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Regulatory Effectiveness of Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements"

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Regulatory Effectiveness of Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements"

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ABSTRACT

As part of the U.S. Nuclear Regulatory Commission's program to assess regulatory effectiveness, the Office of Nuclear Regulatory Research has examined regulations such as the station blackout rule, and anticipated transient without scram rule. As part of this program, the Office of Nuclear Regulatory Research is also reviewing the effectiveness of generic safety issue resolution. One such issue currently being reviewed is Unresolved Safety Issue (USI) A-45 to determine if the requirements are achieving the desired outcomes. It is anticipated that the results of these reviews can be used to improve the effectiveness of NRC requirements and guidance, staff inspection guidance, and oversight decision processes for NRC licensee performance. This report evaluates the effectiveness of the USI A-45 resolution by comparing USI A-45 expectations to outcomes. A set of baseline expectations was established from NUREG-1289, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," November 1988, and NUREG/CR-5230, "Shutdown Decay Heat Removal Analysis: Plant Case Studies and Special Issues," April 1989, and the actual outcomes were obtained from the individual plant examinations and individual plant examination of external events in the areas of total core damage frequency, decay heat removal risk categories and decay heat removal vulnerability. The report concludes that the USI A-45 program expectation regarding decay heat removal-related contribution to core damage frequency was generally met without the imposition of generic hardware fixes expressly for USI A-45. In addition, it is likely that licensees' awareness of the significance of decay heat removal strategies was fostered through the conduct of the individual plant examinations and individual plant examination of external events. Therefore, the USI A-45 resolution approach appears effective in achieving its goals.

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As part of the U.S. Nuclear Regulatory Commission's program to assess regulatory effectiveness, the Office of Nuclear Regulatory Research has examined regulations such as the station blackout rule (SBO) and anticipated transient without scram (ATWS) rule. As part of this program, the Office of Nuclear Regulatory Research is also reviewing the effectiveness of generic safety issue resolution. The resolution of Unresolved Safety Issue (USI) A-45 was reviewed to determine if the resolution achieved the desired outcomes.

In March 1981, the Commission designated "Shutdown Decay Heat Removal Requirements" as USI A-45. At the May 12, 1988, "Briefing to the Commission on the Status of Unresolved Safety/Generic Issues," the staff indicated its intention to incorporate the resolution of USI A-45 in the individual plant examination (IPE) program.

On November 23, 1988, NRC issued Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities – 10 CFR 50.54(f)." This included resolving USI A-45 as part of the overall assessment of plant vulnerabilities to severe accidents and identification of means to reduce or ameliorate their consequences. The letter asked licensees to specifically consider decay heat removal (DHR) vulnerabilities as part of their IPEs. Generic Letter 88-20 Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f)," June 28, 1991, also requested licensees to propose resolving USI A-45 if there were no unresolved vulnerabilities associated with this issue at their plants.

The present evaluation of the resolution of USI A-45 includes performing a review of licensee actions to resolve safety issues regarding USI A-45, and report the findings. Relevant information was reviewed from the generic safety issue program and other generic activities to develop the information needed to document expectations regarding resolution of USI A-45. To assess the regulatory effectiveness of USI A-45, expectations were derived from SECY 88-260, "Shutdown Decay Heat Removal Requirements," September 13, 1988; GL 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities – 10 CFR 50.54(f)," November 1988; and related USI A-45 studies such as NUREG/CR-5230, "Shutdown Decay Heat Removal Analysis: Plant Case Studies and Special Issues," April 1989; and NUREG-1289, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," November 1988. The outcomes were obtained from plant-specific IPE reports, the IPE database, plant-specific modification list, staff evaluation reports, technical evaluation reports, and responses to requests for additional information. Comparison of expectations to the outcomes showed whether the expectations were achieved.

Much was being learned about reactor safety issues during the late 1970's and early 1980's, both from analysis of operating experience and from NRC-sponsored work in risk analysis. The resulting improvements in understanding of the issues and in plants' mitigating systems led to significant redefinition and reformulation of safety issues, as NRC programs adapted in order to respond more efficiently to emergent needs based on then-current knowledge. Many of these developments affected decay heat removal. As a result of these developments, between the original formulation of A-45 and the resolution of A-45 in its final form, significant changes occurred that caused the resolution of A-45 to be very different from the resolutions of station blackout and ATWS.

The USI A-45 resolution differed from the SBO and ATWS resolutions in several ways. SBO and ATWS were addressed through definitive requirements, in that specific capability needs were stated and required by regulation. The resolution of USI A-45 was through generic communications that did not have or establish definitive requirements. The generic communications requested that an IPE and IPEEE be performed to identify and address any DHR "vulnerabilities." A definition of DHR "vulnerability" was not provided. Performance criteria analogous to those for ATWS and SBO were not imposed.

The USI A-45 program expectation regarding reduction of the DHR-related contribution to CDF was generally met without the imposition of generic hardware fixes expressly for USI A-45. Significant reduction in DHR-related risk was achieved as a result of plant changes from the implementation of several regulatory initiatives including USI A-44: Station Blackout, USI A-46: Seismic Qualification of Equipment in Operating Plants, GI 124: Auxiliary Feedwater System Reliability, and GL 89-16: Installation of a Hardened Wetwell Vent. As a result, fewer DHR-related vulnerabilities remained to be discovered through the plant-specific IPEs and IPEEEs. The IPEs were effective at verifying the benefits achieved from the various regulatory initiatives and at identifying unique plant-specific vulnerabilities for internal events. The IPEEEs had a similar effect for external events. DHR challenges that initiate during shutdown modes of operation were not explicitly addressed by either the IPEs or IPEEEs. Therefore, the USI A-45 resolution approach appears effective in achieving its goals.

FOREWORD

This report provides the technical basis for evaluating the effectiveness of the resolution of Unresolved Safety Issue (USI) A-45, Shutdown Decay Heat Removal [DHR] Requirements. The evaluation focused on the approach used to resolve USI A-45, and whether the desired outcome had been achieved in an effective and efficient manner. Expectations for resolution of USI A-45 were based on information reported in NUREG-1289, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Down Decay Heat Removal Requirements," and NUREG/CR-5230, "Shutdown Decay Heat Removal Analysis: Plant Case Studies and Special Issues."

The study utilized information from licensees' Individual Plant Examinations (IPEs) and Individual Plant Examinations of External Events (IPEEEs) to assess the outcome. The report concluded that licensees' IPE and IPEEE programs used to resolve USI A-45, generally met expectations without the imposition of generic hardware fixes to improve DHR reliability. The report notes that DHR enhancements also resulted from the synergistic effect of other regulatory initiatives including resolution of USI A-44 Station Blackout, USI A-46 Seismic Equipment Qualification, GI-124 Auxiliary Feedwater Reliability, and installation of boiling water reactor (BWR) hardened vents. These enhancements taken together with those that stemmed from licensees' IPE/IPEEE programs, indicated that the approach used to resolve USI A-45 had been reasonable and effective.

Certain limitations need to be considered in the overall assessment of the resolution of USI A-45. In many cases, licensees took credit for alternate DHR strategies not required by NRC regulations. The more significant strategies include, for example, strategies to initiate and maintain feed and bleed operation for pressurized-water reactors, use of the firewater system or other sources of water and equipment to remove decay heat, use of system cross-ties between units or trains, and use of a hardened wetwell containment vent for BWRs. Credit for these strategies had in many cases substantially reduced core damage frequency through the introduction of diversity in the DHR function. Strategies that utilize feed and bleed, or steam generator depressurization using condensate for heat removal, for example, had the greatest impact on core damage frequency. These strategies rely on non-safety systems to perform their function. The effectiveness of the resolution of USI A-45, therefore, depends on both licensees' commitment to implement and maintain equipment to enhance DHR, and confirmation of the adequacy of the steps taken to ensure a reliable DHR function. This report could be used to support activities to ensure DHR reliability, (e.g., that the analyzed as-built, as-operated plant including any DHR safety enhancements stemming from the IPE/IPEEE are implemented and maintained).

As described below, this report is consistent with the NRC strategic performance goals that promote maintaining safety; increasing public confidence; and making NRC activities more effective, efficient, and realistic.

Maintaining safety – The study confirms that safety has been maintained or otherwise enhanced with licensees' implementation of the resolution of USI A-45 and associated plant modifications. When considering other factors, such as the continued reduction in transient initiators, the actual risk reduction from the resolution of USI A-45 exceeds that reported in the

IPEs and IPEEEs. If the study had found that safety had not been maintained, appropriate recommendations would have been made.

Public confidence – Public confidence should be improved by confirming that implementation of the resolution of USI A-45 resulted in enhanced safety.

Making NRC activities more effective, efficient, and realistic – The report compares the regulatory expectations to outcomes to assess the effectiveness of the resolution of USI A-45 in achieving its goals, and in identifying areas that may need attention. In this case, the approach taken appeared reasonable given the broad scope of the DHR topic, and the variety of plant designs and many unique, site-specific features which would have made it difficult to identify generic solutions. The quality of probabilistic risk assessments (PRAs) continues to improve and guidance for the use of PRA information for regulatory decision-making is now more fully developed than when Generic Letter 88-20 was issued. Appropriate recommendations would have been made if warranted in this area.

