMP) are not reloaded. Following a loss of the compressors, the IA system would retain sufficient pressure to operate the PCVs for about 1/2 hour. When the air pressure is insufficient to hold open the PCVs, they fail closed and all air flow to the WCCPPS stops. Manual action to regulate WCCPPS air pressure using the PCV bypass lines is feasible (each zone has a pressure relief valve set at 49 psig to prevent overpressurization), but is not considered in the system's design basis. The backup supplies would not prevent this event. The primary source of backup is the nitrogen supply which has sufficient capacity for 24 hours. When the pressure downstream of any PCV drops to 43 psig, the zone's pressure controller (PC) actuates a nitrogen supply valve to deliver nitrogen to a connection just upstream of the PCVs. This gas supply will not operate the PCVs because of intervening check valves. The secondary source of backup air is the plant's SA system which is cross-connected to the IA system. SA would also be lost when an automatic SI actuation signal occurs because the compressors are stripped from the safety related busses and not reloaded. The last backup is a permanent cross connection to the Con Ed IP2 SA system. Operation of this backup requires manual action not considered in the system's design basis. This is considered to be a common cause failure condition.	c	WCCPPPS, IA and SA

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Indian Point 3	28693050	931115	Operations issued a deviation event report to identify the potential inoperability of four Service Water System solenoid operated valves affecting emergency diesel generator operation. The condition resulted from solenoid operated valves rated at less than the instrument air pressure supply and a failure mode that could cause unregulated instrument air. Subsequently, 109 additional valves subject to overpressure and one failure of a valve to vent were identified. The condition resulted from an original design deficiency and a subsequent failure to correct the error in response to regulatory feedback. The system was designed so that the air operated valves fail safe when the instrument air supply is lost. The failure mode of the SOVs due to overpressure from air regulator failure or high setpoint and the effect on the AOVs was not recognized. The licensee did not evaluate GL 91-15 when received. They did not request an adequate review of GL 91-15 because the GL did not require a response to the NRC. There were no procedural requirements in place to request such a review. This is considered to be a common-cause failure condition.	b, c	Service Water System
Indian Point 3	28693053	931202	With the plant in a cold shutdown condition, flow control valves SWN-FCV-1176 and SWN-FCV-1176A were found to be inoperable during a post-work test. Operations declared all three emergency diesel generators inoperable because at least one valve must be operable to allow service water to the generators. The solenoids on AOVs used to control the EDG service water effluent for all three EDGs (two parallel valves in a common header line) were replaced and then tested as part of a normal work package on a safety-related system. The solenoids were incorrectly connected and the valves would, therefore, not stroke open. A series of miscommunications and mistakes (some caused by inadequate procedures) led to the condition where all three emergency diesel generators were inoperable for about 4.5 hours. Although the LER focuses on the mistakes and procedural difficulties, this event illustrates the (unintended) interdependence of critical systems on air-operated valves which are common to parallel trains of safety-related equipment. This is considered to be a common-cause failure condition. See Indian Point 3, Event No. 26449 in Table 4.	b, c	Emergency Diesel Generators

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Indian Point 3	28696004	960215	Two air operated vapor containment isolation diaphragm valves in series were found to be inoperable which violated Technical Specification 3.6.A.1. The original isolation valve design would close at system pressure when there was a differential pressure or accident design pressure, but not against system pressure with no pressure differential. The design and specification of the valves during initial design and construction was not adequate to ensure the valves met their design requirement for containment isolation. RC-AOV-519 and RC-AOV-552 (the valves are designed to positively seal with a differential pressure) would close against a differential pressure of 150 psi (the Primary Water System design pressure) but would not have closed with a line pressure greater than about 120 psig when there is no differential pressure. The original specification for these valves was for a maximum differential pressure of 200 psi, with no reference to a minimum pressure differential or constant line pressure requirement. The valve design had not been modified since the original plant design and construction. The condition could have existed when closing isolation valve RC-AOV-560, downstream and in series with RC-AOV-519 and 552, in a sequence that maintained a line pressure greater than 12 psi. There would then be no differential pressure during stroke testing. Stroke testing in these conditions created another problem if the limit switches are adjusted after the valves are shut but do not fully close. The valves were assumed to be fully closed during the adjustment so subsequent failures to close would be masked and could allow a control room indication that the valves were fully closed when they were not. There was no direct external indication on the valves to show if they were fully closed. This is considered to be a common-cause failure condition.	c	Containment Isolation Diaphragm Valves
Indian Point 3	28696008	960320	The System Engineer for the Isolation Valve Seal Water System (IVSWS) determined that the minimum IVSWS nitrogen tank pressure required to support proper system operation was in question. Deviation Event Report 96-824 was written to document this determination. A subsequent evaluation determined that the nitrogen supply (three bottles on a header) had been less than that required for the IVSWS to operate as designed. This placed the plant in a condition prohibited by Technical Specifications. The cause of the event was the lack of original design basis documentation to identify the requirements for the nitrogen tanks.	c	Isolation Valve Seal Water System

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
LaSalle 1	37396011	960928	While developing an Air operated Valves (AOV) preventative maintenance program, inconsistent testing data were obtained for valves with WKM 70-13-1 pneumatic actuators. The inconsistent results appeared to be related to incorrect effective diaphragm areas (EDA) for the AOV actuators. 36 WKM AOVs were addressed in the LER (18 per unit). 13 AOVs in each unit are part of the primary containment isolation system (PCIS) and 5 are in the reactor core isolation cooling system (RCIC). Two problems associated with the EDA of the actuators of the WKM valves were identified. The first was related to the actual versus the manufacturer's published EDA of the actuator. If the actual EDA is less than what the manufacturer publishes, then the closing (or opening) forces installed in the valve (via spring/spring adjustment) will be less than required. The second problem was stretching of the diaphragm during valve travel resulting in a reduced EDA. This is considered to be a common-cause failure condition.	c	Containment Isolation System Reactor Core Isolation Cooling
LaSalle 2	37495005	950218	The hand switch for the "D" Outboard Main Steam Isolation Valve (MSIV), 2B21-F028D, was placed in the CLOSED position and the valve failed to close. The hand switch was cycled several times from AUTO to CLOSED to AUTO with the same results. The valve was eventually closed by placing the hand switch in the OPEN SLOW TEST position and depressing the Slow Test pushbutton. This slow-closed the valve, and the pilot air was isolated. The cause of the MSIVs failure to close was sticking of the solenoid pilot valve. This sticking prevented the pilot valve from changing state, and did not allow the pilot air to vent. The failure of the pilot air to vent resulted in air ported to the under side of the MSIV operating piston, holding the MSIV in the open position. The sticking solenoid pilot valves were disassembled and inspected. Foreign material was observed on several internal parts of the solenoid pilot valve. The interfacing surface of the core assembly and plug nut of the "B" solenoid appeared to have a thin coat of foreign material. When the core assembly and plug nut were pressed together, which is the normal configuration when the solenoid is energized, the film acted as an adhesive. This adhesive was strong enough that when the plug nut was lifted from the bench, the core assembly adhered to it and was also lifted. The foreign material was determined to be Nyogel 775A. According to ASCO, this material is used during assembly as a thread lubricant on the solenoid cover and at the interface of the solenoid base and housing assembly. This is considered to be a common-cause failure.	b, c	Main Steam Isolation Valves

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Maine Yankee	30996003	960213	Maine Yankee was operating at 2440 MWt (90.3% power) when the reactor automatically scrammed due a loss of load trip from a high steam generator #3 water level. The high water level was due to a faulty positioner on the steam generator #3 Main Feedwater Regulating Valve (MFRV). The faulty positioner caused the main feedwater regulating valve to go to the full open position which resulted in overfeeding the #3 steam generator and the resultant trip on high water level. The faulty positioner on #3 MFRV has been replaced. The positioner on #2 MFRV was also replaced as a precautionary measure, since it had been in service since July 1992. The positioner on #1 MFRV had been replaced in January 1996.	c	Main Feedwater
Millstone 2	33689011	890906	An instrument air check valve in the air supply to the service water supply isolation to one of two Turbine Building Closed Cooling Water headers, was found incorrectly located. The location of the check valve would have caused the accumulator air supply to bleed down on a loss of the normal instrument air supply. The root cause of the condition was the incorrect re-assembly of the instrument air lines to the solenoid valves and the accumulator. It is assumed that the check valve has remained incorrectly located since the disassembly and reassembly of this air line for service water piping replacement.	b, c, Personnel Error	Turbine Building Closed Cooling Water Instrument Air
Millstone 2	33697011	970402	The closing force for multiple dual function (two separate pressure isolation functions) valves had been improperly set, resulting in the valves being incapable of closing to a leak tight condition against normal operating system pressure (NOSP). Eleven of the 23 valves tested were not capable of providing an adequate closing force. This deficiency could have resulted in the potential for a release of radioactive materials to the Auxiliary Building greater than analyzed in the facility Final Safety Analysis Report (FSAR). The closing forces were incorrectly set during the period between October 1986 and March 1997. The cause of this event was an insufficient program to ensure that facility procedures clearly addressed all related design basis functions. The affected valves were to have been adjusted to ensure they properly close against containment design pressure and NOSP. The appropriate procedures were to have been revised to ensure that proper valve control parameters are specified and verified after any maintenance activities are performed that could affect dual function valve closing forces. This condition affected 11 valves in three systems. This is considered to be a common-cause failure condition. Event No. 32070 referred to this event.	c	Containment, CVCS, RC Sample System Liquid Radwaste System

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Millstone 3	42396013	960515	An engineering evaluation determined that a design deficiency in the Residual Heat Removal System (RHS) was a condition that was outside the design basis of the plant. A loss of control air supplied from the non-safety-related instrument air system could cause the RHS control valves to fail open. If this condition occurred during the initial phase of a plant cool down, the Reactor Plant Component Cooling Water System (CCP) temperatures could exceed 125°F. This is the design temperature used in the system stress analysis. If RHS heat exchanger operation was initiated at 350°F RCS temperature, as assumed, then the RHS heat exchanger CCP outlet temperature could be as high as 250°F if the valves failed open. Under the resultant conditions the CCP piping would not meet the ASME Section III, Appendix F stress criteria. The original plant design did not consider that if the RHS flow control valves failed open on a loss of air, it could create unacceptably high RHS heat exchanger CCP discharge temperatures. This is considered to be a common-cause failure condition.	c	Residual Heat Removal System
Millstone 3	42396028	960916	An engineering evaluation identified a failure scenario in which a loss of Instrument Air (IAS) to temperature control valves in the Charging Pump Cooling (CCE) system serving the charging pump lube oil coolers, coincident with 33°F Service Water (SWP) temperature could result in overcooling of both trains of the charging pump lube oil system and challenge charging pump operability. Failure of the air-operated CCE valves to the full open position due to a loss of the non-safety related IAS system would adversely affect both trains of the charging pumps by allowing excessive cooling of the CCE system which cools the lube oil system. This condition alone could have prevented the fulfillment of the safety function of the system. The cause of the charging pump inoperability was inadequate original design. This condition would result from overcooling of the lube oil system from a failure of the non-safety related Instrument Air system coincident with a worst case minimum SWP temperature and maximum flow and heat exchanger cleanliness. Under these conditions, the air-operated CCE valves would fail open and excessive cooling of the lube oil system would occur. This particular combination of conditions was not considered in the initial design. A short term corrective action was committed to in the LER to install a temporary modification to limit the failure position of the three way CCE temperature control valve to ensure sufficient bypass flow (continued on next page)	c	Charging System

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Millstone 3	42396028 (continued)	960916	around the SW heat exchanger to maintain CCE temperature above 45°F. Other long-term actions were also discussed. This is considered to be a common-cause failure condition.	c	Charging System
Millstone 3	42396031	960906	A design engineering review concluded that specific safety related control valves could fail due to exceeding the manufacture's maximum operating pressure differential pressure rating of the Solenoid Operated Valves (SOVs) installed on the control valves. SOVs could fail to perform safety related functions because of excessive operating pressure differentials. This can result from failures of non-qualified air regulators installed in the Instrument Air (IA) system upstream of the SOVs. The failure of an air regulator would, in turn, result in full IA system pressure being applied to the SOV. The SOV can potentially fail to operate properly since they are not rated for full IA system pressure. A total of 48 SOVs which perform safety-related functions had been originally identified in various systems which would be susceptible to such a failure. The cause of this condition was the failure to consider the potential for pressure regulator failure in the original design and selection of SOVs. This is considered to be a common-cause failure condition. The valves were in the following systems: CVCS—5 valves; HVG - 4 valves; HPSI–3 valves; LPSI—2 valves; AFW—2 valves; Reactor Gas Drains—2 valves; Aux. Steam—2 valves; Chg. Pump Cooling—2 valves; Nitrogen Sys.—1 valve; Reactor Plant Vent—12 valves; Main Steam—3 valves; Primary Grade Water—1 valve; Turbine Drains—8 valves; Quench Spray System—1 valve.	b, c	13 Different Systems

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Millstone 3	42396036	960926	The High Pressure Safety Injection (SIH) and Low Pressure Safety Injection (SIL) systems would have been subject to degraded performance due to possible mispositioning of normally closed safety related air operated valves. Mispositioning of these 21 valves can be postulated to occur under post-accident harsh environmental conditions due to failure of non-qualified power and control circuits. As a result the potential diversion of SIH and/or SIL flow under accident conditions could have been more than the margin allowed within the Loss of Coolant Accident analysis. Seventeen additional safety-related air operated and solenoid operated valves were subsequently identified where failures of non-qualified control circuits could degrade performance of a safety system function. These additional valves are located in the following systems: Reactor Plant Component Cooling Water (CCP), Containment Vacuum, Reactor Plant Sampling (SSR), Post Accident Sampling (SSP), and Main Steam to the auxiliary feedwater steam turbine. The cause of the reported conditions was a design mistake. The initial plant design did not adequately consider the potential mispositioning of these valves under harsh environmental conditions or active failure. This is considered to be a common-cause failure condition.	c	High and Low Pressure Safety Injection
Millstone 3	42396040	961024	An engineering evaluation determined that a failure scenario for the Reactor Plant Component Cooling Water (CCP) system had the potential for a loss of system safety function. The failure scenario involves a loss of the non-category 1 Instrument Air System, which would allow CCP valves to reposition to a maximum cooling configuration. Coupled with a low heat load and minimum Service Water (SWP) inlet temperature, the CCP system could reach temperatures lower than values for which they are analyzed, thereby rendering the CCP system, and systems that it serves, potentially inoperable. This was caused by improper initial design of the CCP system. The failure mode described in the LER was a design oversight by the plant's architect engineer that occurred during the original design process. The design basis analysis for the CCP system focused on high CCP heat load conditions. The designers did not analyze for extremely low CCP heat loads concurrent with very low SWP temperatures. This is considered to be a common-cause failure condition.	c	Reactor Plant Component Cooling Water

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Nine Mile Point 1	22096004	960520	Feedwater flow control valve oscillations, caused high reactor water level, which tripped the turbine and resulted in a reactor scram. The cause of the event was a degraded actuator for the #13 feedwater flow control valve (FCV). Upon disassembly of the actuator, worn o-rings and bushings as well as a misaligned stem and bushing were found. Two contributing factors were identified. The first was that the design of the current configuration of the pneumatic controls left little margin for degradation of the actuator and caused the valve to cycle frequently. This cycling may have caused increased wear on the actuator bushings and rubber goods which could lead to volume booster needle valve clogging and cause instability in the control system. The second is the misalignment of the actuator stem. This misalignment caused binding which could also have caused erratic operation. The probable cause of the misalignment was incorrect factory assembly of the actuator, resulting in a misalignment of the internal bushings. The control system design was to have been evaluated to determine if changes were needed to increase reliability. Potential recommended actions included modifications to increase pneumatic volume of sub-components to decrease the gain of the valve controls and limit cycling of the feedwater control valve. This would allow the booster to be less susceptible to wear material and reduce the number of service cycles on the boosters.	c	Main Feedwater
Oconee 2	27093002	930610	An evaluation of air leakage from backup accumulators on AOVs associated with containment integrity revealed that containment integrity could be lost during a postulated Loss of Instrument Air coincident with a Small Break Loss of Coolant Accident. Seal Return Block valves, which are air operated valves outside containment whose purpose is containment isolation for the RCP Seal Return lines, may return to the open position due to accumulator leakage. Due to leakage in the backup accumulator system associated with the valves, the accumulator may not have maintained them closed during a postulated event. In response to Generic Letter 88-14 and other industry guidance, selective air operated valves were tested. During the preparation of the initial test procedures emphasis was placed on ensuring that these valves failed to their required position on a loss of instrument air. It was also recognized that leakage of the accumulator system could result in these valves opening; however, the appropriate acceptance criterion was not established at the time.	c	RCP Seal Return

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Oconee 3	28791007	910703	A reactor trip occurred from 100% power on a loss of main feedwater. Five condensate (in parallel) demineralizer valves failed closed when the master controller for these valves failed. The master controller failed when particles from a degraded seal clogged the Instrument Air flow path. Demineralizer bypass valves could not open to compensate because an Operator had failed to return them to automatic control. This is considered to be a common-cause failure condition. The loss of condensate flow resulted in the trip of condensate booster pumps due to low suction pressure, which then caused a trip of the main feedwater pumps, followed by the reactor trip. Emergency feedwater pumps started, but a solenoid valve failed, requiring manual operator action to control flow to one steam generator. The SOV was a Valcor V-70900-21-3. This model SOV was used in several other applications at the plant and had failed to operate properly in the past. Because it is normally energized and operates at an elevated temperature (approximately 250°F), the licensee believed that the temperature caused degradation leading to the valve sticking open when de-energized. This was an ASP Program Event. See Appendix B.	a, b	Main Feedwater Condensate System Instrument Air
Oyster Creek 1	21985012	850612	As the result of a reactor scram all rods inserted but one of the two Scram Discharge Volumes did not fully isolate. The resulting flow of hot water through the Scram Discharge Volume caused steam and paint fumes to discharge in the reactor building. This, in turn, activated the deluge fire system on one level of the reactor building. One SOV drain valve that isolates the Scram Discharge Volume bottomed out before the valve was fully seated because the stroke adjustment was improperly set. The actuator spring on a second drain valve was improperly sized and opened slightly when pressure from the first valve was applied to its seat. With both valves slightly open, the leak was established. Failure of the SDV to fully isolate would be most severe in the event of a loss of offsite power, which would cause an immediate reactor scram. Feed pumps would not be available and the leak through the SDV would lower the reactor level until Core Spray would be initiated. Operator action to reduce reactor pressure by manual action would be necessary. This was an ASP Program Event. See Appendix B.	c	Scram Discharge Volume

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Palisades	25578003	780108	With the reactor in cold shutdown, CV-3025 (shutdown cooling system HX outlet valve) failed closed, causing a loss of shutdown cooling flow and allowing PCS temperature to rise to approximately 215°F. Condition was attributed to moisture contamination of the air supplied to the AOV. This event is discussed in Section 8.3.2 of this study.	a	Shutdown Cooling (RHR)
Palisades	25581030	810718	CV-3025 failed closed and isolated shutdown cooling system. Condition was again attributed to moisture contamination of the air supplied to the AOV.This event is discussed in Section 8.3.2 of this study.	a	Shutdown Cooling (RHR)
Palisades	25587018	870620	During a power reduction, the "A" train main feedwater regulating valve, failed to close. Shortly after, when the turbine generator was removed from service, a moisture separator and reheater control valve failed to close. The cause of the failure was attributed to a plugged "close" port in the valve positioner. The particle that had plugged the port was not recovered; however, it was believed to have originated from the upstream carbon steel manifold or materials still present from post air system problems. The LER did not indicate the nature of the "air system problems." The failure of the moisture separator reheater control valves to trip closed was attributed to set point drift or in an improper setting within the temperature governed control circuit of the auto control feature. This auto control feature provides for slow closure of the valve for high temperature protection of the low pressure turbine.	a	Main Feedwater Instrument Air
Palisades	25592007	920205	The main steam isolation valve (MSIV) actuator solenoid valves, could have been rendered inoperable by a main steam line break outside of containment because the solenoids were all served by the same power supply. The original plant design placed the MSIV actuator solenoid valves in the same room as the MSIVs. During the 1974 refueling outage a second redundant set of MSIV actuator solenoid valves were installed in the turbine building (a non-harsh environment). However, the redundant solenoid valves located in the non-harsh environment received power from the same electrical circuit as the valves located in the harsh environment. Following a main steam line break outside of containment, a short of the solenoid valves in the harsh environment could short the fuses in the control scheme. These same fuses were in the circuit for the redundant solenoid valves in the non-harsh environment. Thus, a failure of the component in the harsh environment could also have prevented the redundant component from performing its safety function. (continued on next page)	b, c	Main Steam Isolation Valve

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Palisades	25592007 (continued)	920205	The main steam line isolation valve actuation solenoid valves were to have been relocated to a non-harsh environment so that a main steam line break outside of containment would not render the MSIVs closing solenoid valves incapable of performing their design basis function. This modification was to have been completed prior to startup from the then current refueling outage. This is considered to be a common-cause failure condition.	b, c	Main Steam Isolation Valve
Palisades	25592023	920327	It was discovered that previously unidentified junction boxes in the electrical circuits for solenoid valves SV-3084 and SV-3085 and position switches POS-3084 and POS-3085 which control the hot leg injection letdown valves CV-3084 and CV-3085 contained environmentally unqualified electrical connections. (Note that the position switches were not the correct type.) Further, the Air Circuit Assembly (including junction boxes) was found to be not environmentally qualified. The causes of this event included : 1. The failure of the plant designers to ensure that the procurement and installation of the Air Circuit Assembly met the requirements of 10CFR50.49 2. The lack of proper administrative controls for procurement and documentation of environmentally qualified equipment.	b, c	High Pressure Safety Injection
Palisades	25594004	940209	A previously unidentified single failure mechanism was discovered, involving the dependence on non-safety-related instrument air to provide motive power to safety-related AOVs. This condition could have affected the operability of the Engineered Safeguards System (ESS) pumps. The Palisades Engineered Safeguards Systems (ESS) pumps are equipped with seal and bearing cooling. Pump cooling is required for the long term operability of the pumps. ESS pump cooling is provided by: diverting a small portion of each pump's discharge flow through a heat exchanger, where it is cooled by Component Cooling Water (CCW); directing it through cooling jackets around the pump's seal areas; and then returning it to the pump suction. CCW is also directed to cooling jackets around the pump bearing areas. The CCW flow for all pump cooling is controlled by common supply and return AOVs, CV-0913 and CV-0950. (The Palisades CCW system has redundant trains of pumps and pump controls, but uses a single, common, piping system.) These two AOVs are opened automatically on a Safety Injection Signal (SIS). The normal configuration for the CCW to the pump seal cooling had been for (continued on next page)	c	Engineered Safeguards Systems (HPSI, LPSI, CS CCW)

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Palisades	25594004 (continued)	940209	both CV-0913 and CV-0950 to be closed. Both of these AOVs are spring loaded to the open position, requiring availability of both instrument air and electrical power to hold them closed. Other AOVs, which supply service water (SWS) as backup cooling, are also normally closed, but are spring loaded to the closed position, thus requiring availability of both instrument air and electrical power to open them. If either CV-0913 or CV-0950 failed to open on demand following a Loss Of Coolant Accident (LOCA) (possibly due to binding of the valve stem or similar failure), cooling flow to the ESS pumps would not be available from the CCW system. Conceptually, recovery from such a failure would simply require the operators to initiate the alternate cooling by closing whichever CCW valve remained open and opening the SWS valves. However, instrument air is not a safety grade system and therefore might not be available to close either CCW valve or open the SWS valves. The use of non-safety grade instrument air to open the backup (SWS) cooling AOVs implies that credit should not be taken in the design basis for operation of backup cooling. Therefore, if either CV-0913 or CV-0950 should fail to open on demand, all (safety grade) cooling to the ESS pumps, and consequently the functioning of the pumps themselves, could eventually be lost. These conditions lead to the possibility of a single active (common-cause) failure disabling the ESS pumps. Failure of the ESS pumps would be a matter of considerable safety significance. The LER did not explain if or how the SWS backup function would still be an operational option. Palisades decided that operating with the CCW valves normally open meets plant design and licensing requirements and reduced the overall plant risk compared with having the CCW valves normally closed. Accordingly, the CCW valves were aligned so that their normal position is to be open. This is considered to be a common-cause failure condition.	c	Engineered Safeguards Systems (HPSI, LPSI, CS CCW)

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Palisades	25598006	980305	Event No. 33843 for Palisades in Table 4 was originally recorded for this event. The LER was subsequently issued and is discussed here. The original design bases, and subsequent design reviews, for the HP Air System address the Large Break Loss of Coolant Accident (LBLOCA), yet did not address, in detail, the SBLOCA. As a result, the impact of a loss of the HP Air compressors on the ability to supply air needed to align valves for sump recirculation during a SBLOCA, and the need to incorporate manual operator actions in EOPs to assure HP air reliability were not considered. As a result of the design basis reconstitution for the HP Air System, the licensee identified the need for procedural guidance to direct manual operator actions to restore the HP Air System during a SBLOCA. The HP Air System supports ECCS realignment to the recirculation mode by providing air required to operate control valves needed to switch the suction of the ECCS pumps from the Safety Injection Refueling Water Tank (SIRWT) to the containment sump. During a LOCA with LOOP, the HP Air System compressors are load shed from safety-related electrical buses and must be manually repowered at their respective motor control centers (MCCs). When the air compressors are without power, the receiver tanks are not being charged. During this time, the HP Air System pressure experiences a gradual decay due to air bleed-off from pressure regulators and air leakage through seals in piston-driven control valve actuators. The LER focuses on the procedures needed to ensure restoration of HP air; however, the effects of gradual loss of air on the failed position on valves that are supplied by HP air had not been investigated.	c	HP Air System, ECCS
Palo Verde 1	52889005	890412	Atmospheric Dump Valves at PVNGS would not perform their intended safety function because the bonnet pressure was too high. In addition, due to the design of the valve/operator, these AOVs were subject to oscillations that could damage the valve. These conditions are also described in a PVNGS ATMOSPHERIC DUMP VALVE ANALYSIS REPORT, dated March/April 1989. (LER 53089001 described the original performance problems of ADVs during a transient event at Palo Verde 3 that led to the investigation of the ADVs in all three units. This was an ASP Program Event. See Appendix B.) This is considered to be a common-cause failure condition.	c	Containment Main Steam

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Palo Verde 1	52894009	941118	Seal leakage was reported on one of the Fisher air-operated letdown/containment isolation valves reported to be undersized in LER 52895007.	c	Containment
Palo Verde 1	52895007	950512	Three Fisher air-operated letdown/containment isolation valves were found to be undersized and the bench settings were also set too low. These valves were provided by the system vendor (ABB-CE) and the design basis evaluation was found to be at fault. Modifications included spring replacement, reducing the stroke length of the actuator, and modifying the limit switches. Similar conditions were found at Units 2 and 3. The system vendor (ABB-CE) procured valves that had undersized air actuators and bench sets which were too low to provide desired valve seating force for the differential pressure which would be present across the valves during a letdown line break. They could not determine whether or not the procurement of the valves was approved by the Architect Engineer or the owner. The root cause of the design deficiency was attributed to the absence of a detailed design basis evaluation for air operated valves as part of an AOV program. This is considered to be a common-cause failure condition.	c	Containment
Peach Bottom 3	27891017	910924	During routine preventative maintenance on the Main Steam Relief Valve (MSRV) solenoid valves (SOV) the MSRV SOV wiring insulation was discovered to be degraded. Improper installation of the MSRV thermal insulation caused a high temperature environment around the solenoid valves and associated wiring. The MSRV thermal insulation had been improperly installed during the Refueling Outage in 1989. This resulted in an unusually high temperature environment in the immediate vicinity of the SOVs and associated wiring. This high temperature condition caused the MSRV SOV wiring insulation to degrade. The temperature increase from the insulation installation error caused the expiration of the Environmental Qualification (EQ) life of the components after approximately three days of operation. The ADS MSRVs comprise 5 of the 11 MSRVs and three of the ADS MSRVs were affected by this condition. This is considered to be a common-cause failure condition. This was an ASP Program Event. See Appendix B.	Personnel Error, b	Main Steam Relief Valve

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Perry	44087009	870227	Failure of two control air solenoid valves rendered both emergency diesel generators inoperable. The valves were subjected to temperatures near the upper end of the qualified operating range and were continuously energized. These factors are believed to have eventually caused degradation of the valve poppets and failure of the valves. The valve manufacturer was contacted and responded that failures seldom occurred and that the most common failure developed when the poppets, made from Buna-N, were subjected to an incompatible lubricant or excessive heat. These valves are subjected to temperatures near the upper end of the qualified operating range and are continuously energized. These conditions were believed to eventually cause degradation of the material within the valve resulting in air leakage and eventual failure. Both leaking solenoid valves had previously been identified for replacement due to leakage, with work requests initiated but not yet implemented. In addition, several Surveillance Tests had been run successfully subsequent to identifying the leaking valves. The cause of the immediate diesel failures was the accelerated degraded condition of the leaking solenoid valves. At discovery, the conditions were evaluated not to require immediate action, thus expeditious replacement did not occur. Subsequently, it was determined that the control air pressure regulators which supply air to these solenoid valves were malfunctioning and may have contributed to the diesel failure. This is considered to be a common-cause failure condition. This was an ASP Program Event. See Appendix B.	b, c	Emergency Diesel Generators
Perry	44090021	900709	During cooldown of Unit 1, in preparation for the second refueling outage, directions were given to close the main steam isolation valves (MSIV's) in order to maintain control of the reactor cooldown rate. All eight of the MSIV's slow closed properly; however, two of them (1B21-F022C and -F028B) failed to remain closed when their control switches were placed in the 'close' position. These valves later closed on their own, with the control switch left in the 'close' position. Centerior energy letter PY-CEI/OIE-0327 L to the NRC, dated September 11, 1990, documented this event. Similar events occurred on October 29, 1987, November 3, 1987, and November 29, 1987. These earlier events were caused by degraded ethylene propylene diene monomer (EPDM) elastomers in the ASCO 3-way dual coil solenoid valves (October 29, 1987 and November 3, 1987), and by a sliver of EPDM inside a solenoid causing it to stick (November 29, 1987). The elastomers degraded because of locally high temperatures resulting from steam leaks. The EPDM elastomers were replaced with Viton in a complete solenoid valve changeout during (continued on next page)	b, c	Main Steam

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Perry	44090021 (continued)	900709	Refueling Outage 01. The cause of the September 7, 1990 ,events was determined to be failure of the ASCO 3-way dual coil solenoid valves to change position after being de-energized. Specifically, the disc holder seat elastomer failed to shift from the exhaust port.	b, c	Main Steam
Pilgrim	29389002	890110	A shutdown was initiated by the Nuclear Engineering Department upon notification to Station Management of a potential problem with the air supply (air accumulators) for two primary containment air operated valves. The discovery resulted from an analysis of the non-safety-related Instrument Air System conducted per NRC Generic Letter 88-14. Redundant check valves were capable of ensuring the primary containment function. The cause is attributed to insufficient capacity of the original accumulators dating back to original plant construction. This is considered to be a common-cause failure condition.	c	Containment Isolation System
Pilgrim	29389004	890127	An Unusual Event was declared and a shutdown began because two primary containment air operated valves were inoperable. The air operated valves were declared inoperable because of the pressure drop rate of the stored air supply for the valves. The air operated valves were in the closed position at the time of the event. The cause for the pressure drop rate was collective air leakage at some of the air supply connections and leakage past the seat of the pressure relief valve(s) installed in each air supply (Trains 'A' and 'B') to the air operated valves. The air supply had been modified, tested and placed in service just prior to this event using acceptance criteria that established minimum operability without sufficient margin to accommodate increased leakage. This is considered to be a common-cause failure condition.	c	Containment Isolation System

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Pilgrim	29397025	971123	A Technical Specification (TS) required shutdown was completed because two main steam isolation valves (MSIVs) in separate main steam lines were inoperable. The root cause of the failure of MSIVs AO-203-2B and -1C to close via their push button was main closure spring relaxation leading to loss of closure force at the end of the closure stroke. A contributing cause of the failure of MSIV AO-203-1C to fully close was increased friction between the spring plate and actuator stanchions due to improper adjustment or loosening of the cam rollers. A contributing cause of the failure of MSIV AO-203-2B to fully close was improper installation of the setscrew that locks the stem threaded connection together in the hydraulic dashpot. The setscrew was able to back out and was broken off during valve stroking. As a loose part in the hydraulic dashpot, there was evidence of some minor scraping between the hydraulic piston and the cylinder wall that could have increased overall assembly resistance. Corrective action taken included replacement of some closing springs and overhaul of actuators. Corrective action planned includes replacement of closing springs on the remaining MSIVs, evaluation of alternate spring designs, and revision of maintenance procedures. This is considered to be a common-cause failure condition. The main steam isolation valves (MSIVs) form part of the nuclear system primary containment. The MSIVs isolate the main steam pipelines to limit the reactor coolant loss and radioactive material release. There are two MSIVs per main steam line. The inboard valves are located just inside primary containment (AO-203-1A, -1B, -1C, -1D), and the outboard valves are immediately outside of primary containment (AO-203-2A, -2B, -2C, -2D). Either valve is capable of isolating the line. The control systems for the inboard and outboard valves are independent of each other. The MSIVs are twenty inch, Y-pattern valves manufactured by the Atwood and Morrill Company, Model 20849-H. An accumulator is connected to each MSIV between the pneumatic supply and the air cylinder valves. The accumulator functions to store pneumatic energy for closing the MSIV if the supply of pneumatic energy (air or nitrogen) is lost. The accumulator volume provides full valve movement through one-half cycle, open to close. The MSIVs are designed to close via spring force (only) if pneumatic power is not available to assist closing the valves. See Event No. 33360 in Table 4.	c	Main Steam, Containment

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Pilgrim	29397026	971206	A Scram occurred while ascending to power from the outage described in LER 29397025. The cause of the scram was the failure of the "A" feedwater regulating valve (FRV) in the full open position due to the misalignment of the valve clip inside the pilot valve assembly of the positioner. Corrective action taken included replacing the valve positioner and performing calibration adjustments. The Scram root-cause was traced to the opening of the "A" FRV (FV-642A). The opening of FV-642A was caused by the inadvertent misalignment of the pilot valve clip inside the pilot valve assembly of the Bailey positioner for the "A" FRV. The pilot valve clip was misaligned when the positioner was opened during the MSIV forced outage (LER 29397025). After the scram, when the positioner for the "A" FRV was opened, the pilot valve clip was found to be displaced. This caused the positioner's pilot valve stem to move downwards, porting air from the top of the "A" FRV actuator, allowing the valve to go full open. The feedwater regulating valve, FV-642A, is a Copes-Vulcan fourteen inch, 900 psi, double poppet, balanced, hydraulically dampened, diaphragm operated control valve equipped with a D100-160 actuator and a Bailey AV1 positioner. See Event No. 33360 in Table 4.	c	Main Feedwater
Point Beach 1	26697014	970321	Engineers discovered a condition that alone could have prevented the Auxiliary Feedwater (AFW) System from automatically performing its safety-related function during design basis accidents involving a loss of instrument air and reduced steam generator pressures. A loss of instrument air during the accident would cause both motor-driven AFW pump (MDAFWP) flow control valves to fail open. The postulated event scenario is the result of a latent characteristic of the original AFW system design. The original design did not provide ample assurance that the MDAFWPs would automatically function during all design basis events. The expected fix was to furnish a reliable pneumatic supply to the control valves. Event No. 31995 refers. This is considered to be a common-cause failure condition.	c	Auxiliary Feedwater

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Point Beach 1	26698008	980203	A 10 CFR 21 Notification (No. 1998-12-0) was received from Engine Systems, Inc., which indicated that the air start motor SOVs used on Emergency Diesel Generators (EDGs) do not meet the minimum DC voltage requirements when inlet pressures below 200 psig are applied to the SOV. Seven of the SOVs were sold to Wisconsin Electric. The valves were found on all four EDGs. Calculations performed to assess available DC voltage at the air start SOVs indicated that only one of the EDGs (G03) had sufficient post-accident voltage. Since G03 was only connected to one of the safeguards buses, the plant was considered to be outside the design basis. The SOVs rely on system air pressure to assist the coils in overcoming the force of the springs and thus operating the valve plungers. The original valve design assumed air pressure at 200 psig and, therefore, had a weaker spring than (later) supplied with the 275 psig rated valve. The SOVs operate with air pressure under the seat at a minimum of 165 psig for G01 and G02 (Train A) and for G03 and G04 (Train B). The valves are supplied with DC control power. The voltage at the valve is dependent on the battery load during an accident scenario that uses the battery alone under assumed accident load conditions. In the case of these valves, the recommended minimum voltage for the valves was 105 Volts DC. Therefore, the effect of control power voltages less than that recommended by the vendor would not ensure the valve would open due to the stronger spring in the 275 psig rated valve if the air pressure at the valve was less than 200 psig. Initial calculations for the expected at each air start assuming accident DC load conditions resulted in voltages less than 105 V DC for G01, G02 and G04. In summary, the cause of the condition was the use of a 275 psig solenoid valve which does not meet the minimum DC voltage available during an accident, which requires the station batteries for control power. The SOV relies on system air pressure to assist the coil in overcoming the force of the spring and thus operating the valve plunger. Reduced system pressure combined with reduced coil voltage results in the inability of the valve to operate satisfactorily. This problem is not applicable to the original 200 psig valve (identical to the 275 psi valve except for the spring) because it has a weaker spring, and therefore, less air pressure is needed to overcome the lower spring force. There are 335 SOVs, distributed among 17 nuclear plant owners, listed in the Part 21 notice. Clinton LER 46198009 also refers to Engine Systems Part 21 Notification #1998-12-0.This is considered to be a common-cause failure condition.	b, c	EDGs

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
River Bend	45889022	890502	It was identified that the safety related air dryer outlet isolation valves might not perform their design function on the loss of the non-safety- related air dryer or piping. The root cause of this condition was an engineering design oversight in that these valves were installed in the direction of normal air flow (flow above the disc), instead of in the direction required to isolate the post accident safety-related air supply from the non-safety related dryer and associated piping. This LER was included as an example of common-cause failure in AEOD/92-02, "Insights From Common-Mode Failure Events." This was an ASP Program Event. See Appendix B.	c	Instrument Air
Robinson	26194002	940425	During Main Steam Isolation Valve (MSIV) operability testing, all three MSIVs were declared inoperable when they failed to meet the Technical Specification (TS) required closure time. The original MSIV actuator design incorporated minimum design margin. The potential for failure of the MSIVs to meet their TS stroke time requirements undetected was caused by failure of the surveillance and modification testing program to account for the minimal design margin of the actuators. The accumulators were minimally sized to close the valves within five seconds under hot, no load conditions, and to maintain the valves closed if non-safety-related instrument air was not available to the valve actuators. This is considered to be a common-cause-failure condition.	c	Main Steam Isolation Valves
Salem 1	27291030	910920	Both Pressurizer Pressure Operated Relief Valves (PORVs) failed to open during a functional check. The cause was degraded Buna-N diaphragm material causing air leakage from the valve actuators. The initiating cause of both PORVs failing to open was loosening of the fasteners in the valves' actuator diaphragm enclosures. The Buna-N diaphragm material incurred "creep" where the diaphragm changed from its original geometry under load and over time. Elevated ambient temperature degraded the Buna-N diaphragm material resulting in the material taking a permanent set, a loss of resilience, and extrusion of the material from the clamped region. Permanent thinning of the material in the clamped region caused loss of fastener preload. This was observed as loosening of the diaphragm fastener cap screws/nuts, which allowed a leakage pathway for the diaphragm control air. This is considered to be a common-cause-failure condition. This was an ASP Program Event. See Appendix B.	c	Pressurizer Pressure Operated Relief Valves

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Salem 1	27296012	960723	On July 23, 1996, a review determined that the keys/keyways on the actuators for the Residual Heat Removal (RHR) flow control valves (6 AOVs per unit) are subject to failure. The valves were made by Fisher Controls International, Model Type 656/7600. When using the simplified Fisher Catalog 14 methodology, the calculated maximum stem torque exceeds the vendor specified allowable torque. Preliminary calculations of the shaft torque and resulting average shear stresses in the valve stem key and keyway were also performed. The calculations showed that for normal operating conditions, the average shear stress in the key may exceed the material yield stress. The calculations also showed that although the average shear stress in the valve stem was estimated to be less than the minimum material yield stress, the fatigue life of the shaft keyway appeared to be limited to a low number of cycles. Key failures in the past are attributed to an overload during normal operation of these valves. A review of the original design revealed that these valves were installed with little or no design margin and the keys were likely to fail due to low cycle fatigue with stress levels exceeding yield strength. Corrective action was to replace the valves and review Fisher Model 7600 valves for similar concerns. This is considered to be a common-cause failure condition.	c	RHR
San Onofre 2	36196011	961216	On December 10, 1996, the licensee, in developing a test program for air operated valves, applied test equipment to containment isolation valve 2HV0513. It was likely that the actuator settings for valve 2HV0513 would not generate sufficient closing force to overcome internal pressure and packing drag under design basis conditions. Edison concluded that valve 2HV0513 had been inoperable when Unit 2 was in Mode 1. The licensee concluded that either of two separate errors could have caused valve 2HV0513 to have insufficient closing force: (1) vendor setpoint methodology error; or (2) a deficiency in the reassembly procedure. The licensee committed to reset and retest valve 2HV0513 prior to returning Unit 2 to Mode 4. They also were to upgrade its maintenance procedures to conform to the current vendor manual. They performed an analysis on all air-operated containment isolation valves and confirmed that all other Units 2 and 3 air operated containment isolation valves have sufficient actuator closing force to remain operable. The licensee also was to verify actuator settings for Unit 2 air operated containment isolation valves which may not have had their actuator settings established in accordance with the vendor recommendations. Similar actions were to have been completed for Unit 3 during its next refueling outage.	c	PCIS

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Sequoyah 1	32792018	921026	Steam generator (SG) No. 3 was experiencing low level because of irregular feedwater regulating valve (FRV) operation. The unit operator placed the flow indicating controller in manual for SG No. 3 FRV to attempt to gradually raise the level. The automatic-manual switching relay (K-1) for this flow indicating controller did not change state, causing the manual control circuit to be electrically inoperable. This condition resulted in the FRV going to the full open position and not responding to the manual changes to the controller input, thereby causing a reactor trip because of a turbine trip that resulted from a high-high feedwater level in the No. 3 SG. Approximately 1,000 gallons of water was entrained in the nonessential air system which supplies the essential air system. The water saturated the essential air dryers. The saturation of the air dryers caused the nonessential control air system pressure to drop, resulting in an automatic isolation of the essential control air system. Some water was carried over to end-use devices. The first components affected by the water entrainment were the FRVs. The FRV controllers malfunctioned because of the entrained water. This is considered to be a common-cause-failure condition.	a	Main Feedwater Essential Air Non-essential Air System
Sequoyah 1	32797012	970801	While implementing a modification to improve the materiel condition of the control and Service Air System the control air header pressure dropped rapidly after modifications personnel cut into-a six-inch control air header located inside an equipment clearance boundary. The loss of control air pressure resulted in a runback of the Unit 1 and Unit 2 Turbines and instabilities in Unit 1 secondary side flows sufficient to warrant a manual trip of the Unit 1 reactor. The immediate cause of the loss of control air header pressure was corrosion products (rust debris) that inhibited full closure of one of the six-inch gate valves used as a clearance boundary. The corrosion products were assumed to have resulted from water intrusion that had occurred about five years earlier.	c	Control and Service Air System
Shearon Harris	40098001 (Event No. 3513 refers)	980109	A potential failure mechanism existed where a leak in the non-safety Instrument Air System could result in the inoperability of the Steam Generator (S/G) Pre-heater Bypass Isolation Valves (1AF-64, 1AF-102, and 1AF-81. These valves are safety-related containment isolation valves that are required to automatically shut in 10 seconds or less upon receipt of a Main Feedwater Isolation Signal (MFIS). These valves are each positioned by a pneumatic piston-operated actuator and are opened and closed by high pressure air (approximately 150 psig) from the (continued on next page)	c	Steam Generator

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Shearon Harris	40098001 (Event No. 3513 refers) (continued)	980109	actuator's accumulator. The accumulator is maintained at a higher air pressure by an air intensifier pump. However, an air leak was postulated in the non-safety-related IA piping that could possibly reduce the air inlet pressure to just low enough to affect proper operation of the actuator's 3-way and 4-way pilot valves and not be detected by the pressure switches in the main header of the Instrument Air System. If this occurred, the pilot valves would shuttle, causing the accumulator pressure to bleed off, which would prevent the valves from closing as required. Operations personnel would have no indication of accumulator low pressure other than local observations made by an auxiliary operator and possibly dual valve indication in the main control room due to the valves cycling slightly. This potential scenario was reported to the NRC via the emergency notification system on January 9, 1998. (Refer to Event No. 33513.) The cause of this condition was inadequate design control during development of a plant modification implemented in August 1984, which was prior to issuance of the Harris Plant Operating License. Specifically, NRC Information Notice 82-25, "Failure of Hiller Actuators Upon Gradual Loss of Air Pressure," stated that on a gradual loss of control air, the pneumatic control valves may assume some intermediate position and cause the stored air in the accumulator to vent to atmosphere and prevent the actuator from performing its safety function of closing. The investigation for this event also revealed additional missed opportunities to identify this design deficiency in the Instrument Air System. These included: (1) HNP's response development for NRC Generic Letter 88-14 in 1989; (2) Adverse Condition Report #91-314 initiated in June 1991, which identified a leak in the supply air regulator for 1AF-81 that significantly lowered accumulator pressure; and (3) development of PCR-6158 and its associated evaluation (PCR-6066) in 1992, which implemented a modification to the actuator air circuitry to enhance the air intensifier pump and upgrade portions of the control air piping to safety-related. In each of these cases, the "smart" leak scenario described in the LER was either not identified or not considered credible due to the incorrect assumption that the system design met the single failure criteria. This is considered to be a common-cause failure condition that existed for 13 years.	c	Steam Generator

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Three Mile Island 1	28986007	860326	The TMI-1 FSAR states in Subsection 9.10.3.2 that the Two-Hour Backup Air Supply System meets the single failure criteria. As a result of a Safety System Functional Inspection by the Performance Appraisal Team, a concern was developed that the installed Two-Hour Backup Air Supply System may not meet the single failure criteria. If the Backup Air Supply were lost, air pressure to control Emergency Feedwater System Valves and Atmospheric Dump Valves would result, requiring manual action by the operators to control the affected valves. Upon further investigation by GPUN, it was determined that indeed the single failure criteria was not met. The root cause for the above was an error in the original system design. The design verification performed did not identify that the final design was not in accordance with the System Design Description requirements.	c	Two-Hour Backup Air Supply System EFW Atmospheric Dump Valves
Turkey Point 3 & 4	25085020	850723	Turkey Point Units 3 and 4 were notified by the Power Plant Engineering Department of a 10 CFR part 21 deficiency concerning the ability to close the main steam isolation valves (MSIVs). Each MSIV is a check valve installed in the reverse direction. MSIV closure is assisted by instrument air, a partial travel spring and steam flow. Under low steam flow conditions and a loss of non-safety-related instrument air pressure, the accumulator air volume may not be sufficient to close the MSIVs. The inability to close the MSIVs during an uncontrolled steam release could be postulated to result in a loss of the steam generators as a secondary heat sink. The design of the MSIVs was to have been upgraded to ensure that each MSIV will meet the Final Safety Analysis Report closure criteria without steam flow assistance. This is considered to be a common-cause failure condition. Refer to NUREG-1275, Vol. 2 , for additional information.	c	Main Steam Isolation Valves

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Turkey Point 3	25085021	850722	The "B" SG bypass feedwater control valve failed to open following a reactor trip. About 3 hours later, the "C" SG bypass feedwater control valve failed open. After making repairs to correct the problem of the reactor trip, operability checks of AFW flow control valves CV-3-2832 ("B" SG) and CV-3-2833 ("C" SG) for train 2 showed the valves would not close and were declared inoperable. The presence of moisture in the instrument air system caused the formation of corrosion products which, along with the moisture, were supplied to the valve actuators and related control equipment. This event was the common cause of a number of related, simultaneous AOV and AOV-control equipment failures. Refer to Section 5.1.2 of NUREG-1275, Vol. 2 for a detailed discussion of the conditions. The licensee has since resolved these conditions satisfactorily, primarily by dramatically improving instrument air quality. This was an ASP Program Event. See Appendix B. This is considered to be a common-cause failure condition.	a	Auxiliary Feedwater Instrument Air
Vermont Yankee	27198025 (Rev. 2)	981211	On 12/11/98, with the plant at 100% power, it was determined that one Scram Discharge Volume (SDV) drain valve (CRD-LCV-33B) did not meet the stroke time requirements of the In-Service Test (IST) Program. Both the North and the South SDV's have two drain valves in series, CRD-LCV-33A/C on the North and CRD-LCV-33B/D on the South. The drain valve, CRD-LCV-33B, was subsequently declared inoperable. The SDVs are used to limit the loss of and contain the reactor vessel water from all control rod drives during a scram. These volumes are provided in the scram discharge header. During normal plant operation the volumes are empty with all the drain and vent valves open. Upon receipt of a scram signal, the vent and drain valves close. While scrammed, the control rod drive seal leakage continues to flow to the discharge volumes until the discharge volume equals reactor pressure. Following a scram, when the scram signals are cleared and the Reactor Protection System (RPS) logic is manually reset, the vent and drain valves are opened and the SDV drained. (continued on next page)	c	Control Rod Drive

Table 2. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Vermont Yankee	27198025 (Rev. 2) (continued)	981211	According to the LER, the root causes of this event were inadequate actuator sizing calculations (performed by the vendor) under the design conditions specified in the procurement specification, and inadequate manufacturing Quality Assurance controls to ensure specification requirements were maintained. Contributing causes included conflicting information in the design package that was not identified by Vermont Yankee, and closing forces of the actuator at either end of the valve stroke were apparently not as designed. The actuators and valves were installed under Engineering Design Change EDCR 97410 in April of 1998 and were sized in accordance with the vendor's recommendations. These drain valves are required to go shut on a scram signal to isolate the SDV and act as a primary containment isolation valve when the scram valves are open. This issue was addressed by the installation of larger actuators, on December 21, 1998, via the Minor Modification process, which were properly sized to operate the valves under any design conditions at Vermont Yankee. The licensee stated that the LER constituted a Part 21 notification in accordance with 10CFR21.2(c) and NUREG-1022, Revision 1. Further, the licensee informed NRC by Event 35150 dated December 14, 1998, that this event was considered to be a common-cause failure.	c	Control Rod Drive

Notes:

1. The NRC Licensee Event Report (LER) Number consists of the three-digit NRC Docket Number for the plant at which the event occurred, the last two digits of the year in which the LER was generated, and a three-digit sequential number of the LER. This is consistent with the NRC's Sequence Code Search System (SCSS) database designation. The LER system is described in 10 CFR 50.73.

2. Events in the table are classified as follows:

   a. Failures of AOVs or events resulting from contaminated (not clean and dry) air;
   b. Failures of AOVs or events resulting from solenoid failures; and/or
   c. Failures of AOVs or events resulting from design deficiencies, material failure, or insufficient margin.

Table 3. Selected events documented by NRC Licensee Event Reports (LERs) involving air-operated dampers or ventilation components that are powered from air systems.

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Browns Ferry 2	26095008	951004	The inboard and outboard reactor zone isolation dampers failed to close during a weekly routine switch of operating equipment, resulting in the loss of secondary containment. Six SOVs serving six valve operators and dampers were involved. The cause was traced to failure of the solenoid valves which control the dampers. The SOVs failed because of the presence on a black sticky residue (adhesive) that caused the solenoid core to stick to the plugnut. The source of the residue and its composition were under investigation by ASCO when the LER was transmitted to the NRC. Additional SOV failures were discovered after the LER was first transmitted. Browns Ferry 3 LER 26992003, which refers to failures of the outboard containment exhaust damper to close due to unknown-cause SOV failure, is referred to in this LER. The LER indicated that the event was not reportable to the industry's database. This is considered to be a common-cause failure condition.	b, c	Containment
Calvert Cliffs 1	31789018	891106	A condition was discovered that could have prevented the fulfillment of certain systems to remove residual heat and control the release of radioactive material after a Loss of Coolant Accident. During the performance of a test to satisfy specific recommendations in Generic Letter 88-14, it was discovered that many air-operated control valves and piston-operated ventilation dampers which utilize safety-related air accumulators would not have performed as expected after a loss of normal non-safety-related instrument air. The root cause of the event was identified as a lack of an adequate documented design basis combined with inadequacies in the testing and preventative maintenance program for the Instrument Air System. This is considered to be a common-cause failure condition.	c	Instrument Air

Table 3. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Cook 1	31597023 (The event covered by this LER was originally reported as Event No. 32939)	970916	A design change to the bypass dampers for the Engineered Safeguards Features (ESF) ventilation system installed between December 1996, and August 1997, introduced the possibility of a single failure which could result in the loss of both trains of the ESF ventilation system. The loss of the 85 psig air header without concurrent loss of the 20 psig air header would result in the ESF ventilation trains being unable to meet their design function. The ESF vent system charcoal inlet and bypass dampers originally both used a 20 psig air header. The charcoal bypass dampers were normally open and were intended to fail closed. The charcoal inlet dampers were normally closed and intended to fail open. The licensee installed new bypass dampers, which required higher pressure to operate and were therefore transferred to the 85 psig air header. If 85 psig air was lost, the bypass dampers would reposition to the closed position. The inlet dampers would remain closed and this would result in dead-heading of the filter trains and subsequent loss of cooling to ECCS equipment. The root cause of the event is the failure of the design change process to identify the potential adverse impact on the ESF ventilation system created by the modification of the control air supply to the bypass dampers. This is considered to be a common-cause failure condition.	c	ESF Ventilation
Fitzpatrick	33389004	890309	Engineering identified a design deficiency which originated during plant construction. The deficiency would result in loss of area cooling for parts of both safety divisions of safety-related and non-safety-related electrical distribution systems as a result of loss of instrument air to the cooling system temperature control valves. The event was caused by an error during design and construction of the plant. Review of the original design and procurement documentation indicates that the designer selected, from the vendor specification sheet, a valve described as a fail-open valve. It appears that the vendor specification sheet used by the designer contained an error or the designer misinterpreted the specification sheet. The actual valves ordered and installed were of a fail-closed design. A corresponding 10 CFR Part 21 report was not found regarding this condition.	c	Instrument Air Electrical Cooling System

Table 3. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Fort Calhoun	28587025	870923	Diesel Generator #2 automatically shut down on high coolant temperature due to inoperability of the air operated exhaust damper for the diesel radiator. Approximately 14 minutes into a test, DG-2 automatically shut down due to high coolant temperature. The air operated exhaust damper for the diesel generator radiator may not have fully opened automatically as designed when the diesel was running, thus restricting the required air flow through the radiator. The cause of the damper malfunction was postulated to be the presence of residue (see below) causing the pilot valve that directs air flow to sometimes stick. Previously (6/87), water had been introduced into the instrument air system during surveillance test on the fire protection system dry pipe valve for the diesel generator rooms (refer to LER 28587033 in Table 2). An extensive program was undertaken blowdown air operated devices including air operated valves and to cycle those valves as allowed during power operation. After the trip of DG-2, the pilot valve was inspected and cleaned and the accumulator drained. Similar actions were taken for DG-1. This was an ASP Program Event. See Appendix B. Note that the contamination of the air system caused AOV failures after cleanup and after some time had passed. This is considered to be a common-cause failure condition, based on contamination of the instrument air system.	a	Diesel Generator Cooling System
Indian Point 2	24788017	881102	An engineering analysis of the capability of the ventilation system to maintain the EDG building internal temperature at or below the continuous rated temperature of the electronic equipment associated with EDG operation was conducted with unsatisfactory results. The ventilation system is pneumatic operated (i.e., the intake and exhaust dampers are operated by pneumatic actuators and the fans are energized via pneumatic switches). It was postulated that if one of two pressure regulator valves in the supply line from the safety-related air supply to the pneumatic controls failed, it could prevent the ventilation system from operating as intended. This is considered to be a common-cause failure condition.	c	EDG Ventilation System

Table 3. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Indian Point 3	28693045	931023	An investigation identified the loss of Instrument Air as a failure mode of the Central Control Room ventilation system damper actuators that would result in loss of system function. Loss of this function is a condition prohibited by technical specifications. This event was caused by personnel error of an indeterminate origin during original design. The architect engineer did not evaluate this failure mode during initial design or when upgrading the system from a non safety system to a safety system prior to initial operation. This is considered to be a common-cause failure condition.	c	Central Control Room (CCR) Ventilation System
Indian Point 3	28696002	960120	During operator actions to secure Emergency Diesel Generator (EDG) 32, room ventilation did not operate as required. EDG 32 was declared inoperable resulting in a condition prohibited by technical specifications (i.e., two inoperable EDGs). The EDG 32 room ventilation malfunction was caused by debris in the pressure regulator used to supply air to the ventilation control system. The regulator was disassembled and debris was observed, particularly in the valve seating area. Also the valve/stem assembly was found to be loose. The engineers attributed the debris to internal scaling of the carbon steel piping which is connected to the inlet of the regulators. On January 22, 1996, a Deviation Event Report (DER) was issued for the pressure regulator of EDG 33 room ventilation system because it required constant adjustments in order to maintain the output to 100 psig. The regulator was replaced and the old regulator disassembled. The engineers observed debris in the valve seating area and the valve/stem assembly was disengaged from the diaphragm. The EDG 32 regulator that had been replaced on January 21, 1996, was removed and bench tested. The test showed that the replacement regulator for EDG 32 was unable to maintain pressure. Upon disassembly, the replacement regulator for the EDG 32 valve/stem assembly was found disconnected but with no debris and it was replaced with another new regulator. A failure analysis was to be performed on the failed EDG 32 regulator but the results are unknown. The EDG 31 regulator was removed, tested and inspected. Although the regulator maintained air pressure, debris was noted in the internals and the valve/stem was loose. The pressure regulator for EDG 31 was replaced. (continued on next page)	a	Emergency Diesel Generators

Table 3. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Indian Point 3	28696002 (continued)	960120	The regulators are commercial grade and dedicated for Category I service. New York Power Authority (NYPA) reported the loose valve/stem assembly to the manufacturer. NYPA replaced the carbon steel piping from the starting air receiver tank to the regulator, added a filter to the inlet of the regulators, and added a drip leg to piping upstream of the regulators to help trap any debris before it enters the regulator. This is considered to be a common-cause failure condition.	a	Emergency Diesel Generators
Indian Point 3	28693013	930414	While performing System Operating Procedure SOP-CB-10, "Fan Cooler Unit Operations," it was observed that closing the dampers via their respective Central Control Room (CCR) control switches did not result in actuation of the close indication lamps for A and B dampers on 32, 34 and 35 Fan Cooler Units. A mechanic reported that the damper linkages and bearing housings were dirty for all dampers. The pins connecting the linkages were rusty and paint chips were noted on the arms where the linkages connect to the louvers. The rust was cleaned from the pins and dirt and rust was cleaned from the linkages. The linkage arms and pistons were lubricated with Mobil AW-2 Grease. The dampers were operated from the Control Room and found to perform as intended. This is considered to be a common-cause failure condition.	a	Central Control Room (CCR) Ventilation System
Indian Point 3	28693036	930915	A Nuclear Regulatory Commission inspector posed questions regarding the stop nuts installed on damper (DMP) actuators in the Central Control Room (CCR) ventilation system. The inspector also questioned the proper mounting of damper actuators. The Heating, Ventilating and Air Conditioning (HVAC) system engineer produced documentation which indicated that the damper actuators were inadequately mounted to meet seismic qualification. The CCR HVAC system design requires that the system withstand seismic events. The system engineer determined that the inadequate seismic mounting of the damper actuators had existed since initial criticality, which took place on April 6, 1976. This is considered to be a common-cause failure condition.	c	Central Control Room (CCR) Ventilation System

Table 3. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Fermi 2	34197007	970325	One of the two operating Reactor/Auxiliary Building Ventilation System (RBHVAC) exhaust fan discharge dampers closed, resulting in high reactor building pressure and a loss of secondary containment integrity. The cause was age and service related mechanical failure of the SOV that controls opening air to the discharge damper for the exhaust fan. The exhaust port seating material on the SOV was completely disintegrated and allowed air to leak through the exhaust port.	b	Reactor Building Ventilation System
LaSalle 1	37389007	890210	Review of the Control Room HVAC System drawings identified one potential situation where an undetected single failure (a hot short circuit) could result in a failure of the normally closed, fail closed, maximum exhaust air isolation damper, OVC14YA/B in the open position on the operating train. The postulated hot short circuit would cause the damper to move to the fully open position and the Control Room Operators would have no direct indication (e.g., alarm or valve position indication) of the fact that the damper was open. Under this set of circumstances, it is possible that smoke or radioactivity could be drawn into the Control Room in spite of the initiation of the emergency make-up train. The reviewers concluded that the postulated sustained hot short was beyond the design requirements of LaSalle County Station.	b	Control Room HVAC System
LaSalle 1	37397046 and 37398007 (Event No. 33434 also referred.)	971216 and 980319	(LER 37398007) An investigation of the allowable closure time assumptions for the turbine building high energy line break (HELB) check dampers (#1/2VT79YA/B/C) led the licensee to reconsider the original builders assumptions and calculations for the reactor building exhaust isolation dampers (1VR05YA and B). (LER 37397046) In the event of a Main Steam High Energy Line Break, the Turbine Building Ventilation isolation dampers, 1(2)VT079YA, B & C, would not close fast enough to prevent the pressure from exceeding the pressure retaining capability of the walls, floors, and ceilings that separates the VT exhaust tunnel from the safety related High Pressure Core Spray (HPCS) electrical switchgear room. The apparent causes included an invalid calculation assumption dating from the original design. Dampers 1VR05YA and B had malfunctioned and were rebuilt in 1985 (see LER 37385008 and LER 37385011). The licensee indicated that they revised the calculations and assumed a 0.4 second instrument time delay, and a 0.075 second solenoid valve response time. The plant was licensed in 1982.	b, c	Auxiliary Building

Table 3. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
LaSalle 2	37489018	891216	The 250V battery was declared inoperable due to low battery electrolyte temperatures. Since this battery serves as a power supply to the Reactor Core Isolation Cooling system, the RCIC system was also declared inoperable. The cause of this event was the failure of two Division I Switchgear Heat Removal System damper actuators coupled with sub-zero outside air temperatures. The two dampers involved failed to fully close, resulting in sub-zero outside air admission to the switchgear ventilation system.	c	Division I Battery Room Heat Removal System
Millstone 3	42393002	930207	The A Train Control Room Pressurization System failed its 18-month pressurization surveillance test. On February 6th the B Train initially failed its surveillance test. Two common-cause failures were identified by the licensee in the LER, and both trains were declared inoperable pending investigation. The first potential common-cause failure was freezing of the pressure regulating valve caused by moisture in the air banks. The second common-cause failure was pressure oscillations within the control room envelope which caused the differential pressure to periodically fall below the acceptable 0.125 in. Hg. for brief periods of time. The air banks were purged and refilled with dry air to reduce the dew point. Samples of the air in the air bottles of both trains tested at dew points around minus 40°F at atmospheric pressure, which corresponds to approximately 70°F at 2250 psig. Excess moisture could have entered the system due to improper blowing down of the condensate traps or from purifying cartridges which exceeded their useful life. The system design did not include drying capabilities other than the moisture removal capacity of the compressors.	c	Control Room Pressurization System

Table 3. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Prairie Island 1	28295013	950927	During review of the effects of the loss of instrument air and the resultant plant response, it was determined that the control room chilled water system would not function without instrument air (a non-safety-related system). With the chilled water system not functional, the temperature in the control room and relay room would exceed equipment qualification temperatures in approximately 30 minutes unless operator action was taken. Original plant design took credit for instrument air availability after the onset of an event. The instrument air system was assumed to be operable because it is powered by the safeguards diesel generators. Pre-operational testing of the instrument air system demonstrated that all valves fail to their safe position, but did not consider the integrated plant response upon the loss of instrument air. Also, the need for the control room chilled water system to function as essential support equipment was just recently identified. This was reported in 1995. This condition had existed since 1973 when the plant commenced power operation.	c	Room Chilled Water System Instrument Air
Quad Cities 1	25492028	921016	During a review of a Standby Gas Treatment (SBGT) Tech Spec revision, it was revealed that on a loss of instrument air (IA) the Control Room Dose would exceed General Design Criteria (GDC)-19 limits. This conclusion was made assuming that the pneumatically controlled SBGT heaters would fail to start, the air operated flow control valve would fail open, flow would increase to 5100 cfm, and the Control Room Air Filtration Unit (CR AFU) is started in 110 minutes. The apparent cause of the event was an inadequate design for the Standby Gas Treatment System. The pneumatic flow instruments and the Flow Control Valves rely on the non-safety-related Instrument Air system, to function properly. The operation of these components upon a loss of instrument air had not been thoroughly evaluated. The condition was discovered in 1992. This condition had existed since 1972 when the plant commenced commercial power operation. This is considered to be a common-cause failure condition.	c	Control Room

Table 3. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
River Bend	45889024	890406	It was observed that six solenoid operated valves (SOVs) were oriented to isolate normal air flow into the safety-related instrument air system (IAS) accumulator. Since these SOVs are directional, they may not have maintained air in the accumulators as required when in the closed position. This is considered to be a common-cause failure condition.	b, c	Control Building HVAC Auxiliary Building HVAC Fuel Building HVAC
Salem 2	31197002 (Event No. 31783 refers)	970213	Four pressure switches (ABS/PDS) on Salem Unit 2 could not be verified as seismically qualified. In addition, one of the four non-seismically qualified pressure switches was found to have its sensing lines reversed. The pressure switches control the position of backdraft dampers that are boundaries for environmentally isolated contiguous zones which are vented to the atmosphere. Failure of pressure switches could cause failure of damper operators and thus could subject equipment to a harsh environment from steam. In addition, because one of the pressure switches had its sensing lines reversed, its respective isolation damper would not have closed in response to a HELB. The pressure switches were replaced in 1986 under a routine maintenance recurring task. The original pressure switches were seismically qualified but the replacement pressure switches were not. The reason for replacement with non-seismic pressure switches was unknown. The sensing lines for the mechanical penetration area pressure switch were reversed by design in 1978. The design change incorrectly assumed the mechanical penetration air supply duct would be exposed to high energy as a result of a HELB; however, the HELB analysis identified that the mechanical penetration area would be the high energy area.	b, c	Auxiliary Building Ventilation

Table 3. (continued).

Plant	LER Number[1]	Event Date	Description	Classification[2]	System
Summer	39587019	870730	The Licensee identified a design deficiency in the instrument air system which could cause loss of control room ventilation on a loss of instrument air. The air supply to each inlet damper actuator is provided with an air accumulator and check valve to ensure that the dampers remain in the open position to perform their safety-related function in the event of a loss of the instrument air system. While evaluating the system design, it was determined that the instrument air supply check valve for each train of outside air inlet dampers was not designed to provide a leak tight seat under the system air pressure. The event was caused by two improperly designed instrument air check valves installed in the IA system which may not have seated properly at the line pressure within the instrument air system.	c	Instrument Air Control Room Ventilation

Notes:

1. The NRC Licensee Event Report (LER) Number consists of the three-digit NRC Docket Number for the plant at which the event occurred, the last two digits of the year in which the LER was generated, and a three-digit sequential number of the LER. This is consistent with the NRC's Sequence Code Search System (SCSS) database designation. The LER system is described in 10 CFR 50.73.
2. Events in the table are classified as follows:
   a. Failures of AOVs or events resulting from contaminated (not clean and dry) air;
   b. Failures of AOVs or events resulting from solenoid failures; and/or
   c. Failures of AOVs or events resulting from design deficiencies, material failure, or insufficient margin.

Table 4. Selected events described in reports other than Licensee Event Reports..

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Beaver Valley 1	Event No. 34118	980422	Design Change Package (DCP) 2313 was issued for installation of a backup nitrogen supply for PORV PCV-456 after discovering that none existed even though required by (at least) 1982 Updated FSAR commitments, although PCV-456 is not assigned to mitigation of LTOP events. PORVs PCV-455C and -455D had been provided with nitrogen backup and are assigned to mitigation of LTOP events. On March 26, 1998, and again on April 7, 1998, it was discovered that: (1) The installation of the existing tubing to the air regulators associated with PORVs PCV-455C & -455D was too rigid in that it did not allow for sufficient thermal expansion/contraction and seismic movement associated with stroking the PORV under all design conditions; (2) The mounting plate for the Solenoid Operated Valves (SOVs) on the PORVs was inadequately designed; (3) The SOVs were not specifically qualified for excitation resulting from PORV discharge; (4) The existing Nitrogen tubing was field routed and required a level of rework to reestablish seismic adequacy; and 5) The PORV actuators were not adequately protected from a failure of the instrument pressure regulators in that the PORVs could fail in a nonconservative manner by opening in a faster time than what was evaluated in the system transient pipe loading analysis when the Reactor Coolant System (RCS) is at operating temperature and pressure. The licensee stated that the apparent cause of these events was loss of design and licensing basis configuration control which subsequently led to inadequate installation, application and implementation of TS Surveillance Requirements. (LER 33498011 was subsequently issued to cover this event and following developments described herein.)	c	Pressurizer

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Big Rock Point	Event No. 32226	970425	Vent Valve on Control Rod Drive (CRD) System could lift under high reactor pressure condition. Based on a review prompted by NRC Information Notice 96-68, "Incorrect Effective Diaphragm Area Values in Vendor Manual Result in Potential Failure of Pneumatic Diaphragm Actuators," it was discovered that the vent valve on the CRD system (CV-NC11A) would not remain closed on a system pressure in excess of 1585 psig. This determination was based on control air pressure supplied to the valve and the size of the air diaphragm in the valve. The licensee was to modify the system (no details provided) to ensure that the air-operated valve remained closed. No LER was found for this condition. This is considered to be a common-cause failure condition.	c	Control Rod Drive
Cook	Event No. 33832	980304	Pressurizer PORV 2-NRV-152 could have been rendered inoperable because of a leaking air system check valve. It was concluded that the emergency backup air supply for the PORV could have been inoperable by a leaking air system check valve. Details of the time that the check valve was leaking were not provided. An LER could not be found to cover this condition.	c	Pressurizer
Fermi 2	Failure Evaluation of ASCO Solenoid Valves Related to DERs 97-1202, and 97-1200	970917	Analysis of failed solenoid valves which control safety-related AOVs (18 failures in a population of 66 SOVs) indicated problems with lubricants and thread-locking compounds that caused the valves to stick (the core adheres to the wall). The Fermi 2 engineers prepared a comprehensive SOV failure analysis report as part of the resolution of the DER. The failure analysis report included what they refer to as a common-mode failure investigation. The conclusions were: - Sufficient material controls and use instructions were not in place to reduce the possibility of contamination in the plant's pneumatic system. - Definition and understanding was needed for the various licensee-defined categories of materials (in terms of use restrictions) in the plant. (continued on next page)	a,b,c	Various Safety-related Systems

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Fermi 2	Failure Evaluation of ASCO Solenoid Valves Related to DERs 97-1202, and 97-1200 (continued)	970917	- NUREG-1275, Volume 6 findings and recommendations were not thoroughly reviewed for applicability at Fermi 2 and compared to their (then) current practices. - The corrective and preventive maintenance processes did not adequately use actual equipment performance data for SOVs when determining the necessary preventive maintenance schedules. (This was noted to be a problem for other components, such as pumps, valves, and heat exchangers, as well.) This is considered to be a common-cause failure condition. In the SOV failure analysis report, it was also noted that some SOVs were supplied with 125 or 130 volt AC power over their service lifetimes, although the equipment qualification program assumed 120 volt AC power. The report called for further investigation of this consideration.	a,b,c	Various Safety-related Systems
Fermi 2	DER 93-0045	930119	Moisture in the air system (flasks) resulted in corrosion contamination and subsequently caused valve problems.		
Fermi 2	DER 94-0568	941015	Precoat valves were found to be installed backwards since the plant started operations. During investigations that followed, it was found that the air-to-open, spring-to-close valve actuators (G4100F210A and B) were undersized.	a	Various Safety-related and Non-safety-related Systems
Fermi 2	DER 94-0406	940822	The DER referred to Fisher Information Notices (FIN) on valve problems. FIN 94-02 and Supplement 1 notified users of piston rod extension problems. FIN 94-03 regarding Fisher 3570 positioners with Fisher 472 or 473 spring return actuators indicated problems with return to the fail-safe position. FIN 94-04 identified problems with 18 models of piston actuated AOVs. Fermi 2 had 8 valves of these type models. This is considered to be a common-cause failure condition.	c	Various Safety-related and Non-safety-related System

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Hope Creek	Event No. 32836	970828	During a review of the Safety Auxiliaries Cooling System (SACS) design basis, PSE&G identified a potential design deficiency associated with the safety-related Control Area Chilled Water System chiller units that could have prevented the fulfillment of a safety function. In the event of an accident resulting in a loss of instrument air, concurrent with low Ultimate Heat Sink temperature, all of the safety-related chillers could trip. Specifically, the SACS control valves which regulate SACS cooling flow through the chillers are designed to fail open upon loss of (non-safety-related) instrument air. These valves control room cooling for the Emergency Diesel Generators and the Emergency Core Cooling Systems. PSE&G was to evaluate actions to correct the condition as part of its corrective action program. Corrective actions included an operability determination, revising the engineering evaluation, and evaluating and implementing a design change. This condition had existed since initial plant operation (1986). The cause of the failure to recognize and correct this condition was attributed to human error in the original design and in several design reviews. LER 35497020 also referred to this event. This is considered to be a common-cause failure condition.	c	Safety Auxiliaries Cooling
Indian Point 2	Morning Report No. 1-97-0011	970203	The plant shut down due to a feedwater regulating valve failure. Main feedwater regulating valves were stuck open due to serious galling in the cages and plugs. The system was contaminated by abrasive grit used to clean the turbines. This is considered to be a common-cause failure condition.	c	Main Feedwater

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Indian Point 2	Event No. 34103	980420	The available nitrogen supply is insufficient to meet the requirements of the over-pressure protection system design basis. The licensee determined that the available supply of nitrogen may not meet the OPS system design basis of 200 operating cycles in 10 minutes. Additional nitrogen supplies were provided. No LER could be found for this event. The event was not directly related to AOV failures or deteriorated conditions, but was included as an example of inadequate design of a pneumatic supply system that was discovered recently.	c	Various Safety-related and Non-safety-related System
Indian Point 3	Event No. 26449	931202	All three emergency diesel generators (EDGs) were inoperable for approximately four hours. (Originally, the diesels were reported inoperable for approximately five minutes but this was later corrected to four hours.) The solenoids on AOVs used to control the EDG service water effluent for all three EDGs (two parallel valves in a common header line) were replaced and then tested as part of a normal work package on a safety-related system. The valves were stroke-time tested after the solenoids were installed and then energized. The solenoids were incorrectly connected and that the valves would, therefore, not stroke open. LER 28693053 (see the description in Table 2) also refers to this event. See Millstone 2 Event No. 26187 for a similar event. This is considered to be a common-cause failure condition.	b, c	Emergency Diesel Generators

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Limerick 2	Event No. 31739	971206	Six stuck steam flooding HVAC dampers were caused by sticking solenoids. Excessive friction in either the solenoid or the associated bushing appears to be the cause of the failures. The dampers are located in 3 ventilation duct penetrations in the walls surrounding the outboard MSIV room. Each wall penetration has two steam flooding HVAC dampers arranged in series. The dampers are required to close following a HELB in the outboard MSIV room to protect equipment in the adjoining room. The licensee had no analysis for an HELB with steam entering the adjoining room. The adjoining room contains safety-related equipment such as the scram solenoids, scram hydraulic control units, and several ECCS MOV electrical breakers. Shutdown capability could have been adversely affected. The licensee had not determined temperature, pressure, or humidity to be expected from the scenario. This is considered to be a common-cause failure condition. An LER was not submitted for this event.	b, c, d	MSIVs
Millstone 1	Event No. 31910	970307	The emergency service water system was declared inoperable because of potential failure of a non-qualified pressure regulator in the air system that supplies air to the SOVs that actuate the blowdown valve for the ESW strainers. The failure of this regulator would subject the SOVs, which are rated at 85 (psig) to full air system pressure at 100 (psig). Failure of the SOVs could prevent the ESW strainer blowdown valves from opening thus preventing fulfillment of the ESW safety function. Both trains of the ESW system are affected because the regulator provides a common air supply to both SOVs. Other SOVs were to have been investigated for the same or similar problem. This is considered to be a common-cause failure condition.	b, c	Emergency Service Water

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Millstone 2	Event No. 26187	931007	On October 7, 1993, at 1830 hours, with the unit in Mode 4 (Hot Shutdown), the 'B' emergency diesel generator (EDG) was declared inoperable because the solenoid valve 2-DG-95B had a lower rated differential pressure than that produced by the air start system pressure. The "A" EDG was also inoperable at this time due to a failed surveillance test. The solenoid valves 2-DG-96A and 2-DG-96B were ASCO Model 206-381-2RF, rated for a MOPD of 200 psid. These valves were evaluated as adequate for their intended design since they are the equivalent to commercial solenoid valves rated at 300 psid. The nuclear grade solenoid model was derated for the purposes of satisfying the more rigorous seismic requirements. The "B" diesel generator solenoid valve 2-DG-95B was an ASCO Model 206-381-3RF with a differential pressure rating of 150 psid which is below the air pressure it has to work under. The equivalent commercial solenoid for the Model 206-381-3RF has a pressure differential rating of 200 psid. The solenoids were replaced and the EDGs were returned to service about 19 hours later. LER 33693011 refers. See Indian Point 3 Event No. 26449 for a similar event.	b, c	Emergency Diesel Generators

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Oconee 1, 2, and 3	NRC Inspection Report 50-269/99-10, 50-270/99-10, and 50-287/99-10. (Refer to NRC Accession Number 9902090072)	Inspections on Nov. 2-6 and 16–20, 1998, and Jan. 11–15, 1999	The purpose of these inspections was to follow up on an open item regarding potential emergency feedwater (EFW) system design vulnerabilities. The discussion of the design and operation of C-187, a 12-inch AOV in one of the parallel 20-inch lines from the upper surge tank (UST) to the condenser hotwell was of particular interest in this study of AOVs. In 1989, the licensee upgraded the EFW seismic resistance capabilities. As part of that upgrade, normally closed AOV C-187 was declared safety-related and became the single EFW boundary. The licensee incorrectly left AOV C-187 designed to open on a low condenser hotwell level (that would result from a break in any of the non-seismic pipes connected to the hotwell or a break in the main feedwater line) and dump the UST water to the condenser hotwell, thus failing the EFW system. In 1993 and 1994, the licensee modified the controls for AOV C-187 to automatically close at a low UST level; however, the EFW was still left vulnerable to a single failure of the single solenoid valve actuator for C-187. Also, AOV C-187 was still relied upon as a single EFW seismic boundary valve. The licensee's probabilistic risk assessment recognized that failure of AOV C-187 was one of the top contributors to a potential EFW system failure. The PRA stated: "If a main feed line break is assumed, the UST could be drained into the hotwell, thereby failing EFW's initial suction source." The issue remained open as of the publication of this report.	c	Emergency Feedwater

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Oyster Creek (also Browns Ferry, Fitzpatrick, Monticello, Peach Bottom, and Quad Cities)	Event No. 32196 Event No. 32240, Part 21 from ASCO, applies See Morning Report Nos: H-97-0065, dated 6/4/97 and 1-97-0031, dated 4/29/97 E-Mail from David Skeen, dated 5/6/97, 12:35 pm	970422 and 970429	Thirteen of 36 scram solenoid pilot valves (SSPVs) at Oyster Creek exhibited excessive air leakage during performance maintenance testing. The leakage resulted from hardening of the core disks in the valve pilot heads. ASCO later issued a Part 21 report which indicated that nearly 1000 units were fabricated with Buna-N (nitrile) material rather than the specified nuclear grade Viton (fluorocarbon) material. GE indicated that the air leakage would cause control rods to drift closed but blockage of the SSPVs was not considered to be a credible event, and backup scram valves are available. No LERs were found that cover these events; however, LER 27198025 and NRC IN 94-71 refer to similar problems at Vermont Yankee and WNP-2. Event #27149 refers to a similar event at Pilgrim. Contrary to GE's analysis in the cases of Pilgrim/WNP-2, the licensees indicated failures that did result is blockage of SSPVs that could have rod insertion. The plants and number of SSPVs affected were: Browns Ferry - 5 Fitzpatrick - 51 Monticello - 260 Oyster Creek - 300 Peach Bottom - 10 Quad Cities - 372 This is considered to be a common-cause failure condition.	b, c	Scram

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Palisades	Letter from P. Flenner, Consumers Energy, to R. Schaff, USNRC, dated 4/23/97, with attachments Includes Condition Report C-PAL-97-0404	970315	The letter includes condition reports that describe corrosion in air lines which led to degradation or failure of multiple air regulators. This was considered to be a common-cause failure event caused by a high pressure air system that did not meet the required cleanliness criteria, i.e., the system had been contaminated with moisture and resulting corrosion products. The conditions were not reported in an LER. NRC/AEOD was informed of the situation in an event (overnight) report. The attachments to the letter include the licensee's rationale for not reporting the occurrence. See Section 8.3.1 of this report.	a	Various Safety-related and Non-safety-related Systems
Palisades	Palisades Engineering Analysis EA-AOVSYS-ESS-01, Revision 3, dated 9/11/97	970911	This entry in the specified report refers to Design Basis Review (calculations and specifications) for AOVs in the Engineered Safeguards System (ESS). These calculations refer to the CV-3025 AOV discussed in Section 8.3.2 of this study. The CV-3025 AOV has a design basis function to open during entry into the shutdown cooling mode after a small-break LOCA. This valve has a function to be throttled to adjust cooling flow after a small-break LOCA, but the licensee considers this to be beyond the design basis. See LER 25578003 and LER 25581030 in Table 2 for a description of the conditions and events involving the CV-3025 AOV.	NA	Engineered Safeguards and Shutdown Cooling

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Palisades	Event No. 33843	980305	The original design bases, and subsequent design reviews, for the HP Air System address the Large Break Loss of Coolant Accident (LBLOCA), yet did not address, in detail, the SBLOCA. HP Air System testing demonstrated that the HP Air receivers can supply sufficient air for one hour to operate the control valves required to realign ECCS to the recirculation mode when the Recirculation Actuation Signal (RAS) occurs. This bounds the expected time frame for RAS initiation following LBLOCA events. However, SBLOCA events can be postulated, wherein RAS initiation occurs beyond the one hour time frame. The HP Air System compressors must be returned to service within one hour for postulated events where RAS initiation occurs more than one hour into the event. LER 25598006, described in Table 2, was subsequently issued to cover this event.	c	Various Safety-related and Non-safety-related System
Pilgrim	Event No. 33360 (LERs 29397025 and 29397026 were later identified as referring to this event, as noted in the Description column)	971206	(LER 29397025) On November 23, 1997, at 2215 hours, a shutdown was completed as required by Technical Specifications, because two main steam isolation valves (MSIVs) in separate main steam lines were inoperable. The direct cause of the failure of the two MSIVs to close was the relaxation of the closing springs. Corrective action taken for the two MSIVs included replacement of the closing springs and overhaul of the actuators. The cause for the failure of MSIVs AO-203-2B and -1C to close via their push button was main closure spring relaxation leading to loss of closure force at the end of the closure stroke. A contributing cause of the failure of MSIV AO-203-1C to fully close was increased friction between the spring plate and actuator stanchions. This is considered to be a common-cause failure condition. (LER 29397026) During power ascension from the shutdown described above, a scram occurred as the result of a turbine trip due to high reactor water level. The scram was caused by the failure of the "A" feedwater regulating valve (Copes-Vulcan D100-160 diaphragm operated AOV) in the full open position. This was caused by the misalignment of a pilot valve clip in the positioner when the positioner was opened during the forced outage to fix the MSIVs.	c	MSIVs, Main Feedwater

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Valcor Engineer'g Corp. (and Susquehanna)	Event No. 34262 See also Susquehanna Condition Report No. 98-2296	980519	This event report describes a 10 CFR Part 21 report (1998-43-1) of a deficiency in Valcor Model V70900-65-11 solenoid-operated pilot valves. Following 6 to 18 months of continuous energized service, 3 Valcor Model V70900-65-11 air pilot valves failed to stroke closed in service immediately upon de-energization. Delays in closing ranged from one to five minutes. Despite their best efforts, the vendor reported that closing delays have not been replicated outside of the plant systems. The vendor reported that a potential cause of the delayed closing was the possible susceptibility of this particular model to the effects of residual magnetism. Delayed closing could occur when the air gap between the plunger and the stop becomes too small or if the plunger makes contact with the stop. Extended periods of continuously energized service and subsequent compression setting of the O-ring seat may induce a condition in which the air gap is too small. As a result of the small air gap, the valve may then be subject to residual magnetism effects, which would tend to prevent closure of the valve when it is de-energized. Opening operation of the valve was not affected. (continued on next page)	c	Scram
Valcor Engineer'g Corp. (and Susquehanna)	Event No. 34262 See also Susquehanna Condition Report No. 98-2296 (continued)	980519	Further bench testing found additional similar failures. The licensee's failure analysis projected that, in the as-found condition, 59% of the installed valves would fail to perform their design function before the end of their first cycle of operation in the plant (18 months). Forty-six units of this particular model have been delivered, all to Susquehanna. Similar valves sent to Duke Power had the potential for similar failures. All of the valves have been, or were scheduled to be repaired by Valcor. An LER was not found for this condition. This is considered to be a common-cause failure condition.	c	Scram

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Vermont Yankee (10 CFR Part 21 from AV)	Event No. 32253	970425	Vermont Yankee informed the Automatic Valve Co. (THE AUTOMATIC VALVE CO. IS DESIGNATED BY THEMSELVES AS AV, AND IS NOT AVCO AS ERRONEOUSLY NOTED IN MUCH OF THE RELATED CORRESPONDENCE), in September 1996, that a scram solenoid pilot valve (SSPV) was making a "buzzing" noise. On April 24, 1997, during scram time testing, the control rod associated with the noisy SSPV had a slower (but within Tech Spec limits) time than the other control rods being tested. AV determined that the cause of the slow SSPV was the accumulation of debris from the stainless steel plunger resulting from the plunger cycling at 60 Hz for some time. Poor fit and finish between the plunger and guide, and very small tolerances, were thought to be root causes. AV supplied about 90 SSPVs to Vermont Yankee, and Vermont Yankee is the only plant with AV SSPVs. AV supplied SOVs for BWR MSIVs, which they described as similar to the SSPVs, to as many as 25 U.S. nuclear power plants. A search of the SCSS database using the words "scram solenoid" produced 62 LER records, mostly related to slow rod insertion times or failures to insert.	b, c	Scram
Waterford 3	Letter WF3-97-0107 dated May 6, 1997 (NRC Accession Number 9705120329)	961123	NRC Inspection Report 50-382/96-24 identified an unresolved item concerning the closed safety function of certain containment isolation valves that receive an open ESF actuation signal and subsequently fail to open on loss of air. Containment isolation criteria required that containment integrity be maintained for at least 30 days post-accident. Some containment isolation valves are air operated and fail open on loss of air. Since the IA system, which supplies the AOVs is non-safety-related, it cannot be relied upon post-accident. The AOVs were supplied from safety-related air accumulators but the accumulators did not have sufficient capacity to ensure that the AOVs would remain closed for 30 days. Thirteen AOVs were of this type. One other AOV, CVC-209, charging system outside isolation valve, did not have an accumulator nor was it equipped with a Class 1E SOV.	c	Containment

Table 4. (continued).

Plant or Vendor	Event Report Number and Other Related Designations[1]	Event Date	Description	Classification[2]	System
Waterford 3	Letter WF3-97-0107 dated May 6, 1997 (NRC Accession Number 9705120329)	961123	A backup nitrogen supply was to have been provided. No LER could be found for this condition. This is considered to be a common-cause failure condition.	c	Containment
Waterford 3	Event No. 34237 and MR Number 4-98-0028 that updated the Event Report LER 38298010 also refers to this event	980514	Failure of the main nitrogen regulating valve, NG147, a non-safety-related valve, could have affected both trains of several safety-related systems. The configuration of the nitrogen system includes a pressure regulator installed at the outlet of the liquid nitrogen tank which reduces pressure from 2300 to 750 psig at a flow rate of 879 scfm. The design system pressure rating downstream of the pressure regulator was 800 psig with a relief valve installed at a setpoint of 940 psig, at a flow rate capacity of 170 scfm. The nitrogen system provides motive force for operation of valves in the ECCS, EFW, and other systems. Failure of the pressure regulator could have over pressurized the downstream piping and thus, could have either interrupted pneumatic power to the AOVs if the piping failed, or result in damage to the AOVs if the piping did not fail. This is considered to be a common-cause failure condition. The licensee addressed this condition by opening a normally shut manual isolation valve to place a relief valve with a capacity of 1419 scfm in service, thus providing a relieving capacity of 1419 + 170 = 1589 scfm, which is greater than the 879 scfm capacity of the pressure regulator.	c	ECCS, MFW

a. The NRC numbers are 5-digit sequential numbers assigned by the NRC. These events are usually reported in response to the requirements described in 10 CFR 50.72.

b Events in the table are classified as follows:

a. Failures of AOVs or events resulting from contaminated (not clean and dry) air;
b. Failures of AOVs or events resulting from solenoid failures;
c. Failures of AOVs or events resulting from design deficiencies, material failure, or insufficient margin; and/or
d. Failure of AO damper(s) or events related to damper failures..

Table 5. Licensee Event Reports cited in Tables 2, 3, and 7 listed in numerical order.

Plant	LER Number[a]	Event Date	Classification[b]	System
Haddam Neck	21393005	930518	a, b, c	Instrument Air Reactor Coolant Letdown
Haddam Neck	21393007	930525	a, c	Instrument Air Pressurizer Pilot-Operated Relief Valves
Haddam Neck	21394005	940219	c	Pressurizer Pilot-Operated Relief Valves
Haddam Neck	21396012	960611	c	Main Feedwater
Haddam Neck	21396018	960822	c	Main Feedwater
Oyster Creek 1	21985012	850612	c	Scram Discharge Volume
Nine Mile Point 1	22096004	960520	c	Main Feedwater
Dresden 2	23787023	870717	c	Main Feedwater
Dresden 2	23788012	880517	c	Main Steam Isolation Valves
Dresden 2	23798003 (Event No. 33620, dated 1/28/98 followed up by Morning Report H-98-0045, dated 3/6/98 referred to this condition)	980128	c	HPCI
Indian Point 2	24788017	881102	c	EDG Ventilation System
Indian Point 2	24793010	930818	c	Main Steam Power Operated Relief Valves Auxiliary Feedwater System
Dresden 3	24993004	930116	b, c	Scram Header Instrument Air
Dresden 3	24993005	930126	c	Drywell Environmental System
Turkey Point 3 and 4	25085020	850723	c	Main Steam Isolation Valves
Turkey Point 3	25085021	850722	a	Auxiliary Feedwater Instrument Air
Quad Cities 1	25492028	921016	c	Control Room

Table 5. (continued)

Plant	LER Number[a]	Event Date	Classification[b]	System
Palisades	25578003	780108	a	Shutdown Cooling (RHR)
Palisades	25581030	810718	a	Shutdown Cooling (RHR)
Palisades	25587018	870620	a	Main Feedwater Instrument Air
Palisades	25592007	920205	b, c	Main Steam Isolation Valve
Palisades	25592023	920327	b, c	High Pressure Safety Injection
Palisades	25594004	940209	c	Engineered Safeguards Systems (HPSI, LPSI, CS, CCW)
Palisades	25598006	980305	c	HP Air System
Browns Ferry 2	26095008	951004	b, c	Containment
Robinson	26194002	940425	c	Main Steam Isolation Valves
Point Beach 1	26697014	970321	c	Auxiliary Feedwater
Point Beach 1	26698008	980203	b, c	EDGs
Oconee 2	27093002	930610	c	RCP Seal Return
Vermont Yankee	27198025, Revision 2	981211	c	Control Rod Drive
Salem 1	27291030	910920	c	Pressurizer Pressure Operated Relief Valves
Salem 1	27296012	960723	c	RHR
Peach Bottom 3	27891017	910924	Personnel Error, b	Main Steam Relief Valve
Prairie Island 1	28295013	950927	c	Control Room Chilled Water System Instrument Air
Fort Calhoun	28587018	870504	c	Containment Spray High Pressure Safety Injection Low Pressure Safety Injection
Fort Calhoun	28587025	870923	a	Diesel Generator Cooling System

Table 5. (continued)

Plant	LER Number[a]	Event Date	Classification[b]	System
Fort Calhoun	28587033	870706	a, c	Instrument Air Fire Protection System
Fort Calhoun	28588002	880125	c	High Pressure Safety Injection
Fort Calhoun	28588004	880311	c	Instrument Air
Fort Calhoun	28588009	880406	c	High Pressure Safety Injection Component Cooling Water
Fort Calhoun	28588028	881020	a	Main Steam Isolation Valves
Fort Calhoun	28590025	900929	c	Component Cooling Water Raw Water Containment Spray System
Indian Point 3	28688009	881025	b, c	Reactor Coolant System Liquid Waste
Indian Point 3	28693013	930414	a	Central Control Room (CCR) Ventilation System
Indian Point 3	28693035	930916	c	WCCPPPS, IA and SA
Indian Point 3	28693036	930915	c	Central Control Room (CCR) Ventilation System
Indian Point 3	28693045	931023	c	Central Control Room (CCR) Ventilation System
Indian Point 3	28693050	931115	b, c	Service Water System
Indian Point 3	28693053	931202	b, c	Emergency Diesel Generators
Indian Point 3	28696002	960120	a	Emergency Diesel Generators
Indian Point 3	28696004	960215	c	Containment Isolation Diaphragm Valves
Indian Point 3	28696008	960320	c	Isolation Valve Seal Water System
Indian Point 3	28699002	990122	c	Containment

Table 5. (continued)

Plant	LER Number[a]	Event Date	Classification[b]	System
Oconee 3	28791007	910703	a, b	Main Feedwater Condensate System Instrument Air
Three Mile Island 1	28986007	860326	c	Two Hour Backup Air Supply System, EFW, Atmosph. Dump Valves
Pilgrim	29389002	890110	c	Containment Isolation System
Pilgrim	29389004	890127	c	Containment Isolation System
Pilgrim	29397025 (Event No. 33360 refers.)	971123	c	MSIVs, Containment
Pilgrim	29397026 (Event No. 33360 refers.)	971206	c	Main Feedwater
Cooper	29894013	940712	c	RCS, Core Spray
Crystal River 3	30297015	970612	c	Containment
Maine Yankee	30996003	960213	c	Main Feedwater
Salem 2	31197002(Event No. 31783 refers.)	970213	b, c	Auxiliary Building Ventilation
Cook 1	31597023(See Event No. 32939)	970916	c	ESF Ventilation
D. C. Cook	31597026	970925	c	RHR HX Outlet Valves, Steam Generator PORVs
Calvert Cliffs 1	31788009	880824	c	Main Feedwater
Calvert Cliffs 1	31789005	890314	c	Instrument Air
Calvert Cliffs 1	31789018	891106	c	6 Different Systems Instrument Air
Calvert Cliffs 1	31789018	891106	c	Instrument Air
Hatch 1	32192003	920102	b, c	RHR Core Spray
Sequoyah 1	32792018	921026	a	Main Feedwater Essential Air Nonessential Air System
Sequoyah 1	32797012	970801	c	Control and Service Air System

Table 5. (continued)

Plant	LER Number[a]	Event Date	Classification[b]	System
Duane Arnold	33191005	910622	c	Main Steam Isolation Valves
Fitzpatrick	33389004	890309	c	Instrument Air Electrical Cooling System
Beaver Valley 1	33490007	900330	a,c	Main Feedwater Instrument Air
Millstone 2	33689011	890906	b, c, Personnel Error	Turbine Building Closed Cooling Water Instrument Air
Millstone 2	33697011	970402	c	Containment, CVCS, RC Sample System, Liquid Radwaste System
Millstone 2	33698019	980826	c	Auxiliary Feedwater
Fermi 2	34194004	940822	c	Control Rod Drive Hydraulic System Reactor Coolant System
Fermi 2	34197007	970325	b	Reactor Building Ventilation System
Davis Besse	34687015	871207	b, c	Main Steam Main Feedwater Instrument Air
Davis Besse	34688007	880304	c	Decay Heat Removal)Instrument Air
Hope Creek	35486063	860828	b, c	10 Different Systems Instrument Air
Hope Creek	35494017	941110	c	Diesel Denerator Room Cooling
Hope Creek	35497020	970828	c	Safety Auxiliary Cooling
San Onofre 2	36196011	961216	c	PCIS
San Onofre 2 (and 3)	36199003	990210	c	Component Cooling Water
LaSalle 1	37385008	850202	b	Reactor Building Ventilation System
LaSalle 1	37385011	850202	b	Reactor Building Ventilation System

Table 5. (continued)

Plant	LER Number[a]	Event Date	Classification[b]	System
LaSalle 1	37389007	890210	b	Control Room HVAC System
LaSalle 1	37396011	960928	c	Containment Isolation System Reactor Core Isolation Cooling
LaSalle 1	37397046 and 37398007(Event No. 33434 also refers.)	971216 and 980319	b, c	Reactor Building and Auxiliary Building
LaSalle 2	37489018	891216	c	Division I Battery Room Heat Removal System
LaSalle 2	37495005	950218	b, c	Main Steam Isolation Valves
Summer	39587019	870730	c	Instrument Air Control Room Ventilation
Summer	39598009	981006	c	Emergency Feedwater
Shearon Harris	40098001(Event No. 33513 refers.)	980109	c	Steam Generator
Shearon Harris	40098001 (continued)	981009	c	Steam Generator
Catawba 1	41397002	970506	c	Feedwater Containment Isolation Valve
Millstone 3	42393002	930207	c	Control Room Pressurization System
Millstone 3	42396013	960515	c	Residual Heat Removal System
Millstone 3	42396028	960916	c	Charging System
Millstone 3	42396031	960906	b, c	13 Different Systems
Millstone 3	42396036	960926	c	High and Low Pressure Safety Injection
Millstone 3	42396040	961024	c	Reactor Plant Component Cooling Water
Perry	44087009	870227	b, c	Emergency Diesel Generators
Perry	44090021	900709	b, c	Main Steam
Comanche Peak 1	44595005	950831	c	Pressurizer Power Operated Relief Valves
Comanche Peak 1	44598001	980110	c	Feedwater
River Bend	45889022	890502	c	Instrument Air

Table 5. (continued)

Plant	LER Number[a]	Event Date	Classification[b]	System
River Bend	45889024	890406	b, c	Control Building HVAC Auxiliary Building HVAC Fuel Building HVAC
Clinton	46190004	900703	b, c	Misc. Active Safety-related Systems
Clinton	46198009	980203	b, c	EMDs
Palo Verde 1	52889005	890412	c	Containment Main Steam
Palo Verde 1	52894009	941118	c	Containment
Palo Verde 1	52895007	950512	c	Containment

a. The NRC Licensee Event Report (LER) Number consists of the three-digit NRC Docket Number for the plant at which the event occurred, the last two digits of the year in which the LER was generated, and a three-digit sequential number of the LER. This is consistent with the NRC's Sequence Code Search System (SCSS) database designation.

b. Events in the table are classified as follows:
   a. Failures of AOVs or events resulting from contaminated (not clean and dry) air;
   b. Failures of AOVs or events resulting from solenoid failures; and/or
   c. Failures of AOVs or events resulting from design deficiencies, material failure, or insufficient margin.