F Eltawila

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ADS	automatic depressurization system
AFAS	auxiliary feedwater actuation system
AFW	auxiliary feedwater system
ANO-1	Arkansas Nuclear One, Unit 1
ANO-2	Arkansas Nuclear One, Unit 2
ATWS	anticipated transient without scram
B&OTF	Bulletins and Orders Task Force
B&W	Babcock and Wilcox
BWRs	boiling-water reactors
CCWS	component cooling water system
CDF	core damage frequency
CE	Combustion Engineering
CRD	control rod drive
CS	containment spray; core spray
CST	condensate storage tank
CTS	condensate transfer system
DHR	decay heat removal
ECCS	emergency core cooling system
EFW	emergency feedwater
ESFAS	engineered safety features actuation system
EOPs	emergency operating procedures
ESW	emergency service water
FPS	fire protection system
FWIV	feedwater isolation valve
FWCV	feedwater control valve
GL	Generic Letter
GSW	general service water
HGTR	high gas temperature reactor
HPCI	high pressure coolant injection
HPSI	high pressure safety injection
HPCS	high pressure core spray
HPI	high pressure injection

ABBREVIATIONS (Cont.)

IA	instrument air
IC	isolation condenser
IE	NRC Office of Inspection and Enforcement
INPO	Institute of Nuclear Power Operations
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination on External Events
ISLOCA	interfacing systems loss-of-coolant accident
LERF	large early release frequency
LOCA	loss-of-coolant accident
LOOP	loss-of-offsite power
LPCI	low pressure coolant injection
LPCS	low pressure core spray
LPF	low pressure feed
LPI	low pressure injection
LTOP	low temperature overpressure protection
LWRs	light-water reactors
MFW	main feedwater
MSIVs	main steam isolation valve
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
PCS	power conversion system
PDWST	plant dimeneralized water storage tank
PEI	plant emergency instructions
PORV	power operated relief valve
PRA	probabilistic risk assessment
PSW	plant service water
PTS	pressurized thermal shock
PWR	pressurized-water reactor
QDOs	quantitative design objectives
RAI	request for additional information
RBCCW	reactor building component cooling water
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RES	NRC Office of Nuclear Regulatory Research
RHR	residual heat removal
RHRSW	residual heat removal service water
RPS	reactor protection system
RPV	reactor pressure vessel
RY	reactor year

ABBREVIATIONS (Cont.)

SAFW	standby auxiliary feedwater system
SAMA	Severe Accident Mitigation Alternative
SAR	Safety Analysis Report
SBLOCA	small break LOCA
SBO	station blackout
SDC	shutdown cooling system
SDP	Significance Determination Process
SER	Staff evaluation report
SGFW	steam generator feedwater
SGTR	steam generator tube rupture
SOERs	Significant Operating Event Reports
SPAR	Standardized Plant Analysis Risk
SPC	suppression pool cooling
SRVs	safety relief valves
SSCs	systems, structures, and components
SSW	standby service water
SW	service water
SX	shutdown service water
TER	technical evaluation report
TMI-1	Three Mile Island Nuclear Power Plant, Unit 1
TMI-2	Three Mile Island Nuclear Power Plant, Unit 2
TSs	technical specifications
USI	Unresolved Safety Issue
WNP-2	Washington Nuclear Power, Unit 2

1 INTRODUCTION

As part of the U.S. Nuclear Regulatory Commission's program to assess regulatory effectiveness, the Office of Nuclear Regulatory Research (RES) has examined regulations such as the station blackout (SBO) rule and anticipated transients without scram (ATWS) rule. As part of this program, RES is also reviewing the effectiveness of generic safety issue resolution. One such issue currently being reviewed is Unresolved Safety Issue (USI) A-45 to determine if the requirements are achieving the desired outcomes. It is anticipated that the results of these reviews can be used to improve the effectiveness of NRC requirements and guidance, staff inspection guidance, and oversight decision processes for NRC licensee performance.

The work described in this report is an assessment of the effectiveness of the resolution of USI A-45 on decay heat removal (DHR) at U.S. commercial nuclear power plants. Failure of DHR systems can be a significant contributor to core damage frequency (CDF) and the frequency of large releases of radioactive material. Under normal operating conditions of a commercial light-water reactor (LWR), power generated within a reactor is removed as steam to produce electricity via a turbine generator. Following a reactor shutdown, the ongoing radioactive decay of fission products and irradiated core materials produces significant amounts of heat (so-called "decay heat"). The removal of this decay heat from a sub-critical reactor is a vital function in both post accident and routine reactor shutdown scenarios, and under both hot shutdown and cold shutdown conditions. Therefore, all LWRs must have a mechanism of removing decay heat from the reactor coolant system (RCS) and maintaining sufficient water inventory inside the reactor vessel to ensure adequate cooling of the reactor fuel. The risks associated with failure of these functions depend on two factors: the frequency of initiating events that require or jeopardize DHR operations and the probability that required systems will fail to remove the decay heat given an initiating event.

In March 1981, NRC designated "Shutdown Decay Heat Removal Requirements" as USI A-45 in NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants: Special Report to Congress" [Ref. 1]. At the May 12, 1988 Briefing to the Commission on the Status of Unresolved Safety/Generic Issues, the staff indicated its intention to incorporate the resolution of USI A-45 in the Individual Plant Examination (IPE) program. The requirements and staff's justification for implementing this action were first published in SECY 88-260, "Shutdown Decay Heat Removal Requirements," on September 13, 1988 [Ref. 2]. This paper concluded that: (1) risks due to loss of DHR could be "unduly" high for some plants, (2) DHR vulnerabilities and respective corrective actions are strongly plant specific, and (3) detailed plant-specific analyses under the IPE program, including consideration of externally-initiated events in the future, are needed to resolve this issue.

On November 23, 1988, the staff issued Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities" [Ref. 3], in part, to resolve USI A-45. The purpose of this letter was to ensure that the probabilities of severe core damage and large radioactive releases due to failure of DHR systems at U.S. nuclear power plants were consistent with the Commission's Safety Goal Policy Statement. The letter asked the licensees to conduct an IPE and specifically consider DHR vulnerabilities as part of their IPEs. However, the letter did not provide any definition of DHR vulnerability. Many licensees chose to use the Quantitative Design Objectives (QDO) cited in NUREG-1289, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," dated

November 1988 [Ref. 4]. The fundamental QDO chosen for USI A-45, was the CDF due to the failure of the DHR function. A value of 1E-05 per reactor year (RY) was chosen for this QDO. Few, if any, cost-beneficial modifications would be warranted if the CDF were less than this value.

Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f)" [Ref. 5], issued on June 28, 1991, included a request to resolve the external event portion of USI A-45. It requested that if a licensee discovered a potential vulnerability that was topically associated with USI A45 and proposed measures to dispose of the specific safety issue or concluded that no vulnerability existed, then the staff would consider the issue resolved upon review and acceptance of the results from the IPEEE. The generic letter specifically requested the licensee to identify which USIs it was proposing to resolve.

The scope of the present review is therefore to assess the effectiveness of the specific requirements of GL 88-20 and GL 88-20, Supplement 4 identified above in reducing DHR-related CDF, to a value preferably below 1E-05/RY for each plant.

2 BACKGROUND

Much was being learned about reactor safety issues during the late 1970's and early 1980's, both from analysis of operating experience and from NRC-sponsored work in risk analysis. The resulting improvements in understanding of the issues and in plants' mitigating systems led to significant redefinition and reformulation of safety issues, as NRC programs adapted in order to respond more efficiently to emergent needs based on then-current knowledge. Many of these developments affected decay heat removal. As a result of these developments, between the original formulation of A-45 and the resolution of A-45 in its final form, significant changes occurred that caused the resolution of A-45 to be very different from the resolutions of station blackout and ATWS. In order to assess the effectiveness of the resolution of A-45, it is necessary to clarify this history.

2.1 Scope of the USI A-45 Problem

The A-45 Task Action Plan [Ref. 6] uses the following definitions to describe decay heat removal phases following a reactor shutdown.

(a)	Reflood phase	The initial phase of a severe LOCA, when the objective is to reflood the reactor.
(b)	Shutdown decay heat removal (SDHR) phase	The transition from reactor trip to "hot shutdown," excluding the initial reflooding phase in a severe LOCA.
(c)	Residual Heat Removal (RHR) phase	The transition from "hot shutdown" to "cold shutdown" and maintaining cold shutdown conditions.
(d)	Decay Heat Removal (DHR) phase	SDHR and RHR phases combined.