Table 6. Air-operated valves considered by the licensees in the plants visited to be risk significant.

Plant Or Site (See Note 1)	AOV Description and/or Designation	No. of AOVs	Safety Category (Sr = Safety Related, Nsr = Non-Safety-Related)	Risk Ranking or Significance (See Notes)	Remarks
Palo Verde	Charging System Valve CHE-HV0532	3	Passive SR	High	See Note 2. Only single valve listed as having "high risk significance."
Palo Verde	4 Atmospheric Dump Valves per unit. SGA-HV0178, 0179, 0184, and 0185	12	Active SR	High for common-cause failures	See Note 2. See LER 52889005 for a description of the common-cause failure of these AOVs.
Palo Verde	4 Feedwater Isolation Valves per unit. SGA-UV0130, 0135, 0172 and 0175	12	Active SR	High for common-cause failures	See Note 2. These AOVs were analyzed by the licensee because of common cause failures attributed to lack of design margin.
Palo Verde	2 Steam Generator Isolation Valves SGA-0500P and Q	6	Active SR	High for common-cause failures	See Note 2.
Palo Verde	4 Valves SGN-UV170, 171, 180 and 181	12	Active SR	High for common-cause failures	See Note 2.
Palo Verde	2 Valves SGN-FV1113 and 14	6	Passive SR	High for common-cause failures	See Note 2.
Fermi 2	8 MSIVs B2103F022A, B, C, and D, F026 A, B, C, and D	8	SR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	4 Scram Discharge Vol. Vent and Drain Valves C1100F010, 011, 180, and 181	4	SR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.

Table 6. (continued).

Plant Or Site (See Note 1)	AOV Description and/or Designation	No. of AOVs	Safety Category (Sr = Safety Related, Nsr = Non-Safety-Related)	Risk Ranking or Significance (See Notes)	Remarks
Fermi 2	370 Control Rod Drive Scram Inlet and Outlet Valves C1103D	370 (Not added to total)	SR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	1 Drywell Floor Drain Area Sump Pumps C001A and B to Floor Drain Collection Tank CIV G1100F003	1	SR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	1 Drywell Floor Drain Area Sump Pumps C006A and B to Floor Drain Collection Tank CIV G1100F003	1	SR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	1 Pressure Control Valve N20F-400 (condensate polishing demineralizer differential pressure control valve)	1	NSR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	1 Level Control Valve N20F-406 (condensate condenser low-level make-up level control valve emergency supply)	1	NSR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	1 Level Control Valve N21F-403 (level control valve: reactor feed pump start-up control V12-2512)	1	NSR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.

Table 6. (continued).

Plant Or Site (See Note 1)	AOV Description and/or Designation	No. of AOVs	Safety Category (Sr = Safety Related, Nsr = Non-Safety-Related)	Risk Ranking or Significance (See Notes)	Remarks
Fermi 2	1 Temperature Control Valve P43F402 (general service water from turbine building CCW heat exchanger)	1	NSR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.Note: During the visit we were told that this valve had a "surprisingly high" calculated risk significance.
Fermi 2	1 Temperature Control Valve P44F400A (temperature control valve EECW heat exchanger P4400B001 service outlet)	1	SR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	1 Temperature Control Valve P44F400B	1	SR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	2 Division 2 Pump Discharge Primary Containment Valves EESW P45F400A and B	1	SR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	4 EDG Air Coolant System 3-Way Temperature Control Valves (R3000F023A, B, C, and D)	1	SR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	1 Standby Gas Treatment Supply Chamber Purge Isolation ValveT4600F400	1	NSR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.

Table 6. (continued).

Plant Or Site (See Note 1)	AOV Description and/or Designation	No. of AOVs	Safety Category (Sr = Safety Related, Nsr = Non-Safety-Related)	Risk Ranking or Significance (See Notes)	Remarks
Fermi 2	1 Reactor Building HVAC Exhaust System Isolation Valve T4600F407	1	NSR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	1 Secondary Containment to Division 2 Standby Gas Treatment Isolation Valve T4600F408	1	NSR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	1 Standby Gas Treatment Secondary Containment to Division 1 Standby Gas Treatment Isolation Valve T4600F409	1	NSR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	1 Standby Gas Treatment to Torus Air Purge Valve T4600F412	1	NSR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Fermi 2	2 Division 1 Torus Hard Pipe Vent Secondary Control Isolation Valves T4600F421 and 422	2	NSR	Listed as risk significant but no ranking provided	See Note 3. From tabulation of "AOVs That Are Risk Significant" as determined by an Expert Panel.
Palisades	1 LPSI Shutdown Cooling Heat Exchanger Bypass Valve CV-3006	1	SR	High safety significance	See Note 4. From a tabulation of active AOVs modeled in the PSA. No numerical values or ranking were provided.
Palisades	1 Condensate Inlet Containment Isolation Valve CV-2010	1	SR	High safety significance	See Note 4. From a tabulation of active AOVs modeled in the PSA. No numerical values or ranking were provided.

Table 6. (continued).

Plant Or Site (See Note 1)	AOV Description and/or Designation	No. of AOVs	Safety Category (Sr = Safety Related, Nsr = Non-Safety-Related)	Risk Ranking or Significance (See Notes)	Remarks
Palisades	1 Normal Steam toP-8B from Steam Generator A Valve CV-0522B	1	SR	High safety significance	See Note 4. From a tabulation of active AOVs modeled in the PSA. No numerical values or ranking were provided.
Palisades	1 Shutdown Cooling to LPSI Isolation Valve CV-3025	1	SR	High safety significance	See Note 4. From a tabulation of active AOVs modeled in the PSA. No numerical values or ranking were provided.
Palisades	2 Containment Sump Isolation to East Engineered Safeguards Room Valves CV-3029 and 3030	2	SR	High safety significance	See Note 4. From a tabulation of active AOVs modeled in the PSA. No numerical values or ranking were provided.
Palisades	4 Steam Generator E-50B Steam Dump Control Valves CV-0779, 80, 81 and 82	4	SR	High safety significance	See Note 4. From a tabulation of active AOVs modeled in the PSA. No numerical values or ranking were provided.
Palisades	1 Shutdown Cooling Inlet to Shutdown Heat Exchangers CV-3055	1	SR	High safety significance	See Note 4. From a tabulation of active AOVs modeled in the PSA. No numerical values or ranking were provided.
LaSalle 1 and 2	Loss of instrument air accounted for about 1/3 of CDF. See Note 5	Not specified	See Note 5	See Note 5	No individual rankings were provided. See Note 5.
LaSalle 1 and 2	Six of 7 ADS valves fail to function on demand	7 per unit total	SR	RAW = 74.8 F-V = 0.0167	No individual rankings were provided. See Note 5.
Three Mile Island 1	Two Atmospheric Dump Valves for A and B Steam Generator MS-V-004A and B (6" Fisher globe air piston)	2	SR (Category 1)	F-V = MF-V range listed as 0.00136 to 1.14 RAW = L	See Note 6.

Table 6. (continued).

Plant Or Site (See Note 1)	AOV Description and/or Designation	No. of AOVs	Safety Category (Sr = Safety Related, Nsr = Non-Safety-Related)	Risk Ranking or Significance (See Notes)	Remarks
Three Mile Island 1	Containment Isolation IC Isolation coolant return valve IC-V-0003 (6" Tufline plug air piston)	1	SR (Category 1)	RAW = HF-V = MF-V range listed as 0.00116 to 24.2	See Note 6.
Three Mile Island 1	Containment Isolation IC Isolation Coolant Supply IC-V-0004 (6" Tufline plug air piston)	1	SR (Category 1)	RAW = HF-V = MF-V range listed as 0.00116 to 24.2	See Note 6.
Indian Point 3	8 Auxiliary Feedwater Regulating Valves BFD-FCV-405A, B, C, and D and BFD-FCV-406A, B, C, and D (2" Copes-Vulcan D100 diaphragm AOVs)	8	SR	"High PRA"	See Note 7.(RAW = 202.000 associated with BFD-FCV-406 AOVs for CCF of AFW motor-driven pumps.) (RAW = 2.431 for CCF of all 4 PM32 FCVs to open.)(RAW = 2.047 for any one AOV to close on demand.)
Indian Point 3	1 Main Steam to Auxiliary Boiler Feedwater Pump MS-PCV-1139 (2.5" WKM globe valve)	1	SR	"High PRA"	See Note 7.(RAW = 5.956 for Steam Control Valve, PCV-1139 to not open.) (RAW = 9.649 associated with AFW TD Pump #32 fails to continue to run.)
Indian Point 3	2 Condensate Storage Tank to Condensers Level Control Valve CT-LCV-1158-1 and -2 (12" Golden Anderson Butterfly)	1	SR	"High PRA"	See Note 7.(RAW = 1.141 for either valve to not close.)
Indian Point 3	1 Condensate Polisher Facility Inlet Stop CD-AOV-518 (Cameron Iron Works Ball Valve)	1	NSR	"High PRA"	See Note 7.(No RAW value provided.)

Table 6. (continued).

Plant Or Site (See Note 1)	AOV Description and/or Designation	No. of AOVs	Safety Category (Sr = Safety Related, Nsr = Non-Safety-Related)	Risk Ranking or Significance (See Notes)	Remarks
Indian Point 3	3 Heater Drain Tank to Condenser Bypass HD-LCV-7001, 2, and 3 (4" Masoneilan globe valve)	3	NSR	"High PRA"	See Note 7.(No RAW value provided.)
Indian Point 3	4 Steam Generator Atmospheric Dump MS-PCV-1134, 5, 6, and 7 (6" Copes-Vulcan D-100 globe valve)	4	SR	"High PRA"	See Note 7.(RAW = 1.613 for any of the 4 valves to fail to close.)
Indian Point 3	2 Pressurizer PORVs RC-PCV-455C and 456 (3" Copes-Vulcan F series globe valve)	2	SR	"High PRA"	See Note 7.(RAW = 8.217 for PCV-455C to fail to close.) (RAW = 7.476 for PCV-456 to fail to close.) (RAW = 5.666 for PCV-455C does not open.)(RAW = 6.306 for PCV-456 does not open.) (RAW = 6.229 for CCF of PORVs to open.)
Indian Point 3	MSIV 1-31, 1-32, 1-33, and 1-34	4	SR	Each individual valve is listed by the licensee as "No" in the column headed "High PRA"	See Note 7.(RAW = 50.73 associated with CCF of two or more MSIVs.) (RAW = 1.614 for failure of each MSIV to close on demand.)
Indian Point 3	EDG Flow Control Valves FCV-1176 and 1176A	2	Not provided	Not provided	See Note 7.(RAW = 46.97 associated with CCF of EDG flow control valves.)
Indian Point 3	Steam Generator Blowdown AOVs, BD-PCV-1214, 1214A, 1215, 1215A, 1216, 1216A, 1217, and 1217A	8	SR	Each individual valve is listed by the licensee as "No" in the column headed "High PRA"	See Note 7.(RAW = 3.458 associated with CCF of 2 SG Blowdown valves.)

Table 6. (continued).

Plant Or Site (See Note 1)	AOV Description and/or Designation	No. of AOVs	Safety Category (Sr = Safety Related, Nsr = Non-Safety-Related)	Risk Ranking or Significance (See Notes)	Remarks
Indian Point 3	Solenoid valves VS-SOV-1297, 1298, 1303, 1304, 1306, and 1307	6	Not provided	Not provided	See Note 7. (RAW = 1.035 for any SOV's failure to function.)
Turkey Point	3 AOVs FCV-3-6278A, B, and C, Steam Generator Blowdown Control Valves	6	SR	Risk significant	See Note 8. The Expert Panel considered these valves to be not risk significant because manual valves are available as backup.
Turkey Point	3 AOVs CV-3-6275A, B, and C Steam Generator Blowdown Control Valves	6	SR	Risk significant	See Note 8. The Expert Panel considered these valves to be not risk significant because manual valves are available as backup.
Turkey Point	3 AOVs CV-3-2816, 17, and 18 Control Valve for water to steam generator from auxiliary feedwater pump	6	SR	Risk significant	See Note 8.
Turkey Point	3 AOVs CV-3-2831, 32, and 33 Control Valve for water to steam generator from auxiliary feedwater pump	6	SR	Risk significant	See Note 8.
Turkey Point	AOV LCV-3-113B	1	SR	Risk significant	See Note 8.
Turkey Point	AOV LCV-3-114A	1	SR	Risk significant	See Note 8.
Turkey Point	AOV LCV-3-115B	1	SR	Risk significant	See Note 8.
Turkey Point	3 AOVs CV-3-2903, 4, and 5	3	SR	Risk significant	See Note 8.
Turkey Point	AOV CV-4-1605	3	SR	Risk significant	See Note 8.
	Total No. of AOVs	182	(Excluding 370 Control Rod Drive Scram Inlet and Outlet Valves)		

Table 6. (continued).

Notes:

1. Plants are listed in the order of the site visits for the AOV study. There are three units at Palo Verde, two at LaSalle, and two at Turkey Point. (Note: At Turkey Point some systems, e.g. AFW, are shared.)

2. PRA input to the Palo Verde AOV program was provided in 1994 and described 27 AOVs (per unit) as having some impact on core damage. The AOVs were tabulated as "High Risk Significant," "High Risk Significant for Common Mode Failures," or "Low Risk Significant." No risk numbers or rankings within the categories were provided. The "Low Risk Significant" AOVs (10 per unit) are not included in the table above. We have since been told that the information may be obsolete but that no later information is available.

3. Fermi 2 provided a Table of "AOVs That Are Risk Significant." The table listed (according to the summary provided with it) 370 control rod drive AOVs, 22 AOVs "that perform safety-related functions," and 11 AOVs "that perform a non-safety related risk-significant function." All of these AOVs were designated as "Category 1 (high safety significance)." Category 1 was defined as follows:

   - "AOVs falling in this category are those that play an active role within systems that fall within the scope of the Maintenance Rule and that were considered to be risk significant by the Fermi Plant expert panel. Valves in this category are candidates for activities such as review of the design basis and setpoint verification confirmed by periodic diagnostic testing."

   - "Category 2 (less safety significance but having potentially significant economic consequences)" was defined as follows:

   - "Valves in this category may be either safety-related or are not safety-related. These valves either play an active safety-related function of low risk significance or are used in applications that may affect plant availability, capacity factor, or heat rate. AOVs in this category would be subject to design reviews and/or testing on an as-needed basis at the discretion of the plant staff."

   - "Category 3 (less safety significant and having little or no plant economic impact)" was defined as follows:

   - "AOVs in this category may or may not be safety-related, have been determined to be of less safety significance playing no active safety-related role and do not have a significant potential for affecting plant operation. Design basis review or diagnostic testing would not be performed for valves in this category."

4. Three safety significance ratings were described in the Palisades PSA review of AOVs as follows:

   - Non-safety significance rated valves were not modeled in the PSA or were not modeled as active components.

   - Low safety significance rated valves were those modeled in the PSA but not having high importance measures.

Table 6. (continued).

- High safety significance rated valves were modeled in the PSA and had high importance measures. High importance measures were defined as Fussell-Vesely (F-V) >5E-3 or Birnbaum>5.15E-5. These importance measures were considered by the licensee to correspond to Risk-Achievement Worths (RAW) >2.0 and Risk-Reduction Worths (RRW) >1.005 respectively.

5. At LaSalle, no tabulation was provided during the site visit about rankings for AOVs on the basis of risk significance. Two tables of interest here were provided (5a and 5b) in the LaSalle PRA Summary Document. These tables show Component Failure Mode Importances to CDF for RAW $\geq$ 2 and Fussell-Vesely, respectively, for many different components, including AOVs. The components that could be positively identified from the descriptions in these tables as AOVs were included in the table above, along with the values of the importance measures listed.

   NOTE: A list of the top 100 core damage cutsets was included in the LaSalle PRA Summary Document. Transients with loss of instrument air were the largest initiating event category at LaSalle station and contributed 32% of the core-damage frequency. This risk number is noted as the first entry in the table above for LaSalle because it is considered an exceptionally high value and therefore pertinent to the study of AOVs and their risk significance.

6. Two documents were provided during the site visit to Three Mile Island (TMI) 1 that provided data and information on risk-significant AOVs. These were:

   1. "Draft AOV Program Description, Topical Report 118," (not dated) categorized AOVs as follows:

      - Category 1 included AOVs that are required to perform a safety function are called upon during a design basis event. Category 1 AOVs are "...further categorized by an expert panel based on the relationship between the valve's risk ranking (level of safety significance) and amount of available operating margin. Category 1A are High Safety Significant valves or valves that have Low Safety Significance with low operating margin. Category 1B valves are those with Low Safety Significance and medium to high levels of operating margin."

      - Category 2 AOVs were classified as AOVs "...selected by an expert panel which provide significant assistance to the operation of safety systems as delineated in TMI Emergency Operating Procedures. Category 2 valves include valves selected by the Expert Panel that may have a significant effect on the safe operation of the plant. These valves might include valves that are the last line of defense for release of radiation to the environment or may provide emergency boration to the Reactor Coolant System. Additionally, AOVs whose failure may result in a plant trip, power reduction or have economic significance on plant operation are considered for inclusion in this category."

      - Category 3 AOVs were defined as "...valves that have no active safety role and do not have a significant potential for affecting plant operation."

Table 6. (continued).

- A list of Category 1 AOVs was included in the program plan (Appendix A thereto) and is summarized in the table above.

2. A paper entitled "TMI-1 PRA Input to Maintenance Rule Risk Significant System List," developed by C. Adams, Risk Analysis Section (not dated), included Fussel-Vesely and Risk Achievement Worth (see Section 15.61 for further explanation of these terms) and RAW data and rankings for valves including AOVs. (It appears that a portion of the paper was not provided.) Pertinent data was included in the table above. A table of on-line maintenance risk rankings indicated that RAW values of 1 to 3 were considered Low (L), RAW values of 3 to 15 were considered Medium (M), and RAW values of 15 to 30 were considered High (H). RAW values above 30 were not to be allowed without a senior management evaluation and a risk management evaluation. RAW (Risk Achievement Worth was defined as the factor increase in core damage frequency resulting when the structure, system, or component (SSC) is determined to be continuously failed.

   Eight AOVs were categorized as L in the reference and these were not included in the table above. Between 64 and 68 (64 by count and 68 in a summary table) AOVs and SOVs were also listed in the reference as NM (not modeled), N (not ranked) or T (truncated), and these were also not included in the table above.

7. Three lists that included information on the risk significance of AOVs were obtained at the site visit to Indian Point 3. Two of the lists (dated 10-Mar-1998) included sorts of AOVs by system-component and by manufacturer-component and each identified "High PRA" AOVs. No ranking of the AOVs was provided in these two lists, nor was there an indication of what the cutoff for "High PRA" might have been. The "High PRA" AOVs are included in the table above.

   The third list, attached to NYPA Memo from J. Circle to G. Smith dated November 5, 1996, included RRW, RAW, and unavailability estimates for a number of events, including events involving AOV failures. If it was possible to tie the events to a "High PRA" valve, the RAW values were listed in the Remarks column of the above table for the particular valve. In addition, several AOVs and SOVs that did not appear in the two lists described previously were found to be of interest regarding common-cause failures. These valves were also noted in the above table.

   (One item in the third list from IP-3 indicated that the CCF of IA compressors had an associated RAW value of 2.103, but no entry was included in the above table because the event did not refer to an AOV.)

8. An internal memorandum was provided by the Turkey Point engineers that listed 19 risk significant AOVs. These are included in the above table, along with whatever data could be gathered for each AOV. In conversations with the risk analysts at Turkey Point on May 17, 1999, the risk ranking numbers in the internal memorandum could not be confirmed, and were omitted.

Table 7. Some recent (approximately last 5 years) events and conditions involving AOVs and/or air-operated components where the design basis was not met and/or not known.

NOTE: Items in this table marked in the left margin as shown here were not included in tables of events included in the draft (for comment) report dated April 26, 1999.

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Clinton 46198009 980203	Engineering personnel determined during their review of a 10CFR21 notification by Engine Systems, Incorporated, that the air start solenoid pilot valves for the emergency diesel generators (EDG) would not operate as required by the design basis. The solenoid valves, which are used to valve air to the air operated main valves that supply starting air to the start motors on all three EDGs, would not operate reliably at the low end of the design basis air start system pressures and DC voltages. The apparent cause of the inadequate design of the solenoid valve was a failure of the design engineer to consider the full range of design basis design pressures and DC voltages when changing the spring size of the solenoid pilot valve. Corrective action for this event included: changing the spring size in the solenoid valve for the Division III EDG air start System; replacing the Division I and II EDG air start system solenoid valves, or otherwise modifying the system to meet the design basis requirements; and revising the annunciator procedures for the EDG air start system receiver low pressure alarm to require that the EDGs be declared inoperable when supply air pressure drops below 200 psig. This is considered to be a common-cause failure condition.	There are 335 SOVs, distributed among about 17 nuclear plant owners, listed in the Part 21 notice. Point Beach LER 26698008 also refers to Engine Systems Part 21 Notification #1998120, dated January 26, 1998.
Comanche Peak 1 44595005 950831	Engineering personnel identified non-conservatism in the calculation that determined (1) leakage rates for accumulator check valves associated with various air operated valves and the nitrogen accumulators for the pressurizer Power Operated Relief Valves (PORVs), and (2) the pressure switch alarm set points for these accumulators. Engineering personnel performed evaluations which revealed that, with the exception of the PORVs, the valves associated with these accumulators were operable. The PORV accumulator low pressure alarm set points would still ensure operability during Modes 1, 2 and 3. However, for Modes 4, 5 and 6 the set points would not ensure operability for all conditions. TU Electric believed that the cause of the event was non-conservative design in the original calculation used to determine the low pressure alarm set point for the PORV accumulators and the accumulator check valve leakage rates.The PORV nitrogen accumulator low pressure alarm set points were to be raised to 90 psig. A review was to be performed for all safety-related accumulators used for valve actuation to verify that the appropriate set points are being used. This is considered to be a common-cause failure condition.	There are two PORVs per pressurizer, thus 4 for the site.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Cook 1 31597023 970916 (The event covered by this LER was originally reported as Event No. 32939)	A design change to the bypass dampers for the Engineered Safeguards Features (ESF) ventilation system installed between December 1996, and August 1997, introduced the possibility of a single failure which could result in the loss of both trains of the ESF ventilation system. The loss of the 85 psig air header without concurrent loss of the 20 psig air header would result in the ESF ventilation trains being unable to meet their design function. The ESF vent system charcoal inlet and bypass dampers originally both used a 20 psig air header. The charcoal bypass dampers were normally open and were intended to fail closed. The charcoal inlet dampers were normally closed and intended to fail open. The licensee installed new bypass dampers which required higher pressure to operate and were therefore transferred to the 85 psig air header. If 85 psig air was lost, the bypass dampers would reposition to the closed position. The inlet dampers would remain closed and this would result in dead-heading of the filter trains and subsequent loss of cooling to ECCS equipment. The root cause of the event was the failure of the design change process to identify the potential adverse impact on the ESF ventilation system created by the modification of the control air supply to the bypass dampers. This is considered to be a common-cause failure condition.	This event involved dampers only (numbers unspecified). No AOVs were involved.
Cook 1 31597026 970925	Due to a lack of overpressure protection on the 85, 50, or 20 psig control air headers, if a non-safety-related air regulator failed open it would result in an overpressurization of a control air header. This would result in the potential for common-cause failure of both trains of safety related equipment. The lack of overpressure protection on the control air headers due to a regulator failing open had not been identified as a mechanism that could overpressurize the low pressure headers. As a result, single failure of a non-safety related component affecting both trains of safety-related equipment was not identified. There would have been no significant effects for the 85 and 50 psig headers; however, overpressurization of the 20 psig header could have resulted in the degradation of the RHR system and the partial opening of the Unit 2 Steam Generator (SG) Power Operated Relief Valves (PORVs) for the duration of the overpressure event. Due to a single failure being identified that could have potentially resulted in the degradation of both trains of RHR, this event could have been significant. The cause of the lack of overpressure protection on the Control Air System was the fact that a regulator (non-safety-related) failing open was not identified as a mechanism that could overpressurize the low pressure headers. As a result, single failure of a non-safety related component affecting both trains of a safety related system was not identified.	Unknown. The LER and its attached Event Summary refer to "multiple" isolation and relief valves. The text of the LER refers to at least 6 valves by number.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Crystal River 3 30297015 970612	During a review of the differential pressure calculation for the Letdown Line [CB] Inboard Containment Isolation Valves, FPC discovered that the evaluation of the maximum differential pressure (d/P) that these valves could be subject to was in error. These valves are rated to close against a maximum d/P of 1800 psi, but could be subjected to a d/P in excess of 2000 psi in the event of a letdown line rupture downstream of outboard containment isolation valve (MUV-49), concurrent with a failure of MUV-49 to close and operator action in accordance with Emergency Operating Procedure (EOP) 3. Outboard containment isolation valve MUV-49 would not be capable of closing if subjected to 2000 psi d/P. Isolation at Penetration 333 requires either MUV-49 or MUV-40/41/ 505 to close. The cause of this event was the use of inappropriate assumptions in the calculation for the determination of maximum valve d/P. FPC planned to install a new inboard containment isolation valve prior to restart of CR-3. In addition, the air operator on MUV-49 was to be modified to allow closure against the predicted d/P. The closure of these valves is required to mitigate the effects of a Makeup System Letdown Line Failure Accident and in response to a Reactor Building Isolation signal. Consequently, the valves were outside of their design basis and their failure to close in the described scenario during previous operating periods, could have created an non-isolatable loss of reactor coolant to the Auxiliary Building. This is considered to be a common-cause failure condition.	The summary of the event in the LER indicates 8 valves per unit were involved. It was assumed that these were AOVs.
Haddam Neck 21396012 960611	An engineering analysis determined that the main feedwater regulating valves (FRV) would not fully isolate feedwater flow as required for a main steam line break accident in containment. It was determined that the differential pressure across the valves would overcome the valve spring's closing forces. In a design basis steam line break analysis, the feedwater motor operated valves (MOV) are required to isolate; however, in the event of a single failure of the MOV the FRV is credited with isolation. The failure to isolate feedwater for a steam line break in containment could result in exceeding maximum containment design conditions. The cause of this condition was an erroneous assumption in the accident analysis that the FRVs would isolate against a high differential pressure. The differential pressure across the valve would overcome the valve spring's closing force. This is considered to be a common-cause failure condition. See LER 21396018. (Event No. 30619 dated 6/11/96 applies.)	LER text indicated that 4 AOVs were involved.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Haddam Neck 21396018 960822	An engineering analysis revealed that the feedwater regulating bypass valves would not fully isolate feedwater flow as required for a main steam line break inside containment. This condition was discovered during a follow-up to a similar problem with the main feedwater regulating valves (LER 21396012). The failure to isolate feedwater for a steam line break inside containment could result in exceeding maximum containment design conditions. This event did not involve any actual equipment failures. The cause of this condition was an erroneous assumption that the feedwater bypass valves would close and isolate against the differential pressure experienced between the steam generator feed pump and a faulted steam generator. Additionally, credit was taken for isolation of the bypass valves from the control room ten minutes after an accident, however the control air system which operates the valves is not a credited system. The bypass valves (FW-HICV-1301-1,2,3,4) are 1-1/2 inch, air to close, spring to open, manufactured by Masoneilan. They are normally closed during full power operation. On an auxiliary feedwater actuation signal they go full open until manual operator action is taken at the main control board to throttle flow. This is considered to be a common-cause failure condition. See LER 21396012.	LER text indicated that 4 AOVs were involved.
Indian Point 3 28696004 960215	Two air-operated vapor containment isolation diaphragm valves in series were found to be inoperable, which violated Technical Specification 3.6.A.1. The original isolation valve design would close at system pressure when there was a differential pressure or accident design pressure, but not against system pressure with no pressure differential. The design and specification of the valves during initial design and construction was not adequate to ensure the valves met their design requirement for containment isolation. RC-AOV-519 and RC-AOV-552 (the valves are designed to positively seal with a differential pressure) would close against a differential pressure of 150 psi (the Primary Water System design pressure) but would not have closed with a line pressure greater than about 120 psig when there is no differential pressure. The original specification for these valves was for a maximum differential pressure of 200 psi, with no reference to a minimum pressure differential or constant line pressure requirement. The valve design had not been modified since the original plant design and construction. The condition could have existed when closing isolation valve RC-AOV-560, downstream and in series with RC-AOV-519 and 552, in a sequence that maintained a line pressure greater than 12 psi. There would then be no differential pressure during stroke testing. Stroke testing in these conditions created another problem if the limit switches are adjusted after the valves are shut but do not fully close. The valves were assumed to be fully closed during the adjustment so subsequent failures to close would be masked and could allow a control room indication that the valves were fully closed when they were not. There was no direct external indication on the valves to show if they were fully closed. This is considered to be a common-cause failure condition.	2 AOVs.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Indian Point 3 28699002 990122	A design condition that had the potential to place the plant outside its design basis was confirmed to exist. If a single failure of containment isolation valve VS-PCV-1190 to close upon demand occurred, a potential containment release path would exist in the Weld Channel and Containment Penetration Pressurization System (WCCPPS) {BD} supply/containment atmosphere exhaust line. The WCCPPS supplies pressurized air to the spaces between the three in-series containment isolation valves (VS-PCV-1190, -1191 and -1192) in the VC Pressure Relief System. This air is supplied when the isolation valves are closed; closure is determined by the closed position limit switches of the isolation valves being "made." Removal (i.e., exhaust) of WCCPPS occurs when the isolation valve control switches are actuated to initiate operation of the VC Pressure Relief system. Supply of WCCPPS air is directed to each of the spaces between the two isolation valve when the associated three-way solenoid valve {PSV} (PS-SOV-1280) is de-energized; a 1-inch line connects each solenoid valve to its corresponding space between isolation valves. Exhaust of the WCCPPS air occurs through the same 1-inch line back through the solenoid valve's exhaust port to the Primary Auxiliary Building (PAB) {NF} atmosphere. This alignment and flow path occur when the solenoid valve is energized. As part of the initiation of a VC Pressure Relief Operation, PS-SOV-1280 is energized to allow the WCCPPS air in the inter-space between VS-PCV-1190 and -1191 to exhaust into the Piping Penetration area. PS-SOV-1280 will remain in this position until valves VS-PCV-1190 and 1191 are closed. If an automatic containment isolation signal or a manual isolation signal is created, all three isolation valves in the VC Pressure Relief System receive a closure signal. Should inboard isolation valve VS-PCV-1190 (located inside the containment) fail to close under such a condition, an unintended containment atmosphere flow path would exist through the 1-inch line between the VS-PCV-1190/1191 inter-space and the exhaust port of PS-SOV-1280. PS-SOV-1280 is located in the Piping Penetration area of the PAB. This combination of circumstances, although unlikely, would represent a failure to meet the single failure criteria as applied to containment isolation. The most likely cause of the event is an oversight during the initial design phase of the plant when considering single failure. The designers of the VC Pressure Relief System apparently failed to recognize that a single failure of the inboard containment isolation valve (VS-PCV-1190) to close during containment pressure relief could prevent the isolation of the vent path from containment into the PAB during a postulated event.	One AOV.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
LaSalle 1 37396011 960928	While developing an Air-operated Valves (AOV) preventative maintenance program, inconsistent testing data were obtained for valves with WKM 70-13-1 pneumatic actuators. The inconsistent results appeared to be related to incorrect effective diaphragm areas (EDA) for the AOV actuators. 36 WKM AOVs were addressed in the LER (18 per unit). 13 AOVs in each unit are part of the primary containment isolation system (PCIS) and 5 are in the reactor core isolation cooling system (RCIC). Two problems associated with the EDA of the actuators of the WKM valves were identified. The first was related to the actual versus the manufacturer's published EDA of the actuator. If the actual EDA is less than what the manufacturer publishes, then the closing (or opening) forces installed in the valve (via spring/spring adjustment) will be less than required. The second problem was stretching of the diaphragm during valve travel resulting in a reduced EDA. This is considered to be a common-cause failure condition.	36 AOVs (18 per unit).
LaSalle 137397046 and 37398007 971216 and 980319 (Event No. 33 34 also referred)	(LER 37398007) An investigation of the allowable closure time assumptions for the turbine building high-energy line break check dampers (#1/2VT79YA/B/C) led the licensee to reconsider the original builders assumptions and calculations for the reactor building exhaust isolation dampers (1VR05YA and B). (LER 37397046) In the event of a Main Steam High Energy Line Break, the Turbine Building Ventilation isolation dampers, 1(2)VT079YA, B & C, would not close fast enough to prevent the pressure from exceeding the pressure retaining capability of the walls, floors, and ceilings that separates the VT exhaust tunnel from the safety related High Pressure Core Spray (HPCS) electrical switchgear room. The apparent causes included an invalid calculation assumption dating from the original design. Dampers 1VR05YA and B had malfunctioned and were rebuilt in 1985 (see LER 37385008 and LER 37385011). The licensee indicated that they revised the calculations and assumed a 0.4 second instrument time delay, and a 0.075 second solenoid valve response time. The plant was licensed in 1982.	No AOVs. Four air-operated dampers (2 per unit).

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Millstone 2 33697011 970402	The closing force for multiple dual function (two separate pressure isolation functions) valves had been improperly set, resulting in the valves being incapable of closing to a leak tight condition against normal operating system pressure (NOSP). Eleven of the 23 valves tested were not capable of providing an adequate closing force. This deficiency could have resulted in the potential for a release of radioactive materials to the Auxiliary Building greater than analyzed in the facility Final Safety Analysis Report (FSAR). The closing forces were incorrectly set during the period between October 1986 and March 1997. The cause of this event was an insufficient program to ensure that facility procedures clearly addressed all related design basis functions. The affected valves were to have been adjusted to ensure they properly close against containment design pressure and NOSP, and the appropriate procedures were to have been revised to ensure that proper valve control parameters are specified and verified after any maintenance activities are performed that could affect dual function valve closing forces. This condition affected 11 valves in three systems. This is considered to be a common-cause failure condition. Event No. 32070 referred to this event.	Eleven of 23 AOVs. (Event No. 32070 reported 19 of 23 AOVs.)
Millstone 2 3369801 9980826	In 1981, the AFW regulating valves were changed from MOVs to AOVs to provide electrical independence from a failure which could preclude the valves from opening and supplying AFW to the steam generators. Both normally closed valves were designed to open by an Auto AFW Initiation (AAFWI) signal or fail open upon loss of instrument air. In 1992, non-safety grade backup air was installed to provide the ability to close the valves in the event of a beyond design basis main feedwater system line break in the turbine building (TB), i.e., a high energy line break (HELB), coincident with a feedwater check valve failure. This change was made to improve the unit's core melt frequency, however, it was not recognized at that time that more limiting HELBs could affect the TB. The (then) current main steam line break (MSLB) analysis assumed operator action to isolate the faulted S/G ten (10) minutes after the event. However, closure of the AFW regulating valve and isolation of the S/G may not be assured. On September 16, 1998, a NRC review also identified that following a HELB in the TB, a higher-than-previously analyzed ambient temperature may also have challenged the ability of the AFW regulating valve backup air equipment to remotely close and isolate flow. Subsequently, the licensee identified that other components of the AFW regulating valve also may have had been challenged. From 1981 through 1998, when the plant operated in Modes 3 or higher, the capability to manually close the AFW regulating valves to isolate flow out the faulted S/G as analyzed, could not be assured and a potential for operating outside of the plant's design basis existed.	Two AOVs.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Millstone 3 42396013960515	An engineering evaluation determined that a design deficiency in the Residual Heat Removal System (RHS) was a condition that was outside the design basis of the plant. A loss of control air supplied from the non-safety-related instrument air system could cause the RHS control valves to fail open. If this condition occurred during the initial phase of a plant cool down, the Reactor Plant Component Cooling Water System (CCP) temperatures could exceed 125°F. This is the design temperature used in the system stress analysis. If RHS heat exchanger operation was initiated at 350°F RCS temperature, as assumed, then the RHS heat exchanger CCP outlet temperature could be as high as 250°F if the valves failed open. Under the resultant conditions the CCP piping would not meet the ASME Section III, Appendix F stress criteria. The original plant design did not consider that if the RHS flow control valves failed open on a loss of air, it could create unacceptably high RHS heat exchanger CCP discharge temperatures. This is considered to be a common-cause failure condition.	Two AOVs.
Millstone 3 42396028 960916	An engineering evaluation identified a failure scenario in which a loss of Instrument Air (IAS) to temperature control valves in the Charging Pump Cooling (CCE) system serving the charging pump lube oil coolers, coincident with 330°F Service Water (SWP) temperature could result in overcooling of both trains of the charging pump lube oil system and challenge charging pump operability. Failure of the air-operated CCE valves to the full open position due to a loss of the non-safety related IAS system would adversely affect both trains of the charging pumps by allowing excessive cooling of the CCE system which cools the lube oil system. This condition alone could have prevented the fulfillment of the safety function of the system. The cause of the charging pump inoperability was inadequate original design. This condition would result from overcooling of the lube oil system from a failure of the non-safety related Instrument Air system coincident with a worst case minimum SWP temperature and maximum flow and heat exchanger cleanliness. Under these conditions, the air-operated CCE valves would fail open and excessive cooling of the lube oil system would occur. This particular combination of conditions was not considered in the initial design. A short term corrective action was committed to in the LER to install a temporary modification to limit the failure position of the three way CCE temperature control valve to ensure sufficient bypass flow around the SW heat exchanger to maintain CCE temperature above 45°F. Other long-term actions were also discussed. This is considered to be a common-cause failure condition.	The LER indicates "multiple" AOVs.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Millstone 3 42396036 960926	The High Pressure Safety Injection (SIH) and Low Pressure Safety Injection (SIL) systems would have been subject to degraded performance due to possible mispositioning of normally closed safety related air operated valves. Mispositioning of these 38 valves can be postulated to occur under post-accident harsh environmental conditions due to failure of non-qualified power and control circuits. As a result the potential diversion of SIH and/or SIL flow under accident conditions could have been more than the margin allowed within the Loss of Coolant Accident analysis. Seventeen additional safety-related air operated and solenoid operated valves were subsequently identified where failures of non-qualified control circuits could degrade performance of a safety system function. These additional valves are located in the following systems: Reactor Plant Component Cooling Water (CCP), Containment Vacuum, Reactor Plant Sampling (SSR), Post Accident Sampling (SSP), and Main Steam to the auxiliary feedwater steam turbine. The cause of the reported conditions was a design mistake. The initial plant design did not adequately consider the potential mispositioning of these valves under harsh environmental conditions or active failure. This is considered to be a common-cause failure condition.	38 AOVs/SOVs.
Millstone 3 42396040 961024	An engineering evaluation determined that a failure scenario for the Reactor Plant Component Cooling Water (CCP) system had the potential for a loss of system safety function. The failure scenario involves a loss of the non-category 1 Instrument Air System, which would allow CCP valves to reposition to a maximum cooling configuration. Coupled with a low heat load and minimum Service Water (SWP) inlet temperature, the CCP system could reach temperatures lower than values for which they are analyzed, thereby rendering the CCP system, and systems that it serves, potentially inoperable. This was caused by improper initial design of the CCP system. The failure mode described in the LER was a design oversight by the plant's architect engineer that occurred during the original design process. The design basis analysis for the CCP system focused on high CCP heat load conditions. The designers did not analyze for extremely low CCP heat loads concurrent with very low SWP temperatures. This is considered to be a common-cause failure condition.	At least 6 AOVs were involved in this event although the LER does not specify an exact number. The LER summary lists the number of valves as "unknown."

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Palisades 25598006 980305	Event No. 33843 for Palisades in Table 4 was originally recorded for this event. The LER was subsequently issued and is discussed here. The original design bases, and subsequent design reviews, for the HP Air System address the Large Break Loss of Coolant Accident (LBLOCA), yet did not address, in detail, the SBLOCA. As a result, the impact of a loss of the HP Air compressors on the ability to supply air needed to align valves for sump recirculation during a SBLOCA, and the need to incorporate manual operator actions in EOPs to assure HP air reliability were not considered. As a result of the design basis reconstitution for the HP Air System, the licensee identified the need for procedural guidance to direct manual operator actions to restore the HP Air System during a SBLOCA. The HP Air System supports ECCS realignment to the recirculation mode by providing air required to operate control valves needed to switch the suction of the ECCS pumps from the Safety Injection Refueling Water Tank (SIRWT) to the containment sump. During a LOCA with LOOP, the HP Air System compressors are load shed from safety-related electrical buses and must be manually re-powered at their respective motor control centers (MCCs). When the air compressors are without power, the receiver tanks are not being charged. During this time, the HP Air System pressure experiences a gradual decay due to air bleed-off from pressure regulators and air leakage through seals in piston-driven control valve actuators. The LER focuses on the procedures needed to ensure restoration of HP air; however, the effects of gradual loss of air on the failed position on valves that are supplied by HP air had not been investigated.	The LER indicated that "multiple" AOVs were involved in this event but no specific valves or numbers of valves were provided.
Palo Verde 1 None 951126	Three downcomer feedwater isolation valves (DCFWIVs) at PVNGS failed to open, following closure after a main steam isolation signal (MSIS) during a Unit 1 reactor trip on November 26, 1995. LER 52895012 covered the reactor trip but not the DCFWIV failures. The source of the information concerning the DCFWIV failures was a comprehensive report (the copy received during the plant visit was untitled) provided by the licensee during the site visit for this study. There are four DCFWIVs in each unit, two in series to each of the two steam generators. The DCFWIVs are 8-inch flex-wedge gate AOVs. The actuators are Miller Fluid Power single-acting pneumatic cylinders with internal cylinder springs and external spring-and-stanchion setups. The actuators are designed to use pressurized nitrogen to open the valves and spring force to close the valves. The pneumatic control circuits include a 3-way SOV and a 3-way air switch valve. The valves are intended to fail closed on loss of the air signal to the 3-way air switch valve or loss of nitrogen to the operator. The valves are intended to fail open if electric power to the SOVs is lost. The DCFWIVs have three key safety functions: heat removal, trip initiation, and containment isolation. These valves are also Appendix R safe-shutdown components. (continued next page)	Four AOVs per unit, 12 total.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Palo Verde 1 None 951126 (continued)	The most probable root causes for the multiple valve failures to open were: - A lack of prudent actuator design margin. The low actuator margin resulted from using a non-conservative valve factor (0.3) in the original sizing. Recent tests on motor-operated gate valves in response to NRC Generic Letter 89-10 indicate that 0.3 is not a conservative valve factor for flex-wedge gate valves; - Not allowing for the potential effects of thermal binding in the original sizing of the actuator (Pressure locking and thermal binding of power-operated gate valves, including AOVs, isdiscussed in NRC Generic Letter 95-07. The DCFWIVs were excluded from the licensee's GL 95-07 evaluation because they are normally open valves.); and - Not allowing for the potential effects of degradation of the nitrogen supply in the calculation of actuator margin. Results from static diagnostic testing did not reveal why three of the DCFWIVs failed to open and the remaining AOV opened. A dynamic test program to establish the operability of these AOVs was conducted and valve factors of 0.64 and 0.57 were established. From the perspective of this study, the significance of these AOV failures was that the actuators were undersized and had been known to be undersized for some time. This, combined with thermal binding in the gate valves and marginal nitrogen pressure all contributed to the common-cause failure of the DCFWIVs. Another interesting point is that the licensee did not (and did not have to) report the DCFWIV failures to the NRC. The key safety functions of the DCFWIVs are to close upon actuation of a main-steam isolation signal (MSIS) and to open upon demand to establish a flow path from one of the auxiliary feedwater pumps to the steam generators when that pump is chosen to make up steam generator inventory during periods of hot standby and plant cooldown. These AOVs successfully closed on demand from the MSIS, thus satisfying their safety function. However, the AOVs failed to open following cooldown of the unit, and as such, the failures were not classified as safety-related failures. Also, the failures were considered by the licensee as not being Maintenance Rule functional failures. The failure mode that occurred was not considered by PVNGS to be safety significant or subject to review under the requirements of the Maintenance Rule. Three Fisher air-operated letdown/containment isolation valves were found to be undersized and the bench settings were also set too low. These valves were provided by the system vendor (ABB-CE) and the design basis evaluation was found to be incorrect. Modifications included spring replacement, reducing the stroke length of the actuator, and modifying the limit switches. Similar conditions were found at Units 2 and 3. (continued next page)	

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Palo Verde 1 None 951126 (continued)	The system vendor (ABB-CE) procured valves that had undersized air actuators and bench sets which were too low to provide desired valve seating force for the differential pressure which would be present across the valves during a letdown line break. They could not determine whether or not the procurement of the valves was approved by the Architect Engineer or the owner. The root cause of the design deficiency was attributed to the absence of a detailed design basis evaluation for air operated valves as part of an AOV program. This is considered to be a common-cause failure condition.	Three AOVs per unit (9 total).
Palo Verde 1 52895007 950512	Three Fisher air-operated letdown/containment isolation valves were found to be undersized and the bench settings were also set too low. These valves were provided by the system vendor (ABB-CE) and the design basis evaluation was found to be incorrect. Modifications included spring replacement, reducing the stroke length of the actuator, and modifying the limit switches. Similar conditions were found at Units 2 and 3. The system vendor (ABB-CE) procured valves that had undersized air actuators and bench sets which were too low to provide desired valve seating force for the differential pressure which would be present across the valves during a letdown line break. They could not determine whether or not the procurement of the valves was approved by the Architect Engineer or the owner. The root cause of the design deficiency was attributed to the absence of a detailed design basis evaluation for air-operated valves as part of an AOV program. This is considered to be a common-cause failure condition.	Three AOVs per unit (9 total).
Point Beach 1 26697014 970321	Engineers discovered a condition that alone could have prevented the Auxiliary Feedwater (AFW) System from automatically performing its safety-related function during design basis accidents involving a loss of instrument air and reduced steam generator pressures. A loss of instrument air during the accident would cause both motor-driven AFW pump (MDAFWP) flow control valves to fail open. The postulated event scenario is the result of a latent characteristic of the original AFW system design. The original design did not provide ample assurance that the MDAFWPs would automatically function during all design basis events. The expected fix was to furnish a reliable pneumatic supply to the control valves. Event No. 31995 refers. This is considered to be a common-cause failure condition.	Two AOVs.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Salem 1 27296012 960723	On July 23, 1996, a review determined that the keys/keyways on the actuators for the Residual Heat Removal (RHR) flow control valves (6 AOVs per unit) were subject to failure. The valves were made by Fisher Controls International, Model Type 656/7600. When using the simplified Fisher Catalog 14 methodology, the calculated maximum stem torque exceeds the vendor specified allowable torque. Preliminary calculations of the shaft torque and resulting average shear stresses in the valve stem key and keyway were also performed. The calculations showed that for normal operating conditions, the average shear stress in the key may exceed the material yield stress. The calculations also showed that although the average shear stress in the valve stem was estimated to be less than the minimum material yield stress, the fatigue life of the shaft keyway appeared to be limited to a low number of cycles. Key failures in the past are attributed to an overload during normal operation of these valves. A review of the original design revealed that these valves were installed with little or no design margin and the keys were likely to fail due to low cycle fatigue with stress levels exceeding yield strength. Corrective action was to replace the valves and review Fisher Model 7600 valves for similar concerns. This is considered to be a common-cause-failure condition.	Three AOVs per unit (6 total).
San Onofre 2 36196011 961216	On December 10, 1996, the licensee, in developing a test program for air operated valves, applied test equipment to containment isolation valve 2HV0513. It was likely that the actuator settings for valve 2HV0513 would not generate sufficient closing force to overcome internal pressure and packing drag under design basis conditions. The licensee concluded that valve 2HV0513 had been inoperable when Unit 2 was in Mode 1.The licensee concluded that either of two separate errors could have caused valve 2HV0513 to have insufficient closing force: (1) vendor setpoint methodology error, or (2) a deficiency in the reassembly procedure. The licensee committed to reset and retest valve 2HV0513 prior to returning Unit 2 to Mode 4. They also were to upgrade its maintenance procedures to conform to the current vendor manual. They performed an analysis on all air-operated containment isolation valves and confirmed that all other Units 2 and 3 air-operated containment isolation valves have sufficient actuator closing force to remain operable. The licensee also was to verify actuator settings for Unit 2 air-operated containment isolation valves which may not have had their actuator settings established in accordance with the vendor recommendations. Similar actions were to have been completed for Unit 3 during its next refueling outage.	One AOV each in units 2 and 3 (2 total).

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
San Onofre 2 (and 3) 36199003 990210	On February 7, 1999, while Unit 2 was shut down for its Cycle 10 refueling outage, the Component Cooling Water Non-Critical Loop (CCW-NCL) isolation valves failed to satisfy close stroke time requirements during a pneumatic supply test. Subsequent investigations determined that the corresponding valves in Unit 3 might not satisfy stroke time requirements. On February 10, 1999, the non-critical loop valves at Unit 3 were declared inoperable and the unit entered a 72-hour action statement. The cause of this condition was that instrument air failure modes other than a loss of pressure were not adequately considered in the original design of the CCW-NCL valves. Design modifications were implemented on both units to correct a previously unrecognized condition associated with a blockage of the non-seismic pneumatic supply. The San Onofre Nuclear Generating Station (SONGS) Component Cooling Water System (CCW) consists of two independent critical loops (trains) and one non-critical loop (NCL). All three loops are interconnected, such that the NCL can be aligned to either one of the critical loops. A third swing pump can be aligned to either CCW train. The safety function of CCW is to transfer the combined heat load from safety-related systems and components to the Salt Water Cooling System (SWC) during normal and accident conditions, including seismic events. The CCW-NCL isolation valves (HV-6212 and 6218 for Train A, and HV-6213 and HV-6219 for Train B) are designed to close upon receipt of a containment isolation signal or a low-low level signal in the associated CCW train surge tank. These pneumatically actuated valves (28-inch Fisher Type 9241) are each required to close upon loss of pneumatic supply and/or loss of control power to its solenoid valve, and are equipped with twin air receivers to ensure valve closure. The air receivers are aligned to the valve actuator by a pneumatic trip valve that actuates on low supply pressure. The pneumatic system incorporates a 4-way solenoid supply/exhaust valve, two air receiver units, and a pressure actuated trip valve. Normal pneumatic pressure is provided by the instrument air system. De-energizing the solenoid valve vents the top chamber of the piston actuator and supplies air to the bottom chamber through the normal supply port of the trip valve. The differential pressure across the actuator piston causes it to move, thereby closing the valve. The valve actuator is designed to fail in the closed position upon loss of normal supply pressure or loss of control power. The air receivers provide for valve closure upon a loss of air/nitrogen pressure. If the supply pressure drops to less than 75% of the pressure in the air receivers, the differential pressure created across the trip valve piston moves to block the normal supply port and open a flow path from the air receivers to the actuator. If the pneumatic supply pressure remains above this set point, the trip valve allows normal automatic operation with the solenoid valve. In 1995, the non-safety-related nitrogen lines were connected locally to the non-safety-related air supply to increase reliability of cooling to the reactor coolant pump seals. This provided a backup supply to preclude spurious CCW-NCL valve closure. This modification included installation of check valves in both the air and nitrogen systems to isolate one system from another. (continued next page)	Four AOVs per unit (8 total).

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
San Onofre 2 (and 3) 36199003 990210 (continued)	In October 1997, the licensee recognized that CCW-NCL supply and return valves were required to close within a certain sequence and time band to preclude a potential water hammer during postulated accident conditions. In the fall of 1998, the licensee recognized that existing test records did not document the valve closure times using the accumulators only. Testing to obtain these valve closure times was completed during the Unit 2 Cycle 10 refueling outage in early 1999. During the Cycle 10 refueling outage, the closure function of the Unit 2 Train A CCW-NCL isolation valves was tested by various modes of closure. Results using the plant nitrogen and the instrument air system were satisfactory. Closure via the air receivers was tested by isolating the air and nitrogen to the valve actuator and then de-energizing the supply solenoid. During this test, 2HV6212 did not move for several minutes and 2HV6218 closed in approximately 47 seconds. These stroke times exceeded the valve's operability criteria. Follow-up investigation determined that with the pneumatic supply isolated, but with the supply line still pressurized, the trip valve in the actuator pneumatic system would not reliably actuate. With the pneumatic supply isolated, de-energizing the solenoid valve causes the supply volume, trapped upstream of the solenoid valve, to expand into the volume downstream of the solenoid valve. The resulting pressure depends on the volume trapped upstream of the solenoid relative to the volume downstream. The valve will stroke if the trapped upstream volume is either small compared to the downstream volume or very large compared to the downstream volume. If the trapped upstream volume is small compared to the downstream volume, the resulting pressure will be low enough to fully actuate the trip valve. If the trapped volume falls between these limits, the pressure will drop to, or close to, the trip valve set point. At this pressure, the trip valve piston may only partially move, blocking both the normal and accumulator air supply to the NCL valve actuator. As such, the valve may not be able to perform its safety function under all design basis conditions. Based on the results of the investigation described above, the licensee concluded that all 8 CCW-NCL valves could potentially be affected.	

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Shearon Harris 40098001 980109	On January 9, 1998, a condition was identified during operation that results in the plant being potentially outside it's design basis. Specifically, a potential failure mechanism exists where a leak in the non-safety Instrument Air System could result in the inoperability of the Steam Generator Pre-heater Bypass Isolation Valves. These valves are safety-related containment isolation valves that are required by plant procedures to automatically shut in 10 seconds or less upon receipt of a Main Feedwater Isolation Signal. These valves are positioned by a pneumatic piston-operated actuator which is supplied by the non-safety-related Instrument Air System. They are designed to automatically close if control air supply is lost. An air leak was postulated in the Instrument Air system that could possibly reduce the air inlet pressure to just low enough to affect proper operation of the actuator's 3-way and 4-way pilot valves and not be detected by Operations personnel. If this occurred, the pilot valves would shuttle, causing the accumulator pressure to bleed off, which would prevent the valves from closing as required. This potential scenario constituted operation outside the design basis of the plant and was reported to the NRC via the emergency notification system on January 9, 1998, at 1450 hours. The cause of this condition was inadequate design control during development of a plant modification implemented in August 1984 in response to NRC Information Notice 82-25. The investigation for this event also revealed several other missed opportunities to identify this condition during subsequent plant modifications and/or related evaluations.	Three AOVs.
Summer 39598009 981006	It was discovered that there was an issue concerning the ability of a non-safety-related component to affect the operation of the Turbine Driven Emergency Feedwater Pump (TDEFWP). This non-safety- related device (ISY-02034) is a current to pneumatic (I/P) transducer used in the speed control circuitry for the TDEFWP. The licensee conducted an engineering evaluation of the failure of the TDEFWP to achieve rated speed and concluded that there were failure mechanisms that could have prevented the TDEFWP from performing its design function under postulated accident conditions outside containment. The I/P had previously been thought to only fall (on loss of air or power) to full speed. During a steam line break coincident with the loss of offsite power, and a loss of "B" train DC power, there may not have been sufficient EFW flow to the non-faulted steam generators to mitigate the accident. The unanalyzed condition Is the potential failure mechanisms/modes for the I/P converter which allow it to fail in a position which corresponds to less than rated speed. The I/P could fail in such a way that the TDEFWP speed never reaches the required speed and flow. This was determined during the engineering evaluation for condition report 98-0823 after the TDEFW did not achieve rated speed due to internal contamination of the I/P converter. The cause of this event was the belief that the I/P could only fail due to loss of air or power. As a result of this belief, no other failure modes were considered credible. This was an engineering oversight dating back prior to the receipt of the plant operating license. Other I/Ps in the plant were evaluated by the licensee. The licensee discovered that the non-qualified sub-components could potentially fail nonconservatively in a high radiation field that will occur after an accident during the recirculation or cooldown phase.	The LER lists "multiple" flow control valves and an I/P controller.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Three Mile Island 1	Potentially insufficient margins in two AOVs were described by the TMI engineers during the visit for this AOV study. Calculations for five Crane-Aloyco, 2.5 to 6 inch, 1500 and 150 pound class, flex-wedge and split wedge gate AOVs with Miller DA-63-B and A-63-B cylinder actuators were reviewed by one of the TMI engineers. The architect/engineer (AE) requested that the valve manufacturer (Crane-Aloyco) perform thrust calculations on the five AOVs, as part of a limit switch upgrade modification, in order to verify that limit switch installation would not affect valve operability. The resultant thrust calculations, using "present day" methodology, indicated that two of the five valves had negative closing margins for the specified differential pressure (d/P). The AE then requested that the manufacturer re-perform the calculations using the methodology by which the valves were originally sized. The revised Crane-Aloyco calculations, based on the original valve sizing methodology, indicated positive margin in both the opening and closing directions, but using a valve factor of zero. Upon review of the Crane-Aloyco calculations, TMI convened a review group. This group verified that the two containment isolation valves, which had negative margins in the manufacturers' first calculations, were, in fact, operable and would be able to perform their designed safety function because the required operating conditions were less severe than the original design basis conditions. This conclusion was based on TMI's calculations using a 0.75 friction factor and d/P of 1600 psi (that required for containment isolation).	Five AOVs.
Vermont Yankee 27198025 (Rev. 2) 981211	On 12/11/98, with the plant at 100% power, it was determined that one Scram Discharge Volume (SDV) drain valve (CRD-LCV-33B) did not meet the stroke time requirements of the In-Service Test (IST) Program. Both the North and the South SDVs have two drain valves in series, CRD-LCV-33A/C on the North and CRD-LCV-33B/D on the South. The drain valve, CRD-LCV-33B, was subsequently declared inoperable. The SDVs are used to limit the loss of and contain the reactor vessel water from all control rod drives during a scram. These volumes are provided in the scram discharge header. During normal plant operation, the volumes are empty with all the drain and vent valves open. Upon receipt of a scram signal, the vent and drain valves close. While scrammed, the control rod drive seal leakage continues to flow to the discharge volumes until the discharge volume equals reactor pressure. Following a scram, when the scram signals are cleared and the Reactor Protection System (RPS) logic is manually reset, the vent and drain valves are opened and the SDV drained. According to the LER, the root causes of this event were inadequate actuator sizing calculations (performed by the vendor) under the design conditions specified in the procurement specification, and inadequate manufacturing Quality Assurance controls to ensure specification requirements were maintained. Contributing causes included conflicting information in the design package that was not identified by Vermont Yankee, and closing forces of the actuator at either end of the valve stroke were apparently not as designed. (continued next page)	Four AOVs.

Table 7. (continued).

Plant, LER No., and Event Date	Description	Number of AOVs Involved in the Event
Vermont Yankee 27198025 981211 (continued)	(continued from previous page) The actuators and valves were installed under Engineering Design Change EDCR 97410 in April of 1998 and were sized in accordance with the vendor's recommendations. These drain valves are required to go shut on a scram signal to isolate the SDV and act as a primary containment isolation valve when the scram valves are open. This issue was addressed by the installation of larger actuators, on December 21, 1998, via the Minor Modification process, which were properly sized to operate the valves under any design conditions at Vermont Yankee. The licensee stated that the LER constituted a Part 21 notification in accordance with 10CFR21.2(c) and NUREG 1022, Revision 1. Further, the licensee informed NRC by Event 35150 dated December 14, 1998, that this event was considered to be a common-cause failure.	

NOTES:

1. The NRC Licensee Event Report (LER) Number consists of the three-digit NRC Docket Number for the plant at which the event occurred, the last two digits of the year in which the LER was generated, and a three-digit sequential number of the LER. This is consistent with the NRC's Sequence Code Search System (SCSS) database designation. The LER system is described in 10 CFR 50.73.

Appendix A

NRC Generic Communications Related to Air-Operated Valves

Appendix A

NRC Generic Communications Related to Air-Operated Valves

1. NRC GENERIC COMMUNICATIONS RELATED TO AIR-OPERATED VALVES

2. Bulletin 71-001 - PERFORMANCE OF THE MAIN STEAM ISOLATION VALVES

3. Bulletin 73-002 - MALFUNCTION OF CONTAINMENT PURGE SUPPLY VALVE SWITCH

4. Bulletin No. 75-03 - INCORRECT LOWER DISC SPRING AND CLEARANCE DIMENSION IN SERIES 8300 AND 8302 ASCO SOLENOID VALVES

5. Bulletin 76-006 - DIAPHRAGM FAILURES IN AIR OPERATED AUXILIARY ACTUATORS FOR SAFETY/RELIEF VALVES

6. Bulletin 78-004 - ENVIRONMENTAL QUALIFICATION OF CERTAIN STEM MOUNTED LIMIT SWITCHES INSIDE REACTOR CONTAINMENT

7. Bulletin 78-014 - DETERIORATION OF BUNA-N COMPONENTS IN ASCO SOLENOIDS

8. Bulletin 79-001A - ENVIRONMENTAL QUALIFICATION OF CLASS 1E EQUIPMENT (DEFICIENCIES IN THE ENVIRONMENTAL QUALIFICATION OF ASCO SOLENOID VALVES

9. Bulletin 79-001B - ENVIRONMENTAL QUALIFICATION OF CLASS IE EQUIPMENT

10. Bulletin 79-001B Sup. 2 - ENVIRONMENTAL QUALIFICATION OF CLASS 1E EQUIPMENT

11. Bulletin 80-001 - OPERABILITY OF ADS VALVE PNEUMATIC SUPPLY

12. Bulletin 80-014 - DEGRADATION OF BWR SCRAM DISCHARGE VOLUME CAPABILITY

13. Bulletin 80-016 - POTENTIAL MISAPPLICATION OF ROSEMOUNT INC. MODELS 1151 AND 1152 PRESSURE TRANSMITTERS WITH EITHER "A" OR "D" OUTPUT CODES

14. Bulletin 80-017 - FAILURE OF 76 OF 185 CONTROL RODS TO FULLY INSERT DURING A SCRAM AT A BWR

15. Bulletin 80-017 Sup. 2 - FAILURES REVEALED BY TESTING SUBSEQUENT TO FAILURE OF CONTROL RODS TO INSERT DURING A SCRAM AT A BWR

16. Bulletin 80-017 Sup. 3 - FAILURE OF CONTROL RODS TO INSERT DURING A SCRAM AT A BWR

17. Bulletin 80-017 Sup. 4 - FAILURE OF CONTROL RODS TO INSERT DURING A SCRAM AT A BWR

18. Bulletin 80-023 - FAILURES OF SOLENOID VALVES MANUFACTURED BY VALCOR ENGINEERING CORPORATION

19. Bulletin 80-025 - OPERATING PROBLEMS WITH TARGET ROCK SAFETY-RELIEF VALVES AT BWRs

20. Bulletin 86-01 - MINIMUM FLOW LOGIC PROBLEMS THAT COULD DISABLE RHR PUMPS

21. Bulletin 86-03 - POTENTIAL FAILURE OF MULTIPLE ECCS PUMPS DUE TO SINGLE FAILURE OF AIR-OPERATED VALVE IN MINIMUM FLOW LINE

22. Bulletin 88-04 - POTENTIAL SAFETY-RELATED PUMP LOSS

23. Circular 79-018 - PROPER INSTALLATION OF TARGET ROCK SAFETY-RELIEF

24. Circular 79-022 - STROKE TIMES FOR POWER OPERATED RELIEF VALVES

25. Circular 80-008 - BWR TECHNICAL SPECIFICATION INCONSISTENCY - RPS RESPONSE TIME

26. Circular 80-015 - LOSS OF REACTOR COOLANT PUMP COOLING AND NATURAL CIRCULATION COOLDOWN

27. Circular 81-014 - MAIN STEAM ISOLATION VALVE FAILURES TO CLOSE

28. Circular 81-015 - UNNECESSARY RADIATION EXPOSURES TO THE PUBLIC AND WORKERS DURING EVENTS INVOLVING THICKNESS AND LEVEL

29. Generic Letter 79-024 - INADVERTENT REACTOR SCRAM AND SAFETY INJECTION DURING MONTHLY SURVEILLANCE TESTS OF THE SAFEGUARDS

30. Generic Letter 79-045 - TRANSMITTAL OF REPORTS REGARDING FOREIGN REACTOR OPERATING EXPERIENCES

31. Generic Letter 79-046 - CONTAINMENT PURGING AND VENTING DURING NORMAL OPERATION - GUIDELINES FOR VALVE OPERABILITY

32. Generic Letter 79-054 - CONTAINMENT PURGING AND VENTING DURING NORMAL OPERATION

33. Generic Letter 80-004 - OPERABILITY OF ADS VALVE PNEUMATIC SUPPLY

34. Generic Letter 80-012 - ANALYSIS OF A PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

35. Generic Letter 80-029 - MODIFICATIONS TO BOILING WATER REACTOR CONTROL ROD DRIVE SYSTEMS

36. Generic Letter 80-097 - IE Bulletin NO. 80023: FAILURES OF SOLENOID VALVES MANUFACTURED BY VALCOR ENGINEERING CORPORATION

37. Generic Letter 81-009 - BWR SCRAM DISCHARGE SYSTEM

38. Generic Letter 81-014 - SEISMIC QUALIFICATION OF AUXILIARY FEEDWATER SYSTEMS

39. Generic Letter 81-021 - NATURAL CIRCULATION COOLDOWN

40. Generic Letter 87-012 - LOSS OF RESIDUAL HEAT REMOVAL (RHR) WHILE THE REACTOR COOLANT SYSTEM (RCS) IS PARTIALLY FILLED

41. Generic Letter 88-014 - INSTRUMENT AIR SUPPLY SYSTEM PROBLEMS AFFECTING SAFETY- RELATED EQUIPMENT

42. Generic Letter 88-017 - LOSS OF DECAY HEAT REMOVAL

43. Generic Letter 90-006 - RESOLUTION OF GENERIC ISSUE 70, "POWER-OPERATED RELIEF VALVE AND BLOCK VALVE RELIABILITY," AND GENERIC ISSUE 94, "ADDITIONAL LTOP FOR LWRs"

44. Generic Letter 91-015 - OPERATING EXPERIENCE FEEDBACK REPORT, SOLENOID-OPERATED VALVE PROBLEMS AT U.S. REACTORS

45. Generic Letter 95-07 - PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES

46. Generic Letter 96-05 - PERIODIC VERIFICATION OF DESIGN-BASIS CAPABILITY OF SAFETY-RELATED MOTOR-OPERATED VALVES

47. Notice 79-04 - DEGRADATION OF ENGINEERED SAFETY FEATURES

48. Notice 79-08 - INTERCONNECTION OF CONTAMINATED SYSTEMS WITH SERVICE AIR SYSTEMS USED AS THE SOURCE OF BREATHING AIR

49. Notice 79-27 - STEAM GENERATOR TUBE RUPTURES AT TWO POWER PLANTS

50. Notice 80-11 -GENERIC PROBLEMS WITH ASCO VALVES IN NUCLEAR APPLICATIONS INCLUDING FIRE PROTECTION SYSTEMS SSINS No. 6870

51. Notice 80-16 - SHAFT SEAL PACKING CAUSES BINDING IN MAIN STEAM SWING DISC CHECK AND ISOLATION VALVES

52. Notice 80-39 - MALFUNCTION OF SOLENOID VALVES MANUFACTURED BY VALCOR ENGINEERING CORPORATION

53. Notice 80-40 - EXCESSIVE NITROGEN SUPPLY PRESSURE ACTUATES SAFETY-RELIEF VALVE OPERATION TO CAUSE REACTOR DEPRESSURIZATION

54. Notice 81-12 - GUIDANCE ON ORDER ISSUED JANUARY 9, 1981, REGARDING AUTOMATIC CONTROL ROD INSERTION ON LOW CONTROL AIR PRESSURE

55. Notice 81-14 - POTENTIAL OVERSTRESS OF SHAFTS ON FISHER SERIES 9200 BUTTERFLY VALVES WITH EXPANDABLE T RINGS

56. Notice 81-15 - DEGRADATION OF AUTOMATIC ECCS ACTUATION CAPABILITY BY ISOLATION OF INSTRUMENT LINES

57. Notice 81-27 - FLAMMABLE GAS MIXTURES IN THE WASTE GAS DECAY TANKS IN PWR PLANTS

58. Notice 81-29 - EQUIPMENT QUALIFICATION TESTING EXPERIENCE

59. Notice 81-38 - POTENTIALLY SIGNIFICANT EQUIPMENT FAILURES RESULTING FROM CONTAMINATION OF AIR-OPERATED SYSTEMS

60. Notice 82-17 - OVER PRESSURIZATION OF REACTOR COOLANT SYSTEM

61. Notice 82-19 - LOSS OF HIGH HEAD SAFETY INJECTION EMERGENCY BORATION AND REACTOR COOLANT MAKEUP CAPABILITY

62. Notice 82-25 - FAILURES OF HILLER ACTUATORS UPON GRADUAL LOSS OF AIR PRESSURE

63. Notice 82-45 - PWR LOW TEMPERATURE OVERPRESSURE PROTECTION

64. Notice 82-52 - EQUIPMENT ENVIRONMENTAL QUALIFICATION TESTING EXPERIENCE - UPDATING OF TEST SUMMARIES PREVIOUSLY PUBLISHED IN INFORMATION NOTICE 81-29

65. Notice 83-57 - POTENTIAL MISASSEMBLY PROBLEM WITH AUTOMATIC SWITCH COMPANY (ASCO) SOLENOID NP 8316

66. Notice 83-70 - VIBRATION-INDUCED VALVE FAILURES

67. Notice 83-70 Supplement 1 - VIBRATION INDUCED VALVE FAILURES

68. Notice 84-04 - FAILURE OF ELASTOMER SEATED BUTTERFLY VALVES USED ONLY DURING COLD SHUTDOWN

69. Notice 84-12 - FAILURE OF SOFT SEAT VALVE SEALS

70. Notice 84-23 - RESULTS OF THE NRC-SPONSORED QUALIFICATION METHODOLOGY RESEARCH TEST ON ASCO SOLENOID VALVES

71. Notice 84-31 - INCREASED STROKING TIME OF BETTIS ACTUATORS BECAUSE OF SWOLLEN ETHYLENE-PROPYLENE RUBBER SEALS AND SEAL SET

72. Notice 84-48 - FAILURES OF ROCKWELL INTERNATIONAL GLOBE VALVES

73. Notice 84-48 Supplement 1 - FAILURES OF ROCKWELL INTERNATIONAL GLOBE VALVES

74. Notice 84-53 - INFORMATION CONCERNING THE USE OF LOCTITE 242 AND OTHER ANAEROBIC ADHESIVE/SEALANTS

75. Notice 84-68 - POTENTIAL DEFICIENCY IN IMPROPERLY RATED FIELD WIRING TO SOLENOID VALVES

76. Notice 84-74 - ISOLATION OF REACTOR COOL LOW-PRESSURE SYSTEMS OUTS

77. Notice 84-81 - INADVERTENT REDUCTION IN PRIMARY COOLANT INVENTORY IN BOILING WATER REACTORS DURING SHUTDOWN AND STARTUP

78. Notice 85-06 - CONTAMINATION OF BREATHING AIR SYSTEMS

79. Notice 85-08 - INDUSTRY EXPERIENCE ON CERTAIN MATERIALS USED IN SAFETY-RELATED EQUIPMENT

80. Notice 85-17 - POSSIBLE STICKING OF ASCO SOLENOID VALVES

81. Notice 85-17 Supplement 1 - POSSIBLE STICKING OF ASCO SOLENOID VALVES

82. Notice 85-21 - MAIN STEAM ISOLATION VALVE CLOSURE LOGIC

83. Notice 85-26 - VACUUM RELIEF SYSTEM FOR BOILING WATER REACTOR MARK I AND MARK II CONTAINMENTS

84. Notice 85-27 - NOTIFICATIONS TO THE NRC OPERATIONS CENTER AND REPORTING EVENTS IN LICENSEE EVENT REPORTS

85. Notice 85-35 - FAILURE OF AIR CHECK VALVES TO SEAT

86. Notice 85-35 Supplement 1 - FAILURE OF AIR CHECK VALVES TO SEAT

87. Notice 85-47 - POTENTIAL EFFECT OF LINE-INDUCED VIBRATION ON CERTAIN TARGET ROCK SOLENOID-OPERATED VALVES

88. Notice 85-59 - VALVE STEM CORROSION FAILURES

89. Notice 85-67 - VALVE-SHAFT-TO-ACTUATOR KEY MAY FALL OUT OF PLACE WHEN MOUNTED BELOW HORIZONTAL AXIS

90. Notice 85-72 - UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE CONTAINMENT

91. Notice 85-75 - IMPROPERLY INSTALLED INSTRUMENTATION, INADEQUATE QUALITY CONTROL AND INADEQUATE POSTMODIFICATION TESTING

92. Notice 85-84 - INADEQUATE INSERVICE TESTING OF MAIN STEAM ISOLATION VALVES

93. Notice 85-95 - LEAK OF REACTOR WATER TO REACTOR BUILDING CAUSED BY SCRAM SOLENOID VALVE PROBLEM

94. Notice 85-100 - ROSEMOUNT DIFFERENTIAL PRESSURE TRANSMITTER ZERO POINT SHIFT

95. Notice 86-09 - FAILURE OF CHECK AND STOP CHECK VALVES SUBJECTED TO LOW FLOW CONDITIONS

96. Notice 86-16 - FAILURES TO IDENTIFY CONTAINMENT LEAKAGE DUE TO INADEQUATE LOCAL TESTING OF BWR VACUUM RELIEF SYSTEM VALVES

97. Notice 86-50 - INADEQUATE TESTING TO DETECT FAILURES OF SAFETY-RELATED PNEUMATIC COMPONENTS OR SYSTEMS

98. Notice 86-51 - EXCESSIVE PNEUMATIC LEAKAGE IN THE AUTOMATIC DEPRESSURIZATION SYSTEM

99. 86-57 - OPERATING PROBLEMS - VALVES AT NUCLEAR POWER PLANTS

100. Notice 86-72 - FAILURE 17-7 PH STAINLESS STEEL SPRINGS IN VALCOR VALVES DUE TO HYDROGEN EMBRITTLEMENT

101. Notice 86-78 - SCRAM SOLENOID PILOT VALVE (SSPV) REBUILT KIT PROBLEMS

102. Notice 86-82 - FAILURES OF SCRAM DISCHARGE VOLUME VENT AND DRAIN VALVES

103. Notice 86-82 Rev. 1 - FAILURES OF SCRAM DISCHARGE VOLUME VENT AND DRAIN VALVES

104. Notice 86-109 - DIAPHRAGM FAILURE IN SCRAM OUTLET VALVE CAUSING ROD INSERTION

105. Notice 87-02 - INADEQUATE SEISMIC QUALIFICATION OF DIAPHRAGM VALVES BY MATHEMATICAL MODELING AND ANALYSIS

106. Notice 87-17 - RESPONSE TIME OF SCRAM INSTRUMENT VOLUME LEVEL DETECTORS

107. Notice 87-28 - AIR SYSTEMS PROBLEMS AT U.S. LIGHT WATER REACTORS

108. Notice 87-28 Sup. 1 - AIR SYSTEMS PROBLEMS AT U.S. LIGHT WATER REACTORS

109. Notice 87-38 - INADEQUATE OR INADVERTENT BLOCKING OF VALVE MOVEMENT

110. Notice 87-48 - INFORMATION CONCERNING THE USE OF ANAEROBIC ADHESIVE/SEALANTS

111. Notice 88-24 - FAILURES OF AIR-OPERATED VALVES AFFECTING SAFETY-RELATED SYSTEMS

112. Notice 88-43 - SOLENOID VALVE PROBLEMS

113. Notice 88-51 - FAILURES OF MAIN STEAM ISOLATION VALVES

114. Notice 88-86 - OPERATING WITH MULTIPLE GROUNDS IN DIRECT CURRENT DISTRIBUTION SYSTEMS

115. Notice 88-86, Supplement 1 - OPERATING WITH MULTIPLE GROUNDS IN DIRECT CURRENT DISTRIBUTION SYSTEMS

116. Notice 88-94 - POTENTIALLY UNDERSIZED VALVE ACTUATORS

117. Notice 88-97 - POTENTIALLY SUBSTANDARD VALVE REPLACEMENT PARTS

118. Notice 88-97 Sup. 1 - POTENTIALLY SUBSTANDARD VALVE REPLACEMENT PARTS

119. Notice 89-07 - FAILURES OF SMALL-DIAMETER TUBING IN CONTROL AIR, FUEL OIL, AND LUBE OIL SYSTEMS WHICH RENDER EMERGENCY DIESEL

120. Notice 89-26 - INSTRUMENT AIR SUPPLY TO SAFETY-RELATED EQUIPMENT

121. Notice 89-28 - WEIGHT AND CENTER OF GRAVITY DISCREPANCIES FOR COPES-VULCAN AIR-OPERATED VALVES

122. Notice 89-30 - HIGH TEMPERATURE ENVIRONMENTS AT NUCLEAR POWER PLANTS

123. Notice 89-38 - ATMOSPHERIC DUMP VALVE FAILURES AT PALO VERDE UNITS 1, 2, AND 3

124. Notice 89-52 - POTENTIAL FIRE DAMPER OPERATIONAL PROBLEMS

125. Notice 89-54 - POTENTIAL OVER PRESSURIZATION OF THE COMPONENT COOLING WATER SYSTEM

126. Notice 89-66 - QUALIFICATION LIFE OF SOLENOID VALVES

127. Notice 90-06 - POTENTIAL FOR LOSS OF SHUTDOWN COOLING WHILE AT LOW REACTOR COOLANT LEVELS

128. Notice 90-11 - MAINTENANCE DEFICIENCY ASSOCIATED WITH SOLENOID-OPERATED VALVES

129. Notice 90-17 - WEIGHT AND CENTER OF GRAVITY DISCREPANCIES FOR COPES-VULCAN VALVES

130. Notice 90-18 - POTENTIAL PROBLEMS WITH CROSBY SAFETY RELIEF VALVES USED ON DIESEL GENERATOR AIR START RECEIVER TANKS

131. Notice 90-64 - POTENTIAL FOR COMMON-MODE FAILURE OF HIGH PRESSURE SAFETY INJECTION PUMPS OR RELEASE OF REACTOR COOLANT OUTSIDE CONTAINMENT DURING A LOCA

132. Notice 91-58 - DEPENDENCY OF OFFSET DISC BUTTERFLY VALVE'S OPERATION ON ORIENTATION WITH RESPECT TO FLOW

133. Notice 91-83 - SOLENOID-OPERATED VALVE FAILURES RESULTED IN TURBINE OVERSPEED

134. Notice 92-60 - VALVE STEM FAILURE CAUSED BY EMBRITTLEMENT

135. Notice 92-64 - NOZZLE RING SETTINGS ON LOW PRESSURE WATER-RELIEF VALVES

136. Notice 92-67 - DEFICIENCY IN DESIGN MODIFICATIONS TO ADDRESS FAILURES OF HILLER ACTUATORS UPON A GRADUAL LOSS OF AIR PRESSURE

137. Notice 93-35 - INSIGHTS FROM COMMON-CAUSE FAILURE EVENTS

138. Notice 94-06 - POTENTIAL FAILURE OF LONG-TERM EMERGENCY NITROGEN SUPPLY FOR THE AUTOMATIC DEPRESSURIZATION SYSTEM VALVES

139. Notice 94-25 - FAILURE OF CONTAINMENT SPRAY HEADER VALVE TO OPEN DUE TO EXCESSIVE PRESSURE FROM INERTIAL EFFECTS OF WATER

140. Notice 94-44 - MAIN STEAM ISOLATION VALVE FAILURE TO CLOSE ON DEMAND BECAUSE OF INADEQUATE MAINTENANCE AND TESTING

141. Notice 94-55 - PROBLEMS WITH COPES-VULCAN PRESSURIZER POWER- OPERATED RELIEF VALVES

142. Notice 94-61 - CORROSION OF WILLIAM POWELL GATE VALVE DISC HOLDERS

143. Notice 94-61 Sup. 1 - CORROSION OF WILLIAM POWELL GATE VALVE DISC HOLDERS

144. Notice 94-71 - DEGRADATION OF SCRAM SOLENOID PILOT VALVE PRESSURE AND EXHAUST DIAPHRAGMS

145. Notice 95-25 - VALVE FAILURE DURING PATIENT TREATMENT WITH GAMMA STEREOTACTIC RADIOSURGERY UNIT

146. Notice 95-34 - AIR ACTUATOR AND SUPPLY AIR REGULATOR PROBLEMS IN COPES-VULCAN PRESSURIZER POWER-OPERATED RELIEF VALVES

147. Notice 95-47 - UNEXPECTED OPENING OF A SAFETY/RELIEF VALVE AND COMPLICATIONS INVOLVING SUPPRESSION POOL COOLING STRAINER

148. Notice 95-47 Rev. 1 - UNEXPECTED OPENING OF A SAFETY/RELIEF VALVE AND COMPLICATIONS INVOLVING SUPPRESSION POOL COOLING STRAINER

149. Notice 95-53 - FAILURES OF MAIN STEAM ISOLATION VALVES AS A RESULT OF STICKING SOLENOID PILOT VALVES

150. Notice 96-02 - INOPERABILITY OF POWER-OPERATED RELIEF VALVES MASKED BY DOWNSTREAM INDICATIONS DURING TESTING

151. Notice 96-07 - SLOW FIVE PERCENT SCRAM INSERTION TIMES CAUSED BY VITON DIAPHRAGMS IN SCRAM SOLENOID PILOT VALVES

152. Notice 96-42 - UNEXPECTED OPENING OF MULTIPLE SAFETY RELIEF VALVES

153. Notice 96-48 - MOTOR-OPERATED VALVE PERFORMANCE ISSUES (Includes a discussion of the EPRI MOV Performance Prediction Program, which, according to EPRI is applicable to valves with air-operators)

154. Notice 96-68 - INCORRECT EFFECTIVE DIAPHRAGM AREA VALUES IN VENDOR MANUAL RESULT IN POTENTIAL FAILURE OF PNEUMATIC DIAPHRAGM

155. Notice 97-07 - PROBLEMS IDENTIFIED DURING GENERIC LETTER 89-10 CLOSEOUT INSPECTIONS

Appendix B

Air-Operated Valve Event Significance Analysis

Appendix B

Air-Operated Valve Event Significance Analysis

PURPOSE AND BACKGROUND

The purpose of this analysis was to use precursor data to determine the relative significance that operational events involving air-operated valves (AOVs) or AOV-related support systems contributed to the potential for core damage accident sequences.

The precursor results documented in the NRC-sponsored Accident Sequence Precursor Program (ASP) reports (NUREG/CR-4674, "Precursors to Potential Severe Damage Accidents...") for the time period 1984 through 1995 were used to determine AOV event significance. The ASP program involves the systematic review and evaluation of operational events or conditions that have occurred at U.S. commercial light-water reactors, reported under the requirements of 10 CFR 50.72 and 50.73.

The ASP Program uses simplified risk models to provide estimates of operating event significance in terms of the potential for inadequate core cooling and subsequent severe core damage, i.e., as an accident sequence precursor. Identification of an operational event as an accident sequence "precursor" does not of itself imply that a significant potential for core damage existed. It does mean that at least one of a series of protective features designed to prevent core damage was compromised.

An estimate of the likelihood of core damage is then obtained using PRA techniques, conditional on the occurrence of the operational event and the estimated failure probabilities of the remaining mitigative features. This estimate of the likelihood of core damage given such an operational event or condition is defined as the conditional core damage probability (CCDP).

The ASP Program provides a methodology to document and evaluate those operational events or conditions that were determined to be risk-significant. The primary result of the ASP Program is the identification of operational events and conditions with CCDP's greater than or equal to 1.0×10^{-6} (1E-6) that satisfy at least one of four precursor screening criteria:

- A core damage initiator requiring safety system response,
- The failure of a complete system required to mitigate the consequences of a core damage initiator,
- Degradation of more than one system required for mitigation, or
- A trip or loss of feedwater with a degraded mitigating system.

In general, for the nuclear power industry, an accident sequence precursor is a sequence of events that did not progress to core damage, but if additional failures had occurred, would have resulted in inadequate core cooling and subsequent core damage. Those events with conditional probabilities of subsequent severe core damage greater than or equal to 1E-6 are identified and documented as precursors.

Events or conditions originally identified through screening as potential precursors are subjected to an initial engineering evaluation. Generally, this evaluation eliminates events from further consideration as precursors if, for example, they involve only:

- A component failure with no loss of redundancy,
- A short-term loss of redundancy in only one system,
- A structural degradation,
- An event that occurred prior to initial facility startup,
- A design error discovered by re-analysis, or
- An event with no appreciable impact on safety systems.

Events identified for further consideration and detailed analysis include:

- Unexpected initiators,
- All events in which an initiating event occurred and a safety-related component failed,
- All support system failures,
- Any event in which two or more failures occurred, or
- Any event or operating condition that was not predicted or that proceeded differently from the plant design basis.

If an AOV-related event was potentially a contributor to a risk significant conditional core damage sequence (i.e., CCDP greater than or equal to 1E-6), and met the precursor criteria, there was a high likelihood (but not a certainty) that it was identified as a selected precursor event that was analyzed as part of the ASP program for the year in which it occurred. Although the simplified nature of the ASP models and reporting criteria for LERs result in the omission or exclusion of many plant AOVs (and thus, many AOV failures will not have been evaluated), some AOV failures were included in the ASP analyses. Evaluation of these AOV-related events provides a useful starting point to understand the risk significance of AOV failures.

RESULTS OF THE ANALYSIS

Table 1 provides a summary of the findings. Events with CCDPs greater than or equal to 1E-4 have traditionally been considered to be the most important in the ASP Program. Only one event exceeded this range in 1995. Events with CCDPs less than 1E-6 are considered as events of low significance and are not considered in the ASP methodology. There were 288 precursors that had CCDPs greater than or equal to 1E-6 between 1984 and 1995, and 26 of these were AOV-related. There were 89 precursors that had CCDPs greater than or equal to 1E-4 between 1984 and 1995, and 12 of these were AOV-related.

The majority of AOV-related events during this time period involved loss of main feedwater, requiring either automatic or manual trip of the reactor including:

- Several events involving loss of feedwater have occurred due to problems with the instrument air supply to air-operated main feedwater regulating and control valves.
- Loss of feedwater control occurred due to electrical failures in air-operated feedwater regulating valve control circuitry, in some cases, due to moisture intrusion.

A significant number of potential common-cause mechanisms (12 events out of the 26 AOV-related events had some common-cause implications) were observed for this set of AOV-related failures including:

- Water or other contamination in instrument air supply lines to air-operated components (4 events),
- Design and installation errors that caused air-operated component failures or operation in a non-fail-safe configuration (4 events), and
- Moisture in electrical components such as feedwater regulating valve pneumatic controllers (1 event).

The AOV-related failures analyzed in this report impacted many plant safety and non-safety related systems including:

- Scram system (1 event),
- Main feedwater and auxiliary feedwater systems (8 events),
- Main condenser and MSIVs (2 events),
- Core Spray (1 event),
- Service water/ Component Cooling Water (2 events),
- Diesel-Generators (2 events),
- ADS (3 events),
- Chemical Volume and Control system (1 event),
- PORVs (4 events), and
- ECCS Recirculation (1 event).

Table 2 provides totals of the number of AOV-related precursor events by year that were analyzed in the ASP reports. The relative ranking of the AOV-related events based on CCDP, as compared to all precursor events analyzed for a given year, is also included in Table 2. Twenty-six events (about 9%) of a total of 288 events were AOV-related.

Table 3 provides yearly totals of the number of AOV-related precursor events with CCDPs greater than or equal to 1E-4, as compared to all precursor events with CCDPs greater than or equal to 1E-4. Twelve events (about 13%) of the total of 89 events with CCDPs greater than or equal to 1E-4 were AOV-related.

As noted in the precursor reports, a direct comparison of CCDP results between years should be approached with caution because CCDP estimates have become more refined over time due to analysis differences in the selection and modeling of events.

Following are descriptions of five AOV-related events which were evaluated in the ASP Program and identified as precursors of high risk significance.

Turkey Point 3 (LER No. 250/85-021)

This event occurred at Turkey Point 3 in 1985, and had an estimated CCDP of 8.96E-4, which ranked it as the fourth highest event out of forty precursor events identified for that year. The event involved multiple AOV failures, primarily due to moisture in the instrument air system.

While in hot standby and recovering from a prior reactor trip caused by a lightning strike, a Main Feedwater bypass valve failed to open. A low steam generator level resulted in an Auxiliary Feedwater auto-start signal, but AFW trains A and C failed on demand due to governor problems. Train A was restarted but during the subsequent recovery, a MFW bypass valve failed to close, resulting in a high SG level and trip of the only operating MFW pump. AFW auto-started but train B's flow control valve failed open. Later, during restart, two AFW train flow control valves failed to open during testing. The air-operated valves failed because of moisture in the instrument air lines.

Oyster Creek 1 (LER No. 219/85-012)

This event occurred at Oyster Creek 1 in 1985, and had an estimated CCDP of 2.28E-4, which ranked it as the seventh highest event out of forty precursor events identified for that year.

Subsequent to a reactor scram, one of the two Scram Discharge Volumes (SDV) did not fully isolate. The resulting flow of hot water from the reactor through the Scram Discharge Volume caused steam and paint fumes to discharge in the reactor building. This, in turn, activated the deluge fire system on one level of the reactor building. The plant experienced a non-isolatable leak of reactor fluid outside primary containment.

One SDV drain valve that isolates the Scram Discharge Volume bottomed out before the valve was fully seated because the stroke adjustment was improperly set. The actuator spring on a second drain valve was improperly sized and opened slightly when pressure from the first valve was applied to its seat. With both valves partly open, the leak was established.

Fort Calhoun (LER No. 285/87-025)

This event occurred at Fort Calhoun in 1987, and was estimated to have a CCDP of 6.2E-4, which ranked it as the second highest event out of thirty-three precursor events identified for that year.

During conversion of the diesel generator (DG) room fire system to a dry-pipe water system, water inadvertently entered the Instrument Air system, due to check valve problems. Upon discovery, the system was repaired and the instrument air system blown down to remove the water. Individual components with safety-related air accumulators were also blown down and drained. However, the emergency DG radiator exhaust damper accumulators, and the AOVs which control the radiator dampers were overlooked.

During a subsequent test, the DG 2 automatically shut down due to high coolant temperature. The AOV controlling radiator exhaust dampers failed to fully open because of the prior water intrusion incident. The lack of full air flow through the radiator resulted in the high coolant temperature. The pilot orifice valve was found to be blocked by foreign material, from the interaction of O-ring lubricant and water. The backup air accumulator for the pilot valve was found to contain 50% water (2 quarts). Water was subsequently found in the accumulator for DG 1.

If a loss of offsite power event had occurred while the DG radiator damper AOVs and their accumulators were inoperable, both DGs would have run to destruction because the high coolant temperature shutoff would have been bypassed during a real demand.

Peach Bottom 3 (LER No. 278/91-017)

This event occurred at Peach Bottom 3 in 1991, and was estimated to have a CCDP of 3.3E-4, which ranked it as the seventh highest event out of twenty-seven precursor events identified for that year.