In accordance with the Task Action Plan, the major focus is equipment and system reliability associated with the SDHR phase. This includes those components and systems required to maintain primary and/or secondary coolant inventory control and to transfer heat from the reactor coolant system to an ultimate heat sink following shutdown of the reactor for normal events, off-normal transient events (e.g., loss of offsite power, loss of main feedwater) and small-break LOCAs. It does not address those emergency core cooling systems required only to maintain coolant inventory and dissipate heat during the first 10 minutes following medium or large LOCAs. It does include the supporting systems that would be required for successful decay heat removal in various modes.

The Task Action Plan also addresses RHR phases (c) and (d). It states that the main problems are "(i) to ensure adequate reliability in the electrical and mechanical equipment of the RHR systems during prolonged exposure to a hostile environment, such as would be encountered after a LOCA, whether small or large, and (ii) to ensure adequate reliability of the RHR systems after being subjected to severely disturbed conditions, such as earthquakes, floods or fires."

2.2 Evolution of USI A-45

The starting point for NRC's study of DHR is WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants" [Ref. 7]. The study was sponsored by the U.S. Atomic Energy Commission to estimate the public risks that could be involved in potential accidents in commercial nuclear power plants and was completed in October 1975. The study gave special emphasis to the safety and adequacy of DHR systems at nuclear power plants. The computed CDF for pressurized-water reactors (PWRs) was 6E-05/RY. Small break loss of coolant accidents (SBLOCAs) sequences and transient events initiated by the loss-of-offsite power (LOOP) followed by the failure of DHR systems were found to be substantial contributors to the overall core melt probability. On the other hand, the calculated total CDF for boiling-water reactors (BWRs) was 3E-05/RY. Failure to rapidly shut down the reactor and failure of the DHR systems after shutdowns were major contributors to core melt frequency. Therefore, the study concluded that the CDF for the PWR and the BWR combined would be an average value of 5E-05/RY [Ref. 6].

On March 28, 1979, the Three Mile Island Nuclear Power Plant, Unit 2 (TMI-2) experienced severe core damage, which resulted from a series of events initiated by a loss of feedwater transient. Following the accident at TMI-2, the NRC issued NRC Office of Inspection and Enforcement (IE) Bulletin No. 79-05, "Nuclear Incident at Three Mile Island," on April 1, 1979. This bulletin was directed to the Babcock and Wilcox (B&W) PWR facilities. One purpose of the bulletin was to direct the attention of the licensees toward understanding the sequence of the TMI-2 events to ensure against such an accident at their facility. A supplement to this bulletin, IE Bulletin 79-05A, "Nuclear Incident at Three Mile Island – Supplement," was issued on April 5, 1979 which required the licensees to take certain actions [Ref. 8].

On April 21, 1979, the NRC issued another supplement to the bulletin, IE Bulletin 79-05B, "Nuclear Incident at Three Mile Island – Supplement" [Ref. 9]. One of the actions in this bulletin required the B&W operating facilities to develop procedures and train operations personnel on methods of establishing and maintaining natural circulation. These procedures and corresponding training would include means of monitoring heat removal efficiency by available plant instrumentation and a method of assuring that the primary coolant system would be subcooled by at least 50°F before natural circulation was initiated.

Earlier reliability studies (WASH-1400) and related experience from the accident at TMI-2 confirmed that the loss of the capability to remove heat through a steam generator was a significant contributor to the probability of a core-melt event. Following the TMI-2 accident, the NRC Office of Nuclear Reactor Regulation (NRR) established the Bulletins and Orders Task Force (B&OTF) in early May 1979; its work was completed on December 31, 1979. This task force was accountable for assessing and governing TMI-2 related staff activities associated with the IE bulletins, Commission Orders, and generic evaluations of loss-of-feedwater transients and SBLOCAs for all operating plants to assure continued safe operation. NUREG-0645, "Report of the Bulletins and Orders Task Force," summarizes the results of the work performed [Ref. 10]. NUREG-0578, "TMI-2 Lessons Learned Task Force," published in July 1979, offered short-term recommendations in the areas of design and analysis to licensees of all nuclear power plants. Some of the actions required the licensees to provide the reactor operator with the necessary instrumentation, procedures and training to readily recognize and implement actions avoiding conditions of "inadequate core cooling" [Ref. 11 Section 2.1.3.b and 2.1.9]. Others directed the licensees to implement modifications for the automatic initiation of the auxiliary feedwater system (AFW) [Ref. 11 Section 2.1.7.a & b].

Because loss of decay heat removal seemed to be a potentially significant contributor to the total risk associated with nuclear power, a program was initiated to evaluate the safety adequacy of the DHR function in operating LWRs and assess the value and impact of alternative means of DHR in both boiling and pressurized water reactors. In March 1981, the Commission identified "Shutdown Decay Heat Removal Requirements" as USI A-45 in NUREG-0705.

2.3 Loss of Decay Heat Removal During Shutdown Modes of Operation

Between 1977 and 1990, there were numerous losses of the DHR system during mid-loop operation. Loss of DHR during nonpower operation and the consequences of such a loss became an issue of increasing concern. Numerous industry and NRC publications addressed the subject.

On June 11, 1980, the NRC issued GL 80-053 [Ref. 12]. The basis for this request was a number of events that occurred at operating PWR facilities where DHR capability had been seriously degraded due to lack of administrative controls when the plants were in shutdown modes of operation. A significant total loss of DHR capability occurred at Davis-Besse Unit 1 on April 19, 1980, which was described in IE Information Notice No. 80-20, "Loss of Decay Heat Removal Capability at Davis-Besse Unit 1 While in a Refueling Mode," dated May 8, 1980. Factors contributing to the event involved (1) extensive maintenance activities, which led to a loss of redundancy in the DHR capability and (2) lack of procedures and/or administrative controls. To resolve this problem, GL 80-053, "Decay Heat Removal Capability," June 11, 1980, required the licensees to amend the technical specifications (TSs) for their facilities that provide for redundancy in DHR capability for their plant(s) in all modes of operation.

Nonetheless, loss of DHR during nonpower operation continued to be of increasing concern. The report of the Diablo Canyon event of April 10, 1987 (NUREG-1269, "Loss of Residual Heat

Removal System") stated that operation at reduced RCS inventory was a particularly sensitive condition and identified many generic weaknesses in DHR. DHR problems continued to occur as illustrated by: (1) the inadequacies demonstrated by many licensees in their response to GL 87-12, (2) the event at Waterford on May 12, 1988, (3) the event at Sequoyah on May 23, 1988, and (4) the DHR perturbations due to an "inadequate" level at San Onofre on July 7, 1988. This resulted in the NRC issuance of GL 88-17, "Loss of Decay Heat Removal – 10 CFR 50.54(f)," on October 17, 1988, drawing the attention of the operators to the risks linked to shutdown states and requesting them to take corrective measures (improvement of water level measurement instrumentation and exclusion of certain configurations with the RCS open) [Ref. 13].

Based on the A-45 Task Action Plan, the focus of A-45 is on challenges to decay heat removal that are initiated from an at-power transient. This includes the evaluation of reliability of decay heat removal equipment and systems during the RHR phase. However, the emphasis is on the reliability of RHR systems after being subject to the initial transient and not on those conditions associated with refueling operations. Although improvements evaluated by A-45 will likely benefit the resolution of issues associated with mid-loop operation and other non-power operation, these issues were not directly targeted by A-45.

2.4 NRC Sponsored Studies and Generic Letter 88-20

Following the identification of USI A-45, several reports were issued to the NRC addressing the vulnerabilities of DHR systems in commercial boiling and pressurized water reactors. NUREG/CR-2799, "Evaluation of Events Involving Decay Heat Removal Systems in Nuclear Power Plants," prepared by Oak Ridge National Laboratory, was issued in July 1982. This report covered 327 events involving DHR in U.S. Nuclear Power Plants from June 1979 through June 1981 [Ref. 14]. However, only 38 events in these 2 years met the criteria for safety significance. The most frequent event involving a significant problem with DHR was the cavitation of residual heat removal (RHR) pumps. Davis-Besse 1 had several instances in which an inadvertent signal to the safety features actuation system caused the operating DHR pumps (terminology used for RHR pumps at Davis-Besse) to align to the dry sump causing pump cavitation. Steam bubble formation in the reactor vessel head during natural circulation cooldown at St. Lucie 1 was another major event.

Another report published by the Nuclear Safety Analysis Center, NSAC-52, "Residual Heat Removal Experience Review and Analysis" [Ref. 15], documents some of the reports issued by the NRC on DHR. One such report, NUREG-0606, "Unresolved Safety Issues Summary – The Aqua Book," was issued in August 1982. In this report, the NRC supported the position that upgraded and/or alternative means of DHR could substantially increase a plants' capability to deal with a broader spectrum of transients and accidents and, therefore, could significantly reduce the overall risk to the public. The NRC reaffirmed that Task A-45 would investigate an alternative means of DHR in both BWR and PWR plants, including but not limited to using existing equipment where possible. Emphasis was also made on the further development of Regulatory Guide 1.139, "Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown," Revision 1 [Ref. 16].

On August 8, 1985, the NRC issued its "Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants" (Federal Register, 50FR32138). The policy statement explicitly stated the Commission's strategy to address severe accident issues for all U.S. commercial nuclear power plants. The Commission mandated a methodical safety examination of plants to study specific accident vulnerabilities and desirable, cost effective changes to ensure that the plants do not pose any excessive risk to public health and safety.

To take corrective actions, the NRC initiated a program to evaluate the adequacy of current designs to ensure that LWRs did not pose unacceptable risk as a result of DHR system failures. The results of this program were documented in two reports: NUREG/CR-5230 [Ref. 17], "Shutdown Decay Heat Removal Analysis: Plant Case Studies and Special Issues," issued in April 1989 and NUREG-1289, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements" [Ref. 4], issued in November 1988. This latter report formed the basis for the regulatory analysis for USI A-45 and is explained in more detail in Section 2.5.

During the 341st meeting of the Advisory Committee on Reactor Safeguards (ACRS) in September 1988, the NRC staff discussed the proposed resolution of USI A-45. The staff concluded that because the IPEs provide an examination of the DHR systems and their importance as well as the importance of systems performing other functions, USI A-45 should be subsumed in the IPE program under the Severe Accident Policy. ACRS supported the staff's decision to incorporate A-45 in the IPE program and the decision that the IPEs should cover a broad range of conspicuous safety issues. In addition, the ACRS made three important recommendations [Ref. 18] to the staff to be considered as a part of the A-45 resolution. These recommendations are:

1. Certification of feed and bleed capability at each plant

   By "certification," the ACRS intended that the licensees provide documentation regarding their plant's feed and bleed capability. The documentation was required to show that adequate analysis had been done to the satisfaction of the NRC staff and that feed and bleed could sufficiently be relied upon as an emergency means for cooling the core.

2. Direct use of feedback from experience

   The Institute of Nuclear Power Operations (INPO) had done extensive research on the safety and adequacy of DHR systems. The recommendations proposed by INPO were documented in Significant Operating Event Reports (SOERs). The ACRS suggested that the licensees make use of these recommendations in the course of their IPEs.

3. Protection against sabotage

   One original purpose of A-45 was to consider the safety of DHR systems against sabotage. In order to provide such protection it was suggested that dedicated add-on systems be provided at each utility. However, after several case studies, NUREG/CR-5230 and NUREG-1289, the staff concluded that implementing such a system would not be cost beneficial. The staff also considered that sabotage would not be included as part of the IPE program as it was not considered useful to treat sabotage in a risk perspective. The ACRS recommended that the design for sabotage resistance should be considered in the advanced reactors program.

2.4.1 Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities"

On November 13, 1988, NRC issued GL 88-20. The letter requested licensees to perform a systematic examination to identify any plant-specific vulnerability to severe accidents and report the results to the Commission. The general purposes of this examination, termed as an IPE, were for each utility (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant, (3) to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases and (4) if necessary, reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents [Ref. 3].

The letter also asked the licensees to use the screening criteria based on the procedure established in NUREG/CR–2300 [Ref. 19] to determine important functional sequences and functional failures that might lead to core damage or poor containment performance. These screening criteria are as follows:

1. Any functional sequence that contributes 1E-06/RY or more to core damage,

2. Any functional sequence that contributes 5 percent or more to the total CDF,

3. Any functional sequence that has a CDF greater than or equal to 1E-06/RY and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to the BWR-3 or PWR-4 release categories of WASH-1400,

4. Functional sequences that contribute to a containment bypass frequency in excess of 1E-07/RY, or

5. Any functional sequences that the utility determines from previous applicable PRAs or by utility engineering judgment to be important contributors to CDF or poor containment performance.

GL 88-20 also requested licensees to specifically consider DHR vulnerabilities as part of their IPEs. Each plant's IPE was required to include an examination of its DHR system as well as systems used for the other safety functions for the purpose of identifying severe accident vulnerabilities. The letter requested only Level 1, "determination of CDF based on system and human factor evaluations," and Level 2, "determination of the physical and chemical phenomena affecting containment performance and other mitigating features and the behavior and release of fission products to the environment," internal events initiated at full power. Perspectives regarding low-power, shutdown or external events were not covered. Extension to Level 3, "determination of the offsite transport, deposition, and health effects of fission product releases," was also not considered part of the IPEs.

Identification of plant-specific vulnerabilities to severe accidents due to externally-initiated events (internal fires, high winds/tornadoes, transportation accidents, external floods and earthquakes) were deferred to the Individual Plant Examination on External Events (IPEEE) program and the requirements were issued in Supplement 4 of GL 88-20. A comparable examination of shutdown risk has not been requested.

2.4.2 Generic Letter 88-20 Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f)"

On June 28, 1991, NRC issued GL 88-20, Supplement 4 [Ref. 5]. The letter stated that current risk from external events could be a significant contributor to core damage. It requested that each licensee perform an individual plant examination of external events to identify vulnerabilities, if any, to severe accidents and report the results together with any licensee-determined improvements and corrective actions to the Commission. The general purpose of the IPEEE was similar to that of the internal event IPE.

The resolution of the external event portion of USI A-45 was included in the scope of GL 88-20, Supplement 4. The letter requested that if a licensee discovered a potential vulnerability that was topically associated with USI A45 and proposed measures to dispose of the specific safety issue or concluded that no vulnerability existed, then the staff would consider the issue resolved upon review and acceptance of the results from the IPEEE. The generic letter specifically requested the licensee to identify those USIs it was proposing to resolve.

2.5 NUREG-1289, "Regulatory and Backfit Analysis, USI A-45, Shutdown Decay Heat Removal Requirements"

The primary objective of USI A-45 was to evaluate the safety adequacy of DHR systems in existing LWR power plants and to assess the value and impact of alternative measures for improving the overall reliability of the DHR function.

The NRC initiated a program to evaluate the safety adequacy of DHR systems in LWR nuclear power plants and to assess the value and impact of alternative measures for improving the overall reliability of the DHR function. To provide the technical data required to meet these objectives, this program examined the state of DHR system reliability in a sample of existing plants. The findings of the program were completed in August 1987 and documented in NUREG/CR-5230 [Ref. 17], "Shutdown Decay Heat Removal Analysis: Plant Case Studies and Special Issues," in April 1989. The NRC also issued NUREG-1289 [Ref. 4], "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," in November 1988. This report formed the basis for the regulatory analysis for USI A-45. It included (1) the proposed technical resolution of USI A-45, described in Section 2.3.4, (2) six alternative resolutions considered by the NRC, (3) an assessment of the value and impact of all alternatives considered, and (4) the decision rationale.

2.5.1 NUREG-1289 Definition of Decay Heat Removal

In NUREG-1289, the staff defines the systems related to the DHR function as "those components and systems required to maintain primary and secondary coolant inventory control and to remove heat from the RCS following shutdown of the reactor for normal events or abnormal transients such as the loss of main feedwater, LOOP and SBLOCAs." Initiating events such as ATWS and interfacing system loss-of-coolant accidents (ISLOCAs) or those emergency core cooling systems (ECCS) required only during the reflood phase to maintain coolant inventory following either a medium or large LOCA were not considered as part of the USI A-45 program. However, important support systems such as the component cooling water (CCW) system, essential service water system and emergency onsite AC and DC power systems required for various modes of DHR were considered as part of the USI A-45 program. The transition from reactor trip to hot shutdown was also included in the USI A-45 study. The reliability of the reactor protection system (RPS) was not addressed, and successful shutdown of the reactor was assumed.

2.5.2 Program Approach

The scope of USI A-45 included the entire population of existing LWR plants in the U.S. commercial nuclear power industry. Therefore, the population of plants to be considered for analysis started with 173 units in 1981 [Ref. 17]. Since it was not practical to examine every unit, 70 units were excluded from the study. The basis for exclusion was because they were of a special type e.g., high gas temperature reactors (HGTR) or included in the Systematic Evaluation Program; or were units for which sufficiently similar ones remained in the database; or the units were not far enough along in the construction process to have information available. Therefore, after the initial screening process about 100 units formed the base of the study [Ref. 17].

The preliminary initiatives in evaluating the state of DHR included characterizing the units in terms of their physical parameters (e.g., number and location of safety pumps, number of redundant power trains) and developing a set of qualitative screening questions against which the characteristics could be compared. These qualitative screening questions were based on a detailed review of available probabilistic risk assessments (PRAs), regulatory guidance, and topical studies such as AFW studies, and provided a qualitative screen designed to reveal potential vulnerabilities in DHR capabilities both for design basis events and for beyond design basis situations. The screening was only a tool to highlight plants with potential DHR vulnerabilities for further study.

The initial qualitative screening identified 20 plants, which included all vendor types, as candidates for further study. From this group, six plants were selected after considering vendor, product line, and other issues in which that particular plant might be involved, such as its operational status and utility willingness to participate [Ref. 17].

The six plants selected for the study included four PWR plants (Point Beach, Turkey Point, St. Lucie and ANO-1) and two BWR plants (Quad Cities and Cooper). These plants were identified through a qualitative screening process as plants with potential DHR vulnerabilities [Ref. 4].