Improperly installed insulation on the automatic depressurization system (ADS)/safety relief valves (SRVs) resulted in damage to SRV control wiring. The ADS main steam SRVs comprise 5 of the 11 MSRVs. An investigation determined that the MSRV thermal insulation was installed incorrectly during a previous refueling outage. The temperature increase from the installation error caused the expiration of the equipment qualification life of the components after approximately 3 days of operation.

Haddam Neck (LER No. 213/94-005)

This event occurred at Haddam Neck in 1994, and was estimated to have a CCDP of 1.4E-4, which ranked it as the second highest event out of the nine precursor events identified for that year.

During testing, it was discovered that the air operators for the pressurizer power-operated relief valves (PORVs) were experiencing control air leaks and that the PORVs could not be operated properly from their safety-grade control air supply. Investigation revealed that repairs to fix a prior PORV failure (see 1993 event in Table 1) were made incorrectly during the previous refueling outage. The PORV diaphragms were not seated correctly and were coated with a lubricant rather than a required sealant. Substantial air leaks resulted, and the PORVs could not be opened more than 50%.

The LER for the event indicates that two safety functions were potentially compromised by the PORV failures: feed-and-bleed cooling and high-pressure safety injection (HPSI) makeup during certain small-break LOCAs.

Table B-1. Summary of AOV-related precursor analysis results.

Year	LER Number/Plant Name	CCDP Contribution: Events with CCDP ≥ 1e-4	CCDP Contribution: Events With CCDP ≥ 1e-6 & <1e-4	Ranking	Plant Status at Time of Event; Method of Discovery	Brief Event Description
1984	259/84-032/BROWNS FERRY 1		6.60e-06	28/33	At Power; Operational event	Low Pressure Core Spray testable check valve's air operator was installed backwards and resulted in inadvertent over pressure and leakage from system relief valve during MOV test while at power.
1984	325/84-006/BRUNSWICK 1	2.60e-04		6/33	At Power; Operational event	Operators working on instrument air system caused MFW to trip off and resulted in a plant trip; would also have affected Unit 2 had it been operating.
1984	325/84-014/BRUNSWICK 1	1.20e-04		15/33	At Power; Operational event	Following reactor trip and Group 1 isolation signal, a MSIV failed to close when its air-operated solenoid pilot valve failed.
1984	331/84-001/DUANE ARNOLD	1.20e-04		16/33	At Power; Operational event	Loss of feedwater occurred when FW recirculation valve failed open due to a broken air supply line fitting.
1984	387/84-010/SUSQUEHANNAH 1	1.40e-04		13/33	At Power; Testing	Manual scram was required when ADS SRV failed to close during testing when its air-operated solenoid pilot valve failed.
1885	219/85-012/OYSTER CREEK 1	2.28e-04		7/40	At Power; Operational event	One SDV drain valve that isolates the Scram Discharge Volume bottomed out before the valve was fully seated because the stroke adjustment was improperly set. The actuator spring on a second drain valve was improperly sized and opened slightly when pressure from the first valve was applied to its seat. With both valves slightly open, the leak was established.
1985	220/85-021/NINE MILE POINT 1		7.25e-06	25/40	At Power; Operational event	A problem with Instrument Air caused malfunction of a feedwater flow control valve resulting in a reactor trip.

Table B-1. (continued).

Year	LER Number/Plant Name	CCDP Contribution: Events with CCDP ≥ 1e-4	CCDP Contribution: Events With CCDP ≥ 1e-6 & <1e-4	Ranking	Plant Status at Time of Event; Method of Discovery	Brief Event Description
1985	250/85-021/TURKEY POINT 3	8.96e-04		4/40	Routine Shutdown; Operational event	Aux. Feedwater flow control valves and MFW bypass valves malfunctioned due to moisture in Instrument Air lines.
1986	247/86-017/INDIAN POINT 2	1.00e-04		6/18	At Power; Operational event	Safety Injection Actuation & Reactor Trip was caused by condenser steam dump valves opening on a faulty steam dump controller signal.
1987	285/87-025/FORT CALHOUN	6.20e-04		2/33	At Power; Testing	DG2 shutdown due to high coolant temperature caused by a stuck air flow pilot valve for the radiator exhaust dampers. Water was found in the air accumulator of DG 1 & 2 pilot valves.
1987	440/87-009/PERRY 1	2.30e-04		8/33	At Power; Testing	Control air solenoid valves failed resulting in inoperability of two DGs.
1988	280/88-011/SURRY 1		1.50e-05	17/32	Routine Shutdown; Operational event	Two PORVs failed to open due to improper torque on actuator diaphragm bolts.
1988	346/88-007/ DAVIS BESSE 1		1.60e-06	31/32	At Power; Maintenance	Leakage of air from service water valve accumulators after prolonged loss of instrument air could have resulted in loss of SW.
1988	369/88-007/ McGUIRE 1		1.00e-06	32/32	At Power; Operational event	A FW regulating valve failed closed due to controller failure resulting in need for reactor trip.
1989	317/89-005/CALVERT CLIFFS 1		1.40e-06	27/30	Routine Shutdown; Testing	Instrument air boundary check valve leak could result in saltwater pump runout during containment sump recirculation after a LOOP.
1989	338/89-005/ NORTH ANNA 1	1.90e-04		5/30	At Power; Operational event	Reactor tripped due to MFW regulating valve closure caused by loss of instrument air supply.

Table B-1. (continued).

Year	LER Number/Plant Name	CCDP Contribution: Events with CCDP ≥ 1e-4	CCDP Contribution: Events With CCDP ≥ 1e-6 & <1e-4	Ranking	Plant Status at Time of Event; Method of Discovery	Brief Event Description
1989	530/89-001PALO VERDE 3		4.90e-05	9/30	At Power; Operational event	ADVs at PV3 did not actuate from either the control room or the remote shutdown panel following a transient and reactor trip. One ADV was manually opened and another was damaged in an unsuccessful attempt to open it. Investigation of the ADVs at PV 1, 2, and 3 revealed common-cause design problems with all the ADVs, as well as moisture contamination in the air systems and inadequate accumulator capacity.
1989	458/89-022/ RIVER BEND 1		1.30e-05	18/30	Refueling; Testing	Instrument air solenoid isolation valves were installed backward and would impact long-term ADS operations.
1990	206/90-006/SAN ONOFRE 1		6.00e-05	9/28	At Power; Testing	Chemical volume and control system pneumatic valve installed to fail open instead of closed on loss of IA. This could have resulted in gas binding of charging pumps.
1990	285/90-025/FORT CALHOUN		1.70e-06	24/28	At Power; Operational Event	Due to cross ties between CCW and Raw Water system by normally closed, air-operated valves, loss of IA would cause valves to open and drain CCW.
1990	293/90-013/PILGRIM		8.40e-05	8/28	At Power; Operational event	Manual scram after moisture caused an electrical short led to loss of the feedwater regulating valves.
1991	287/91-007/OCONEE 3		1.80e-05	19/27	At Power; Operational event	Reactor trip occurred after LOFW from clogged instrument air flow in master valve controller for condensate demineralizer system.
1991	272/91-030/SALEM 1		4.40e-06	24/27	Routine Shutdown; Testing	Both PORVs failed due to leakage from their air-operated actuators.

Table B-1. (continued).

Year	LER Number/Plant Name	CCDP Contribution: Events with CCDP ≥ 1e-4	CCDP Contribution: Events With CCDP ≥ 1e-6 & <1e-4	Ranking	Plant Status at Time of Event; Method of Discovery	Brief Event Description
1991	278/91-017/PEACH BOTTOM 3	3.30e-04		7/27	Refueling; Maintenance	Improperly installed insulation on ADS SRVs resulted in damage to the SRV control wiring.
1993	213/93-007/HADDAM NECK		6.50e-05	5/16	At Power; Maintenance	Leak in diaphragm assembly of PORV would have prevented feed & bleed cooling following loss of containment air compressors.
1994	213/94-005/HADDAM NECK	1.40e-04		2/9	At Power; Testing	During testing it was determined that both PORVs were leaking from their air-operated actuators.

Table B-2. Totals of ASP analyzed LER events and AOV-related events with CCDP ≥ 10^{-6} per year.

Year	Total No. of All Events With CCDP ≥ 10^{-6}	Total No. of AOV-Related Events With CCDP ≥ 10^{-6} Year	AOV Event Ranking Within Year
1984	33	5	6, 13, 15, 16, 28
1985	40	3	4, 7, 25
1986	18	1	6
1987	18	2	2, 8
1988	32	3	17, 31, 32
1989	30	4	5, 9, 18, 27
1990	28	3	8, 9, 24
1991	27	3	7, 19, 24
1992	27	0	—
1993	16	1	5
1994	9	1	2
1995	10	0	—
Totals	288	26	

Table 3. Totals of ASP analyzed LER events and AOV-related events with CCDP $\geq 10^{-4}$ per year.

Year	Total No. of All Events With CCDP $\geq 10^{-4}$	Total No. of Aov-Related Events With CCDP $\geq 10^{-4}$
1984	17	4
1985	10	2
1986	5	1
1987	10	2
1988	7	0
1989	7	1
1990	6	0
1991	13	1
1992	7	0
1993	4	0
1994	2	1
1995	1	0
Totals	89	12

Appendix C

Trip Reports

Appendix C

TRIP No. 1 REPORT
STUDY OF AIR-OPERATED VALVES
PALO VERDE, OCTOBER 27 AND 28, 1997

We had two days of meetings and interviews with the engineers at Palo Verde who are concerned with AOVs. We also were shown portions of the air system and valves served by it.

Palo Verde has a dedicated AOV service group comprised of engineers and technicians. The engineers provided us with two notebooks, prepared for our visit, that described their program and some of the pertinent problems and studies regarding AOVs. These are being reviewed as part of this AOV study. The contents of the notebooks included a description of their AOV program, examples of AOV sizing calculations, and several root cause analyses of failures of AOVs to perform as intended.

The cooperation, courtesy, and knowledgeable responses from the members of the Palo Verde staff were noted and appreciated by those of us who are involved in this study of AOVs.

Each of the plants at Palo Verde has an independent service air system that also provides a source to the instrument air system. The systems are described in a "Compressed Gas System Evaluation and Analysis Report," (13-MS-A20, Revision 2, dated June 15, 1989, Revision 3, undated). This report is used as a working document for analyzing and maintaining the air system. The air systems each consist of an instrument air subsystem and nitrogen subsystem. The nitrogen system is used when the air header pressure drops below 85 psig. The air/nitrogen systems are not safety-related and safety-related items supplied by them are augmented by backup accumulators.

The AOVs at Palo Verde are categorized as I, II, or III. Category I includes active, safety-related AOVs. Category II includes safety-related AOVs not in category I. Category III includes all other AOVs. There are 41 category I AOVs in each unit (123 total for the site) and 131 category II AOVs (393 for the site). We were told that there are a total of approximately 8400 AOVs on the site. For comparison, there are 831 motor-operated valves on the site (277 per unit), of which 336 are safety-related and in the Generic Letter 89-10 program (112 per unit).

The AOV engineers provided detailed root-cause analysis reports and information on four of the major problems that they had encountered involving AOVs at Palo Verde. These were:

- Atmospheric Dump Valve (ADV) Failures (see LER 52889005),
- Letdown Containment Isolation Valve Leakage,
- Downcomer Feedwater Isolation Valve Failures, and
- Failure of a Vacuum Breaker Solenoid and Subsequent Investigations Involving SOVs.

The ADVs were subjected to evaluation, as well as diagnostic and dynamic testing after the failures occurred. Excessive piston ring leakage, combined with inadequate pilot valve relieving capacity, created high forces in the valve bonnet (also called the balance chamber) that could not be overcome by the actuator. Other problems were also found that compromised the operability of the ADVs. These included:

- Valve oscillations caused by lower than required nitrogen pressure (the regulators exhibited excessive leakage);

- Positioners that were not adjusted and/or maintained properly;

- Springs that were left on the valve operators should have been removed prior to startup;

- An actuator piston that was fitted with a non-qualified Buna-N rather than Viton O-ring;

- Air and nitrogen quality that was suspect (particulate contamination); and

- Several non-qualified pressure gages that were left installed on the positioners.

PVNGS completed an investigation, in 1995, of the recurring seat leakage over several years, of three letdown containment isolation valves. LER 52895007 (a previous LER 52894009 also applies) and PVNGS Condition Report 95Q028 of 5/11/95 describe the results of the investigation. The most probable causes of the seat leakage problems were:

- undersized pneumatic actuators resulting from not accounting for the high frictional loads of graphite style packing during original sizing of the actuators; and

- not maintaining the specified bench set on the spring-and-diaphragm actuators.

Three downcomer feedwater isolation valves (DCFWIVs) at PVNGS failed to open, following closure after a main steam isolation signal (MSIS) during a Unit 1 reactor trip on 11/26/95. The most probable root causes for the multiple valve failures to open were:

- a lack of prudent actuator design margin. The low actuator margin resulted from using a non-conservative valve factor (0.3) in the original actuator/valve manufacturer (Anchor Darling) sizing. Recent tests on motor-operated gate valves indicate that 0.3 is not a conservative valve factor for flex-wedge gate valves.

- not allowing for the potential effects of thermal binding in the original sizing of the actuator.[1]

- not allowing for the potential effects of degradation of the nitrogen supply in the calculation of actuator margin in the original valve/actuator manufacturer (Anchor Darling) sizing.

On November 26, 1995, a loss of condenser vacuum condition occurred at Unit 1 of PVNGS and caused a main turbine trip. A reactor trip followed shortly thereafter. The event was initiated when a condenser vacuum breaker inadvertently opened. Two similar SOVs on Unit 2 were found to have air leaks. The root cause of the SOV failure was attributed to aging.

Palo Verde conducted extensive research, root cause analyses, static testing, diagnostic testing, and dynamic testing in their analysis of the events that are described briefly above.

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[1]Note: Pressure locking and thermal binding of power-operated gate valves, including AOVs, is discussed in NRC Generic Letter 95-07. GL 95-07 was not applied by PVNGS to gate valves whose safety-related function is to close, such as the DCFWIVs.

Among the observations made during the visit to PVNGS for this study, it was noted that several safety-related AOVs had low margins, according to the calculations that were furnished for discussion. Specifically:

> Globe valves: The licensee's engineers were using the port area times the differential pressure (DP) times 1.0 in their thrust estimate calculations. Instrument Society of America guidance is to use the port area times d/P times 1.0. Motor-operated valve testing for other types of globe valves would indicate a valve factor of 0.9 to 1.1. Depending on the particular valve design, the licensee thrust estimates may be as much as 10% low.
>
> It is important to note that the air-operated globe valves used at Palo Verde have not been dynamically tested in a Generic Letter 89-10 type of program to verify valve factors. Also the licensee has only found physical valve/actuator performance issues with actuators (for globe valves) which were not sized in accordance with the original equipment manufacture's published recommendations.
>
> Gate Valves: The licensee's engineers were using the seat area (as provided by the valve manufacturer) times d/P times a valve factor of 0.6 in their thrust estimate calculations. A valve factor of 0.6 is reasonable for cold water systems. INEEL and EPRI test results both support the use of a mean seat area [0.5 X (inside diameter + outside diameter)].
>
> NOTE: Palo Verde suggested the above wording in their comments on the draft report and provided the following explanation as justification for the gate valve discussion: The Palo Verde engineers had originally used the port area for the thrust estimates but had revised the calculation to use the seat area prior to the site visit. The calculation shown to the authors of this study included an error in that some text explaining the calculation indicated the port area was used rather than the seat area. This text was missed as part of the calculation revision to use seat area and has since been revised. The quantitative part of the calculation reviewed by the authors actually used the seat area even though the text indicated the port area had been used.

The quality of the air supply was monitored intermittently for moisture content.

The following tabular summary was prepared to describe the information gathered during the subject visit. The NRC, with assistance from the INEEL, is studying the performance of air-operated valves (AOVs) in commercial nuclear power plants. The information was collected in accordance with the Program Plan dated 10/22/97 (INEEL Letter to H. Ornstein, NRC, from J. Bryce, 10/23/97, Job Code E8238, Task Order 15 - JHB-167-97).

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO PALO VERDE NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
1	Dates.	October 27 and 28, 1997.
2	Names of Interviewers.	Hal Ornstein, NRC/AEOD, 301/415-7574 Owen Rothberg, INEEL/Rockville, 301/816-7773 John Watkins, INEEL/Idaho Falls, 208/526-0567
3	Plant Name & Docket No.	Palo Verde 1, 2, and 3, Dockets 50-528, 529, and 530 respectively.
4	Person(s) Interviewed, Title(s), Phone Number(s), E-Mail address, short description of organization(s) and duties.	Martin Grissom, Licensing, 602/393-5744 Mike Renfroe, Design Engineering, 602/393-1914 Sonja Waters, Design Engineering, 602/393-1935 Steve Quan, IST, 602/393-6215 Mike Hooshmand, Valve Services Engineering, 602/393-1090 Dave Stricker, Valve Services Engineering, 602/393-1938, FAX 602/393-1854 Jim Minnicks, Valve Services Department, 602/393-1070 Benny Malekzadeh, Valve Services Engineering, 602/393-1026 Katie Clifton, Valve Services Engineering, 602/393-1085 Lonnie Bullington, PRA, 602/393-6523 Tim Mitchell, HVAC Maintenance Engineering, 602/393-3541 John Glover, IA Systems Engineering, 602/393/6254 Steve Coppock, Technical Assistant Vice President, Engineering William Ide, Vice-President Engineering
5	If necessary, and if person(s) interviewed can do so, obtain any missing information not provided prior to the site visit, as described in the outline for Task 4 above. Note what information was provided.	A notebook was provided that describes the AOV Program at PVNGS and provides detailed valve lists, sample calculations, failure analyses reports for 1995, and other data. A notebook was provided that describes significant events and issues involving AOVs at Palo Verde. A copy of the Palo Verde "Compressed Gas System Evaluation and Analysis Report" was provided. This document was originally prepared to evaluate and analyze a March 1989 event involving loss of power to the compressed gas system. We were told that this has become something of a "living" reference document for the technical staff involved with compressed gas systems.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO PALO VERDE NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
6	Describe plant events involving AOVs and provide reference information, if possible. Recent: Recurring: Significant:	In March 1989, a reactor trip resulted in loss of IA compressors and degradation of air pressure. Backup nitrogen did not restore pressure. Investigation revealed a number of problems including leaks and unanalyzed demands on the IA system. See reports furnished by PVNGS. The ADVs did not function as designed and a separate report noted a number of design and QA problems with the valves. These problems are considered significant and resulted in a 10 CFR Part 21 report by the licensee. In October 1995 a report describing excessive seat leakage of three letdown isolation valves, in events over the previous three years, was prepared. The root cause problem appeared to be undersized actuators attributable to packing loads from graphite packing not originally accounted for. A copy was furnished during the trip. An investigation of the failure of three downcomer feedwater isolation valves to open in November 1995 was provided. The root cause appeared to be undersized actuators attributable to a not conservatively chosen valve factor. In November 1995, a loss of condenser vacuum led to a reactor trip at Unit 1. This event was caused by a condenser vacuum breaker failing open, which was caused by leakage of the SOVs. The SOV failures were attributed to aging.
7	Describe AOV or air system actual or detected potential failures at the plant? Provide reference information, if possible.	See Component Failure Analysis Report (CFAR) of 9/21/95 for an analysis of failures. In addition, it was determined that several AOVs have margins that are below standard (DCFWIVs). PVNGS is examining options to increase the margin. However, globe and gate valve thrust prediction methods that had not been validated and that had been shown to be nonconservative per MOV testing were, at the time of the visit, being used.
8	Describe actions taken after events or failures involving AOVs or the air system. Provide reference information, if possible.	See the Compressed Gas System Evaluation and Analysis Report that was generated after the March 1989 event and is maintained as a working document by the plant staff concerned with compressed gas systems. A number of recommendations mostly having to do with keeping the air system clean were included. In-plant dynamic diagnostic testing of an AOV was performed.
9	Were there any actual or potential common mode or common cause failures in the air system or AOVs at the plant? Describe and provide reference information, if possible.	The March 1989 event revealed a number of common cause failure mechanisms including debris in the IA and nitrogen backup system, unanalyzed demands on the IA system, and faulty check valves in the accumulators. Globe and gate valve thrust prediction methods that had not been validated and that may have been nonconservative per MOV testing were, at the time of the visit, being used.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO PALO VERDE NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
		Packing adjustments were poorly controlled and their effect on operability was not always known. Performance assumptions (such as the thrust or diaphragm area versus stroke) of actuators is being questioned by some in the industry. The diagnostic test equipment uses the effective diaphragm area as an assumption. The same assumption is used in the actuator verification calculation. This unverified assumption could lead to a common-cause nonconservative seating load.
10	Describe root cause analysis procedures for the plant. Provide reference information, if possible.	Formal root cause analyses are conducted by PVNGS and examples were furnished. The engineering staff has applied standard engineering methods to determine root causes of valve, IA, and nitrogen systems deficiencies and these appear to be well researched and comprehensive.
11	Describe root-cause analyses performed for air system or AOV failures at the plant. Provide reference information, if possible.	Refer to the documentation and calculations provided by PVNGS. The root causes of AOV failures appear to have been competently investigated. A dynamic test with diagnostic equipment was performed on one valve as part of this effort. SOV failures were not tracked, in some cases, because PVNGS replaced failed SOVs as piece-parts.
12	Describe maintenance procedures for the air system. Provide reference information, if possible.	Air quality with respect to water vapor and hydrocarbons is monitored semi-annually (recently changed from quarterly and considering annually). Particulate contamination is controlled by scheduled filter replacement and monitoring three micron filters in the air system. Dew point monitoring is available at the dryer outlets via a tell-tale color change; however, the operators are not required to record color changes on their rounds sheets.
13	Describe maintenance procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	A dedicated team of engineers and technicians is assigned to maintenance of AOVs. The AOVs are monitored in accordance with their ranking as Category 1 (active safety-related), Category 2 (inactive safety related), or Category 3 (non-safety-related). In addition, PVNGS has categorized AOVs as either active safety-related, passive safety-related, or important-to-safety, in response to the Maintenance Rule, 10 CFR 50.65.
14	Describe IST procedures for the air system. Provide reference information, if possible.	The air system is monitored semi-annually for particulates, water contamination, and hydrocarbons. The accumulators for the four ADVs are monitored under plant surveillance test 73SP-9SG05. The remaining accumulators are not monitored.
15	Describe IST procedures for AOVs. Provide reference information, if possible. Safety-related:	ASME Section XI, stroke-timing testing is conducted for safety-related AOVs. Most of these tests are conducted without process fluid loads (pressure and/or flow). The ADVs have been stroked with fluid flow or pressure in the lines although it is not known if this is a standard surveillance procedure.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO PALO VERDE NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
	Important non-safety-related: Non-safety-related:	No regular IST schedule was observed for non-safety-related AOVs.
16	Describe diagnostic systems, if any, used for AOVs. Provide reference information, if possible. Description of system: Specifications: Data collected and frequency of collection: Vendor assistance provided, if any:	The Fisher FlowScanner AOV diagnostic system was used at Palo Verde at the time of the visit. The engineers indicated that the system was somewhat obsolete and PVNGS is investigating purchase of a more modern AOV diagnostic system. Several vendors have presented systems but the plant engineers had not made a decision at the time of the visit. A detailed description of the diagnostic systems used for AOVs was not obtained, although the diagnostic system in use at the time of the visit was discussed with the plant engineers and the training mock-up was viewed. A copy of the procedure used to bench test AOVs was provided. The Fisher FlowScanner system is used routinely to determine various operating characteristics of safety-related and non-safety-related AOVs at PVNGS.
17	Describe design (and analysis) procedures for AOVs. Describe how design basis is established and maintained for AOVs. Provide reference information, if possible.	Design bases estimates and calculations of margin were provided by PVNGS for active safety-related AOVs. The demand calculations are based on estimates of the loads to be encountered. In several cases, the margins appear to be quite narrow (low or negative on Calculation Sheet 13-MC-ZZ-219, page 4 of 18) for the downcomer feedwater isolation valves. The PVNGS engineering staff was working on the problem. Globe and gate valve thrust prediction methods that had not been validated and that may have been nonconservative per MOV testing were, at the time of the visit, being used.
18	Describe analyses and/or testing for verification of operability during postulated transient or accident conditions. Provide reference information, if possible.	PVNGS engineers conducted a number of analyses to estimate the margins available for AOVs at design basis conditions. They have also done some bench testing and diagnostic evaluations that resulted in hardware modifications. They are working to improve the margins on AOVs that they consider important.
19	Describe training for installation, maintenance, and testing of AOVs. Provide reference information, if possible.	Training on mock-ups is in place. No flow loop is available on-site. Diagnostic testing is done using the Fisher FlowScanner system. Valves are tested in accordance with IST requirements and sometimes when anomalous behavior is noted.
20	Describe databases used to track maintenance, failures, and events regarding AOVs. Provide reference information if possible.	PVNGS maintains records of valve behavior in several databases but does not maintain a maintenance/performance history database on each valve. The engineers are aware of the performance and performance characteristics of the hardware in the plant.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO PALO VERDE NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
	On site: Company wide: Industry:	
21	Describe the impact of the Maintenance Rule, 10 CFR 50.65 on AOV and air system maintenance and testing. Provide reference information if possible.	The Maintenance Rule required ranking of systems within the plant in terms of risk significance. Several AOVs that had not been considered significant in the past are now regarded as being important and the AOV group is more focused on the performance of these valves than previously.
22	Is PRA data used for predictive maintenance or replacement of AOVs? If so, how?	According to the "AOV Maintenance Program Description Outline" provided during the visit, PRA rankings are to be used to identify important AOVs. The schedule for accomplishing this was not defined as of the time of the visit.
23	Are AOVs serviced on site, serviced offsite, or replaced as piece-parts if found to require service?	AOVs are generally serviced on site. SOVs are generally replaced as piece-parts when they fail. On occasion, for important events or if assistance is needed to solve a problem, valves are sent to the manufacturer for repair or analysis.
24	Identify and describe the most common recurring maintenance problem(s) and failures regarding AOVs and the air system. What did you see? Provide reference information if possible.	The failure incident report showed a number of problems and concerns regarding the air system and AOVs, particularly with regard to ADVs. We were concerned that there was no continuous monitoring of moisture in the air system, although there are indications that such monitoring could improve equipment performance.
25	Interviewer comments regarding actual valves viewed during the visit, in the plant, undergoing maintenance or replacement, or in the material, if applicable to this interview.	The station air system and components in the turbine building were viewed. Several turbine control and feedwater control valves were also viewed. The shops and training facilities were visited.
26	Has the plant made changes to valves or systems that include AOVs, or replaced AOVs with different models of AOVs or different valves that are not AOVs? If so, describe the changes and the circumstances. What prompted the change?	Yes. See engineering evaluation reports provided on atmospheric dump valves, letdown isolation valves, downcomer feedwater isolation valves, and loss of condenser vacuum.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO PALO VERDE NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
	Was the change made for this plant only?	
27	Does the plant follow EPRI/NMAC guidelines for maintaining AOVs and the air system(s)? If not, describe differences and reasons for the differences. Provide reference information, if possible.	The AOV engineers at Palo Verde are familiar with the EPRI/NMAC guidelines and other industry literature. According to their remarks, the EPRI/NMAC guidelines provided several insights for their program.
28	What is the plant doing or planning to do in response to the recent Industry correspondence on AOVs. Provide reference information, if possible.	PVNGS is actively involved in improvements to their air system and AOVs. They have dedicated a specific group of engineers and technicians. They are aware of the industry work on the subject and indicated that they are open to suggestions for improvement.
29	Do you have any suggestions for improving the performance of AOVs, particularly in the areas of surveillance, testing, or maintenance?	The PVNGS personnel consider that they are aggressively pursuing performance and maintenance of AOVs in the plant. AOVs on auxiliary equipment such as the diesel generators and in the HVAC system are not under the direct control of the AOV group. They are covered separately by the EDG and HVAC engineers respectively.
30	Provide a list of 10 CFR 50.59 and 10 CFR 50.72 reports on AOVs and AOV support systems (air or inert gas supply, etc.) that have been issued for this plant.	A number of LERs and plant internal evaluations were discussed during the visit. A notebook was provided by the plant engineers describing analyses of the atmospheric dump valves, failures of letdown isolation valves, downcomer feedwater isolation valves, and condenser vacuum breaker event.
31	What thrust or torque margins are expected for AOVs? Are different margins used for safety-related, important non-safety-related, or non-safety-related AOVs?	The plant engineers indicated that PVNGS does not have a specific numerical requirement for a margin. A margin in the range of 25% was informally discussed by the engineers as possibly being a desirable high-end goal. (Note: This question was asked during the interviews to get some idea of how knowledgeable engineers viewed expectations of valve margins and was not an attempt to establish some arbitrary standard for any margin.) In some cases, the valves fall far short of this margin, although we were assured that the valves meet all regulatory requirements. Refer to Anchor Darling Gate Valve calculations (13-MC-ZZ-219, page 4). Revision 1 of the calculations stated that the selected valves had been re-analyzed using new valve data provided by Anchor Darling. The calculations refer to a "port" area in some places and a "seat/port" area in others. If the port area was used, the resulting thrust requirements may be under predicted.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO PALO VERDE NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
		Globe and gate valve thrust prediction methods that had not been validated and that may have been nonconservative per MOV testing were, at the time of the visit, being used.
32	What maintenance or surveillance is done to AOV accumulators to ensure air/nitrogen quality and pressure? Were seismic considerations and size verified?	The accumulators for the ADVs (4 per plant) are monitored under plant surveillance test 73SP-9SG05. The remaining accumulators are not monitored. Several accumulator check valves have been repaired or replaced because of excessive seat leakage.
33	Describe problems with pressure regulators, if any.	None noted during visit.
34	Describe problems with feedwater regulating valves, if any.	None noted during visit.
35	What, if any, is your involvement with the AOV Users Group? Describe.	The plant engineers are actively involved in the AOV Users Group. Dave Stricker is the present Chairman and former Vice-Chairman of the AOVUG. Sonja Waters is the current Vice-Chairman and former Engineering and Design Committee Chairman of the AOVUG.

TRIP No. 2 REPORT
STUDY OF AIR-OPERATED VALVES
FERMI 2, NOVEMBER 3 AND 4, 1997

We had two days of meetings and interviews with the engineers at Fermi 2 who are concerned with AOVs. We also were shown portions of the air system and valves served by it.

The AOVs at Fermi 2 are under the cognizance of the Plant Support Engineering (PSE) group. This group is supported by engineering services provided by Tenera and Sargent & Lundy. Detroit Edison is also a member of a group of utilities that pool their resources to devise integrated solutions to problems and concerns. This group of utilities is know as the Utilities Service Alliance, USA, made up of Fermi 2, WNP 2, Ft. Calhoun, Clinton, Cooper, Palisades, and Wolf Creek. AOVs are being studied at Fermi as one of the USA's group efforts.

The engineers provided us with a notebook, prepared for our visit, that described their program and some of the pertinent problems and studies regarding AOVs. These are being reviewed as part of this AOV study. The contents of the notebook included a description of their AOV program, and several root cause analyses of failures of AOVs to perform as intended.

The cooperation, courtesy, and knowledgeable responses from the members of the Fermi 2 staff were noted and appreciated by those of us who are involved in this study of AOVs.

The compressed air systems at Fermi 2 are described in Section 9.3.1 of the Fermi 2 FSAR. The station air system and the control air system provide clean, dry, oil-free compressed air for plant operation. The control air system consists of an interruptible air system (IAS) and a non-interruptible air system (NIAS). The control air system provides air with a dewpoint of -40^0F (at pressure). The station air system and IAS are non-safety-related, non-seismic, and designed to ASME Code Section VIII and ANSI B31.1.0. The NIAS is safety-related, seismic category 1, and designed to ASME Section 3, Class 3 requirements. The NIAS is required for safe-shutdown and for control during long-term recovery.

The station air and IAS equipment is located in the turbine building and the NIAS equipment is located in the auxiliary building. The station air system consists of three compressors, two 150 ft^3 capacity receivers, and an associated distribution network (piping, valves, and fittings). The station air compressors and their associated coolers are cooled by the turbine building closed cooling water system. A station air connection supplies the IAS. The IAS includes two 100% redundant dryers, each with its own pre-filter, after-filter, and instrumentation, IAS receiver, and an associated distribution network. The NIAS consists of two, 100% capacity 100 scfm compressors, two parallel trains of oil filters, air dryers, and after-filters, two parallel air receivers, and an associated distribution network. The control air compressors and after-coolers are cooled by the reactor building closed cooling water system or the emergency equipment cooling water system.

The source of non-interruptible (NIAS) and interruptible (IAS) control air during normal plant operation is from the station air system through interconnections between the station air system and control air system. One interconnection supplies Division 1 and 2 of the NIAS and another interconnection supplies the IAS. The air is filtered and dried. The FSAR indicated that the air for the NIAS is filtered to remove particles greater than 0.5 microns nominal and 0.9 microns absolute, but filtering requirements for the air supplied by the IAS or station air was not included. There are no alarms for high dew point.

The NIAS supplies control air to the:

- standby gas treatment system,
- control center air conditioning system,
- main steam isolation valve leakage control system,
- primary containment atmosphere monitoring system,
- emergency equipment cooling water system,
- primary containment pneumatic supply system,
- torus to reactor building vacuum relief system, and
- railroad bay airlock door seals.

In addition, Division 1 of the NIAS supplies control air to the:

- primary containment isolation of drywell equipment and floor drain sump pump discharge lines, and
- back-up supply for Division 1 (nitrogen) pneumatic supply to the primary containment.

In addition, Division 2 of the NIAS supplies AOVs in the following systems:

- high pressure coolant injection system,
- reactor core isolation cooling system,
- standby gas treatment primary containment isolation valves which support torus venting, and
- torus vent secondary containment isolation valves.

All other control air users are connected to the IAS.

Any one of three station air compressors will be operating during normal operation. One of the other two station air compressors is on "auto" and the other is "off." Normal operating pressure is 100 psig. If the station air header pressure drops below 95 psig, the compressor on "auto" is to automatically start and if the pressure drops to 90 psig, an alarm is initiated in the control room. The compressor in "off" can be manually started form the control room. If the station air header pressure decreases to 85 psig the station air header isolates, air is supplied only to the IAS and NIAS, and an alarm is initiated in the control room. If station air supply pressure to either division of the NIAS decrease to 85 psig, its division's control air compressor automatically starts. If the pressure continues to decrease, the station air supply isolates from the NIAS and alarms at 75 psig, and each division of NIAS is supplied by its own control air compressor from then on.

There is a normally locked-closed intertie between the NIAS divisions and each division may be supplied from the other. There is also a normally closed IAS inter-system tie to Division 2 of the NIAS which may be opened (for maintenance, etc.). In this case, a loss of offsite power or loss of header pressure would render NIAS Division 2 inoperable, so an intertie auto isolation valve is installed to automatically close in order to maintain Division 2 NIAS receiver integrity.

The primary containment is normally supplied with nitrogen from a separate nitrogen inerting system (FSAR section 9.3.6). An intertie is provided to permit Division 1of the NIAS to be used as an emergency backup to Division 1 of the containment pneumatic supply system. Bottled nitrogen can also be used.

The control air compressors are automatically started upon loss of off-site power, with power supplied by the emergency diesel generators (EDGs). Each division of the NIAS has enough receiver capacity to supply control air for 10 minutes after loss of offsite power and before the EDGs come on line. Control air accumulators are also individually provided for safety-related equipment.

Another separate system is dedicated to provide starting air to the EDGs.

The AOVs at Fermi 2 are categorized as 1, 2 or 3. Category 1 includes "AOVs having relatively high safety significance." Category 2 includes "less safety significant AOVs that support safety related functions or have relatively high economic consequences." Category 3 includes "AOVs having limited safety or economic consequences." These definitions were taken from Figure 1, page 6, under tab 3 in the notebook of handout materials provided by the licensee. The definition for Category 3 AOVs in Figure 1, page 6, did not exactly match the definition found on page 3, tab 3. Further, the scope depicted in Figure 1 of those AOVs covered by the Maintenance Rule, 10 CFR 50.65, appeared to be incomplete. These observations were passed on to the licensee's engineers for their evaluation.

There are 29 Category 1 AOVs and 34 Category 2 AOVs out of approximately 1100 (our own rough estimate) AOVs in the plant. These valves are equipped with actuators from 37 different manufacturers. In addition, there are approximately 2482 solenoid operated valves in the plant, of which 1442 are classified as class QA-1. By way of comparison, there are 147 motor-operated valves in the Fermi 2 MOV program.

The air systems are not equipped with devices to monitor dew point and air quality is monitored at "approximately" six month intervals.

Fermi 2 is heavily involved in compliance with the Maintenance Rule, 10 CFR 50.65, at this time. They are reviewing AOVs within the context of the Maintenance Rule to determine important (risk significant) AOVs. They are conducting design basis reviews to determine if the AOVs were designed with sufficient margin. They are going to follow up with static diagnostic testing to verify the condition of the AOVs. They do not appear to be particularly concerned about air system quality and believe it to be acceptable. However, there had been previous air system quality problems (moisture) which resulted in corrosion products in the air system and thus affected plant operation. Fermi 2 made instrument air system modifications, including the installation of new dryers, and this appears to have improved the situation.

Fermi 2 is the lead BWR plant in an Electric Power Research Institute (EPRI) program to develop an overall Air-Operated Valve Program document. Detroit Edison is to conduct design basis system level (d/P, flow, temperature) and component level (required thrust or torque and actuator output capability or margin) evaluations for their Category 1 AOVs using the methods defined in the AOV Program document. This effort is a follow up to the EPRI Performance Prediction Program (PPM) that was devised for motor-operated valves and is planned to be used for AOVs, to the extent practical. EPRI and its contractors are working with the Detroit Edison staff to categorize and analyze the capabilities of selected, important AOVs.

We noted from the statement of work for the EPRI project that the experience gained during the project is planned to be used in the preparation of a comprehensive AOV Evaluation Guide to be developed by EPRI (but which is beyond the scope of the project).

The AOV engineers pointed out a particular AOV (PF45F400A or B, Division 2 Pump Discharge PCV, EESW) during a tour of the plant. They indicated that their recent risk analyses, performed in response to the Maintenance Rule, 10 CFR 50.65, revealed for the first time that these AOVs were particularly risk significant. Subsequent investigations by the licensee confirmed this conclusion. These valves receive an opening signal at pump start and it is important that the valves re-close to prevent diversion of flow. This observation was considered to be germane to the current study of AOVs and indicated the significance of verifying the design basis.

Fermi does not have a dedicated diagnostic system for AOVs at this time. They are working with the Utilities Service Alliance to pool their resources and devise a common program. Their plans for obtaining a diagnostic system for AOVs did not appear to be firm at the time of our visit.

The following tabular summary was prepared to describe the information gathered during the subject visit. The NRC, with assistance from the INEEL, is studying the performance of air-operated valves (AOVs) in commercial nuclear power plants. The information was collected in accordance with the Program Plan dated 10/22/97 (INEEL Letter to H. Ornstein, NRC, from J. Bryce, 10/23/97, Job Code E8238, Task Order 15 - JHB-167-97).

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE ENRICO FERMI 2 NUCLEAR PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
1	Date	November 3 and 4, 1997.
2	Name of Interviewer.	Owen Rothberg, INEEL/LMITCO, 301/816-7773 John Watkins, INEEL/LMITCO, 208/526-0567 Hal Ornstein, NRC/AEOD, 301/415-7574 Joe Colaccino, NRC/NRR, 301/415-2753
3	Plant Name & Docket No.	Enrico Fermi 2 Nuclear Power Plant, Docket No. 50-341.
4	Person(s) Interviewed, Title(s), Phone Number(s), E-Mail address, short description of organization(s) and duties. NOTE: The telephone area code for Fermi was recently changed to 734.	Alan T. Goldsby, Valve Engineer, Detroit Edison, 313/586-1777 Dan Thomas, DECO Maintenance Engineer, 313/586-5598 J. O'Donnell, DECO Maintenance, 313/586-5209 A. (Inadi) Nayaknadi, DECO Plant Engineer, 313/586-1195 John Wald, DECO/ISI/PEP, 313/586-1619 Joe Pendergast, DECO Licensing Engineer, 313/586-1682 Linda Boguci, DECO Risk Assessment Engineer, 313/586-1317 Jorge Ramirez, PSA Consultant, 313/586-1466 John Tibai, Maintenance Rule Principal Engineer, 313/586-4289 Edward J. Vinsilo, I&C Maintenance Supervisor for AOVs, 313/586-4936 Jaime L. Perez, Maintenance-Training, 313/586-4341 Randy Kendrick, Maintenance Engineering, 313/586-5384 Roger Tasell, DECO Plant Support Engineer, 313/586-1768 Greg Lane, DECO Maintenance (SOVs), 313/586-1952 John Tansek, DECO Chemistry, 313/586-5388 Mark E. Soave, PSE-EQ, 313/586-1362 Donald Cobb, OPS Steven Booker, Electrical Maintenance Kenneth Howard, Maintenance Support Engineer Ron Mathews, Maintenance Gabe Verespej, Electrical Maintenance David Roe, Electrical Maintenance John A. Hughes, NQA

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE ENRICO FERMI 2 NUCLEAR PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
		Rodney Johnson, Licensing Neil O'Keefe, NRC Resident Inspector, 313/586-2798 George Pickard, Air System Engineer (contacted 6/2/98)
5	If necessary, and if person(s) interviewed can do so, obtain any missing information not provided prior to the site visit, as described in the outline for Task 4 above. Note what information was provided.	A notebook was provided to us when we arrived, the contents of which are: 1. AOV Program Document 2. Failure Data 3. PRA Application and Data Use 4. Maintenance Data and Diagnostic Systems in Use 5. Design Data 6. Overall Number of AOVs 7. Events Involving AOVs 8. AOV Failures 9. Actions Taken After AOV Events 10. Root Cause Analyses 11. Maintenance and IST Procedures for AOVs 12. Diagnostic Systems Used For AOVs 13. Design (and Analysis) Procedures for AOVs 14. Training for Installation, Maintenance and Testing of AOVs. A memorandum titled "Failure Evaluation of ASCO Solenoid Valves for Fermi 2...", dated September 17, 1997, was provided. This memorandum refers to DER 97-1202, "Inadvertent Closure of T4901F466," a copy of which we were also given.
6	Describe plant events involving AOVs and provide reference information, if possible. Recent: Recurring: Significant:	Failure of a number of solenoid valves, as described in Fermi 2 DER 97-1202, involved use of thread-locking compounds which subsequently migrated to the working parts of the valves and caused them to stick in various positions. This was a recurring and significant problem that had occurred at several other plants and is a common-cause failure mechanism. Spring preload settings on four AOVs were found deficient, as documented in LER 34194004. This may be a recurring deficiency and is a significant common-cause failure mechanism, depending on the valve function.
7	Describe AOV or air system actual or detected potential failures at the plant? Provide reference	The air system has been, and continues to be, upgraded to minimize moisture intrusion. A number of Fermi 2 Deviation Event Reports (DERs 96-0730, 93-0045, and 88-1696) indicated that there were air quality problems that led to moisture in the air lines and subsequent common-

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE ENRICO FERMI 2 NUCLEAR PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
	information, if possible.	cause failures of AOVs. Fermi 2 engineers do not monitor air quality for particulate or moisture contamination on a continuous basis. They do not have automatic moisture monitoring devices in the air system and, therefore, must rely on examination of drain samples. Moisture content for the interruptible portion of the air system is monitored at monthly intervals. The non-interruptible portion of the air system is monitored for moisture content at quarterly intervals.
8	Describe actions taken after events or failures involving AOVs or the air system. Provide reference information, if possible.	The use of thread locking compound was severely restricted by the licensee, training of maintenance personnel was initiated, and the potential use of similar materials that might cause problems was investigated. Lab analysis of adjacent valves was to be performed to estimate the extent of migration. The spring preload problem was resolved by re-engineering.
9	Were there any actual or potential common mode or common cause failures in the air system or AOVs at the plant? Describe and provide reference information, if possible.	Solenoid valve problems involving use of thread locking compound constitute a common-cause failure occurrence which has been a concern in the industry for some time. This type of failure mechanism was documented in NUREG-1275, Vol. 2, 12/87, and in Vol. 6, 2/91. See item 6, 8, and DER 97-1202. Several DERs (for example, 96-0730, 93-0045, and 88-1696) indicated that the air system had been contaminated with moisture and several AOV problems resulted. LER 34194004 indicated that several AOVs might not be capable of meeting their design basis demands because of inadequate control of the valve actuator settings, which resulted in insufficient preload settings in the actuator settings.
10	Describe root cause analysis procedures for the plant. Provide reference information, if possible.	Root-cause analyses are noted in LERs. An example of a detailed root-cause analysis for the failure evaluation of solenoid valves was included under Tab 9 of the notebook which was provided to us by the Fermi 2 engineers. DER 97-1202 refers.
11	Describe root-cause analyses performed for air system or AOV failures at the plant. Provide reference information, if possible.	Root-cause analyses are noted in the materials provided and in several LERs. See item 10.
12	Describe maintenance procedures for the air system. Provide reference information, if possible.	Maintenance was discussed with the plant representatives. Preventive maintenance consists of changing filters and monitoring air quality on approximately six month intervals. Other maintenance is in response to failures or defects.

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13	Describe maintenance procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	The plant engineers are approaching AOVs based on their experiences with MOVs. They have selected 41 AOVs for their initial design basis review. The engineers were in the process of selecting a diagnostic system vendor for AOVs at the time of our visit. There was no diagnostic equipment available on site. At the time of the visit, DECO was negotiating with a contractor who was to evaluate design basis capabilities of 41 important (category 1) AOVs.
14	Describe IST procedures for the air system. Provide reference information, if possible.	No IST is done for the air system.
15	Describe IST procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	IST is done on safety-related AOVs to meet the requirements of 10CFR. No dynamic testing of AOVs has been done. Other AOVs are stroke/time tested prior to return to service after maintenance.
16	Describe diagnostic systems, if any, used for AOVs. Provide reference information, if possible. Description of system: Specifications: Data collected and frequency of collection: Vendor assistance provided, if any:	Fermi did not have a dedicated diagnostic system for AOVs at the time of our visit. They are working with a group of other utilities (the Utilities Service Alliance, USA, made up of Fermi 2, WNP 2, Ft. Calhoun, Clinton, Cooper, and Wolf Creek) to pool their resources and devise a common program. Their plans for obtaining a diagnostic system for AOVs did not appear to be firm at the time of our visit.
17	Describe design (and analysis) procedures for AOVs. Describe how design basis is established and maintained for AOVs. Provide reference information, if possible.	The design of AOVs is being reviewed by Tenera and Sargent & Lundy under contract with Detroit Edison and with support from EPRI. EPRI is providing technical as well as project management input toward developing sizing criteria for AOVs. Tenera reviewed the safety-related and non-safety-related AOVs and categorized them into Category I, II, and III based on PRA and safety significant functions. Sargent & Lundy will be performing the design basis calculations of Category I and II valves. Listings of valves by category were provided. The engineering staff learned recently about the high risk significance of a number of AOVs and

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		has modified their surveillance accordingly.
18	Describe analyses and/or testing for verification of operability during postulated transient or accident conditions. Provide reference information, if possible.	No dynamic testing was being done although the Fermi 2 program plan for AOVs indicates that dynamic testing of some AOVs was being considered. Fermi will depend on a verification by S&L/Tenera and some static testing. The IST and maintenance procedures are outlined under Tab 11 of the notebook that was provided to us by the Fermi 2 engineers.
19	Describe training for installation, maintenance, and testing of AOVs. Provide reference information, if possible.	The training program was described and the training facility was visited. Fermi has a test loop on site and several impressive mock-ups. Refer to Tab 14 of the notebook that was provided to us by the Fermi 2 engineers.
20	Describe databases used to track maintenance, failures, and events regarding AOVs. Provide reference information if possible. On site: Company wide: Industry:	We were shown the databases and information that the plant uses to track failures and events. There are a number of initiatives that the plant is following regarding design verification of AOVs.
21	Describe the impact of the Maintenance Rule, 10 CFR 50.65 on AOV and air system maintenance and testing. Provide reference information if possible.	Many of the recent activities regarding review of AOV design and maintenance are being motivated by compliance with the Maintenance Rule. Fermi 2 is using risk assessments as tools to determine those valves and systems that need to be tracked under the Maintenance Rule. Their design verification initiatives appear to be directly related to the Maintenance Rule efforts. They appear to be enthusiastic about the application of these tools and believe that their efforts will improve performance and save money as well. The AOV engineers pointed out a particular AOV (PF45F400A or B, Division 2 Pump Discharge PCV, EESW) during a tour of the plant. They indicated that their recent risk analyses, performed in response to the Maintenance Rule, 10 CFR 50.65, revealed for the first time that these AOVs were particularly risk significant. Subsequent investigations by the licensee confirmed this conclusion. These valves receive an opening signal at pump start and it is important that the valves re-close to prevent diversion of flow. This observation was considered to be germane to the current study of AOVs and indicated the significance of verifying the design basis.
22	Is PRA data used for predictive maintenance or replacement of AOVs? If	Does not appear to be done yet. We were told that they plan to do so. Study of predictive maintenance program(s) is ongoing.