In reaching its proposed resolution of USI A-45, the staff considered six specific alternatives.

1. No action
2. Limited scope PRA as a basis for modification
3. Application of specified system modifications to all plants
4. Depressurization and cooling capability
5. Dedicated hot-shutdown capability
6. Dedicated cold-shutdown capability

Following the identification of the six alternatives, plant analysis was done for each alternative to establish its value in terms of reduced CDF and reduced public risk. This information was then used to generate value-impact assessments for each of the alternatives and the results obtained were used as technical bases to develop the generic insights necessary for the resolution of the USI A-45 issue.

As a result of the insights gained during this study, the staff selected Alternative 2 as the necessary precursor to any of the other alternatives. In its evaluation of the issue, the staff concluded that a generic resolution (e.g., a dedicated DHR system) was not cost effective.

NUREG-1289 stated that "a limited scope plant specific PRA could demonstrate the adequacy of the existing DHR function by documenting that its contribution to CDF was relatively low, on the order of 1E-05/RY or less." For the purpose of the resolution of USI A-45, a limited scope PRA was defined as one that considered at least the following initiating events [Ref. 4]:

1. Small LOCAs
2. LOOP transients
3. Transients caused by the loss of the power conversion system (PCS)
4. Transients with offsite power and PCS initially available
5. Transients caused by the loss of an AC or DC bus

The following initiating events were not included in a limited scope PRA:

1. Large and medium LOCAs
2. Reactor vessel ruptures
3. ISLOCAs
4. Steam generator tube rupture (SGTR)
5. ATWS
6. Special emergencies

Issues such as pressurized thermal shock (PTS) were also excluded from the USI A-45 analyses [Ref. 4].

NUREG-1289 also includes a discussion of resolving A-45 through the IPE program. It states that it is considered likely that the analysis described in Alternative 2 "will be performed as part of the overall IPE program and need not be performed separately and in addition to the IPE." It also states "It is believed that such a combination will allow a more efficient as well as a more comprehensive evaluation to be performed compared with separate evaluations for DHR-related events and for other events under the IPE program."

NUREG-1289 notes that the analysis conducted as part of the case studies developed under USI A-45 have indicated that seven special emergencies have a significant potential for influencing the decay heat removal core damage frequency: earthquake, fire, internal flooding, external flooding, winds, lightning, and sabotage. It states that these events, characterized as "special emergencies," may be treated using simplified procedures.

2.5.3 Major Findings of the Program

NUREG-1289 identified support system failures as an important part of the USI A-45 analyses. USI A-45 identified support system failures as a significant contributor to CDF at most plants. These failures were primarily due to 1) limited redundancy, 2) considerable sharing of systems, 3) lack of separation and independence between trains, and 4) poor overall arrangement of equipment from a safety viewpoint. Lack of physical separation and lack of protection of redundant safeguard trains would create vulnerabilities in that single events such as a fire or flood could disable multiple trains of safeguard equipment, resulting in an inability to cool the plant. Insider sabotage could also be a factor in disabling plant equipment. Another weakness identified in NUREG-1289 was the lack of independence (i.e., considerable sharing and interconnection between redundant safeguard trains). This lack of independence was found to create single point vulnerabilities, which could disable the DHR path if one system failed [Ref. 4]

Other major insights gained from the USI A-45 case study include the following [Ref. 17]:

1. For PWRs, the effect of feed and bleed on the total plant CDF was examined as a sensitivity issue in the USI A-45 case studies. Feed and bleed was found to have a significant impact on the total PWR plant CDF. It was determined that the probability of core damage would increase by 4E-05/RY without feed and bleed except for Arkansas Nuclear One, Unit 1 (ANO-1) where it would increase by 1E-03/RY. The CDFs for internal events only, with and without feed and bleed from the plant case studies are shown in Table 1.

Table 1 Effect of Feed and Bleed on the Total Plant CDF

Plant	CDF/RY w/o F&B	CDF/RY w/F&B	ΔCDF/RY
Point Beach	1.87E-04	1.39E-04	4.8E-05
Turkey Point	1E-04	7.1E-05	2.9E-04
St. Lucie	4.8E-05	1.4E-05	3.4E-05
ANO-1	1.23E-03	8.8E-05	1.15E-03

It can be inferred from the above table that a feed and bleed capability could result in a 25 to 90 percent reduction in the plant CDF for internal events. The case studies also revealed that feed and bleed could be successful in plants provided the process is initiated early enough. The required initiation time could be as little as 5 to 8 minutes into an event depending upon when reactor scram occurs and the particular plant type [Ref. 17]. This puts considerable pressure and reliance on the operators.

2. The CDF without secondary blowdown would increase by about 4E-05/RY except for St. Lucie where it would only increase by 3E-06/RY. This led to the conclusion that the ability to reduce the primary pressure below the low-pressure injection (LPI) system operation pressure was important [Ref. 17].

3. For BWRs, Primary containment venting, which is often a last resort for DHR, was found to have relatively little effect on the total plant CDF. This is because venting affects the long term, slowly developing accident sequences in which there is significant time for plant personnel to restore failed equipment prior to venting [Ref. 17]. In cases where the DHR accident sequences dominate the CDF, failures leading to core damage occur prior to the containment pressure and temperature reaching levels that would call for venting to prevent gross containment failure. The possibility of venting the containment subsequent to core melting was not considered in the case studies because of the uncertainties associated with radionuclide scrubbing capabilities of the suppression

pool. It was estimated that if no credit for primary containment venting were given in the study, the total CDF would increase by less than 30 percent [Ref. 17].

4. With credit for feed and bleed and containment venting, the total CDF for PWRs and BWRs ranged from 7E-05/RY to 3E-04/RY with an average CDF value of 2E-04/RY indicating that some improvement was desirable. Without taking credit for feed and bleed and containment venting the CDF ranged from 1E-04/RY to 1E-03/RY with an average value of 4E-04/RY [Ref. 2].

2.5.4 Quantitative Design Objective for USI A-45

The results obtained from the USI A-45 case studies suggested that the mean value of DHR CDF for most LWR plants would be in the range of 1E-04/RY to 3E-04/RY with a few exceptions extending the total range from 3E-05/RY to 6E-04/RY. The study also indicated that if events that A-45 did not consider (e.g., ATWS, ISLOCAs and large LOCAs) were included, these values would be much higher.

The results obtained from the A-45 PRAs indicated that in the existing population of around 100 plants, the chance of at least one accident leading to core damage in the next 10 years could be near 1 in 5 and in the next 30 years (the assumed average lifetime of the existing plants before plant life extension) could be near 1 in 2 [Ref. 4]. These results did not include:

1. Unquantifiable contributions to CDF from events related to DHR failure such as operator errors of commission and unidentified "common cause" events. The staff estimated that the sum of these unquantifiable contributions may be approximately twice the quantifiable portion of the CDF related to DHR failure; i.e., the total (quantifiable plus unquantifiable) CDF related to DHR failure may be three times the result actually obtained by the PRA, which by definition represented only the quantifiable portion of the frequency [Ref. 4].

2. Contributions from events not related to DHR failures. Based on the results of numerous PRAs, the staff believed that the CDF due to DHR failure represented at least one third of the total CDF from all causes; i.e., the total CDF was three times the quantifiable plus unquantifiable CDF related to DHR failure. This factor allowed for contributions from such causes as large LOCAs, vessel ruptures, ISLOCA and ATWS without exceeding the total 1E-04/RY goal for all core damage events [Ref. 4].

It was determined that if the mean CDF of events (quantifiable plus unquantifiable) related to DHR were reduced to 3E-05/RY, the chance of such an accident in the next 10 years would be reduced to about 1 in 30, and in the next 30 years, this chance would be about 1 in 10.

Based on these results, the NRC took initiatives to develop a set of qualitative safety goals and QDOs. The principal QDO selected for USI A-45 was the CDF due to failure of the DHR function. An interim value of 1E-05/RY was recommended for examining the DHR related risk at individual plants [Ref. 4]. On several recent occasions, the staff also applied an additional objective that the total CDF from all causes should not be greater than 1E-04/RY. Therefore, the NRC staff selected the goal that the quantifiable contribution to CDF related to DHR failure should not be greater than 1E-05/RY in order to provide a margin for the unquantifiable contribution. This goal was considered to be appropriate since the application of one factor of three to this goal accounting for unquantifiables and another factor of three accounting for events not related to DHR failures would allow the above quoted total CDF of not greater than 1E-04/RY to be satisfied. Thus the average value of 2E-04/RY obtained from the A-45 case studies, which was considered representative of the CDF related to DHR failure in present plants, was found to be well above the goal of 1E-05/RY selected by the staff.

Another important insight from the experience gained from application of PRAs to US LWRs in USI A-45 suggested that when the systematic examinations for severe accident vulnerabilities would be completed, the existing plants would fall into three broad categories. The staff used the quantitative values listed in Table 2 as a basis for categorization of these events.