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	so, how?	
23	Are AOVs serviced on site, serviced offsite, or replaced as piece-parts if found to require service?	AOVs are generally serviced on site. Solenoids are replaced as piece parts and are shipped offsite for analysis, when appropriate.
24	Identify and describe the most common recurring maintenance problem(s) and failures regarding AOVs and the air system. What did you see? Provide reference information if possible.	Fermi has had a common-cause failure problem with certain solenoid valves, related to use of thread locking compounds that migrate toward the working parts and foul them. Refer to DER 97-1202 (Tab 9 in the notebook provided to us by the Fermi 2 engineers). Previous problems were related to moisture in the air system, prior to recent and ongoing plant modifications. Several DERs (for example, 96-0730, 93-0045, and 88-1696) indicated that the air system was contaminated with moisture and several AOV problems resulted.
25	Interviewer comments regarding actual valves viewed during the visit, in the plant, undergoing maintenance or replacement, or in the plant stock system, if applicable to this interview.	We viewed a diesel generator service water isolation valve, the diesel generators and their directly attached valves, the training facility for AOVs, a flow loop, and several mock-ups including a full-size MSIV. The AOV engineers pointed out a particular AOV (PF45F400A or B, Division 2 Pump Discharge PCV, EESW) during a tour of the plant. They indicated that their recent risk analyses, performed in response to the Maintenance Rule, 10 CFR 50.65, revealed for the first time that this AOV was particularly risk significant because it may be called upon to open under accident loads. This AOV was to be evaluated as one of the Category 1 valves in the EPRI sponsored AOV program. This observation was considered to be germane to the current study of AOVs and indicated the significance of verifying the design basis. Several AOVs (diaphragm and spring types) were noted to be mounted with their actuators horizontal or at an angle with the vertical. We were assured that this was acceptable, although perhaps unusual.
26	Has the plant made changes to valves or systems that include AOVs, or replaced AOVs with different models of AOVs or different valves that are not AOVs? If so, describe the changes and the circumstances. What prompted the change? Was the change made for this plant only?	Solenoid valves were changed to different models and the addition of live-load packing was made on some valves. Changes were prompted by recurring solenoid failures and packing leakage. Recurring problems with contaminated air supplied to valves required changes to the air system and monitoring techniques. The NRC, in Generic Letter 88-14, provided guidance to all licensees regarding the necessity to verify the quality of air provided by the air systems so that safety-related equipment would perform as expected. Fermi 2 upgraded their air system in response to GL 88-14.
27	Does the plant follow EPRI/NMAC guidelines for	Yes. The plant follows the recommended actions that are applicable to its AOVs.

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	maintaining AOVs and the air system(s)? If not, describe differences and reasons for the differences. Provide reference information, if possible.	
28	What is the plant doing or planning to do in response to the recent Industry correspondence on AOVs. Provide reference information, if possible.	Fermi had not made a special response to the industry correspondence as of the time of the visit.
29	Do you have any suggestions for improving the performance of AOVs, particularly in the areas of surveillance, testing, or maintenance?	The AOV engineers are aware of the importance of insights from their Maintenance Rule ranking investigations and the need for confirming AOV setups (post-maintenance testing).
30	Provide a list of 10 CFR 50.59 and 10 CFR 50.72 reports on AOVs and AOV support systems (air or inert gas supply, etc.) that have been issued for this plant.	A number of inspection reports, DERs, LERs, and responses to NRC Bulletins or other NRC correspondence were provided.
31	What thrust or torque margins are expected for AOVs? Are different margins used for safety-related, important non-safety-related, or non-safety-related AOVs?	This information was not well documented during our visit, but following the design basis verification currently in progress, they would like to have at least 25% margin. We were told that some AOVs (no details) are suspected to have somewhat less, but still acceptable margins, and these valves were being investigated. Currently, they adjust the packing load based on the packing nut torque. No specific testing to determine packing drag is performed at the present time; only a stroke/time test.
32	What maintenance or surveillance is done to AOV accumulators to ensure air/nitrogen quality and pressure? Were seismic considerations and size verified?	No special inspection or surveillance is done.
33	Describe problems with pressure regulators, if any.	DERs 88-1696 and 96-0730 specifically refer to contamination of the air supply. DER 95-0663 refers to repetitive failures of a pressure regulator. Pressure regulator problems are to be expected if clean, dry air is not consistently provided by the air system.

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34	Describe problems with feedwater regulating valves, if any.	None were discussed during the visit.
35	What, if any, is your involvement with the AOV Users Group? Describe.	Randy Kendrick of Fermi is a member on the AOV Users Group (AUG) and is active and aware of their work.

TRIP No. 3 REPORT
STUDY OF AIR-OPERATED VALVES
PALISADES, NOVEMBER 18 AND 19, 1997

We had two days of meetings with the engineers and technicians at Palisades concerned with the air system and air-operated valves. We did not have an opportunity to view equipment in the plant; however, we did tour the diagnostic facilities that were used to evaluate power operated valves at Palisades.

Palisades has a dedicated AOV Program Engineer under the Manager of System Engineering. At the time of the visit, Palisades was a member of a group of utilities that pool their resources to devise integrated solutions to problems and concerns. This group of utilities is known as the Utilities Service Alliance, USA, and was made up of Fermi 2, WNP2, Palisades, Ft. Calhoun, Clinton, Cooper, and Wolf Creek. The Palisades AOV Program was part of this group effort. Palisades is still participating with this group regarding their AOV program, but to a lesser extent than previously since the completion of their design basis reviews.

The engineers at Palisades provided us with a notebook prepared for our visit that described their program and some of the pertinent problems regarding AOVs. The contents of the notebook also included a description of the Palisades organization, a document entitled "Air Operated Valve Program" (that specifically excludes HVAC dampers from its scope), and several lists of AOVs, as well as lists of AOV failures and deficiencies that have occurred. We were provided with several summary charts to indicate the pertinent Probabilistic Safety Assessment (PSA) results for Palisades. We were provided with a copy of an engineering analysis entitled "System Level Design Basis Review for Air Operated Valves (AOV) in the Engineered Safeguards System (ESS)." Copies of the hand-written logs associated with the 1978 and 1981 incidents involving failure of AOV CV-3025 in the shutdown cooling mode were also provided.

We were provided with a summary of the results and goals associated with the Maintenance Rule review. The primary goal is to implement an action plan for the improvement of the high pressure air system. A closeout memo for the previously implemented plan for improvement of instrument air compressors, summarizing what had been accomplished, was also provided. In addition, we were provided with a copy of another action plan for improvement of the plant air system that was not associated with actions required by the Maintenance Rule. The reason for the distinction was not clear.

In early December, the plant engineers forwarded a document entitled "Compressed Air System Safety System Design Confirmation Report" (SSDC), dated November 21, 1997. This report was prepared by the plant for the purpose of an internal review and was mentioned during our visit.

The cooperation, courtesy, and knowledgeable responses from the members of the Palisades staff were noted and appreciated by those of us who are involved in this study of AOVs.

Service and instrument air is provided by three compressors, each with a separate receiver. The receivers are connected to the compressed air header, which branches to an instrument air header and a service air header. The instrument air header is equipped with a single desiccant dryer and, pre-filters, and post-filters. There are two additional compressors that can be connected to the instrument air system.

High pressure compressed air is provided by three high pressure compressors, each with its own refrigeration type dryer and air receiver.

Nitrogen is supplied from bottles or in bulk as a backup. Two banks of 2000 psig nitrogen bottles provide limited backup for the auxiliary feedwater system valves. Four other nitrogen backup stations, each consisting of 2000 psig nitrogen bottles, are located in the auxiliary and turbine buildings to provide for operation of certain safety-related AOVs. A bulk nitrogen backup system of the instrument air system provides for the operation of the atmospheric dump valves (ADVs).

The instrument air system and the high pressure air system have had significant design and operational weaknesses, which could have led to safety significant events (such as the losses of shutdown cooling in 1978 and 1981, due to failure of the single CV-3025 AOV) and common-cause failures of AOVs and other pneumatic equipment. Those weaknesses included:

- Lack of redundant dryers for the instrument air system. The dryer was being bypassed when serviced and the system was left without drying capability during that time.

- Use of refrigerant dryers on the high pressure air system. These dryers lack the capability to lower the dew point sufficiently to ensure that a supply of sufficiently dry air is provided.

- Misplacement of filters. Several filters were noted to have been placed downstream of the pressure regulators that they are intended to serve.

- Deterioration of piping and equipment served by the high pressure air system. Contaminated air regulators and corroded piping were reported.

A bank of air bottles provides 1800 psig air to backup the high pressure air system for the operation of AOV CV-3018, to meet the fire protection requirements of Appendix R of 10 CFR.

The Condensate Demineralizer Building compressed air needs are supplied by either of two air compressors, each with an integral intercooler and separate aftercooler and receiver. Service air is piped directly from the receivers, while instrument air is routed from the receivers to a dryer and then to the instruments.

The operation and control of the air and nitrogen systems is described in the FSAR.

The AOVs at Palisades are classified as Category 1, 2, or 3. Category 1 valves include safety-related AOVs with an active safety function, AOVs that are important to safety based on their PSA risk significance, or AOVs designated by an Expert Panel. Category 2 valves may be safety-related and of low risk significance or non-safety related and used in critical applications that could affect plant availability, capacity factor, heat rate, or maintenance costs. The remaining AOVs are included in Category 3. There are 111 Category 1 AOVs, and 42 Category 2 AOVs. There are approximately 714 AOVs, total, in the Palisades plant. For comparison, there are 54 motor-operated valves (MOVs) in the plant, 30 of which are covered by NRC Generic Letter 89-10.

Palisades is now involved in efforts in response to the requirements of the Maintenance Rule, 10 CFR 50.65, and those efforts complement their AOV program. Specifically the Palisades staff is reviewing the importance of their AOVs from several different perspectives, including PRA and Expert Panel insights. Palisades has a formal AOV maintenance program and is in the early stages of implementing a plan for improvement of the performance of AOVs similar to that previously invoked and accomplished for motor-operated valves (MOVs), and based on experience with MOVs. Palisades is also part of an EPRI pilot program on AOVs similar to the one described to us in more detail at Fermi 2, and using EPRI's Performance Prediction Program devised for motor-operated valves. Palisades and EPRI are collaborating to develop design basis AOV calculations.

The goal of the Palisades AOV Program is to ensure that the program valves are capable of performing their design basis functions. Both static and dynamic testing have been performed to compare actual valve performance to assumptions made in the AOV calculations.

Implementation of the Maintenance Rule made the plant engineers consider each AOV and rank the valves in terms of risk significance, in accordance with industry guidelines. The ranking process resulted in about 11 of 84 active AOVs in the PSA model being categorized as "high safety significance." Those 11 AOVs are listed in item 21 in the table that follows.

After reviewing their program and the air system (including recent incidents), we made several observations about the air quality and the failures that they had experienced. Our comments on the Palisades AOVs and air system are summarized in the table that follows.

The following tabular summary was prepared to describe the information gathered during the subject visit. The NRC, with assistance from the INEEL, is studying the performance of air-operated valves (AOVs) in commercial nuclear power plants. The information was collected in accordance with the Program Plan dated 10/22/97 (INEEL Letter to H. Ornstein, NRC, from J. Bryce, 10/23/97, Job Code E8238, Task Order 15 - JHB-167-97).

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE PALISADES NUCLEAR PLANT		
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1	Date.	November 18 and 19, 1997.
2	Name of Interviewer.	Owen Rothberg, INEEL/LMITCO, 301/816-7773 John Watkins, INEEL/LMITCO, 208/526-0567 Hal Ornstein, NRC/AEOD, 301/415-7574 Gerry Weidenhamer, NRC/RES, 301/415-6015
3	Plant Name & Docket No.	Palisades Nuclear Plant, 50-255.
4	Person(s) Interviewed, Title(s), Phone Number(s), E-Mail address, short description of organization(s) and duties.	Plant address is Palisades Nuclear Plant, 27780 Blue Star Memorial Highway, Covert, Michigan (MI), 49043. (See the phone listing in the materials provided for phones not shown). Philip D. Flenner, Senior Licensing Engineer, 616/764-2544 Gary W. Foster, Component Engineer, 616/674-2684 Robert A. Gambrill III, Component Engineering Supervisor, CE, Engineering Programs, 616/674-2497 Thomas E. Bordine, Licensing Manager, CE Daniel Mauck, Crane-Movats, Component Engineer Chet Cynoski, CYNCOM, Consulting Engineer Leslie Bradshaw, Valve Engineer, CE Ronald Penna, Alpine Enterprises, AOV Technical Specialist Judy K. Ford, Manager of Engineering Programs, Engineering Programs Dept., CE, 616/764-2340 Kerry A. Toner, Licensing Supervisor, CE Ken T. Speicher, Systems Engineer CAS, CE Robert A. White, Reliability Engineering Supervisor, CE, 616/ 764-2860 Paul F. Prescot, NRC Resident Inspector, 616/764-2741 Eric Grindahl, Diesel Generator Engineer, CE R. A. Fenech, Sr. V.P. Generation, CE T. J. Palmisano, Site V.P., Palisades, CE Bill Beach, NRC, Region III Melvin Leach, NRC, Region III Gregory B. Szczotka, Manager NAPD, CE

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		Kurt Haas, Director of Engineering (Acting), CE Rob McCaleb, NPAD, CE Phillip Young, Project Engineer, DE&S Ken Squibbs, System Engineer Supervisor, CE Paul Fitton, System Engineer Manager, CE R. A. Vincent, Licensing Supervisor, CE Daniel G. Malone, Operations Supt., CE
5	If necessary, and if person(s) interviewed can do so, obtain any missing information not provided prior to the site visit, as described in the outline for Task 4 above. Note what information was provided.	We were given the following materials when we arrived and during the course of the discussions: 1. A notebook containing organization charts, the Air-Operated Valve Program Plan, an analysis of the AOV Program scope, a list of Maintenance Rule (10 CFR 50.65) evaluations, and several valve lists. 2. A design basis review of AOVs in the Engineered Safeguards System. 3. Action plans for air systems improvements and instrument air compressor improvements.
6	Describe plant events involving AOVs and provide reference information, if possible. Recent: Recurring: Significant:	LERs 25578003 and 25581030 were concerned with the significant 1978 and 1981 failure-to-open of a particular shutdown cooling system AOV (CV-3025) when activated at shutdown. The valve is also required to open to mitigate the effects of a small-break LOCA. The single active failure of the valve on each occasion resulted in a rise in core temperature and apparent boiling or near-boiling conditions in the core. We do not know the exact amount of time that it would take to uncover the core, but it is believed to be only a few hours. Although the valve had not failed when called to open at shutdown during the last 16 years, we became concerned that the quality of the air provided by the instrument air system is such that the event could occur again. We expressed our concerns at the exit meeting. Another significant and recent event of interest was the common-cause contamination of nine high-pressure air regulators caused by rust in the air lines. This situation was originally reported to AEOD in April 1997 but was not covered by an LER. (Palisades Condition Report C-PAL-97-0404, dated 3/18/97 and INPO OE 8335, dated 4/22/97 refer to this event. Note that the authors did not have access to the INPO OE report.) The conclusion was that the high pressure air system, which serves many ECCS components, has not been operating in accordance with industry standards. As a result, the pneumatic equipment (including AOVs) serving the ECCS is highly susceptible to common-cause failures.

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7	Describe AOV or air-system actual or detected potential failures at the plant? Provide reference information, if possible.	In discussions with the plant staff in charge of the air system , it became clear that the quality of the air delivered by the instrument air and high pressure air systems at Palisades was suspect. In addition to the problems noted in item 6 above, we were told that the instrument air system has only one dryer and it is bypassed when required by plant operations. According to Palisades, this is less than one day per year. We were also informed that the high-pressure air system's refrigerant dryers do not work properly or reliably, resulting in frequent instances of air system contamination (water and/or rust). In addition, several filters in the high pressure air system had been installed downstream of pressure regulators served by the air system rather than upstream. As a result, corrosion products have produced common-cause pressure regulator contamination.
8	Describe actions taken after events or failures involving AOVs or the air system. Provide reference information, if possible.	The plant has focused on the high-pressure air system as part of its Maintenance Rule reviews. The plant conducted a review of their air systems and have added several goals under the Maintenance Rule, some of which are still in the process of being implemented. The high-pressure air system was on the Maintenance Rule "a(1)" list due to concerns about long-term performance. Our additional comments prompted plant management to assure us of renewed and additional attention to the quality of air.
9	Were there any actual or potential common mode or common cause failures in the air system or AOVs at the plant? Describe and provide reference information, if possible.	Yes. See item 6, above. According to the comments on the draft of this report from Palisades, actions to prevent reoccurrence have been taken.
10	Describe root cause analysis procedures for the plant. Provide reference information, if possible.	Not described.
11	Describe root-cause analyses performed for air system or AOV failures at the plant. Provide reference information, if possible.	Not described.
12	Describe maintenance procedures for the air system. Provide reference information, if possible.	Maintenance and operating practices for the air system were discussed at length. The air system is maintained by an engineer and station maintenance personnel dedicated to this system. The station AOV program is managed by several valve engineers and technicians. Dampers for ventilation and diesel generator AOVs are maintained by others. See items 6 and 7, above.

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		Air quality with respect to water and particulate content is monitored "periodically."
13	Describe maintenance procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	Palisades has a formal AOV maintenance program and is in the early stages of implementing a plan for improvement of the performance of AOVs, similar to that previously invoked and accomplished for motor-operated valves (MOVs) and based on experience with MOVs. (See the material in the notebook provided by the plant engineers.) Diagnostic testing equipment specifically adapted to AOVs is being evaluated. The prioritization of AOV importance is being established from studies implemented to meet the requirements of the Maintenance Rule. It appears that the AOVs are getting the deserved attention.
14	Describe IST procedures for the air system. Provide reference information, if possible.	IST is performed on the required ASME Class 1, 2, and 3 components, which does not include much of the instrument air and high-pressure air systems. Post-maintenance testing is done on components and portions of the system affected by repairs or maintenance.
15	Describe IST procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	Periodic testing of AOVs is done on Category 1 AOVs and consists of ASME stroke/time testing (no load). Diagnostic testing and bench testing are done on AOVs, although apparently not periodically. According to comments provided by Palisades on the draft, special dynamic testing of AOVs is performed as needed to verify that AOVs can perform their design function. No details of this testing were obtained during the site visit.
16	Describe diagnostic systems, if any, used for AOVs. Provide reference information, if possible. Description of system: Specifications: Data collected and frequency of collection: Vendor assistance provided, if any:	Palisades uses a MOVATS universal diagnostic system for AOVs and is trying to adapt Liberty Technologies Easy Torque/Thrust (ETT) sensor for AOVs. They have become quite adept at diagnostic testing because of their MOV experience and believe that they can determine margins using such tools. The new AOV diagnostic system is in the early stages of evolution and Palisades is evaluating the software. See Palisades AOV Program, Procedure EM-28-03, which was included in the notebook provided to us, for further discussion.
17	Describe design (and analysis) procedures for AOVs. Describe how design basis is established and maintained for AOVs. Provide reference information, if possible.	We were provided with a copy of a document entitled "System Level Design Basis Review for Air Operated Valves (AOV) in the Engineered Safeguards System (ESS)." Similar documents for other systems were observed during the visit. Palisades is reviewing their design bases for AOVs to ensure that they are accurate and complete. As part of that process, they are reviewing or revising the original calculations for the valves. In some cases plant calculations are being generated for the first

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		time. Palisades is conducting system and component level design basis reviews of their AOVs, as described in their Program Plan.
18	Describe analyses and/or testing for verification of operability during postulated transient or accident conditions. Provide reference information, if possible.	Analyses are described in item 17, above. No specific test program is in place for verifying design-basis operability of AOVs, other than diagnostic testing and comparisons with the EPRI PPM methodology. According to recent comments from Palisades, dynamic testing is performed if the valves to not exhibit adequate margin. Special test procedures are prepared for the dynamic tests to allow Operations to manipulate the plant to maximize differential pressure and flow. Details and results of such tests were not obtained during the site visit.
19	Describe training for installation, maintenance, and testing of AOVs. Provide reference information, if possible.	Training for valve disassembly, diagnostic testing, and valve maintenance is provided. We viewed part of the Palisades facility for diagnostic testing and the procedure was demonstrated. The engineers were quite knowledgeable about the various diagnostic systems available. Palisades personnel made presentations regarding diagnostic testing at the AUG meetings, to the benefit of other plants.
20	Describe databases used to track maintenance, failures, and events regarding AOVs. Provide reference information if possible. On site: Company wide: Industry:	Palisades maintains a number of computerized, on-line databases to track failures and events. These can be sorted for particular valves and valve types. Engineers and technicians are familiar with the operation and performance of equipment in the plant on a detailed level.
21	Describe the impact of the Maintenance Rule, 10 CFR 50.65 on AOV and air system maintenance and testing. Provide reference information if possible.	Implementation of the Maintenance Rule made the plant engineers consider each AOV and rank the valves in terms of risk significance, in accordance with industry guidelines. The ranking process resulted in about 11 of 75 active AOVs in the PSA model being categorized as having "high safety significance." Those 11 AOVs are: - CV-3006, LPSI Shutdown Cooling Exchanger Bypass - CV-2010, Condensate Inlet Containment Isolation - CV-0522B, Normal Steam to P-8B from Steam Generator "A" - CV-3025, Shutdown Cooling to LPSI Isolation - CV-3029, Containment Sump Isolation to East Engineered Safeguards Room - CV-3030, Containment Sump Isolation to East Engineered Safeguards Room - CV-0779, Steam Generator E-50B Steam Dump Control

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
		- CV-0780, Steam Generator E-50B Steam Dump Control - CV-0781, Steam Generator E-50B Steam Dump Control - CV-0782, Steam Generator E-50B Steam Dump Control - CV-3055, Shutdown Cooling Heat Exchanger Bypass
22	Is PRA data used for predictive maintenance or replacement of AOVs? If so, how?	According to recent comments from Palisades, PRA is receiving limited use as an input to determine predictive maintenance activities on AOVs. Corrective maintenance appeared to be the norm. Selected AOVs do have maintenance procedures that discuss overhauls and predictive maintenance checks.
23	Are AOVs serviced on site, serviced offsite, or replaced as piece-parts if found to require service?	No standard procedure is used. Maintenance, repair, or replacement methods depend on the circumstances.
24	Identify and describe the most common recurring maintenance problem(s) and failures regarding AOVs and the air system. What did you see? Provide reference information if possible.	There is only one dryer for the instrument air system. As a result, when the dryer must be taken off line for any reason the compressor after-cooler is relied upon for air drying. A backup dryer is available but, because of the expense, had yet to be installed. Air quality was monitored intermittently. The high-pressure air system serves both safety-related ECCS equipment and non-safety-related equipment. Refrigerant dryers for the non-safety-related high pressure air system tend to freeze up and so it is necessary to cycle the compressors to prevent that. An adjustment was made to the drain valve timer to ensure that moisture does not remain in the air dryer exchanger. Recently, several regulators were found contaminated with corrosion products which were caused by moisture in the air system. In addition, several filters are installed downstream of the air regulators in the high pressure air system rather than upstream of them. One valve CV-3025 (high risk significance AOV, see item 21), used for shutdown cooling, failed in 1978 and again in 1981 in single-failure incidents that led to boiling, or near-boiling, conditions in the reactor during shutdown. Although the valve had been modified (provided with a hand wheel), and has not failed since 1981, the quality of the air in the air system is suspect, and therefore, so is the potential performance of this valve. This AOV is now stroked quarterly. Previously, CV-3025 was stroked on a cold shutdown frequency. See item 6 in this table for additional information concerning the history of this AOV.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE PALISADES NUCLEAR PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
24	Identify and describe the most common recurring maintenance problem(s) and failures regarding AOVs and the air system.	The Compressed Air System SSDC provides detailed information on the licensee's assessment of the Palisades air systems. It was noted that the CV-3025 valve was not mentioned in Attachment B of that report, which describes air system functional requirements. Several SOV related events have occurred. One event of interest, documented in LER 25592007, occurred on February 5, 1992 while the plant was operating at 100% power. As a result of an ongoing Equipment Classification (Q-list) review program, it was determined that the main steam isolation valve (MSIV) actuator solenoid valves could be rendered inoperable by a main steam line break outside of containment. There were several contributing factors related to the cause of the MSIVs solenoids not meeting the EQ rule (10 CFR 50.49) requirements regarding electrical isolation: - The redundant set of solenoid valves were installed in a non-harsh environment to ensure that the main steam isolation valves would still close in the event of a main steam line break outside of containment. This modification, however, used the same power source as the original SOVs without ensuring appropriate isolation of non-qualified equipment on the power circuit. This resulted in the second set of solenoid valves not being completely redundant.
	Identify and describe the most common recurring maintenance problem(s) and failures regarding AOVs and the air system.	- In 1981 the plant environmental qualification (EQ) project evaluators believed the FSAR to be correct. They failed to realize that the FSAR was incorrect and that the non-harsh environment solenoid valves were truly not redundant. Based on the erroneous information, the MSIV solenoids were removed from the EQ list. Another event that described inadequate environmental qualification of SOVs which are piece-parts of AOVs is documented in LER 25592016. SOVs and position switches for the control valves which control the service water flow from the CAW Heat Exchangers were not environmentally qualified in accordance with 10 CFR 50.49. SOVs SV-0823A, SV-0823B, SV-0826A, SV-0826B, and position switches POS-0823 and POS-0826 provide control and indication for control valves CV-0823 and CV-0826 which control the service water flow from the CAW Water Heat Exchangers E-54A and E-54B, respectively. These components were not qualified for a high-energy line break outside containment and were not on the EQ list. Furthermore, they were not electrically isolated from environmentally qualified instruments in the same electrical scheme. The root cause for this event was attributed an inadequate engineering analysis.
25	Interviewer comments regarding actual valves viewed during the visit, in	We did not have time to view specific AOVs in the plant itself; however, we saw several AOVs in a test/training facility and viewed diagnostic testing devices that the plant uses.

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	the plant, undergoing maintenance or replacement, or in the plant stock system, if applicable to this interview.	
26	Has the plant made changes to valves or systems that include AOVs, or replaced AOVs with different models of AOVs or different valves that are not AOVs? If so, describe the changes and the circumstances. What prompted the change? Was the change made for this plant only?	In the specific case of CV-3025, a handwheel was added as an emergency operation measure. Based on poor performance of the original valves, and the guidance in NUREG 1275, Vol. 6, several solenoid valves throughout the plant were replaced with different models. According to comments received from Palisades on the draft of this report, a number of changes were made to the air system and procedures related to the air system. Refer to Attachment 4 to the letter from N. Haskell of Consumers Energy to the NRC dated July 30, 1999, for a list of those changes.
27	Does the plant follow EPRI/NMAC guidelines for maintaining AOVs and the air system(s)? If not, describe differences and reasons for the differences. Provide reference information, if possible.	Industry guidance, including EPRI/NMAC guidelines, are consulted when formulating plant procedures. Explicit compliance with industry guidance could not be determined. Palisades is part of an EPRI pilot program on AOVs similar to the one described to us in more detail at Fermi 2, and is using EPRI's Performance Prediction Program devised for motor-operated valves. Palisades and EPRI are involved in a collaborative effort to develop design basis AOV calculations.
28	What is the plant doing or planning to do in response to the recent Industry correspondence on AOVs. Provide reference information, if possible.	See item 27.
29	Do you have any suggestions for improving the performance of AOVs, particularly in the areas of surveillance, testing, or maintenance?	Plant engineers discussed several updates to the air system that had been proposed to management to improve air quality and the subsequent performance of air-operated equipment. Among those were installation of a replacement dryer and relocation of filters in the high pressure air system. Recent information received from Palisades in response to the draft of this report indicated that a replacement dryer is to be installed and filters are to be relocated in the near future. See item 24.
30	Provide a list of 10 CFR 50.59 and 10 CFR 50.72 reports on AOVs and AOV support systems (air or inert gas supply, etc.) that have been issued for this plant.	None provided beyond the LERs previously found by AEOD/INEEL.

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31	What thrust or torque margins are expected for AOVs? Are different margins used for safety-related, important non-safety-related, or non-safety-related AOVs?	A number of "20% " was discussed as an acceptable margin. Details on existing margins for AOVs were not provided. Note: This question was asked to get an idea of what engineers considered to be an acceptable margin. There was no attempt to establish any sort of commitment to a particular value.
32	What maintenance or surveillance is done to AOV accumulators to ensure air/nitrogen quality and pressure? Were seismic considerations and size verified?	Surveillance on containment isolation valve accumulators are performed every outage. Accumulator leakage is monitored and trended to ensure that the accumulator can maintain the valve closed for up to 4 hours. Seismic design is in accordance with the design basis and FSAR commitments.
33	Describe problems with pressure regulators, if any.	A significant and recent event of interest was the common-cause contamination of nine high-pressure air regulators caused by rust in the air lines. This situation was originally reported to AEOD in April 1997 but was not covered by an LER.
34	Describe problems with feedwater regulating valves, if any.	Not discussed.
35	What, if any, is your involvement with the AOV Users Group? Describe.	Gary Foster and Bob Gambrill are active participants in the AOV Users Group. Palisades personnel made presentations regarding diagnostic testing at AUG meetings.

TRIP No. 4 REPORT
STUDY OF AIR-OPERATED VALVES
LaSALLE, DECEMBER 17 AND 18, 1997

We had two days of meetings with the engineers and technicians at LaSalle concerned with the air system and air-operated valves. We were escorted through the rooms in the plant that housed the station air system equipment. The plant was shut down, and been shut down for an extended period, at the time of our visit. We did not have an opportunity to visit the training facilities or see diagnostic equipment used by the plant.

LaSalle has several engineers and technicians dedicated to the station air system and another group dedicated to the diesel air start system. A third group is concerned with maintenance and service of ventilation dampers. LaSalle also has an Equipment Qualification Department (EQ), an Inservice Testing Department (IST), and a group dedicated to implementation of the Maintenance Rule. The interfaces and divisions of responsibilities between these groups was not clear to us.

The engineers at LaSalle provided us with a draft "Administrative Procedure for Air Operated Valve Program," a draft "Engineering Qualification Guide Air Operated Valve (AOV) Engineer Qualification," and several lists of AOVs in groupings by system. In addition, we were provided with a copy of a training document entitled "LaSalle A.O.V. Seminar" that describes procedures for setting and maintaining AOVs. An internal document, "Subject: Elastomer Evaluation Guide - NDIT MSD-94-048," was provided as part of the discussion of elastomeric materials during the visit. Results of database searches in the LaSalle internal maintenance (Problem Identification Form, e.g., PIF) database for "AOV" and "air-operate" were also made and provided to us during the visit.

The AOV program plan is in the draft and planning stage at LaSalle and they want to have it in place by restart. (They were projecting a 4/98 restart date at the time of our visit.) No ranking of AOVs at LaSalle, in terms of importance, was presented.

There are several compressed gas systems used to power AOVs at LaSalle; these are station air (consisting of service air and instrument air, which are shared by the two units) and the drywell pneumatic systems, one for each unit. The instrument air system is used to supply air to operate AOVs outside containment and the drywell pneumatic systems supply nitrogen to operate AOVs inside each drywell. The instrument air system supplies both units. The licensee noted that the EDGs are not dependent on the operation of any AOVs. The EDGs are dependent on stand-alone SOVs (one per EDG).

The instrument air system is equipped with continuous dew point monitors at the receivers and alarms in the control room. This was the first plant that we visited that had such equipment. The LaSalle personnel noted that early in the plant's life, they had numerous problems because they had not paid much attention to the dew point monitors. However, when actions were taken to fix the causes of high dew point, they experienced a dramatic improvement, and in recent years have had very few problems that can be attributed to poor air system quality. The drywell pneumatic system and the diesel starting air system do not have dew point monitors or alarms.

No compressors or their associated equipment are required to safely shut down either unit following a postulated LOCA and/or loss of offsite electrical power. AOVs requiring pneumatic supply for safe reactor shutdown are provided with individual pneumatic accumulators.

There are a total of 90 safety-related AOVs for both units at LaSalle. In addition, each unit has 370 safety-related control-rod drive valves (740 total for both units). There are 1575 non-safety-related AOVs at the LaSalle plant (both units). For comparison, there are a total of 200 motor-operated valves in the Generic Letter 89-10 program for both LaSalle units.

We obtained a copy of the LaSalle Summary PRA dated March 1996. According to the Summary PRA, "(t)ransients with loss of instrument air, (T11), are the largest initiating event category, contributing 32% of the CDF. These transients are significant because venting containment cannot be performed without instrument air. Failure to vent results in the loss of the ADS function (and subsequent loss of the low pressure injection systems) and eventual containment failure, causing potential loss of injection systems in the reactor building due to severe environments."

"Loss of offsite power (LOSP) events are the second highest contributor to CDF. Single unit LOSP events contribute 6.5% of CDF and dual unit LOSP events contribute 22.9%. If AC power can be restored to the emergency busses by the diesel generators or crossties, then the plant response is similar to transient events. If both diesel generators or the crossties become unavailable, the unit is considered to be in a station blackout sequence. The core damage contribution of those SBO sequences (subset of LOSP) is 17.2%."

It is interesting to note that a loss of offsite power to Unit 1 or to both units leads directly to a loss of instrument air and the CDF resulting from such an event should be at least as significant as a loss of instrument air alone. The "Accident Sequence Event Descriptions" listed in the tables in the Summary PRA include the event described as "LOSS OF INSTRUMENT AIR 1E OR LOSP AT UNIT 1;" however, the Summary PRA does not provide a discussion of the relationship or dependencies between the two events.

A summary card, distributed by Commenwealth Edison is reproduced below and shows the following information regarding the PRA for LaSalle:

FRONT OF CARD		BACK OF CARD	
Initiating Events Initiator	%CDF		
Loss of Instrument Air	32.1	LaSalle Station	
Dual Unit LOSP	22.9	Key PRA Results	
Transient with Bypass	8.0		
Transient w/out Bypass	7.4		
Single Unit LOSP	6.5		
Loss of Division 1 AC	4.8		
Loss of Division 2 AC	4.7		
MSIV Closure	4.3		
Other	9.3		
Key Operator Actions		Key Equipment System	RAW
Initiate ADS		DC Division 1Y	31.7
Vent Containment		DC Division 1X	25.1
Depressurize with bypass valves only		0DG01P	24.5
FW cntrl to lower power < bypass		DG"B"CWP	22.0

FRONT OF CARD	BACK OF CARD		
Capacity (ATWS)	1(2)DG01P		20.2
ADS to Restart pumps given FWLC-A	DC Division 2Y		20.2
fails (ATWS)	HPCS	18.1	
Restore RPV level to mix boron	Instrument Air		16.9
Initiate SBLC	RHR "B"		16.2
	RHR "A"		15.7
	Containment Vent		7.5
	SBLC	7.4	
	CD	5.4	
	RCIC	5.1	
	DG "B"	4.1	
	DG "A"		3.5
	LPCS	3.0	
	DG "0"	2.5	

After the visit we received NRC Event No. 33434 dated 12/19/97, "Inadequate Turbine Building Vent System Exhaust ..." This report was discussed with plant representatives by phone on 1/7/98 (H. Ornstein & O. Rothberg / Shafique R. Khan & Vince Guterrez). The Event Report involves a transient analysis performed to predict pressures in the turbine building ventilation exhaust tunnel downstream of a postulated high energy line break. A particular set of ventilation system air-operated dampers, whose safety function is to remain closed, could remain open under certain circumstances involving failure of non-safety-related solenoids that control the dampers. In this circumstance, the generated pressures could exceed the structural capability of block walls in the area and this could result in damage to vital equipment in the area. During a recent walk-down they found a failed SOV which could be a precursor. LaSalle intends to make hardware modifications so that failure of the solenoids cannot compromise the safety function of the dampers. This recently discovered postulated event describes a condition where failures of non-safety-related AOVs can affect or interfere with equipment that has to perform a safety function under certain conditions. We are concerned about the importance of a number of events that involve air-operated damper (AOV) failures in nuclear plants and about how the dampers are designed and maintained.

The cooperation, courtesy, and knowledgeable responses from the members of the LaSalle engineering and technical staff were noted and appreciated by those of us who are involved in this study of AOVs.

The following tabular summary was prepared to describe the information gathered during the subject visit. The NRC, with assistance from the INEEL, is studying the performance of air-operated valves (AOVs) in commercial nuclear power plants. The information was collected in accordance with the Program Plan dated 10/22/97 (INEEL Letter to H. Ornstein, NRC, from J. Bryce, 10/23/97, Job Code E8238, Task Order 15 - JHB-167-97).

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE LaSALLE NUCLEAR PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
1	Date.	December 17 and 18, 1997.
2	Name of Interviewer.	Owen Rothberg, INEEL/LMITCO, 301/816-7773 Mark Holbrook, INEEL/LMITCO, 208/526-4362 Hal Ornstein, NRC/AEOD, 301/415-7574 Joe Colaccino, NRC/NRR, 301/415-2753
3	Plant Name & Docket No.	LaSalle County Station, Docket No.'s 50-373 and 50-374.
4	Person(s) Interviewed, Title(s), Phone Number(s), E-Mail address, short description of organization(s) and duties.	Steve Shields, Com. Edison/AIRC Coordinator Guy Campbell, Com. Edison/EPM Baron Westphal, Com. Edison/Pump and Valve Supervisor, x2770 John Kowalski, Com. Edison/AOV Mech. Engr. Tim Sandness, Com. Edison/FLS Ernie Bianchetta, Com. Edison/Principal Instructor Paul Templet, Com. Edison/FLS Mark A. Smith, Com. Edison/AOV Coordinator, x2323 Shafique R. Khan, Sargent & Lundy/Sr. Project Engineer, 312/ 269-7482 Ivo Garza, Com. Edison/Power Operated Valves Steven Smalley, Com. Edison/PCM, Maintenance Rule Rodney Delap, Com. Edison/SA/IA Engineer Keith Tabel, Com. Edison/SA/IA Group Leader Roy Linthicum, Sargent & Lundy/Project Engineer Al Carroll, Com. Edison/Coordinator Robert Tjernlund, Com. Edison/Design Engineer, x2918 Robert Cockrel, Com. Edison/DG System Engineer Fred Darim, Com. Edison/Plant General Manager Michael Reynolds, Com. Edison/IMA Len North, Com. Edison/ImCST John Pollock, Com. Edison/Support Engineering Supervisor Bob Janacek, Com. Edison/PRA (by phone) LaSalle Switchboard: 815/357-6761 LaSalle Mailing Address is: 2601North 21st Rd., Marsailles, IL 61341

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE LaSALLE NUCLEAR PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5	If necessary, and if person(s) interviewed can do so, obtain any missing information not provided prior to the site visit, as described in the outline for Task 4 above. Note what information was provided.	Several LERs and event descriptions that appeared pertinent were selected for discussion with the LaSalle engineering staff prior to the trip. Several of these are discussed under item 6, below. The engineers at LaSalle provided us with a draft "Administrative Procedure for Air Operated Valve Program," a draft "Engineering Qualification Guide Air Operated Valve (AOV) Engineer Qualification," and several lists of AOVs in groupings by system. In addition we were provided with a copy of a training document entitled "LaSalle A.O.V. Seminar" that describes procedures for setting and maintaining AOVs. An internal document, "Subject: Elastomer Evaluation Guide - NDIT MSD-94-048," was provided as part of the discussion of elastomeric materials during the visit. Results of database searches in the LaSalle internal maintenance (Problem Identification Form, e.g., PIF) database for "AOV" and "air-operate" were also made and provided to us during the visit. Copies of P&IDs for the instrument air and drywell pneumatic system were also provided to us. The FSAR Section 9.3.1 on Process Auxiliaries was reviewed after the site visit.
6	Describe plant events involving AOVs and provide reference information, if possible. Recent: Recurring: Significant:	LER # 37385008, "Secondary Containment Isolation Dampers Found Inoperable" LER # 37385011, "Reactor Scram Caused by High Temperature in the Main Steam Tunnel Caused by Solenoid Valve Repairs" LER # 37387032, "Reactor Scram on Low Reactor Level Due To Difficulty ... Feedwater Regulating Valve" LER # 37389007, "Potential Loss of Control Room Isolation Due to Possible Failure of Exhaust Purge Dampers..." LER # 37391007, "RWCU Isolation Due to Leaky Filter/Demineralizer Valve" LER # 37396011, "Pneumatic Valves With Less-Than-Designed Diaphragm Area Results in Inadequate Valve Closing Forces..." LER # 37486015, "RWCU ... Isolation Due to Relief Valve Lifting" LER # 37488010, "Shutdown Due to ADS Nitrogen Backup Pressure Regulator Failure" LER # 37489018, "Battery Low Temperatures Caused by Failed Damper and Low Outside Air Temperatures" LER # 37490011, "Loss of Unit 2 North Bank ADS Backup Pressure Supply" LER # 37492016, "Reactor Scram on Loss of Air..." LER # 37493005, "2A Diesel Generator Air Start System Below 200 psig..."

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
		LER # 37495005, "Failure of Outboard MSIV..." NRC Inspection Report No. 50-373/97011 dated October 15, 1997, discusses leaks in Scram Service Pilot Valves. This report was discovered after the trip and was not discussed with plant representatives. NRC Event No. 33434 dated 12/19/97, "Inadequate Turbine Building Vent System Exhaust ...," was discovered after the trip and was discussed with plant representatives by phone on 1/7/98 (H. Ornstein & O. Rothberg / Shafique R. Khan & Vince Guterrez). LaSalle provided a copy of an Engineering Design Change Package to NRC after the phone call (received on 1/23/98).
7	Describe AOV or air system actual or detected potential failures at the plant? Provide reference information, if possible.	LER # 37396011 refers to a situation discovered by the LaSalle engineers in which certain AOVs did not have effective diaphragm areas (EDA) that conformed to those the valve manufacturer publishes. A "Part 21" notification was issued by LaSalle on 10/4/96. A total of 36 AOVs are involved at LaSalle. This is a generic problem applicable to similar or other Anchor Darling/WKM valves at other plants. As learned from other AOV Users Group participants, the problem has also been observed with other AOV manufacturers. Several LERs (37385008, 37385011, 37389007, 37489018) and NRC Event Report 33434 refer to various problems with air-operated dampers. LER 37492016 describes a SCRAM caused by loss of the instrument air system. Two station air compressors tripped on high lube oil temperature, which was caused by loss of cooling water, which was caused by personnel closing the TBCCW supply and return valves for one unit which were cross-tied to another unit. LER 37493005 describes a diesel generator failure caused by moisture in the air lines. This system is not equipped with moisture sensors or alarms.
8	Describe actions taken after events or failures involving AOVs or the air system. Provide reference information, if possible.	Refer to the LERs in item 7.
9	Were there any actual or potential common mode or common cause failures in the air system or AOVs at the plant? Describe and provide reference information, if possible.	Common cause failures were indicated in LER 37396011 that refers to incorrect calculation of effective diaphragm areas on 36 AOVs. Loss of instrument air or either of the drywell pneumatic systems each has a number of common cause ramifications. Refer to the LERs described above.
10	Describe root cause analysis procedures for the plant.	Root-cause analysis procedures were not described. However, extensive follow-up to AOV anomalies were performed, including use of diagnostic

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE LaSALLE NUCLEAR PLANT		
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	Provide reference information, if possible.	equipment and associated analyses tools. The discovery of the problem of the EDA of certain AOVs, described in item 9 of this table, is an example.
11	Describe root-cause analyses performed for air system or AOV failures at the plant. Provide reference information, if possible.	Root causes are described in most (since 1986) of the LERs noted above. It appears that the root causes were logically determined and addressed.
12	Describe maintenance procedures for the air system. Provide reference information, if possible.	The instrument air receiver moisture traps are blown down once per shift except during the summer when it is done twice per shift. Evidence of contamination is sought. Filters are changed on a regular schedule and there is a regular schedule for dryer maintenance. The licensee indicated that AOVs, and pneumatic equipment in general, were performing much better as a result of their heightened awareness and improved IA system maintenance. There were numerous air system contamination events in the 1980s that were attributed to inadequate dryer maintenance. LaSalle was the first plant visited that continuously monitors air quality and our preliminary conclusion is that the reliability of AOVs and other pneumatic equipment is improved (because of better quality air) over some of the other plants visited.
13	Describe maintenance procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	We did not visit the training or maintenance facilities at LaSalle. LaSalle has several engineering and maintenance groups. They are knowledgeable concerning the equipment and have expertise in maintaining it; however, their approach to maintenance appears to be reactive. This will probably change because of the impact of the Maintenance Rule, 10 CFR 50.65. LaSalle is in the preliminary stages of implementation of the Maintenance Rule.
14	Describe IST procedures for the air system. Provide reference information, if possible.	No IST is done on the air system except for post-maintenance and post-repair testing.
15	Describe IST procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	Section XI of the ASME Code (1989 edition with no addenda) is used to stroke/time test safety-related AOVs. No IST is performed on other AOVs.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE LaSALLE NUCLEAR PLANT		
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16	Describe diagnostic systems, if any, used for AOVs. Provide reference information, if possible. Description of system: Specifications: Data collected and frequency of collection: Vendor assistance provided, if any:	LaSalle uses a recent version of Fisher FlowScanner for diagnostic testing of AOVs. In addition, the plant engineers have adapted some of the techniques developed for MOVs to predict the margins on AOVs. Using these combined techniques, the engineers discovered the discrepancies described in LER # 37396011 and NRC Information Notice 96-68. For example, certain AOVs did not have effective diaphragm areas (EDA) that conformed to those the valve manufacturer publishes. The pneumatic diaphragm actuators are WKM Model 70-13 in sizes 35, 70, 140, and 280. LaSalle has 36 of these actuators total in both units. Thirteen of the valves, per unit (26 total) are in the RCIC system.
17	Describe design (and analysis) procedures for AOVs. Describe how design basis is established and maintained for AOVs. Provide reference information, if possible.	Procedures are described in the draft AOV program for a design basis review. A detailed description of the design and analysis procedures was not provided.
18	Describe analyses and/or testing for verification of operability during postulated transient or accident conditions. Provide reference information, if possible.	Procedures are described in the draft AOV program for a design basis review. A detailed description of the design and analysis procedures was not provided.
19	Describe training for installation, maintenance, and testing of AOVs. Provide reference information, if possible.	Training was described in general terms. All technicians are trained to assemble/disassemble AOVs using hands-on techniques. Diagnostic testing of AOVs is performed by a trained group. LaSalle Station has been ComEd's lead plant in the field of AOV diagnostics and LaSalle has been "lending out" AOV technical personnel to help upgrade those activities at the other ComEd plants. LaSalle personnel made presentations at the AUG and international (ICONE) meetings, to the benefit of other plants.
20	Describe databases used to track maintenance, failures, and events regarding AOVs. Provide reference information if possible. On site: Company wide: Industry:	The LaSalle plant uses a number of computerized databases to track maintenance, events, and failures. These appear to provide necessary information. We asked during the visit, and in a telephone conversation shortly thereafter, about sharing information between the CECO plants and the industry in general.

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
21	Describe the impact of the Maintenance Rule, 10 CFR 50.65 on AOV and air system maintenance and testing. Provide reference information if possible.	LaSalle will, in accordance with the Maintenance Rule, examine the performance of certain AOVs and the supporting air supply system and establish performance goals. This, in turn, requires specific actions to improve performance. At the time of our visit, the plant did not appear to be as far along in the process as some of the other plants that we visited.
22	Is PRA data used for predictive maintenance or replacement of AOVs? If so, how?	PRA data is not being used to make a decision on the AOV program at this time. Again, it is expected that PRA and failure data will be used by the plant in accordance with the Maintenance Rule.
23	Are AOVs serviced on site, serviced offsite, or replaced as piece-parts if found to require service?	AOVs are serviced on site, under the supervision of an AOV coordinator. Solenoids are normally replaced as piece-parts and are not repaired.
24	Identify and describe the most common recurring maintenance problem(s) and failures regarding AOVs and the air system. What did you see? Provide reference information if possible.	Problems with air-operated dampers appear common, although no common cause was identified. Problems with solenoids, related to organic materials and lubrication incompatibilities, as documented in NUREG-1275, Vol. 6, also appeared to be common. Contamination and/or moisture in the instrument air system, in contrast to plants previously visited, has NOT been a problem in recent years. The operators experienced air system and AOV problems when the plant first started up until they paid attention to instrument air quality, as was being reported on the dew point monitoring equipment.
25	Interviewer comments regarding actual valves viewed during the visit, in the plant, undergoing maintenance or replacement, or in the plant stock system, if applicable to this interview.	We toured the instrument air room in the plant and were impressed by the cleanliness, large scale, and attention to detailed design that was apparent in the installation. The engineers were cognizant of the design and function of the air system, and its requirements.
26	Has the plant made changes to valves or systems that include AOVs, or replaced AOVs with different models of AOVs or different valves that are not AOVs? If so, describe the changes and the circumstances. What prompted the change?	Plant changed ASCO 8323 (see LER 37495005) solenoids to Valcor solenoids on MSIVs after repeated failures related to lubricant/thread-locking compound-related problems. Change out of the ASCO 8323 SOVs were done in a staggered sequence as suggested in NUREG-1275, Vol. 6. It was noted that the Valcor valves are to be installed, per the manufacturer's instructions, using Graphoil tape; however, LaSalle uses a different thread-locking compound. We questioned the use of thread-locking compound and were told that the engineers would investigate.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE LaSALLE NUCLEAR PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
	Was the change made for this plant only?	
27	Does the plant follow EPRI/NMAC guidelines for maintaining AOVs and the air system(s)? If not, describe differences and reasons for the differences. Provide reference information, if possible.	EPRI/NMAC guidelines are used for reference, along with information from a number of industry sources.
28	What is the plant doing or planning to do in response to the recent Industry correspondence on AOVs. Provide reference information, if possible.	We were provided with the draft of a document entitled "LaSalle Administrative Procedure - Air Operated Valve Program." The purpose of the procedure, as stated therein, is to categorize AOVs based on their safety significance and functional requirements, prioritize AOVs based on past performance and industry data, establish AOV design bases by reviewing design data, determine as-built configuration by performing walkdowns, tracking, and trending. The plan appeared to be comprehensive and was based on information from a number of industry organizations including the AOV Users Group, NSAC, NMAC, and EPRI.
29	Do you have any suggestions for improving the performance of AOVs, particularly in the areas of surveillance, testing, or maintenance?	The use of a comprehensive program for AOVs, that includes prioritization based on safety significance, failure experience, diagnostic evaluations, industry experience, and test data should be used to improve the performance of AOVs. LaSalle is in the process of constructing and implementing such a program.
30	Provide a list of 10 CFR 50.59 and 10 CFR 50.72 reports on AOVs and AOV support systems (air or inert gas supply, etc.) that have been issued for this plant.	A number of LERs (noted previously in items 6 and 7) and information from the LaSalle PIF database were reviewed.
31	What thrust or torque margins are expected for AOVs? Are different margins used for safety-related, important non-safety-related, or non-safety-related AOVs?	Engineers would like to have 20% margin in their calculations. The 20% figure is an informal judgement call and is not a plant requirement. The question was asked in order to survey knowledgeable engineers about what they thought was an acceptable margin and was not done to establish any commitment.
32	What maintenance or surveillance is done to AOV accumulators to ensure air/nitrogen quality and pressure? Were seismic	The accumulators are blown down every refueling cycle and the check valves are verified to be functioning at that time. Seismic design, along with other design considerations, are to be reviewed as part of a comprehensive program.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE LaSALLE NUCLEAR PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
	considerations and size verified?	
33	Describe problems with pressure regulators, if any.	LER # 37490011, "Loss of Unit 2 North Bank ADS Backup Pressure Supply" indicated the failure or misoperation of a pressure regulator used to maintain several ADS accumulators. The North ADS Bottle Bank pressure regulator was not opening to control header pressure. A manual stop valve was found to have been left in the closed (wrong) position. The drywell pneumatic compressors failed to maintain system pressure during the event. The stop valve mispositioning affected the three ADS valves supplied by the North Bottle Bank. There are a total of seven ADS valves.
34	Describe problems with feedwater regulating valves, if any.	LER # 37387032, "Reactor Scram on Low Reactor Level Due To Difficulty Controlling Reactor Water Level with the Feedwater Regulating Valve at Low Flow, Low Power Conditions" refers to a problem with the system design. Hardware modifications were described in the LER including installation of a motor operator on the FRV inlet isolation valve along with a control switch in the control room, as well as installation of a bypass line and regulating valve around the existing FRV. It is assumed that these modifications were made; however, we did not verify this during our visit.
35	What, if any, is your involvement with the AOV Users Group? Describe.	The plant has a engineer who is a representative heavily involved in the AOV Users group. LaSalle AOV engineers and their contractors have made presentations at AOV Users Group and other industry meetings to feed back information on their AOV diagnostics and the EDA problems that were discovered with their Anchor Darling/WKM/BSB valves.

Trip No. 5 REPORT
STUDY OF AIR-OPERATED VALVES
THREE MILE ISLAND, UNIT 1, FEBRUARY 12 AND 13, 1998

We had two days of meetings and interviews with the engineers at Three Mile Island (TMI) who are concerned with AOVs and the air systems. We were shown portions of the air systems, as well as some of the valves served by them.

TMI has several engineers dedicated to the service air system and AOVs. Several types of air-operated components such as air-operated dampers, solenoid-operated valves in various services, and air-operated components attached to the diesel generators are served by system engineers for the particular systems. We noted that the engineers involved with the air systems that we spoke to seem knowledgeable about their own systems and equipment as well as the other air systems and equipment outside of their cognizance.

We were provided with a number of documents and reports including a draft "Air Operated Valve Program Description;" an installation specification for "Installation of Air Filters on Critical Plant Control Valves;" a copy of the TMI letter to the NRC responding to Generic Letter 88-14 on instrument air system problems; several lists of plant events and maintenance reports; and a package of information (including sorted lists of valves) used to describe the methods and techniques for risk analysis and risk-based ranking of AOVs and other equipment.

We were also provided with a copy of a report of the B&W Owners Group (B&WOG) entitled "Instrument Air System Review Report." This was part of an effort by the B&WOG for B&W plants that corresponded to the work in NUREG-1275, Volume 2, and Generic Letter 88-14. We spoke with Gordon Skillman, who was a contributor to the report.

The purpose of the instrument and control air system at TMI is to continuously deliver clean, dry air at 100 psig (-40^{0}F dew point, filtered to 0.9 micron particle size). The instrument and control air system consists of three unlubricated air compressors, each discharging through a separate after cooler and air receiver. Two of the air compressors utilize a common air dryer (the "old" portion of the instrument and control air system). The third air compressor uses a separate air dryer (the "new" portion of the instrument and control air system). The "new" portion of the instrument and control air system was added to address problems with maintaining air quality, particularly moisture, that the plant experienced after startup. The "new" portion is now run continuously and the "old" portion is only run for testing or when the "new" portion is taken off line for service.

Backup for the instrument and control air system is provided by a connection to the plant service air system. Air will automatically flow from the plant service air system at a pressure of approximately 70 psig through an oil removal filter when instrument air pressure drops sufficiently, in order to supply components needed to shut down the plant. A two-hour backup air supply is also available to provide compressed air for operation of components within the main steam and emergency feedwater systems if the plant instrument air system is not available. We were told that no AOVs are powered from the 200 psig diesel air start system.

The instrument and control air system is equipped with continuous dew point monitors and alarms in the control room. The instrument and control air system is arranged to be restarted and run off the diesel generators, from a switch in the control room, in the event of a loss of offsite power. The piping for the "new" instrument and control air system is stainless steel up to the dryers and copper downstream. The piping for the "old" instrument and control air system is carbon steel up to the dryers and copper downstream. All of the receivers are carbon steel with a coated interior. We were told that there is evidence that the coating may have worn over time. A number of air filters have been added directly upstream of critical plant control valves in order to minimize the risk of particulate contamination of these components. The performance of the instrument air system is described in Item 7 in the table below.

The TMI engineers described to us a particularly interesting event involving insufficient margins in two AOVs. Calculations for five Crane-Aloyco, 2.5 to 6 inch, 1500 and 150 pound class, flex-wedge and split wedge gate AOVs with Miller DA-63-B and A-63-B cylinder actuators were reviewed by one of the TMI engineers. The AE requested that the valve manufacturer (Crane-Aloyco) perform thrust calculations on the five AOVs, as part of a limit switch upgrade modification, in order to verify that limit switch installation would not affect valve operability. The resultant thrust calculations, using "present day" methodology, indicated that two of the five valves had negative closing margins for the specified differential pressure (d/P). The architect/engineer (AE) then requested that the manufacturer redo the calculations using the methodology by which the valves were originally sized. The revised Crane-Aloyco calculations, based on the original valve sizing methodology, indicated positive margins in both the opening and closing directions, but using a valve factor of zero. Upon review of the Crane-Aloyco calculations, TMI convened a review group which verified that the two containment isolation valves which had negative margins in the manufacturers' first calculations were, in fact, operable and would be able to perform their designed safety function. This conclusion was based on TMI's calculations using a 0.75 friction factor and d/P of 1600 psi (that d/P required for containment isolation).

A design modification was proposed to increase the closing margin for one of the valves because normal operating d/P (2375 psi) was greater than the d/p needed to perform its safety function (1600 psi). The proposed modification included the installation of an accumulator and piping to provide air assistance to the spring. Subsequently, TMI performed an analysis of the valve using the EPRI PPM methodology (2375 psi and 0.67 friction factor) and confirmed the operability of the valve. Design problems with a pressure booster for the AOV were also noted.

It was learned that Crystal River 3 (a similar B&W plant with the same AE, Gilbert Associates, as TMI-1) had a similar AOV margin problem. Both plants have been exchanging information on methods to increase the available margins for the AOVs. These conditions were not previously covered by an LER or other correspondence with the NRC. As a result of our visit, the TMI engineers agreed to investigate to determine if there are Part 21 concerns to be addressed, either by TMI or the valve manufacturer (Crane-Aloyco). There may be generic or common-cause issues with these types of AOVs, related to the design or the OEM calculations, that need to be addressed.

Several common-cause solenoid valve (piece parts of AOVs) failures, involving hardening of O-rings were reported. These failures are described in the Table, below.

There are 910 AOVs in the TMI plant. Of these, 98 are safety-related (Q class or Class 1), 328 are Class 2, and 484 are Class 3, in accordance with the lists provided. By way of comparison, there are 81 motor-operated valves in the TMI Generic Letter 89-10 program.

The cooperation, courtesy, and knowledgeable responses from the members of the TMI technical staff were noted and appreciated by those of us who are involved in this study of AOVs.

The following tabular summary was prepared to describe the information gathered during the subject visit. The NRC, with assistance from the INEEL, is studying the performance of air-operated valves (AOVs) in commercial nuclear power plants. The information was collected in accordance with the Program Plan dated 10/22/97 (INEEL Letter to H. Ornstein, NRC, from J. Bryce, 10/23/97, Job Code E8238, Task Order 15 - JHB-167-97).

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
1	Dates.	February 12 and 13, 1998.
2	Names of Interviewers.	Hal Ornstein, NRC/AEOD, 301/415-7574 Owen Rothberg, INEEL/Rockville, 301/816-7773 John Watkins, INEEL/Idaho Falls, 208/526-0567 David Terao, NRC/NRR, 301/415-3317
3	Plant Name & Docket No.	Three Mile Island, Unit 1, Docket No. 50-289.
4	Persons Interviewed, Titles, Phone Numbers, E-Mail address, short description of organizations, and duties.	Harold Wilson, Supervisor, Maintenance Assessment, TMI, 717/948-8050, FAX 717/948-8598 David Atherholt, Plant Maintenance Director, TMI, 717/948-8838, FAX 717/948-8598, datherholt@gpu.com Charles Hartman, GPUN/Engineering Dept., 717/948-8150 James Gilles, GPUN/Engineering Dept., 717/948-8840 Howard Crawford, Manager - Programs Engineering Division, TMI, 717-948-8412, FAX 717/948-6822, hcrawford@gpu.com Greg Gurican, Nuclear Licensing Engineer, TMI, 717/948-8753, FAX 717/948-2820, ggurican@gpu.com Patrick Bennett, GPUN System Engineer, 717/948-8232 Charles Adams, GPUN, IOSRG (PRA), 717/948-8055 Mark Fauber, GPUN Licensing, (Conducted Tour of Plant) Gordon Skillman, GPUN, (Stopped by and told us of B&W Owners Report on Instrument Air System Review) The plant mailing address is: GPU Nuclear Inc. Route 441 South Post Office Box 480 Middletown, PA 17057-0480
5	If necessary, and if person(s) interviewed can do so, obtain any missing information not provided prior to the site visit, as described in the outline for Task 4 of the program plan. Note what information was provided.	No information was provided by TMI prior to our visit. A copy of our program plan was provided to TMI. We arrived with the following materials: - A copy of Section 9.10.1 of the FSAR (Update-12, 3/94), INSTRUMENT AND CONTROL AIR SYSTEM - Reduced GPU Drawing No. 302-271, Rev 62, INSTRUMENT & STATION SERVICE AIR Flow Diagram (continued next page)

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5	(continued)	- Reduced GPU Drawing No. 302-272, Rev. 16, BACKUP INSTRUMENT AIR Flow Diagram - Reduced GPU Drawing No. 302-273, Rev. 18, EMERGENCY FEEDWATER & MAIN STEAM VALVE - 2-Hour Backup Supply Air Flow Diagram - LER 28986002 Summary, Anticipatory Reactor Trip, Turbine Trip, High Moisture Separator Level - LER 28986007, Failure to Meet Design Criteria on the Two Hour Backup Air Supply - LER 28987008, Reactor Trip, Turbine Trip, High Moisture Separator Level - A list of LERs taken from the Sequence Code Search System (SCSS) database The following materials were provided by TMI during our visit: - AIR OPERATED VALVE PROGRAM DESCRIPTION Draft Topical Report 118 - Letter from GPUN to NRC dated 2/24/89 (C311-89-2016) indicating TMI's response to Generic Letter 88-14 - Letter from GPUN to NRC dated 2/26/92 (C311-92-2031) indicating TMI's completion of actions for their Generic Letter 88-14 commitments - A computer list of MAINTENANCE TREND ACTION NOTICE (MTAN) REPORTS dated 2/4/98 with AOV items marked (individual MTANs noted below) - A computer list of POTENTIAL FAILURE OF AIR OPERATED VALVES, SOLENOID VALVES, AND AIR SYSTEM AT TMI prepared 2/13/98 - Citations for the ORAM User's Manual, Version 1.5 (DOS), EPRI TR-102819, Nov. 1993 and the ORAM SENTINEL Users Manual, Version 3.0, TR 107018, Sept. 1997, ALL MODES MAINTENANCE AND SAFETY FUNCTION ADVISOR - Cover page and Introduction to the TMI OUTAGE FUEL PROTECTION CRITERIA, Topical Report 097, Revision 3, dated 1/16/95 (Howard Crawford) - Exhibit 5 to TMI Administrative Procedure 10780, Rev. 13, TMI PSA SUMMARY RESULTS, Contains listing of core damage sequences and initiating events in order of importance (continued next page)

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5	(continued)	Materials provided by TMI (continued): - An undated paper (one figure is dated 7/15/93), TMI-1 PRA INPUT TO MAINTENANCE RULE - RISK SIGNIFICANT SYSTEM LIST, Developed by Risk Analysis Section, C. Adams - A list of valves headed TABLE-1, INSERVICE TESTING VALVES RANKED BY FUSSEL-VESELY (WITHOUT COMMON CAUSE) - A list of valves headed TABLE-2, TMI-1 INSERVICE TESTING VALVES RANKED BY RISK ACHIEVEMENT (WITHOUT COMMON CAUSE) - A list of valves headed "FVRArk" indicating Fussel-Vesely Importance and Rank - A list of valves headed TABLE-3, INSERVICE TESTING VALVES RANKED BY FUSSEL-VESELY (WITH COMMON CAUSE) - A list of valves headed TABLE-4, TMI-1 INSERVICE TESTING VALVES RANKED BY RISK ACHIEVEMENT (WITH COMMON CAUSE) - A list of valves headed SHEET 6, showing a summary of the rankings (H i.e., high, M i.e., medium, L i.e., low, N i.e., none, T i.e., truncated, and NM i.e., not modeled) in Tables 1 through 4 for each valve - A list of classes of valves headed SHEET 7, summarizing total number of valves in the rankings (H i.e., high, M i.e., medium, L i.e., low, N i.e., none, T i.e., truncated, and NM i.e., not modeled) in Tables 1 through 4 - MTAN No. 93-15 dated 12/21/93 to investigate the reliability of make-up valve MU-V-0017 as the result of a history of slow or sluggish operation - MTAN No. 93-03 dated 7/29/93 to investigate the failures of several solenoid failures for valves CM-V-0002, 3, 4, 7, and 8 (includes test data) - TMI-1 SPECIFICATION T1-IS-123267-001, REGULATORY REQUIRED INSTALLATION SPECIFICATION FOR INSTALLATION OF AIR FILTERS ON CRITICAL PLANT CONTROL VALVES (continued next page)

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5	(continued)	Materials provided by TMI (continued): - Several diagrams and excerpted pages from TMI Report 643, REQUIRED THRUST ANALYSIS, for limit switch modifications to valves MU-V-3, MU-V-26, WDL-V-304, WDL-V-534, and WDL-V-535 - INSTRUMENT AIR SYSTEM REVIEW REPORT, B&W Owners Group, Safety and Performance Program Report No. 47-1165965-00 dated 12/86 Materials provided by TMI by mail after the visit: - A list of SAFETY EVALUATIONS REPORTS with those pertinent to the AOV study marked. Attached were copies of GPUN forms entitled "Technical Functions, Safety/Environmental Determination and 50.59 Review (EP-016)" and numbered with "SE" numbers. The attached forms were: - SE No. SE-CMR-113202-397, Replacement of instrument air dewpoint recorder, 1/28/93 - SE No. SE-CMR-113202, Replacement of Failed IWT Regulators, 2/5/90 - SE No. 115101-002, Instrument Air As-Found Conditions, 5/21/91 - SE No. 123267-001, Installation of Air Filters on Critical Plant Control Valves, 8/11/89 - SE No. SE-128961-002, Two Hour Backup Instrument Air Switching Valve Modification, 3/16/87 - SE No. 412512-005, Vent Valve Installation for Testing Air Accumulators, 1/22/90 - SE No. 412012-002, EF and MS 2 Hour Air Supply, Truck Connection Modification, 9/23/83 - The following tabular listings dated 7/19/93: - Table 1, Modified Master Frequency File (TMI-1 NOHUMAN Model) Materials provided by TMI by mail after the visit : - The following tabular listings dated 7/19/93: - Table 2, Initiating Event Contributions (TMI-1 NOHUMAN Model) - Table 3, TMI-1 NOHUMAN Model top 50 Output Sequence Lists (continued on next page)

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5	(continued)	- Table 4, Split Fraction Importance (Sorted by Importance) - Table 5, Split Fraction Importance (Sorted by Achievement) - Table 6, Split Fraction Importance (Sorted by Risk Reduction) - Table 7, Split Fraction Importance (Sorted by Fraction Designator) - Table 8, Top Event Importance (Sorted by Importance) - Table 9, Top Event Importance (Sorted by Risk Achievement) - Table 10, Top Event Importance (Sorted by Risk Reduction Worth)
6	Describe plant events involving AOVs and provide reference information, if possible. Recent: Recurring: Significant:	- LER 28986002 described a reactor trip caused by dirt and rust in the air ports of a valve controller. - LER 28986007 indicated that a backup air supply might not have met the single failure criterion. - LER 28987002 described a reactor trip caused by failure of an air-operated controller due to undefined causes. - MTAN No. 93-15, dated 12/21/93, documents the licensee's effort to investigate the reliability of make-up valve MU-V-0017 as the result of a history of slow or sluggish operation. The valve and operator were rebuilt and several items, including the positioner, were replaced. - MTAN No. 93-03 dated 7/29/93 documents the licensee's effort to investigate the failures of several solenoid failures for valves CM-V-0002, 3, 4, 7, and 8, sample supply and return valves for the Reactor Building air monitor (included test data). Several SOVs were replaced with nuclear grade valves and several were rebuilt. The root cause was reported as hardening of the O-rings, although the mechanism that caused the deterioration was not reported. This was a common-cause failure mechanism. - Several diagrams and excerpted pages TMI Report 643, REQUIRED THRUST ANALYSIS, for limit switch modifications to valves MU-V-3, MU-V-26, WDL-V-304, WDL-V-534, and WDL-V-535. Deficient design margins of the AOVs were calculated during a design review (see the summary at the beginning of this trip report). The engineers were asked if they made an effort to determine if AOVs might have inadequate design margins or suspect calculations. They said that they did a search and did not find any. They also said that Crystal River 3 (similar plant with same AE) had a similar problem and the two plants discussed ways of increasing the margins of the affected AOVs. - Several 50.59 reports (see item 5 above for a list) describe a number of modifications and additions to the air systems prompted by a desire to improve performance of the equipment served.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
7	Describe AOV or air system actual or detected potential failures at the plant? Provide reference information, if possible.	Prior to 1990, the instrument air system performed poorly from the standpoint of providing clean, dry air. There was only one dryer, which had to be bypassed in order to service it. In response to GL 88-14, NUREG-1275, Vol. 2, and the B&WOG Instrument Air System Review Report, 47-11659565-00, TMI installed an additional instrument air system at that time. The "new" instrument air system, equipped with its own dryer, became the main supplier of instrument air for the plant and the "old" system is still used for backup. All of the air systems, including the service air system, are equipped with dryers and both of the instrument air systems have continuous dew point monitors and alarms in the control room. The "old" instrument air system has carbon steel piping upstream of the dryers while the "new" system has stainless steel piping. Both systems use copper piping downstream of the dryers. Receivers for both instrument air systems are carbon steel and the plant installed filters directly upstream of "critical" valves in order to protect against any particulate contamination. In addition, instrument air can be restored after a loss of offsite power event by operating a switch in the control room that powers the instrument air system from the diesel generators. A backup supply of air bottles provides two hours of air for the main steam and emergency feedwater systems. These features ensure reliable performance of the instrument air system, as well as the quality of the air provided. As a result of all the improvements in the instrument air system design, operation and maintenance, TMI has had few problems and no reportable problems related to air quality or quantity in the last few years.
8	Describe actions taken after events or failures involving AOVs or the air system. Provide reference information, if possible.	See item No.s 6 and 7 for a discussion of problems and actions taken. The TMI engineers were preparing an Air-Operated Valve Program at the time of the visit, and furnished us with a draft copy. We do not know when the plan will be finished or put into operation. This plan is similar to several that we have seen in visits to other plants and is related to the licensee's effort to meet the requirements of the Maintenance Rule, 10 CFR 50.65. The scope of the program includes: - Categorizing AOVs by importance. - Determining design-bases of AOVs. - Identifying methods to determine required and actual AOV capabilities (i.e., margins). - Methods for making any required hardware, software, or procedural modifications to ensure design basis operability of AOVs. - Determining test methods and test accuracy for AOVs. - Developing training methods, standards, and devices. - Upgrading and improving periodic maintenance and testing of AOVs to ensure long-term operability. - Developing methods and data to evaluate AOV failures, diagnostic system performance, and preventative maintenance frequency.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
9	Were there any actual or potential common mode or common cause failures in the air system or AOVs at the plant? Describe and provide reference information, if possible.	Solenoid valve failures described in MTAN No. 93-03 (see ITEM No.s 5 and 6) were caused by hardening of the O-rings. The root cause for the deterioration of the O-rings was described as aging, without further details. No further problems were reported since the corrective actions were completed.
10	Describe root cause analysis procedures for the plant. Provide reference information, if possible.	No specific or formal root cause analysis procedures were described or presented to us. From examination of the licensee's materials, it appears that root cause analyses are routinely attempted for all failures and events. Our impression was that the methods and results have improved over the years.
11	Describe root-cause analyses performed for air system or AOV failures at the plant. Provide reference information, if possible.	One root cause analysis described to us involved the calculation of insufficient margins in two Crane-Aloyco AOVs. Diagrams and excerpted pages from TMI Report 643, REQUIRED THRUST ANALYSIS, were provided to us and described the investigation of margins for the AOVs. (Refer to ITEM No.s 5 and 6.) Deficient design margins for the AOVs were calculated during a design review (see the summary at the beginning of this trip report). As noted above, Crystal River 3 had the same or a similar problem. This may be a common-cause (AE and/or valve manufacturer) problem.
12	Describe maintenance procedures for the air system. Provide reference information, if possible.	Preventive maintenance of the air system consists of filter replacement and dryer maintenance on a regular schedule. Dew point monitors are checked on a regular schedule and we saw information being recorded on automatic recorders attached to the monitors.
13	Describe maintenance procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	TMI has a system for investigating trends of recurring and common-cause failures. We were given several MAINTENANCE TREND ACTION NOTICEs that are pertinent examples of their system and applicable to our interest. A list of MTANs was also provided. TMI considers valves in all categories. The plant also provides standard preventive and corrective maintenance for AOVs. The plant engineer in charge of the air system has been involved with this system for quite some time and appears to be quite knowledgeable concerning AOV and air system maintenance problems. The maintenance of air-operated dampers and the AOVs for the diesel generators is done by the engineers and technicians in charge of the particular system. The TMI AOV program plan specifically excludes air-operated dampers. Preventive (and predictive?) maintenance of AOVs is to be accomplished as part of the Air-Operated Valve Program (see ITEM No. 8).
14	Describe IST procedures for the air system. Provide reference information, if possible.	No inservice testing of the air system was indicated, other than post-maintenance testing. We did note that records of air system moisture content are produced and we were told trends are monitored.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
15	Describe IST procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	Inservice testing of safety-related AOVs is accomplished in accordance with the applicable requirements of the ASME Code (stroke-timing testing under no-load conditions for those safety-related AOVs not exempted for some acceptable reason). No IST is accomplished on other AOVs.
16	Describe diagnostic systems, if any, used for AOVs. Provide reference information, if possible. Description of system: Specifications: Data collected and frequency of collection: Vendor assistance provided, if any:	TMI uses the MOVATS universal diagnostic system for their MOVs. The plant engineers said that they would use a similar or the same system, adapted to AOVs, but they did not provide specific plans or details for their approach to diagnostic testing of AOVs. The TMI draft AOV program plan indicates that diagnostic testing of AOVs is planned.
17	Describe design (and analysis) procedures for AOVs. Describe how design basis is established and maintained for AOVs. Provide reference information, if possible.	The draft AOV program plan provides for a review of the design bases and design calculations for AOVs. The engineers reviewed design calculations as a result of discovery of suspect design margins in two AOVs (see the discussion of Report 643 in Item 6 of this table).
18	Describe analyses and/or testing for verification of operability during postulated transient or accident conditions. Provide reference information, if possible.	We were not told of any testing done by GPU or TMI. Analyses of AOV designs were previously described (see previous entries this table) in TMI Report 643, REQUIRED THRUST ANALYSIS for two AOVs that appeared to have deficient design margins.
19	Describe training for installation, maintenance, and testing of AOVs. Provide reference information, if possible.	Training regarding AOVs was not described in any detail. The AOV draft program plan describes training practices to be used but our impression is that they are still in the planning stages of specific training on diagnostic testing of AOVs. Other maintenance training for AOVs involves hands-on, supervised training on sample valves and equipment, in accordance with standard training methods.
20	Describe databases used to track maintenance, failures, and events regarding AOVs.	TMI uses internal databases such as the MTAN and SE reports, as described in Item 5 of this table. The databases are used to track recurring failures that require additional maintenance and/or modifications. TMI is

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
	Provide reference information if possible. On site: Company wide: Industry:	recording failures and analyzing the data for trends and common-cause events.
21	Describe the impact of the Maintenance Rule, 10 CFR 50.65 on AOV and air system maintenance and testing. Provide reference information if possible.	As with the other plants that we visited, TMI is actively involved in compliance with the Maintenance Rule. Their model for guidance is Regulatory Guide 1.160 and NUMARC 93-01. TMI has ranked events in terms of calculated core-damage frequency (CDF) and is ranking systems and components, including AOVs, in terms of their importance to risk. An undated paper (one figure is dated 7/15/93), TMI-1 PRA INPUT TO MAINTENANCE RULE - RISK SIGNIFICANT SYSTEM LIST, developed by Risk Analysis Section, C. Adams, includes a description of the plant's compliance with the requirements of the Maintenance Rule.
22	Is PRA data used for predictive maintenance or replacement of AOVs? If so, how?	Valves have been categorized, in terms of importance to safety and operations, in accordance with the TMI-1 PRA INPUT TO MAINTENANCE RULE - RISK SIGNIFICANT SYSTEM LIST, developed by Risk Analysis Section, C. Adams. It was noted that 5.3% of calculated core-damage frequency was attributed to loss of instrument air. Loss of instrument air was ranked sixth in importance of initiating events, in terms of CDF (small LOCA was the highest at 18.8% of CDF).
23	Are AOVs serviced on site, serviced offsite, or replaced as piece-parts if found to require service?	The methods of repair vary with the circumstances. Vendor input is sought if the engineers need additional information about the design history, etc. AOVs and SOVs have been sent out in the past for repair or failure research.
24	Identify and describe the most common recurring maintenance problem(s) and failures regarding AOVs and the air system. What did you see? Provide reference information if possible.	The list of Maintenance Trend Action Notices that were provided to us indicate the valves and systems that have the most attention due to repeated or common-cause failures. MTAN 93-03 on solenoid valves and MTAN 93-15 on a sluggish valve were discussed with us. In addition, the problem described in TMI Report 643, REQUIRED THRUST ANALYSIS, for two AOVs that had been calculated to have insufficient design margins, although not covered by an MTAN, is significant because of the generic considerations.
25	Interviewer comments regarding actual valves viewed during the visit, in the plant, undergoing maintenance or replacement, or in the plant stock system, if applicable to this interview.	No AOVs undergoing maintenance were seen during our brief visit inside the plant.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY
SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION

ITEM No.	INFORMATION	RESPONSE OR INFORMATION
26	Has the plant made changes to valves or systems that include AOVs, or replaced AOVs with different models of AOVs or different valves that are not AOVs? If so, describe the changes and the circumstances. What prompted the change? Was the change made for this plant only?	Several SOVs were changed to improve reliability. The old valves had deteriorated O-rings. See MTAN 93-03. As noted above, a second instrument air system and filters upstream of "critical" valves were added. This was prompted by the lack of redundancy and poor performance of the "old" service air system.
27	Does the plant follow EPRI/NMAC guidelines for maintaining AOVs and the air system(s)? If not, describe differences and reasons for the differences. Provide reference information, if possible.	The recommendations of EPRI/NMAC, along with other industry groups, is reviewed for applicability to the plant. TMI has incorporated a number of improvements to enhance the quality of instrument air.
28	What is the plant doing or planning to do in response to the recent Industry correspondence on AOVs. Provide reference information, if possible.	TMI reviewed the Industry correspondence no specific response is planned.
29	Do you have any suggestions for improving the performance of AOVs, particularly in the areas of surveillance, testing, or maintenance?	The B&W Owner's Group Report (47-1165965-00) dated December 1986, INSTRUMENT AIR SYSTEM REVIEW REPORT, was provided to us. Gordon Skillman, who was part of the B&W Safety and Performance Improvement Program team discussed the report with us. A number of recommendations in the report are pertinent to all plants as well as B&W designs. TMI used a number of the recommendations in modifications to its air system. The specific recommendations for TMI included: - In a major loss-of-air event (e.g., air pipe break), high air flow rates pass through the driers and filters. The desiccant could break up and clog the filters. A bypass procedure or automatic bypass of the dryers and filters, in a major loss-of-air event, was recommended. - The local AOV accumulators were not being checked for operability. The recommendation was to inspect the accumulators and their associated check valves. - Local accumulators had no water blowdown method available to remove water accumulation. Any water present reduces the

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
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		available air volume. The recommendation was to check the accumulators for water (at the next outage). - Failure of an air compressor could lead to a rapid decrease in air system pressure if the system has no reverse flow check valve between each compressor and the rest of the system. It was recommended that a particular valve (IA-V12) should be positioned normally closed. Refer to item 32 for additional comments on the actions taken in response to the B&WOG report.
30	Provide a list of 10 CFR 50.59 and 10 CFR 50.72 reports on AOVs and AOV support systems (air or inert gas supply, etc.) that have been issued for this plant.	A list of 50.59 reports is provided in Item 5 of this table. No 50.72 reports were provided.
31	What thrust or torque margins are expected for AOVs? Are different margins used for safety-related, important non-safety-related, or non-safety-related AOVs?	TMI expects margins better than 10%. They are in the process of categorizing valves as those having 10% or greater margin (acceptable), those with 5% to 10% margin (to be improved, if practical), and those with less than 5% margin (unacceptable, must be improved). They are in the planning stages of this process, for the most part. Refer to the TMI program plan for AOVs.
32	What maintenance or surveillance is done to AOV accumulators to ensure air/nitrogen quality and pressure? Were seismic considerations and size verified?	No maintenance, other than corrective maintenance is performed on accumulators. The B&WOG Report on air systems noted the need to ensure that accumulators are clean and dry. Generic Letter 88-14 requires verification of the capability of air-operated components to perform their safety function. A TMI-1 50.59 report, SE 412512-005, indicates that several vent valves were added to the air system for testing the functional capability of accumulators. However, no inspection of the interior of the accumulators or means for blowdown is indicated. The engineers at TMI said that they added drains to those accumulators that had moisture in them when they applied the recommendations in the B&WOG report; however, they had made no modifications to other accumulators.
33	Describe problems with pressure regulators, if any.	SE No. SE-CMR-113202, Replacement of Failed IWT Regulators, 2/5/90 indicated that several failed pressure regulators were replaced although the cause of failure was not noted. Several of the 50.59 reports noted previously above describe the installation of filters upstream of critical valves. These would also protect the positioners and regulators associated with the valves.

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THREE MILE ISLAND, UNIT 1, NUCLEAR GENERATING STATION		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
34	Describe problems with feedwater regulating valves, if any.	There were no problems since the instrument air system improvements were made.
35	What, if any, is your involvement with the AOV Users Group? Describe.	The plant engineers have a representative in the AOVUG who is also an active participant in the Joint Owners Group AOV efforts.

Trip No. 6 REPORT
STUDY OF AIR-OPERATED VALVES
INDIAN POINT, UNIT 3, MARCH 10 AND 11, 1998

We had two days of meetings with the engineers concerned with the air systems and air-operated valves (AOVs) at Indian Point 3 (IP3). We were escorted through portions of the plant and saw some of the air systems and AOVs in the plant. Indian Point 2 (IP2) is adjacent to IP3 but is owned and operated by another company, Consolidated Edison. Indian Point 1 (IP1), also adjacent to IP3, is shut down. IP3 was operating at full power at the time of our visit.

IP3 has maintenance engineers and system engineers who are charged with maintaining the operational readiness of the air system and the operability of AOVs for the plant.

One portion of the instrument air (IA) system at IP3 is served by two 225 scfm compressors (IAC #31 and #32) that serve a common receiver. Two banks of two desiccant-type dryers are downstream of the receiver. A cross connection to the station air system is upstream of the dryers to allow use of station air in an emergency. A third 350 scfm IA compressor (IAC #33) was added to the system and has its own receiver and a desiccant-type dryer. Two other IA compressors (designated Admin #31 and #32) were also added to the system and each of these has its own receiver and one desiccant-type dryer. The #31 and #32 IA system compressors and dryers can be loaded on to the station diesel generators in the event of a loss of offsite power .

The service air (SA) system is served by a compressor, a backup diesel powered compressor, and a tie-in from IP2. The diesel SA compressor has a deliquescent-type dryer downstream. A separate polisher building was added and contains a separate polisher air system. It is served by two redundant 400 scfm compressors (Polisher #31 and #32). The polisher air system has one air receiver and a bank of two refrigerant-type air dryers.

The components or systems essential to plant safety and serviced by the IA system are:

1. containment isolation valves,
2. cooling water valves for containment building fan coolers,
3. condensate storage tank shut-off valves,
4. auxiliary boiler feed pump control valves,
5. steam dump valves to atmosphere,
6. low pressure steam dump valves to main condenser,
7. weld channel and containment penetration pressurization system (WCCPPS),
8. emergency diesel cooling water valves,
9. spray additive tank outlet valves,
10. boron injection tank recirculation valves, and
11. control room air conditioning dampers.

In the event of low pressure in the IA system, air is automatically supplied to the IA system from the SA system. In the event of low pressure at any or all of the components listed as items 4, 7, or 11, dry nitrogen cylinders automatically supply gas pressure to those components required for safe shutdown or continued plant operation. In addition, a manual supply of dry nitrogen is available, in the event of low IA system pressure, to operate the steam dump valves. In the event of an IA system line rupture in the conventional plant, a restriction orifice limits the flow to 225 scfm. One compressor supplies air to compensate for the break and the spare compressor supplies the primary plant.

Originally, the IP3 IA system had carbon steel piping upstream of the dryers and copper piping downstream. A major portion of the carbon steel piping was replaced with stainless steel, with the exception of the pipe connections near the aftercoolers, and a section of pipe for the IA containment isolation valve.

The weld channel and containment penetration pressurization system (WCCPPS) is an unusual innovation. This system provides gas to the containment penetrations and the liner weld channels at higher than containment pressure in order to provide an additional barrier against containment leakage. IP2 has a similar system. Zion also has such a system, although of different design. We were told that moisture intrusion from the instrument air system, prior to actions taken by the licensee to improve air system quality, caused some contamination of this system, some corrosion damage to valves and piping, and back leakage of moisture into portions of the nitrogen supply piping. These problems were corrected by upgrading the original IA system.

The IA system at IP3 is equipped with continuous dew point monitors with digital readout on a small instrument adjacent to the compressors. This readout is monitored frequently and anomalies are noted. In addition, if the dew point exceeds -15^{0}F, an alarm is activated in the control room.

IP3 has 578 AOVs in their plant. Of these 263 AOVs are classified as safety-related. Also, 215 of the 578 AOVs are classified as within the scope of the Maintenance Rule. By way of comparison, 89 motor-operated valves are within the scope of Generic Letter 89-10.

IP3 initiated a program plan to improve the performance of their AOVs based on failure rates that they estimated to be twice the industry average. This plan includes categorization of AOVs by importance, engineering evaluations to establish design requirements, and extensive use of diagnostic testing to identify maintenance requirements. They intend to implement their program fully by 1999.

The cooperation, courtesy, and knowledgeable responses from the members of the IP3 engineering and technical staff were noted and appreciated by those of us who are involved in this study of AOVs.

The following tabular summary was prepared to describe the information gathered during the subject visit. The NRC, with assistance from the INEEL, is studying the performance of air-operated valves (AOVs) in commercial nuclear power plants. The information was collected in accordance with the Program Plan dated 10/22/97 (INEEL Letter to H. Ornstein, NRC, from J. Bryce, 10/23/97, Job Code E8238, Task Order 15 - JHB-167-97).

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO INDIAN POINT, UNIT 3, NUCLEAR POWER PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
1	Date.	March 10 and 11, 1998.
2	Name of Interviewer.	Hal Ornstein, NRC/AEOD, 301/415-7574 Owen Rothberg, INEEL/Rockville, 301/816-7773 John Watkins, INEEL/Idaho Falls, 208/526-0567 Jit Vora, NRC/RESEARCH, 301/415-5833
3	Plant Name & Docket No.	Indian Point, Unit 3, Docket No. 50-289.
4	Person(s) Interviewed, Title(s), Phone Number(s), E-Mail address, short description of organization(s) and duties.	Plant Address is: Indian Point 3 Nuclear Power Plant, P.O. Box 215, Buchanan, NY 10511. Phone numbers listed below are 914/736-xxxx unless otherwise noted. K. Peters, Licensing Manager, IP3, x8029 J. Comiotes, General Manager, Operations, IP3, x8002 J. Odendahl, Acting General Manager, IP3, x8701 M. Carmichael, QA Manager, IP3, x8501 R. Schmitt, Maintenance Eng. Supervisor, IP3, x8632 K. Eslinger, Systems Eng. Supervisor, IP3, x8993 M. Dinelli, Performance Engineer, IP3, x8315 Jim Werner, Systems Engineer, IP3, x8319, FAX 914/734-6031 Joe DeRoy, Director of Engineering, IP3, x8006 Bob Barrett, Site Executive Officer, IP3, x8001 Victor P. Rizzo, Maintenance Engineer, IP3, x8536 Tat Chan, System Engineer, IP3, x8874 Michael J. Dries, System Engineer, IP3, x8382 Becky Green, ORG Engineer, IP3 Bob Dolansky, IST Engineer, IP3, x8458, FAX x8350 Andrew Mihalik, Senior Systems Engineer, x8362 Anthony Dicescaro, Engineer involved with Maintenance Rule implementation David Lew, NRC Senior Resident Inspector, IP3, P.O. Box 337, 914/739-8565, FAX 914/739-8624 Laura Dudes, NRC Resident Inspector, IP3, P.O. Box 337, 914/739-8565, FAX 914/739-8624

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5	If necessary, and if person(s) interviewed can do so, obtain any missing information not provided prior to the site visit, as described in the outline for Task 4 of the program plan. Note what information was provided.	A copy of our program plan was provided to IP3 prior to the visit. Slides of the IP3 AOV Program Overview by Victor Rizzo, IP3, and slides of the IP3 Field Testing Experience by Rusty Gasser, Crane-Movats, provided to the NRC TM at a MOV/AOV conference in early December 1997, were reviewed before the visit and used in the discussions at the plant. We arrived with copies of the following materials: - Rev. 2 of Section 9.6.3 of the IP3 FSAR, Compressed Air System (including a flow diagram of the instrument air (IA) system) - General facility description taken from the NRC Internet home page - LER 28685002, Reactor Trip Resulting from Feedwater Regulating Valve Positioner Failure - LER 28685006, Reactor Trip Resulting from Feedwater Regulating Valve Positioner Failure - LER 28688006, Reactor Trip Resulting from failure of Main Turbine Air Pilot Valve Failure Due To Foreign Materials on the Valve Internals - LER 28688009, Failures of ASCO Solenoid Valves Due to Lubricant Problems - LER 28693013, Damper Failures Caused by Dirty Linkages - LER 28693036, Improper Seismic Mounting of Damper Actuators in the Control Room - LER 28693045, Failure Mode of Central Control Room Dampers Upon Loss of Instrument Air LER 28693050, Solenoid Valves Can Be Over-Pressurized - LER 28696002, Loss of EDG After Loss of Offsite Power Due to Pressure Regulator Failure for Ventilation Control System - LER 28696004, Two Diaphragm AOVs in Series Discovered to be Incapable of Closing at System Pressure - LER 28696008, Nitrogen Pressure Below Minimum to Adequately Supply Isolation Valve Seal Water System - LER 28696013, AOV Prevented From Fully Opening Due to Crud Buildup Inside Valve (continued next page)

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5 (cont.)	Information Provided (continued)	The engineers at IP3 provided us with the following information: - An informal sketch of the Unit Air System (for information only) - IP3 Drawing #9321-F-20363, Rev. 52, Flow Diagram, Instrument Air, Sheet 1 - IP3 Drawing #9321-F-20363, Rev. 4, Flow Diagram, Instrument Air, Sheet 2 - IP3 Drawing #9321-F-27233, Rev. 36, Flow Diagram, Nitrogen to Nuclear Equipment - IP3 Drawing #9321-F-27623, Rev. 39, Flow Diagram, Penetration & Liner Weld Joint Channel pressurization System - NYPA Memorandum RET-96-311 dated 11/5/96, IP3 Maintenance Rule Basis Documentation Revised Importance Ranking with a 47-page table attached entitled IP3 IPE Risk Achievement/Reduction Worths - Table entitled Air Operated Valve Listing by system-component dated 3/10/98, 65 pages - Table entitled Air Operated Valve Listing by manufacturer-component dated 3/10/98, 65 pages - IP3 Action Plan IDSE-APL-96-061, ACTS No. 96-22957, Rev. 0, dated 1/6/97, entitled The Return of 32 IA Compressor to Maintenance Rule Status "(a)(2)" and the Resolution of 32 & 32 IA Compressor Replacement or System Upgrade - Customer Service Guide and Technical Data for Drain-All Instrument Air Drain Traps, (3/17/94) - IP3 Engineering Memo dated 5/5/95, Subject: Evaluation of PCV-1310A and B (MSIVs) Operation and Performance This memo describes the resolution of a concern with the operation of the MSIVs The plant engineers sent us a box of event reports and other data after the trip (5/28/98). A number of work requests were included that they considered to be of interest for this study of AOVs. These were: - WR 93-10265-00, Replace diaphragm on aux. feed regulating control valve operator (PCV-1310A). - WR 93-10284-03, FCV-447-SOV-1 could be over-pressurized. - WR 94-00005-03, CVCS SOV-LCV-112A could be over-pressurized. (continued next page)

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5 (cont.)	Information Provided (continued)	- WR 94-02963-04, Diaphragm leak in 32 cation bed first injection AOV. - WR 94-04801-00, PCV-1193 in WCCPPS does not respond to adjustments. Regulator does not control pressure. - WR 95-04694-02, Follow-up to WR-04801-00 on PCV-1193. - WR 96-00020-00, Air leak on air-regulator to AOV FCV-1173. - WR 96-01430-00, Valve 4EX-2 operator stuck. - WR 96-01717-01, Excessive air pressure required to stroke AOV LCV01127C, heater drain tank to condenser #32 bypass. - WR 96-01770-01, Repair pressure regulator for PCV-1200 in WCCPPS. - WR 96-02488-02, SG#31 MSIV supply SOV air leaks. - WR 96-03865-00, IA regulator to MS-FCV-1155 has a diaphragm leak. - WR 96-05736-00, #31 anion bed rinse control valve PM inspection request on AOV. - WR 96-06664-00, CVCS FCV-111A strokes slowly. - WR 97-00797-00, #32 condenser high level control valve packing leak. (Package contains IP3 Generic Valve Packing Procedure). - WR 97-04166-06, Need to blow down piping in WCCPPS. Repaired and cleaned pressure regulator. - WR 97-04166-07, Need to blow down piping in WCCPPS. Rebuilt pressure regulator. - WR 97-04684-00, SGBDR recovery outlet PCV erratic behavior causes flow and pressure spikes. (Package contains Copes-Vulcan instruction manual for diaphragm-actuated control valve.) - WR 97-05214-32, #32 cation bed backwash control valve air operator, bad diaphragm. - WR 97-05643-01, Valve BFD-FCV-1116 identified to be leaking. - WR 97-05953-46, Primary water containment isolation valve failed stroke test. - WR 97-05957-35, Pressure regulator gage for N2 supply broken. - WR 97-05957-81, Cation regeneration vessel backwash inlet valve positioner leaking air. - WR 97-05957-82, Resin mixing and hold vessel rinse inlet valve positioner leaking air. - WR 97-05957-84, Cation regeneration vessel rinse inlet valve leaking air on positioner. (continued next page)

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5(cont.)	Information Provided (continued)	- WR 97-05957-85, Resin mixing and hold bottom water inlet valve positioner leaking air. - WR 97-06959-01, Resin mixing and hold vessel rinse and drain outlet valve leaked air. - WR 97-06959-40, SGBDR purification system bypass valve supply pressure regulator broken. - WR 98-00506-00, #32 anion bed rinse control valve air operator flange leak.
6	Describe plant events involving AOVs and provide reference information, if possible. Recent: Recurring: Significant:	We discussed the LERs and other events listed in item 5, above. Problems included solenoid valve lubrication, dirt in working parts of valves, and mechanical parts failures. Problems with packing friction and other maintenance adjustments are common.
		The plant engineers sent us a box of event reports and other data after the trip (5/28/98). Events or conditions of interest for this study of AOVs included the following items: - DER 93-734, SOVs can fail due to over-pressurization. This DER and associated LER 28693050 indicate that a number of non-safety-related IA regulators were installed. Over-pressurization of SOVs [exceeding designed maximum operating pressure differential (MOPD)] was not recognized by the plant designers as a failure mechanism. Generic Letter 91-15, Information Notice 88-24, and NUREG-1275, Vol. 6, described the problem but the plant engineers did not recognize the problem and make the necessary changes promptly. In addition to replacement of SOVs 1276 and 1276A, corrective actions were taken for 109 SOVs having insufficient MOPD. - DER 94-097, Failure of Appendix R emergency diesel air-start motor supply valves, AOV-112 and -113 due to inadequate operator pressure rating. This was an original plant design mistake. - DER 94-109, AOV-113 was reinstalled in wrong orientation (includes an analysis of the event and critique of procedures). - DER 94-397, The controls (SOVs) for AOVs FCV-1204 and -1205, quench-water supply valves to the 31 heater drain pump, were not the proper design for the intended function and had to be replaced. - DER 94-894, Feedwater valve stuck open. - DER 94-1105, Improper maintenance and reassembly of AOV BFD-FCV-406 C and D. - DER 95-0033, Critique of post-installation tests of diaphragms on Copes-Vulcan D-100 AOVs.