Table 2 NUREG-1289 DHR Vulnerability Classification Scheme

Category	Classification of Level 1 DHR Vulnerability	Criterion (/RY)
C1	Frequency of core damage due to failures of DHR function acceptably small, or reducible to an acceptable level by simple improvements	Less than 3.0E-05
C2	DHR performance characteristics intermediate between categories 1 and 3	Less than 3.0E-04 but greater than 3.0E-05
C3	Frequency of core damage so large that prompt action to reduce the probability of core damage to an unacceptable level is necessary	Greater than 3.0E-04

The DHR criterion was structured such that at least the initial vulnerability screening is based solely on the CDF as the figure of merit. This was based on the NRC's conclusion that if the plant were to fall into Category 1 of the NRC's vulnerability classification scheme, little if any cost beneficial modification would be warranted.

2.5.5 Assessment of Other Generic Issues

NUREG-1289 included a discussion of how the implementation of other related generic issues could impact the benefit of modifications designed to resolve USI A-45. Several of these generic issues were being resolved coincident with the implementation of the IPE program. Three significant issues, USI A-44: Station Blackout, USI A-46: Seismic Qualification of Equipment in Operating Plants, and Generic Issue 124: Auxiliary Feedwater System Reliability, have directly impacted the potential benefit of the resolutions of USI A-45. These issues are described below.

2.5.5.1 USI A-44: Station Blackout

"Station blackout" is defined as the loss of all AC power and the unavailability of the redundant onsite emergency AC power. NUREG-1289 provides a discussion of the relationship between A-44 and A-45. It states ,"The plant-specific modifications considered in the USI A-45 program based on PRA results (Alternative 2) often addressed the vulnerabilities to station blackout. Virtually all the reduction in core damage frequency from USI A-45 alternatives that reduce vulnerabilities to long-tem station blackout events would be accomplished by USI A-44. For the plants evaluated in USI A-45, USI A-44 would achieve about 40 to 70 percent of the estimated reduction in core damage frequency associated with modifications considered in USI A-45."

2.5.5.2 USI A-46: Seismic Qualification of Equipment in Operating Plants

NUREG-1289 states "This issue examines the need to verify the seismic adequacy of mechanical and electrical equipment required to safely bring the reactor and plant to a safe shutdown condition and to maintain it in a safe condition. The specific objective of the A-46 task is to develop viable, cost-effective alternatives to current seismic qualification requirements to be applied to operating nuclear power plants." NUREG-1289 assesses the reduction in benefit of implementing USI A-46 as opposed to the reduction in benefit of USI A-45 from the implementation of USI A-45. It concludes that a total industry USI-A46 cost savings of $14 million dollars could be achieved if Alternative 6, a Seismic Category 1 alternate decay heat removal system, is selected as the resolution for USI A-45.

2.5.5.3 Generic Issue 124: Auxiliary Feedwater System Reliability

Generic Issue 124 is concerned with the failure probability associated with AFW systems and resulted in a specified AFW system reliability criteria. NUREG-1289 notes that "implementation of an USI A-45 independent and dedicated DHR would sufficiently improve overall AFWS reliability to render most hardware modifications proposed in GI-124 unnecessary."

2.5.6 Feed and Bleed Alternative 4

Alternative 4 in NUREG-1289 evaluated depressurization and cooling capabilities of the plants. For PWRs, the decay heat removal methods examined were bleed and feed or secondary side blowdown, and for BWRs, containment venting was examined. This discussion is included in order to gain a perspective on potential feed and bleed enhancements considered by NUREG-1289.

As indicated in NUREG-1289, many PWRs have some capability to cool the reactor through bleed and feed; however, successful completion of this process is limited in some plants by the pressure relief capacity of the PORVs (or other type of relief valve) and the makeup capacity of the HPI system at high pressure. As noted in the report, bleed and feed cooling could be enhanced by improving the capability of either the PORVs or the HPI system. Suggested improvements to the PORVs included: increasing the size of the existing valve, adding an additional PORV, installing a special line and valve for bleed operations, or installing an air-operated device on the existing safety valves. Improvements to the HPI system included upgrading or replacing the existing HPI pump with one of higher flow capacity and discharge pressure.

The value-impact analysis considered three cases associated with the bleed and feed option for Alternative 4. Case 1 applied to exisiting PWRs where some capability to bleed and feed already exists, but it is decided to increase the relief capacity of the pressurizer by adding air-operated assist devices for the existing safety valves. Case 2 considered adding additional relief capacity to the pressurizer for depressurization and using the present HPI system for RCS inventory makeup. Case 3 considered adding additional relief capacity to the pressurizer for depressurization and adding a new HPI pump and piping system for RCS inventory makeup. The costs associated with Case 1 were the least at $650K (at that time). These case studies were performed assuming that bleed and feed capability is added to a plant that did not already have that capability. The study was only able to justify Case 1 for one plant based solely on quantitative cost-benefit considerations and considering only averted offsite dose.

The focus of Alternative 4 is whether additional enhancements to the current bleed and feed capability should be considered. Therefore, an expectation was that the plant-specific IPEs would consider such enhancements in the assessment of USI A-45.

3 ASSESSMENT OF THE REGULATORY EFFECTIVENESS OF USI A-45

The scope of the present USI A-45 assessment is to determine whether the resolution approach was reasonable and whether certain areas may exist that need the staff's attention. The expectations for this assessment are derived from SECY 88-260, GL 88-20 and related USI A-45 studies such as NUREG/CR-5230 and NUREG-1289. The assessment uses plant-specific IPE reports, plant-specific modification list, staff evaluation reports (SERs), technical evaluation reports (TERs), NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," dated April 2002 [Ref. 20], and responses to requests for additional information (RAI) to make conclusions about the effectiveness of the resolution of USI A-45.

3.1 Method for Assessing Regulatory Effectiveness of USI A-45

The evaluation of the resolution of USI A-45 includes performing a review of licensees' actions to resolve safety issues regarding USI A-45, and report the findings. Relevant information was reviewed from the generic safety issue program and other generic activities to develop the information needed to document NRC's expectations regarding resolution of USI A-45. Expectations were established from USI A-45 case studies documented in NUREG/CR-5230 and NUREG-1289. IPE reports, IPE Insights Report, NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance" [Ref. 21], and previous contractor reports prepared in support of resolution of this issue were also reviewed in order to draw conclusions regarding outcomes of efforts to resolve the technical concerns of USI A-45. The review of the IPEEEs was limited to the summary information and observations contained in NUREG-1742. This limited review appears warranted based on the insights gained from the detailed plant-specific IPE reports.

The overall question is:

- Were the IPE and IPEEE programs effective in resolving USI A-45?

Two facets of this question are addressed:

- Were expectations regarding DHR risk reduction achieved as a result of the program?
- Was this approach (plant-specific IPEs, plant-specific IPEEEs, self-assessment of vulnerabilities, licensee initiative to devise and implement plant modifications as deemed appropriate) effective in achieving this DHR risk reduction?

The first question is answered below (to the extent that it can be answered based on a review of IPE results and the IPEEE summary information contained in NUREG-1742). The second question is more difficult. Given that the approach was predicated on there being no apparent generic fixes to impose (unlike SBO and ATWS), part of answering this question involves determining whether there were generic fixes that could appropriately have been imposed. This question was addressed by taking up the following questions:

- Did any identifiable class of plants make similar DHR improvements?
  - Which DHR systems do plants credit? (Design basis strategies as well as alternative methods of core cooling and containment heat removal)
  - Within a class of similar plants, does the credited complement of plant features correlate meaningfully with the kinds of DHR improvements reported?

A finding that certain design features correlate with certain kinds of fixes would suggest that a generic requirement of some kind might have been a more effective approach.

The first step in assessing the regulatory effectiveness of USI A-45 involved the classification of BWRs and PWRs. The BWRs were classified according to NUREG-1560 and included BWR 1/2/3, BWR 3/4 and BWR 5/6 plants. The PWRs were also classified according to NUREG-1560 and included Combustion Engineering (CE), B&W, Westinghouse 2-Loop, Westinghouse 3-Loop and Westinghouse 4-Loop plants.

The second step involved a review of the IPE submittals. One purpose of reviewing the IPE submittals was to determine the approach taken by each class of plants to effectively resolve USI A-45. The DHR systems were identified for each plant. The DHR systems analyzed for BWRs include the high pressure coolant injection systems (HPCI and HPCS), low pressure coolant injection systems (LPCI), low pressure core spray (LPCS), automatic depressurization system (ADS), heat removal systems (isolation condenser and various modes of RHR) and containment venting. For PWRs, the DHR systems analyzed include the high pressure injection (HPI) system, the AFW system, systems used for feed and bleed, secondary side feed using the condensate pumps, and containment cooling systems.