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
6 (cont.)	Describe plant events (continued)	- DER 95-0233, Investigation of the use of Teflon tape on pipe threads in the plant. Includes a number of background references. Contamination of stainless steel piping from migration of flourides in Teflon products and blockage of small orifices from "plate-out" of Teflon on hot surfaces are major concerns. - DER 95-0395, Investigation of the (incorrectly) reported failure of the station air/instrument air backup valve (PCV-1142). - DER 95-1302, Failure of PCV-1310A, Main Steam Isolation Valve to Aux. FW Pump, to stroke. See DER 96-2747. - DER 95-1392, Spring washers were shown on a drawing for SWN-PCV-1205 through 1210 but were deleted by the manufacturer. The washers are not needed and are not installed. - DER 95-2244, High failure rate for instrument air dewpoint probes was investigated. - DER 95-2317, Failure of CH-AOV-204B, normal charging isolation valve to open. Operator tapped the SOV and the valve operated normally afterward. Results of investigation not included. - DER 95-2517, Fail-safe position of valve WD-AOV-1610 listed in the FSAR did not agree with system drawings. - DER 96-0064, Normal charging valve, CH-AOV-204B, failed to open. No analysis or cause included. - DER 96-2056, Investigation of crud buildup in service water system valves (PCV-1296 and 1297, 2 inch WKM Type 70-28-1D pressure control valves) which resulted in inadequate service water supply to the control room AC system. LER 28696013 was included. NRC Generic Letter 89-13, "Actions for Low-Flow SWS Lines Susceptible to Crudding" applies. DERs 96-1267, 96-1422, 96-1756, 96-1763, 96-1776, 96-1779, and 96-1780 were noted as related. - DER 96-0393, AOV, BFD-FCV-406A, B, and C, feedwater regulating valves did not stroke smoothly without system pressure and did not open fully during testing. The I/P controllers were recalibrated and the "popping" operation of the valves was investigated and subsequently considered satisfactory. See DER 96-0812 for a comprehensive history of these and related AOVs. - DER 96-0416, An exhaustive discussion of the merits and objections to protecting accumulator vent valve, HCV-943, from foreign material intrusion. DERs 95-1299 and 94-0809, both referring to the same issue, were included. (continued next page)

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO INDIAN POINT, UNIT 3, NUCLEAR POWER PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
6 (cont.)	Describe plant events (continued)	- DER 96-0557, An AOV, DA-AOV-112, DG starting air inlet control valve, failed because a label was applied by the licensee over the (very small) vent port. Other similar valves were investigated to ensure that the problem was resolved. Receipt inspection procedures were modified. Drawings of the valve were included. - DER 96-0687, A nitrogen pressure regulator for AOV PCV-1196 in the WCCPPS was replaced with a regulator from IP-2. Debris and corrosion products in the re-used regulator caused it to leak during a post-installation test. It was inferred that the deteriorated condition should have been discovered prior to installation. - DER 96-0812, This DER contains a history of the leakage problems associated with the auxiliary feed pump discharge flow control valves FCV-406C and D, and the plant operation problems caused by their excessive leakage. It appears that the basic problem is that the operators are undersized for maximum pressure conditions. See DER 96-0393 for another problem with these or related AOVs. DER 96-1029 indicated that FCV-406-A and D stuck on opening with no pressure in the line. DER 96-0696, Fine mesh screens were apparently supposed to have been provided in the inlets of Weld Channel and Containment Penetration Pressurization System (WCCPPS) valves PS-PCV-1194, 1196, 1198, and 1200. The screens were found to be not required because filters were provided when regulators were changed. The design-basis documentation was updated. - DER 96-0905, Failure of main turbine stop valve equalizing valve to close. The horizontal orientation of the valve caused stem binding. The AOV manufacturer (Copes- Vulcan) recommended additional bushings and packing redesign to reduce stem binding. - - DER 96-1029, AOV BFD-FCV-406A was reported to have stuck closed on a demand to open with no pressure or little open-direction pressure on the valve seat. DER 96-1367 (copy not provided) was referenced in this package and was described as the document for tracking implementation of an AOV program. - DER 96-1267, CCRAC pressure control valve, SWN-PCV-1297 moved to mid-position on demand to move to full open and resulted in a 72-hour LCO condition. A blocked sensing line in the pneumatic controller was cleared. (continued next page)

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO INDIAN POINT, UNIT 3, NUCLEAR POWER PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
6 (cont.)	Describe plant events (continued)	- DER 96-1422, Repair of sensing line and pneumatic controller for PCV-1297 (DER 96-1267). Dirt from the service water system clogged the line. In addition, a mistake was found in the calculated effective pressure drop on the manufacturer's data sheet. This AOV is a WKM 2 inch model 70-28-1D pressure control valve. Refer to DER 96-1776. See DER 96-1756 for repair of the controller. - DER 96-1674, No. 32 EDG starting air pressure regulator DA-PCV-14-3 was not maintaining pressure. Foreign material was found in the space between the diaphragm and the valve seat. Debris from the air start system (corrosion product perhaps?) was suspected. The foreign material was removed by cycling the valve open and closed. The corrective actions note that the air tank was scheduled to be cleaned and piping had been changed to stainless steel to prevent foreign material intrusion in the future. - DER 96-1756, Repair of pneumatic controller PC-1432 for AOV PCV-1297 (see DERs 96-1267 and -1422). Several components and linkages were broken and an orifice was clogged. - DER 96-1776, Valve internals were removed from PCV-1297 in order to obtain required flow. Investigation confirmed that the manufacturer's original data sheet was not correct. See DER 96-1267 and -1422. LER 28696013 refers to, and was included in this package, but does not explicitly mention the mistake in the manufacturer's data sheet. - DER 96-1779, Critique of valve vendors data and plant procedures related to insufficient flow in AOV PCV-1297. See DERs 96-1776, -1422, and -1267. - DER 96-1780, AOV PCV-1297 was not stroke tested prior to retest. See DERs 96-1267, -1422, -1776, and -1779. Package includes control room AC diagrams and a copy of the General Troubleshooting Procedure (IC-AD-13) for IP3. - DER 96-1973, The FSAR indicated that a single failure in the Weld Channel and Containment Penetration Pressurization System (WCCPPS) would not prevent fulfillment of the design function of the system. On 8/30/96, the plant engineers realized that the FSAR was incorrect; a single failure of the air or nitrogen regulators would render the WCCPPS incapable of performing its design basis function. The primary function of the WCCPPS is to prevent leakage from the containment to the surrounding environment. An operability determination investigation is included in the package and the conclusion was that the single air or nitrogen regulator failure would not impact public health and safety. (continued next page)

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
6 (cont.)	Describe plant events (continued)	- DER 96-1985, Evaluation of the differences and similarities between PCV-1310B and A. PCV-1310A did not open properly but PCV-1310B did. These are two identical AOVs in series. These are steam isolation valves to the 32 auxiliary boiler feed pump turbine. See DER 95-1302 and 96-2747. - DER 96-2056, Debris found in PCV-1296. See DER 96-1267, 96-1780, and others that refer to these AC compressor control valves. - DER 96-2239, Failure of nitrogen pressure regulator in Weld Channel and Containment Penetration Pressurization System (WCCPPS). LER 28696010 was included. Particulate contamination in the system was suspected to have caused the regulator failure. - DER 96-2747, Investigation of the failure-to-open of PCV-1310A, Steam Isolation Valve to Aux. FW Pump, failure to open during a stroke-timing test. This DER included a detailed analysis of the operation, maintenance, and design of the AOV. Other DER reports in the box sent on 5/28/98 included: 94-0398, 94-0765, 95-0384, 95-0561, 96-1763, 96-2241, and 96-2649. Also included in the box sent on 5/28/98, along with the event reports, were: - Listing of DER history on IA systems through 3/2/98 - Listing of DER history on AOVs through 3/2/98 - Listing of maintenance work history on air operators through 3/2/98 - Listing of maintenance work history on SOVs through 3/2/98 - Listing of DER history on SOVs through 3/2/98 - Listing of maintenance work history on compressors through 3/2/98 - Listing of maintenance work history on regulators through 3/2/98 - Listing of maintenance work history on valves
7	Describe AOV or air system actual or detected potential failures at the plant? Provide reference information, if possible.	IP3 initiated an AOV program because their failure rate was found to be about twice the industry average over the past few years. See the slides of the IP3 AOV Program Overview by Victor Rizzo, IP3, and slides of the IP3 Field Testing Experience by Rusty Gasser, Crane-Movats, provided to the NRC TM at a MOV/AOV conference in December 1997. Their program includes engineering reviews, diagnostic testing, and equipment overhaul. IP3 made a number of repairs and adjustments to their AOVs based on this program.
8	Describe actions taken after events or failures involving AOVs or the air system.	Before the late 1980s, IP3 had numerous problems attributable to contaminated air (mostly moisture) in the past. A number of improvements were made including on-line dew point monitoring and

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	Provide reference information, if possible.	alarms in the control room, use of automatic drains, and replacement of carbon steel piping with stainless steel. It appears that IP3 now provides clean, dry air to AOVs in the plant, thereby eliminating the potential for common-cause AOV failures from instrument air contamination.
9	Were there any actual or potential common mode or common cause failures in the air system or AOVs at the plant? Describe and provide reference information, if possible.	One very recent event that occurred at Dresden and Quad Cities involving potential failure of AOV diaphragms on Copes-Vulcan D100 valve operators was discussed with the maintenance engineers. The problem is that the elastomer covering over the fibers in the diaphragms is too thin and the diaphragms wear out as the valve is operated. IP3 engineers had not been aware of this problem and since they have a large number of this type of valve they plan to investigate immediately. This is a potential common-cause failure mechanism. Another problem, mentioned above, involved the lubricant used on ASCO solenoid valves. This problem was reported at other plants and is covered in NRC correspondence. There appears to be no way to determine the internal condition of AOV accumulators, including the presence of corrosion products or moisture. This is a common problem at most, if not all, of the plants visited.
10	Describe root cause analysis procedures for the plant. Provide reference information, if possible.	IP3 has a formal root cause analysis procedure (Administrative Procedure 8.2) based on standard industry practices and references. Except for some of the early ones, LERs identify root causes.
11	Describe root-cause analyses performed for air system or AOV failures at the plant. Provide reference information, if possible.	Several root-cause analyses were described to us including the IP3 determination of problems having to do with ASCO solenoid valves and their subsequent replacement. The plant's adoption of diagnostic system technology indicates their dedication to determining root causes of AOV failures.
12	Describe maintenance procedures for the air system. Provide reference information, if possible.	The plant has upgraded their air system over the past few years to improve capacity and ensure air quality. As noted above, the air system has been equipped with adequate drying capacity and filtration, both general and local, to the valves. A major portion of the carbon steel piping between the 31 and 32 IA compressors and the copper air headers was replaced with stainless steel. One of the plant's objectives was to reduce the number of compressors and dryers, while increasing capacity, and to replace compressors with less maintenance-intensive ones.
13	Describe maintenance procedures for AOVs. Provide reference information, if possible. Safety-related:	AOV maintenance procedures are to be based on diagnostic testing. IP3 has a two-phase program in place to determine the condition of their AOVs. The first phase is a pilot program for selected AOVs and is based on their refueling outage needs, their priorities from the Maintenance Rule, safety-related importance, and plant efficiency. The second phase, as yet

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	Important non-safety-related: Non-safety-related:	to be fully implemented, includes a comprehensive program of industry involvement, engineering reviews, diagnostic verification, and training.
14	Describe IST procedures for the air system. Provide reference information, if possible.	No IST is done for the air system because it is considered to be a non-safety related system.
15	Describe IST procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	IST procedures for AOVs consist of ASME stroke-time testing of safety-related AOVs. IP3 uses the provisions of Section XI of the 1983/Summer 1983 ASME Code. They are also implementing provisions of OM-10 as allowed by NUREG-1482. No other IST is mandated. The remark was made that it is rare to pick up an incipient valve failure during a stroke-timing test. The IP3 program for AOV diagnostic testing is not mandated by IST considerations. Rather it is being driven by high failure rates and their Maintenance Rule and QA requirements.
16	Describe diagnostic systems, if any, used for AOVs. Provide reference information, if possible. Description of system: Specifications: Data collected and frequency of collection: Vendor assistance provided, if any:	IP3 uses the CRANE/MOVATS diagnostic system for AOVs. Their AOV program includes heavy dependence on diagnostic testing to identify maintenance and adjustment problems with AOVs. They are doing engineering evaluations to determine design basis requirements and it is not clear how they plan to relate that to available design margins. They will have to rely on the manufacturer's data to determine the designed capability of their AOVs. Their program will at least ensure that their AOVs are capable of opening and closing without loss of margin due to misadjustment or deterioration. IP3 cannot assess margins for AOVs without knowing the design bases. If the manufacturer's data and methods are used without verification, the design bases and the margins for the AOVs may not be conservatively estimated.
17	Describe design (and analysis) procedures for AOVs. Describe how design basis is established and maintained for AOVs. Provide reference information, if possible.	The AOV program plan calls for a review of the design basis and a design review. Additional calculations may be required. The design basis was established when the plant was designed and the calculations of the valve manufacturers will have to be reviewed to try to establish if acceptable margins were provided. This effort is not complete at IP3.
18	Describe analyses and/or testing for verification of operability during postulated transient or accident conditions. Provide reference information, if possible.	No testing is being done or is planned to determine design-basis operability. Analyses, or examination of existing analyses, is planned as part of the AOV program plan for a number of safety-related and important valves and operators.

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19	Describe training for installation, maintenance, and testing of AOVs. Provide reference information, if possible.	Training improvement is part of the AOV program plan. The training was described as mostly "on-the-job" with close supervision.
20	Describe databases used to track maintenance, failures, and events regarding AOVs. Provide reference information if possible. On site: Company wide: Industry:	Copies of several printouts from the databases for AOVs were provided. Sorts by manufacturer and system were provided and it is possible to construct sorts in a wide variety, including safety-related, containment isolation, PRA ranking, etc.
21	Describe the impact of the Maintenance Rule, 10 CFR 50.65 on AOV and air system maintenance and testing. Provide reference information if possible.	As with all of the plants visited, the impact of the Maintenance Rule is apparent. IP3 has had to determine those valves important from a risk perspective and compare those valves with their safety-related AOVs. In addition, the plant realized that a number of AOVs are significant from an economic perspective. The AOV program plan is based heavily on the rankings needed to satisfy the requirements of the Maintenance Rule. No (a)(1) goals for the air system or AOVs were in place in the summary dated 12/30/97 that we looked at. An action plan, dated 1/6/97, for return of the #32 IA compressor to (a)(2) status was reviewed.
22	Is PRA data used for predictive maintenance or replacement of AOVs? If so, how?	Predictive maintenance is being considered by the plant and PRA data is now used for ranking the importance of AOVs. The AOV program plan does not include predictive maintenance or replacement of AOVs based on PRA data or techniques.
23	Are AOVs serviced on site, serviced offsite, or replaced as piece-parts if found to require service?	IP3 services valves on site or offsite, depending on the individual circumstances. Solenoids are generally replaced as piece-parts, although some have been sent to the manufacturer for analysis (lubricant problem).
24	Identify and describe the most common recurring maintenance problem(s) and failures regarding AOVs and the air system. What did you see? Provide reference information if possible.	The most common AOV problems, as identified from the diagnostic field test experience on 10 AOVs, appear to be related to packing, seating/unseating difficulties and positioner calibration.
25	Interviewer comments regarding actual valves	Although a number of AOVs were viewed, there were none shown to us that were undergoing replacement or maintenance. The plant was at full

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	viewed during the visit, in the plant, undergoing maintenance or replacement, or in the plant stock system, if applicable to this interview.	power at the time.
26	Has the plant made changes to valves or systems that include AOVs, or replaced AOVs with different models of AOVs or different valves that are not AOVs? If so, describe the changes and the circumstances. What prompted the change? Was the change made for this plant only?	The plant has not changed their AOVs although several ASCO solenoids were replaced with Valcor solenoids in an effort to improve performance (sticking). The changes were considered successful.
27	Does the plant follow EPRI/NMAC guidelines for maintaining AOVs and the air system(s)? If not, describe differences and reasons for the differences. Provide reference information, if possible.	The EPRI/NMAC guidelines were consulted during the preparation of the AOV program plan at IP3.
28	What is the plant doing or planning to do in response to the recent Industry correspondence on AOVs. Provide reference information, if possible.	Although not the driving force for preparation of the IP3 AOV program plan, the recent industry correspondence and interest in AOVs did serve to focus management attention on the issue.
29	Do you have any suggestions for improving the performance of AOVs, particularly in the areas of surveillance, testing, or maintenance?	The preparation of a comprehensive program plan appears to be the centerpiece of the IP3 effort to improve AOV performance. Their objective is to reduce AOV failures and corrective maintenance.
30	Provide a list of 10 CFR 50.59 and 10 CFR 50.72 reports on AOVs and AOV support systems (air or	No reports were reviewed.

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	inert gas supply, etc.) that have been issued for this plant.	
31	What thrust or torque margins are expected for AOVs? Are different margins used for safety-related, important non-safety-related, or non-safety-related AOVs?	Various figures were discussed from 0 to 20%. Certainly no negative margins would be tolerated. The plan is to look at the results and decide if a margin is acceptable based on the individual circumstance. Referring to item 16 in this table, if the design bases are not conservatively established then the margins will not be meaningful.
32	What maintenance or surveillance is done to AOV accumulators to ensure air/nitrogen quality and pressure? Were seismic considerations and size verified?	No specific maintenance is done on AOV accumulators. There is no direct way to inspect the conditions inside the accumulators.
33	Describe problems with pressure regulators, if any.	Refer to DERs included in supplemental material forwarded on 5/28/98.
34	Describe problems with feedwater regulating valves, if any.	Two 1985 LERs indicated problems with FRVs. These problems were related to the poor quality of the air. No recent incidents were discovered. Refer also to DERs included in supplemental material forwarded on 5/28/98.
35	What, if any, is your involvement with the AOV Users Group? Describe.	The maintenance engineers are active in the industry groups as part of their implementation of the AOV program plan. This plan was recently presented and discussed at an AOV Users Group meeting.

Trip No. 7 REPORT
STUDY OF AIR-OPERATED VALVES (AOVs)
TURKEY POINT, UNITS 3 AND 4, MARCH 24 AND 25, 1998

We had two days of meetings and interviews with the engineers at Turkey Point 3 & 4 (TP) who are concerned with AOVs and the air systems. We were shown portions of the air system, as well as some of the valves served by it.

TP has several engineers dedicated to the air system and the operability of AOVs. However, diesel generator AOVs and dampers are under the cognizance of system engineers and technicians assigned to care for the particular system.

A number of reports and training aids were provided. These are described in the tabular listing below. The plant air system is described in TP System Description #155, which was provided.

The instrument air (IA) system is intended to provide clean, dry, oil free air at 100 psig for instrumentation and control as well as pneumatic actuators throughout the plant. This is a non-safety-related system. Electrically driven compressors supply each unit and diesel driven compressors are also available for each unit. Present plant configuration is one electrically driven compressor (3CM and/or 4CM) supplying air to both units, with the remaining electrically driven compressor and the diesel driven compressors (3CD and 4CD) on standby. Service air is available as a backup in case of a loss of IA. The IA header is divided into branch lines that supply the steam dump valve accumulators, turbine area, intake structure and water treatment plant, auxiliary building and control room, and the containment and blowdown area. Air supplied to the intake structure/water treatment plant and the auxiliary building/control room is supplied from both units to preclude losing air to those areas if IA from a single unit is lost. Temporary diesel compressors can also be attached to the IA system.

The IA system is equipped with filters for the system and for individual AOVs. Continuous digital readout dew point monitors are installed which are not alarmed; however, the dryers signal a local alarm for high moisture on outlet. (There is no alarm in the control room for high dew point but there is an alarm for loss of IA.) Automatic drain traps are installed. The dryers are powered from emergency power to ensure that dry air will be provided under accident conditions. Air system samples are taken for particulates, dew point, and hydrocarbons/toxins at 20 different locations on the system. The distribution piping for the system is galvanized steel with the instrument supply tubing being stainless steel or copper.

TP had problems with its IA system some years ago related to considerable amounts of water in the instrument air system. These problems prompted action by the management to improve the quality of IA, and are considered by the operators to be very successful, based on performance since the modifications were made.

There are 174 AOVs at TP classified as Category 1 (98 active and 76 passive) and 53 AOVs classified as Category 2 (34 active and 19 passive). Of the 227 Category 1 and 2 AOVs, 191 are classified as safety-related, 34 as quality-related, and 2 as non-nuclear-safety. The total number of AOVs at TP is 836. By way of comparison, there are 111 motor-operated valves in the TP Generic Letter 89-10 program.

Several functional failures and degraded conditions of AOVs were identified by the engineers. These are discussed in Item 7 of the table below. Corrective actions were taken for the degraded conditions.

Condition Report 96-1598 indicated that the plant engineers examined the 10 CFR Part 21 notification from LaSalle regarding mistakes made by the AOV manufacturer in computing pneumatic actuator diaphragm areas for WKM Model 70-13 pneumatic actuators. CR 96-1598 applies to BS&B actuators at Turkey Point. CR 96-1598 does not mention NRC Information Notice 96-68, which also describes this deficiency. The engineers reviewed diaphragm areas and calculations for 28 (including 20 safety-related) AOVs. Their conclusion was that the valves at Turkey Point that might be affected by the Part 21 notification would be capable of performing their intended functions.

The cooperation, courtesy, and knowledgeable responses from the members of the Turkey Point technical staff were noted and appreciated by those of us who are involved in this study of AOVs.

The following tabular summary was prepared to describe the information gathered during the subject visit. The NRC, with assistance from the INEEL, is studying the performance of air-operated valves (AOVs) in commercial nuclear power plants. The information was collected in accordance with the Program Plan dated 10/22/97 (INEEL Letter to H. Ornstein, NRC, from J. Bryce, 10/23/97, Job Code E8238, Task Order 15 - JHB-167-97).

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
1	Date.	March 24 and 25, 1998.
2	Name of Interviewers.	Owen Rothberg, INEEL/LMITCO, 301/816-7773 Mark Holbrook, INEEL/LMITCO, 208/526-4362 Hal Ornstein, NRC/AEOD, 301/415-7574 Jerry Jackson, NRC/RESEARCH, 301/415-6656 Kahtan N. Jabbour, NRC/NRR/PM, 301/415-1496 (sat in on many of the sessions)
3	Plant Name & Docket No.	Turkey Point Nuclear Plant, Units 3 and 4. Docket Nos. 50-250 and 50-251.
4	Person(s) Interviewed, Title(s), Phone Number(s), E-Mail address, short description of organization(s) and duties.	Olga Hanek, TP Licensing Engineer, 305/246-6607 Inman Lanier, CSI Valve Engineer, Juno Tim Miller, TP AOV Engineer, 305/246-6620 Hal McKaig, TP Component Engineering Supervisor, 305/246-6739 Gary Hollinger, TP Licensing Manager Dan Tomaszewski, TP System Engineering Manager, 305/ 246-6158 T. V. Abbatiello, TP Quality Manager Steve Hellriega, TP Works Control Supervisor Tom Carter, TP Maintenance Devin Ryan, TP Air Systems Engineer, 305/246-6612 Tom Jernigan, TP Plant General Manager Jalal Zamanali, FPL Supervising Engineer, PRA, 561/694-3857 E. A. Thompson, TP Engineering Manager Scott Turner, St. Lucie Senior Engineer J. W. York, NRC Resident Inspector Rogerio Reyes, NRC Senior Resident Inspector, 305/245-7669

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5	If necessary, and if person(s) interviewed can do so, obtain any missing information not provided prior to the site visit, as described in the outline for Task 4 above. Note what information was provided.	We arrived with the following materials: - LER 25084031, Containment Isolation Valve for the nitrogen supply system to the Safety Injection Accumulators did not close. Solenoid valve failed. - LER 25085002, Containment Isolation Valve for Steam Generator Blowdown Isolation would not close because dirt had clogged the SOVs. - LER 25085020, MSIVs might not close because of insufficient instrument air combined with low steam flow. - 25085021, eedwater Control Valve and Aux. Feed Flow Control Valve malfunctioned due to moisture in the instrument air system. - LER 25085024, Aux. Feed Pump Discharge Control Valve failed to close fully because of a misadjusted positioner. - LER 25086005, MSIV partially closed and would not close fully because of malfunctioning SOV. - LER 25086028, Potential loss of SI pumps due to potential failure of AOVs from a loss of instrument air. - LER 25086031, Unit Shutdown due to Aux. Feed Flow Control Valve malfunction due to SOV failure. - LER 25086036, Both Emergency Diesel Generators out of service due to governor solenoid out of adjustment. - LER 25087002, Containment Isolation Valve was out of service due to a failed solenoid coil. - LER 25088002, Reactor Coolant System pressure decreased due to failure of Control Valve's controller. LER 25092005, MSIVs declared inoperable due to misalignment of valves during nitrogen bottle replacement. - LER 25094001, SOV failed during surveillance, 3rd of this type. SOVs replaced and configuration changed. - LER 25185014, Containment and control room isolation occurred and pressurizer relief tank drained due to leakage of Pressure Control Valve. - LER 25185021, Turbine runback occurred due to erratic behavior of Main Feedwater Flow Control Valve. - LER 25186018, Auxiliary Feedwater declared out of service due to Aux Feed Control Valve Failure to open. (continued next page)

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5 (cont.)	Information obtained (continued)	- LER 25186025, Reactor Trip due to Feedwater FCV failing closed due to a failed solenoid coil. - LER 25188009, Failure of Differential Pressure Switch on Condensate Polishing Vessel Resulted in SG Feed Pump Trip and Aux. Feed Initiation. - LER 25191006, Autostart of Aux. Feed Pumps Following a low suction pressure trip of MF pump due to malfunction of Condensate Polishing Vessel inlet Valve close limit switch. - LER 25192007, Auto Aux. Feed start on MFP trip due to malfunction of limit switch on AOV. - USNRC FACILITY STATISTICS AND GENERAL INFORMATION, taken from the NRC Bulletin Board on the INTERNET. We were provided with the following materials by the site engineers: - Slides describing the Turkey Point AOV Program, including diagrams of the instrument air system and instrument air dryers. - A list, generated from the licensee's database of safety-related AOVs at Turkey Point showing descriptive information and categories. - TP Student Handout 1710800, Pneumatic Measurement and Control Applications. - TP Student Lab Exercise Guide 1708805, Troubleshoot a Pneumatic Pressure Measurement Channel. - TP Student Lab Exercise Guide 1708808, Calibrate a Pneumatic Level Measurement Channel. - TP Student Lab Exercise Guide 1708809, Troubleshoot a Pneumatic Level Measurement Channel. - TP Student Lab Exercise Guide 1708810, Align a Pneumatic Controller. TP Student Lab Exercise Guide 1708812, Align a Control Valve and Actuator. - TP Student Lab Exercise Guide 1708829, Calibrate a Positioner (Fisher 3582/3583). TP Student Lab Exercise Guide 1708524, Pressure Process Lab Orientation. (continued next page)

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE TURKEY POINT NUCLEAR PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5 (cont.)	Information obtained (continued)	- TP Student Handout 1710600, Process Control Fundamentals. - TP Student Lab Exercise Guide 1708601, Determine the Characteristics of a Temperature Process. - TP Student Lab Exercise Guide 1708602, Determine the Characteristics of a Pressure Process. - TP Student Lab Exercise Guide 1708603, Determine the Characteristics of a Flow Process. - TP Student Lab Exercise Guide 1708604, Determine the Characteristics of a Level Process. - TP Student Lab Exercise Guide 1708606, Determine the Characteristics of Controlling a Level Process With a Two-Position Controller. - TP Student Lab Exercise Guide 1708609, Analyze the Effects of Changes on a Proportional Controller for a Flow Process. - TP Student Lab Exercise Guide 1708613, Analyze the Effects of Changes on a Proportional Plus Integral Controller For a Level Process. - TP Student Lab Exercise Guide 1708615, Analyze the Effects of Changes on a Proportional Plus Integral Plus Derivative Controller for a Pressure Process. - TP Student Lab Exercise Guide 1708617, Bench Check a Controller. - A chart and card entitled "Turkey Point System Importance," (showing instrument air ranked 7 of 17 systems in risk significance and showing IA as about 0.4 of the highest risk significance, by scale). - Plant Event Report for all plants, dated 3/23/98. - FP 0-OSP-200.1, Schedule of Plant Checks and Surveillances. - FP 4-OSP-072.4, MSIV Air Accumulator Check Valve Test. - FP 3-OSP-072.3, MSIV N-sub-2 Backup System Consumption Test. - FP 3-OSP-072.2, MSIV N-sub-2 Backup Periodic Test. - FP 3-OSP-072, MSIV Closure Test. - FP 4-OSP-072, MSIV Closure Test. - FP 3-OSP-075.3, AFW Nitrogen Backup System Low Pressure Alarm Setpoint Verification. - FP 3-OSP-075.6, Auxiliary Feedwater Train 1 Backup Nitrogen Test. (continued next page)

TOPICS TO REVIEW FOR AIR OPERATED VALVE STUDY SITE VISIT TO THE TURKEY POINT NUCLEAR PLANT		
ITEM No.	INFORMATION	RESPONSE OR INFORMATION
5 (cont.)	Information obtained (continued)	- FP 3-OSP-075.7, Auxiliary Feedwater Train 2 Backup Nitrogen Test. - FP System Description No. 155, Plant Air Systems. - FP 0-ADM-518, Condition Reports (Preparation Instruction). - FP 3-OSP-041.4, Overpressure Mitigating System Nitrogen Backup Leak and Functional Test. - FP 0-ADM-059, Root Cause Analysis. - FP 3-OP-041.4, Overpressure Mitigating System. - FP 3-OSP-075.10, AFW Flow Control Valve Test. - FP 0-OSP-200.5, Miscellaneous Tests, Checks and Operating Evolutions. - FP 4-OSP-200.3, Secondary Plant Periodic Tests. - FP 0-PPM-041.4, RCS Pressurizer Main Spray Valves Overhaul. - Memo from Jalal Zamanali, FPL to Tim Miller (ENG-NR-98-023) Risk Ranking of the AOVs for Turkey Point Nuclear Plant, dated March 3, 1998. - TP3 Drawing 5613-M-3030, Component Cooling Water. - TP3 Drawing 5614-M-3041, RCS PORV Control. - TP3 Drawing 5613-M-3075, Aux. Feed System Nitrogen Supplies to AFW Control Valves. - TP3 Drawing 5614-M-3072, Main Steam System MSIV Control. - TP3 Drawing 5613-M-3013, Instrument Air System Air Compressors. - Summary of Condition Reports Involving AOVs, Tracking and History. - Condition Report 96-0304, 3A TPCW HX Inlet Isolation Valve (POV-3-4882) failed to fully close. - Condition Report 96-0535, Pilot Lockup Valves stuck closed. - Condition Report 96-1202, RCS letdown isolation valve failed to close. - Condition Report 96-1598, Investigation of Part 21 notification on vendor mistakes in computing AOV disk areas. - Condition Report 97-0754, AFW FCV failed stroke-timing test. - Condition Report 98-141, Auxiliary Spray to Pressurizer Spray isolation valve SOV failure. - - Two Reports entitled "Instrument Air Compressor Upgrade" dated 6/92 and 12/93. (continued next page)

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
6	Describe plant events involving AOVs and provide reference information, if possible. Recent: Recurring: Significant:	Feedwater bypass failures in 1985 due to moisture in the Instrument Air system; valve failures caused a S/G water level transient; moisture also caused AFW valves to fail during the same time period. These multiple, simultaneous AOV failures affected all of the three trains of AFW on site. This was a significant common-cause failure event. It was found to be an important accident sequence precursor (ASP) event (see Appendix C) and also caused steam generator water level transients.
7	Describe AOV or air system actual or detected potential failures at the plant? Provide reference information, if possible.	Five AOV functional failures have been identified since January 1994, as follows: 1. Pilot Operated Lockup Valves (POLVs) for Emergency Containment Cooler (ECC) Valves. The function of the POLVs is to open the ECC outlet valves when air pressure drops to about 45 psig, Unit 4, or 60 psig, Unit 3. The ECC outlet valves are moved to the open position on loss of IA because the actuators require air pressure to function and the IA system is not safety-related. Failure of the POLVs to shift on loss of IA pressure could result in insufficient CCW flow to support the containment temperature/pressure control design basis safety function. Therefore, the POLVs are considered safety-related. There are 12 POLVs (6 per unit) at Turkey Point. One failure of a POLV to shift was confirmed in 1996. As-found examination of the other POLVs indicated that 3 of the 12 were stuck. Several contributors to failure, included O-ring distortion and grease caking, were identified as causing excessive drag. These are considered to be common-cause, common-mode failures. Increased exercise and spring modifications were implemented as fixes and the failures had not recurred. (Condition Report 96-0535 applies.) 2. Intake cooling water isolation valve failures. POV-3-4882 and POV-4-4883 (safety-related AOVs) failed to fully close in two separate incidents. The problems were considered not reportable under 10 CFR 50.72 or 50.73. Minor wear was found in the housing and cover guides. Corrosion was found on the lower portion of the yoke and housing below the O-ring seal. The corrosion was identified as the root cause of failure. (Condition Reports 96-0304 and 96-0735 apply.) POV-4882 is a risk significant, key component. These are considered to be common-cause failures. The PM frequency and monitoring were increased and the licensee indicated these measures had been effective.

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ITEM No.	INFORMATION	RESPONSE OR INFORMATION
7 (cont.)	(continued) Describe AOV or air system actual or detected potential failures at the plant? Provide reference information, if possible	3. RCS letdown isolation valve failure. CV-3-204 (safety-related AOV) failed to close when remotely operated from the control room. The SOV pilot was determined to be defective and was replaced; however, no definitive root cause was identified. A small amount of Teflon tape was found in the body of the SOV. Condition Report 96-1202 applies. 4. AFW control valve exceeded allowable stroke time. CV-3-2832 (safety-related AOV) was stroked a number of times and the stroke times were found to be erratic and slow. The most likely problem was wear and debris generated from that wear in the positioner. Condition Report 97-0754 applies. 5. Potential failure of auxiliary spray to pressurizer spray line. CV-3-311 (safety-related AOV) was reported to have a solenoid pilot blowing air. The SOV was replaced but no definite root cause was identified. Degraded AOV Conditions: 1. Copes Vulcan AOVs (PORVs) had diaphragm problems due to a high ambient temperature environment. This was termed “diaphragm creep.” A short time after installing a new diaphragm, it leaked and the hold-down bolts were found to be loose. A new silicon rubber diaphragm was installed to correct the problem. This is a common-cause failure mechanism. Condition Report 94-1192 applies. 2. The pressurizer spray valves were moved to a cooler environment in the 1980s in order to improve operation. No reference was found.
8	Describe actions taken after events or failures involving AOVs or the air system. Provide reference information, if possible.	TP added auto drain traps to several low points in the instrument air system. The licensee performs air system samples at 20 different locations for particulates, dewpoint, and hydrocarbons/toxins. The air dryers are powered from emergency power to provide assurance that dry air will be available during accident conditions. FPL added continuous on-line dew point monitoring as well as alarms on the outlets of the dryers. FPL also installed new oil-free rotary screw air compressors which are much more efficient and require less maintenance, and added 2 diesel-driven air compressors (all with auto-start capability) to increase air system reliability.
9	Were there any actual or potential common mode or common cause failures in the air system or AOVs at the plant? Describe and provide reference information, if possible.	Turkey Point has several Copes-Vulcan AOVs; these valves may potentially be affected by the diaphragm creep problem recently identified at Dresden. Refer to Dresden LER 23798003 dated 2/27/98, Dresden 50.72 Report 33620 dated 1/28/98, and Dresden MR No. H-98-0045 dated 3/6/98.

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10	Describe root cause analysis procedures for the plant. Provide reference information, if possible.	Refer to FP&L 0-ADM-059, Root Cause Analysis. This Procedure describes a formal process for determining root causes of failures.
11	Describe root-cause analyses performed for air system or AOV failures at the plant. Provide reference information, if possible.	A formal root cause analysis program is in place and is described in FP&L 0-ADM-059. An examination of the LERs for the plant indicates that root causes for events are investigated and reported.
12	Describe maintenance procedures for the air system. Provide reference information, if possible.	Periodic maintenance is performed on the air compressors (quarterly and semi-annually) and controls (annually, auto-start pressure switches semi-annually); air dryer (desiccant) heaters and controls (annually); after-filter (annually), and point of use filters are routinely changed out. Dryer performance testing is done quarterly. Air quality sampling is performed every 18 months. Water traps are blown down and the dew point is checked each shift.
13	Describe maintenance procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	TP did maintenance in the past based on experience; are moving toward a more standardized set of activities; depends on environment; will do routine actuator overhauls in some cases.
14	Describe IST procedures for the air system. Provide reference information, if possible.	No formal IST program is used for the IA system, although a number of system and equipment checks are performed on a routine basis. The IA system is considered to be a non-safety-related system.
15	Describe IST procedures for AOVs. Provide reference information, if possible. Safety-related: Important non-safety-related: Non-safety-related:	Stroke time testing is performed in accordance with ASME Section XI code requirements.
16	Describe diagnostic systems, if any, used for AOVs. Provide reference information, if possible. Description of system:	Turkey Point uses the Fisher FlowScan system. This system has been used in the past as a troubleshooting tool. The licensee also recently purchased the Crane·MOVATS Universal Diagnostic System (UDS) to provide direct force measurements. This system uses Teledyne strain gauges.

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	Specifications: Data collected and frequency of collection: Vendor assistance provided, if any:	Crane MOVATS recently provided UDS training to Turkey Point personnel. Turkey Point engineers plan to use the UDS as a monitoring tool for AOVs.
17	Describe design (and analysis) procedures for AOVs. Describe how design basis is established and maintained for AOVs. Provide reference information, if possible.	As of the time of the site visit, TP is using the original air settings that were provided when the plant was constructed. The licensee believes that this approach is adequate based on experience (i.e., few design-related problems have been observed). FPL engineers will develop design-basis requirements for a sample of their Category 1 AOVs to see if there are any problems. FPL will also look at specific valves if a design-related failure occurs. However, the licensee does not intend (at this time) to do a design-basis review for all program AOVs. For example, none of the Category 2 valves will be reviewed unless there are problems identified as part of the Category 1 AOV review. FPL engineers indicated that they also plan to review any future JOG recommendations to develop design-basis requirements as part of any industry-wide AOV program. This approach assumes that the fluid conditions experienced by an AOV during normal operations are in fact the worst-case conditions that a given valve will ever have to operate under. This assumption cannot be verified unless a design-basis review is performed. This approach also assumes that the valve vendors' sizing methodologies are conservative. Industry MOV experience has shown that, in many cases, the valve vendors have a poor understanding of how the valves will perform under design-basis conditions.
18	Describe analyses and/or testing for verification of operability during postulated transient or accident conditions. Provide reference information, if possible.	No specific test program is in place for verifying design-basis operability of AOVs, other than diagnostic testing under static conditions and comparisons to existing vendor requirements. See Item 17.
19	Describe training for installation, maintenance, and testing of AOVs. Provide reference information, if possible.	Training for use of the Fisher FlowScan diagnostic system is addressed by on-the-job training in the plant. Three days of training are set aside to cover use of the new MOVATS UDS system. New I&C personnel receive eight weeks of Lab training on in-plant equipment. AOVs are covered during part of this lab period. Training includes disassembly, reassembly, and actuator setup of the more common AOV actuators found in the plant. Industry events and emerging technical issues are addressed during Annual Continuing Training sessions, which last two weeks.

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20	Describe databases used to track maintenance, failures, and events regarding AOVs. Provide reference information if possible. On site: Company wide: Industry:	TP provided several runs from their database of AOVs and AOV events for the site. They track data from their own plant and from the industry.
21	Describe the impact of the Maintenance Rule, 10 CFR 50.65 on AOV and air system maintenance and testing. Provide reference information if possible.	The licensee has reviewed the plant systems to determine those functions that are risk significant. Using these functions, the licensee reviewed the AOV population to identify those valves that are needed to support the risk-significant functions. These valves were identified as Category 1 AOVs. If a valve was safety related, but not identified as Category 1, it was classified as Category 2. Other AOVs that were not Category 1, but were considered important to plant operation were also classified as Category 2.
22	Is PRA data used for predictive maintenance or replacement of AOVs? If so, how?	Predictive maintenance of AOVs was not discussed.
23	Are AOVs serviced on site, serviced offsite, or replaced as piece-parts if found to require service?	AOVs are typically serviced on site by licensee maintenance personnel. Some outside personnel are added during outages to complete the increased work scope. These additional personnel receive training and are supervised by licensee maintenance personnel.
24	Identify and describe the most common recurring maintenance problem(s) and failures regarding AOVs and the air system. What did you see? Provide reference information if possible.	TP noted some environmental effects on the PORVs (i.e., "diaphragm creep") which was caused by high environmental temperature conditions.
25	Interviewer comments regarding actual valves viewed during the visit, in the plant, undergoing maintenance or replacement, or in the plant stock system, if applicable to this interview.	We noticed several "green tags" on valves and pressure switches denoting maintenance outstanding issues. Some of these tags dated back to 1997. Time did not allow further investigation to determine how effective this tag system is for scheduling needed work activities.

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26	Has the plant made changes to valves or systems that include AOVs, or replaced AOVs with different models of AOVs or different valves that are not AOVs? If so, describe the changes and the circumstances. What prompted the change? Was the change made for this plant only?	The Feedwater regulating valves were modified to put a Fisher actuator on a Copes-Vulcan valve when the original actuator proved unreliable.
27	Does the plant follow EPRI/NMAC guidelines for maintaining AOVs and the air system(s)? If not, describe differences and reasons for the differences. Provide reference information, if possible.	The plant personnel are aware of the industry efforts toward AOVs. They have consulted the industry guidelines as reference material.
28	What is the plant doing or planning to do in response to the recent Industry correspondence on AOVs. Provide reference information, if possible.	At the time of the visit, we were told that TP had no plans to change their existing approach to evaluation or maintenance of AOVs. Since that time, we have been told that TP is putting together an AOV program.
29	Do you have any suggestions for improving the performance of AOVs, particularly in the areas of surveillance, testing, or maintenance?	TP relies heavily on personnel training to ensure proper maintenance and adjustment of AOVs, and this was emphasized. As noted in item 5 in this table, TP has specific procedures in place to control maintenance on safety-related and important non-safety-related AOVs.
30	Provide a list of 10 CFR 50.59 and 10 CFR 50.72 reports on AOVs and AOV support systems (air or inert gas supply, etc.) that have been issued for this plant.	A list was requested and was provided near the end of the plant visit. See item 7 in this table.

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31	What thrust or torque margins are expected for AOVs? Are different margins used for safety-related, important non-safety-related, or non-safety-related AOVs?	Turkey Point has not established any formal margin guidance for AOVs. Licensee personnel indicated that an informal target of 20% margin is desired. However, this margin is not known to exist for all AOVs in the program because calculations have (as of the time of the visit) not been completed.
32	What maintenance or surveillance is done to AOV accumulators to ensure air/nitrogen quality and pressure? Were seismic considerations and size verified?	Turkey Point has several AOVs that have nitrogen or air accumulators (e.g., MSIVs, AFW regulating valves, steam dump valves, and PORVs). These accumulators are tested via consumption tests and intersystem (i.e., nitrogen/IA) check valve isolation tests.
33	Describe problems with pressure regulators, if any.	In the discussions, it was noted that settings for pressure regulators and controllers need to be formally tracked and monitored in order to be assured that settings are proper. An extensive set of training materials was provided related to troubleshooting and adjustment of AOVs and their control hardware. Settings are tracked and controlled by plant internal procedures.
34	Describe problems with Feedwater regulating valves, if any.	A large amount of water in the instrument air system in 1985 caused several Feedwater and Aux. Feedwater problems as identified in the LERs. These events led to improvements in the supply and quality of IA at TP.
35	What, if any, is your involvement with the AOV Users Group? Describe.	FPL personnel routinely participate in the AOV Users' Group meetings. They were also to have participated in the summer of 1998 ASME/NRC pump and valve symposium.

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10. SUPPLEMENTARY NOTES

H. Ornstein, NRC Project Manager

11. ABSTRACT *(200 words or less)*

A study of air-operated valves in nuclear power plant applications was conducted for the NRC Office of Research (the project was initiated by NRC/AEOD). The results of the study were based on visits to seven nuclear power plant sites, literature studies, and examination of event records in databases available to the NRC. The purpose is to provide information to the NRC staff concerning capabilities and performance of air-operated valves (AOVs). Descriptions of air systems and AOVs were studied along with the support systems and equipment. System and equipment that contain AOVs and solenoid-operated valves (SOVs) were studied to determine their dependencies. Applications of AOVs and SOVs were listed along with current NRC requirements.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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