The third step involved identifying DHR-related plant modifications from over 500 improvements proposed by licensees as addressed in the IPE submittals. Some of these modifications were suggested as a result of the SBO rule and other regulatory activities. To assess the effectiveness of the resolution of USI A-45, enhancements already implemented or planned specifically to improve the reliability of DHR systems at U.S. commercial nuclear power plants and not undertaken as a result of the SBO rule or other requirements were identified from among the 500. Also, modifications made to support systems such as the AC/DC power systems or service water (SW) system that could impact many systems were not considered.

The information gathered as a results of the IPE submittal review and DHR-related plant modification was documented in a series of tables included with a letter from R.W. Youngblood, ISL, to J.V. Kauffman, NRC, "Transmittal of Tables Summarizing IPE Submittal Results," dated July 15, 2003 [Ref. 22].

Since all plants do not report the DHR CDF in their IPE, the IPE database was used to determine the CDF contribution of DHR accident sequences to the total CDF based on NUREG-1289's definition of DHR. Three DHR CDFs are computed from the IPE database. The first DHR CDF has accident sequences with SBO included and is considered to contribute to the "IPE DATABASE DHR CDF." Then accident sequences with SBO as attributes are removed, and the remaining DHR accident sequences are considered to contribute to "IPE DATABASE NON-SBO DHR CDF." Finally, those accident sequences with loss of HPI, but not loss of AFW capability are removed. The remaining non-SBO DHR accident sequences are considered to contribute to "IPE DATABASE NON-SBO, NON-HPI DHR CDF." The results obtained are documented in Table 4, "DHR CDF for BWRs" and Table 5, "DHR CDF for PWRs." This table also lists the "IPE DHR CDF" which is the DHR CDF reported in the IPEs for some plants. The rest of the results are based on the information available in the IPE database. These results were used to classify plants into one of the three DHR risk categories defined in NUREG-1289 and to compute the average DHR CDF for BWRs and PWRs.

It is important to note that in most cases the IPE submittals only report the CDF changes based on the overall proposed modifications. Most plants do not give the net CDF effect from specific enhancements in their IPE and therefore, the net CDF from specific DHR modifications cannot be determined.

Finally, for the purposes of the USI A-45 assessment, another table, "Summary of USI A-45 Expectations and Outcomes," (Table 6) was developed for both the BWRs and PWRs. This table presents expectations (desired outcomes) derived from NUREG-1289 and the actual outcomes in terms of DHR CDF obtained from the IPE database. The table could be used as a guide to determine if the IPE program was an effective way to resolve USI A-45 and if the expectations of USI A-45 were met.

Following the completion of the review of the effectiveness of the IPE in resolving the internal events portion of USI-A45, a review of the summary information and insights contained in NUREG-1742 was performed for the external events portion. This more limited review is thought to be adequate since no generic internal event fixes were discovered during the detailed reviews of plant-specific IPEs.

3.2 Evaluation of USI A-45 in Response to Generic Letter 88-20

The DHR function includes removing heat from the core. This is a broad scope. Because several generic issues touch on DHR, it is necessary to develop a clear rationale for associating particular plant modifications with A-45. This section summarizes the rationale used in the present effort for BWRs and PWRs.

3.2.1 Review of BWR Submittals

The primary method for removing decay heat in BWRs while at high pressure is through the steam lines to the turbine condenser. The condensate is normally returned to the reactor vessel by the MFW system. In some early BWRs, an isolation condenser (IC) can provide heat removal. ICs can condense steam from the core and return condensate directly to the core, rejecting the heat to an external pool of water and thereby accomplishing the DHR function in a single step.

However, most BWRs have a steam turbine driven reactor core isolation cooling (RCIC) system that controls primary system inventory during abnormal transients or if AC power is lost. If the condenser is unavailable, energy is removed through the safety/relief valves (SRVs) to the suppression pool.

Alternative emergency cooling in the form of either a HPCI system or a high pressure core spray (HPCS) system is also provided on most BWRs. These systems can provide makeup to the reactor vessel from either the condensate storage tank (CST) or the suppression pool.

At low RCS pressure, decay heat is removed by the RHR system. If normal methods of core cooling and pressure reduction are not available, the pressure can be reduced by the ADS, which opens the SRVs and rejects energy to the suppression pool. At low pressure, long term cooling is initiated using the RHR system which circulates water through the core, passing it through heat exchangers to remove decay heat and maintain cold shutdown conditions.

3.2.1.1 BWR 1/2/3 Plant Response

The feature that differentiates BWR 1/2/3 plants from the other BWR types is the availability of the IC [Ref. 21]. The IC provides a heat sink for the reactor through natural circulation if it is isolated from the main condenser or if a loss of all feedwater occurs. As part of the NRC's program to assess the regulatory effectiveness of USI A-45, the following BWR 1/2/3 plants were analyzed:

1. Nine Mile Point Unit 1
2. Oyster Creek
3. Millstone Unit 1
4. Dresden Units 2 & 3

Nine-Mile Point 1 and Oyster Creek are BWR 2 plants having Mark I containments. Dresden Units 2 & 3 and Millstone 1 are BWR 3 plants and also have Mark I containments.

The DHR evaluation for the BWR 1/2/3 plants yields the following results [Refs. 23-26]:

1. DHR systems available at plants in this group, other than normal heat removal paths, include the IC, various LPCI modes and containment venting systems. The availability of systems such as motor-driven feedwater pumps and ICs reduces the importance of transients with loss of injection as compared to the other BWR groups. The primary means of removing decay heat from the RCS is through the main steam lines to the condenser. If the main condenser is unavailable and the main steam isolation valves (MSIVs) are closed, decay heat is removed through the IC. In this path, the decay heat is discharged to the atmosphere via boil-off of shell side inventory. If the IC fails then the decay heat is discharged into the containment through the operation of relief or safety valves or through the break in the event of a LOCA and is then removed by the containment sprays (CSs). A last resort for DHR is containment venting.

2. For BWR 1/2/3 plants, transients leading to loss of DHR systems involve 1) loss of containment heat removal with a subsequent loss of coolant inventory makeup and 2) failure of the IC. Sequences involving loss of DHR include accidents in which the coolant injection succeeds initially but is followed by a failure of all containment heat

removal. As an accident of this type progresses, direct heat removal from the vessel fails, resulting in relief valve openings that carry the steam generated in the vessel to the suppression pool. As the suppression pool temperature reaches saturation, containment pressure also rises, and if a DHR path is not established, failure of containment due to high pressure eventually occurs. Coolant injection systems also fail either because of high suppression pool temperatures or because of adverse environments created in the containment or the reactor building following containment venting/failure. Also, an overpressure failure of the containment results in substantial flashing of saturated water inside the containment and the ECCS suction lines. In addition, containment failure leads to the release of large quantities of steam into the reactor building. Both of these events can jeopardize the operability of coolant makeup systems located in the reactor building, thus leading to the potential of core damage.

3. All plants in this group voluntarily credit the use of multiple feedwater injection systems (e.g., the control rod drive (CRD) and FPS as alternate injection sources) and containment venting.

4. The Millstone 1 plant does not identify the DHR CDF in its IPE. Since all plants do not report the DHR CDF, the number of plants falling into a particular DHR risk category cannot be determined. For this purpose a query was developed using the IPE database to determine the "IPE DATABASE DHR CDF" for all the plants and categorize them into one of the three DHR risk categories. The results of this query are presented in Section 4, Table 4.

   According to the query results obtained from the IPE database, the CDF due to the failure of DHR systems at the BWR 1/2/3-plant group falls into Category 1 of the NRC's vulnerability classification scheme. Also, the total plant CDF for this type meets the staff's objective of an overall CDF of less than or equal to 1E-04/RY.

5. None of the plants in this group identify any DHR vulnerabilities. To resolve USI A-45, licensees in this group made certain hardware modifications to improve the reliability of both DHR and coolant injection systems.

   Millstone implemented the bulk of hardware improvements to improve the performance of its DHR systems in this group of BWR plants. In addition to the hardware changes, some plants implemented procedural changes to improve the reliability of their DHR systems.

6. Out of the four plants analyzed, two plants, Nine Mile Point 1 and Oyster Creek, installed the hardened vent and credited it in their IPE. Dresden 2 & 3 installed the hardened vent but did not credit it in the IPE. Millstone 1 also addressed the proposed Mark I containment performance improvements in its IPE. The IPE analysis showed that less than 1 percent reduction in the CDF could be achieved from the proposed hardened wetwell vent (less than 1E-07/RY). However, the licensee installed the hardened vent in 1994 after the completion of the IPE.

7. All BWR 1/2/3 plants do not identify the impact of modifications on the original total and DHR CDF in the IPE.

8. In general, improvements were made on a plant-specific basis and comparable modifications were not made among the various plants.

3.2.1.2 BWR 3/4 Plant Response

The feature that differentiates BWR 3/4 plants from the BWR 1/2/3 type is the availability of the RCIC system [Ref. 18]. As part of the NRC's program to assess the regulatory effectiveness of USI A-45, the following BWR 3/4 plants were analyzed:

1. Quad Cities 1 & 2
2. Fitzpatrick
3. Brunswick 1 & 2
4. Cooper
5. Pilgrim 1
6. Monticello
7. Fermi 2
8. Duane Arnold
9. Browns Ferry 2
10. Vermont Yankee
11. Hatch 1 & 2
12. Peach Bottom 2 & 3
13. Hope Creek
14. Limerick 1 & 2
15. Susquehanna 1 & 2

All the plants in this class have Mark I containments except Limerick and Susquehanna, which have Mark II containments.

The DHR evaluation for the BWR 3/4 plants yields the following results [Refs. 27-41]:

9. DHR systems available for this plant group include the PCS, RHR and containment venting. Some plants also have alternative coolant injection systems to supply water to the reactor vessel when the HPCI and RCIC systems have failed. Some units use the diesel-driven firewater system. The ability of this system to inject coolant water varies from one plant to the other and is not modeled as a preferred source of injection in cases of early failure of RCIC and HPCI. Diesel-driven firewater is most important for sequences with delayed failure of HPCI and RCIC because there is sufficient time to configure the firewater system for injection in those cases.

10. For BWR 3/4 plants, loss of DHR transient sequences involve accidents in which coolant injection succeeds initially but containment heat removal fails. In this condition, the suppression pool heats up leading to containment pressurization after suppression pool saturation is reached and the containment eventually fails if it is not vented in time. Coolant injection also eventually fails as a result of high suppression pool temperature or adverse conditions in the suppression pool or steam in the reactor building.

11. Only one plant, Browns Ferry, does not take credit for containment venting in its IPE. However, all plants in this group voluntarily credit the use of feedwater and multiple HPI systems (such as RCIC and HPCI) and containment venting. The availability of motor-

driven feedwater pumps also reduces the CDF from LOCAs as these pumps can continue to operate during transients with MSIV closure. All plants have available injection systems other than ECCS that can have all required equipment including associated support equipment located outside the containment and reactor building. Thus these systems are not subject to the potential harsh environments. Examples of such systems include the CRD and the condensate systems. Plants with such systems that experience no adverse effects following containment failure have lower CDFs.

12. The CDF due to the failure of DHR systems at the BWR 3/4 plant group falls into Category 1 (less than 3E-05/RY) of the NRC's vulnerability classification scheme.

13. None of the plants in this group identify any DHR vulnerabilities. One plant, Susquehanna, did not make any DHR modifications. To resolve USI A-45, most of the licensees installed the hardened wetwell vent to improve the reliability of DHR systems.

    Other modifications made by plants in this group include procedural changes such as allowing the tripping of RHR/CS pumps upon loss of HVAC to insure operation of one pump without room cooling and modifications to power the SRV solenoid valves from the DC instrument bus.

14. Most licensees address the Mark I improvements in response to the GL 88-20, Supplement 1 and credit the installation of a hardened vent in the IPE. The impact of venting on the CDF varies across plants depending on the available systems for both DHR and coolant injection. Use of these vents can reduce the CDF for this accident class. However, some licensees, such as Browns Ferry, Cooper, and Monticello have not modeled these vents in their IPEs. Therefore, there are differences in the results from plant to plant because of differences in the treatment of venting and its effects.

15. All BWR 3/4 plants do not identify the impact of modifications on the original total and DHR CDF in the IPE.

16. In general, improvements were made on a plant-specific basis and comparable modifications were not made among the various plants.

    Only one plant, Brunswick gives the impact of the modifications on DHR CDF. The IPE determined two changes having a significant effect on the DHR: installation of a hardened wetwell vent and taking credit for CRD pump injection at low pressures. Based on these two changes, the IPE reported a revised CDF of 1.8E-06/RY and a revised overall CDF of 1.1E-05/RY.

3.2.1.3 BWR 5/6 Plant Response

The main feature that differentiates these plants from the BWR 3/4 type is the replacement of the HPCI system containing a turbine-driven pump with a HPCS system containing a motor-driven pump [Ref. 16]. As part of the NRC's program to assess the regulatory effectiveness of USI A-45, the following BWR 5/6 plants were analyzed:

1. Washington Nuclear Power 2 (WNP-2)
2. Nine Mile Point 2
3. Perry 1
4. Clinton
5. River Bend
6. Grand Gulf

WNP-2 and Nine-Mile Point 2 are BWR 5 plants and have Mark II containments. The remaining are BWR 6 plants and have Mark III containments.

The DHR evaluation for the BWR 5/6 plants yields the following results [Refs. 42-48]:

- DHR systems available at plants in this group include the PCS, RHR systems and containment venting systems. The primary means of removing decay heat from the RCS is through the main steam lines to the condenser. If the main condenser and feedwater are not available systems such as RCIC, HPCS, LPCI and low pressure core spray (LPCS) or other alternate coolant injection systems are used for level control and suppression pool cooling (SPC) mode of RHR is used for DHR until the shutdown cooling (SDC) mode of RHR can be used. If SPC and SDC mode of RHR fails, then either containment spray or containment venting removes the decay heat.

- For BWR 5/6, sequences involving loss of DHR include accidents in which the coolant injection succeeds initially but is followed by a failure of all containment heat removal. In this situation the suppression pool heats up leading to containment pressurization and failure if it is not vented in time.

- For BWR 5/6 transients with loss of coolant injection involve the loss of support systems as initiating events. For example, LOOP and loss of AC bus initiators are significant contributors at many BWR 5/6 nuclear power plants such as Grand Gulf and Nine-Mile Point 2 because of the isolation of non-safety related support systems. These support systems are essential for the continued operation of coolant injection systems such as feedwater, condensate and CRD systems. LOOP is important at some plants because it causes the failure of the PCS including motor-driven feedwater and condensate pumps and can also cause the failure of the CRD system depending on whether the CRD system and required support systems are powered by emergency buses. Failures such as the loss of DC power, instrument air (IA) and cooling water system are also important at some plants because plant-specific alignments of these systems negatively affect coolant injection systems by isolating the systems through loss of control power or essential cooling, or through impacts on other support systems.

  Other support system failures involve the loss of DC power, cooling water systems, reactor building component cooling water (RBCCW), ESW, plant service water (PSW), shutdown service water (SX) and IA. The failure of these support systems varies across the plants because the design of these systems and the dependence of DHR and other support systems on them are highly plant specific.

- All plants in this group voluntarily credit the use of alternate injection sources. All plants in this group have alternate injection sources such as, FPS, condensate transfer system (CTS), CRD, standby service water (SSW) cross-tie, and ESW B/RHR B cross-tie that provide coolant makeup to the reactor vessel during accidents in which normal sources of emergency injection have failed. These alternate injection sources have been credited in the IPE. BWR 5/6 plants also have motor-driven feedwater pumps that can supply water to the reactor regardless of the availability of motive steam and the main condenser, which are required for the operation of the turbine driven reactor feedwater pumps. Thus the feedwater system can provide core-cooling water for transients with main steam line isolation.

- All plants credit containment venting in their IPEs. Perry evaluated the possibility of installing a passive containment vent as a means to improve the reliability of its DHR systems.

  The Perry IPE performed a sensitivity analysis on the impact of containment failure to the loss of reactor pressure vessel (RPV) injection and subsequent core damage. The analysis identified that the addition of a passive containment vent path independent of AC power would reduce the frequency of core damage and also result in reduction of offsite releases. Perry evaluated the possibility of installing the passive containment vent path and found that the overall CDF excluding flooding would be reduced from approximately 1.2E-05/RY to 9E-06/RY. The frequency of core damage sequences with containment failure would also reduce from 2E-06/RY to 4.2E-07/RY. However, Perry did not credit the installation of a passive containment vent in its IPE. One plant, WNP-2, considered the installation of the hardened vent. However, the IPE analysis showed that the DHR risk would not be reduced due to the installation of the vent and that it would have very limited benefit in reducing offsite consequences.

- The Grand Gulf plant does not identify the DHR CDF in its IPE. Since all plants do not report the DHR CDF, the number of plants falling into a particular DHR risk category cannot be determined. For this purpose a query was developed using the IPE database to determine the "IPE DATABASE DHR CDF" for all the plants and categorize them into one of the three DHR risk categories. The results of this query are presented in Section 4, Table 4.

  According to the query results obtained from the IPE database, the CDF due to the failure of DHR systems at the BWR 5/6 plant group falls into Category 1 of the NRC's vulnerability classification scheme. Also, the total plant CDF for this type meets the staff's objective of an overall CDF of less than or equal to 1E-04/RY.

- None of the plants in this group identify any DHR vulnerabilities. Four plants, Clinton, River Bend, Nine Mile Point 2 and WNP-2, do not identify any DHR modifications. To resolve USI A-45, other licensees in this group made improvements/modifications to procedures and system hardware that have the potential to improve the reliability of DHR systems and therefore, reduce the CDF contribution from this accident class.

- Some licensees made DHR modifications and credited them in their IPEs. One plant in this group improved the reliability of its HPCS system by modifying its maintenance activities and system surveillance procedure to include testing the pump suction line from the suppression pool. Another licensee installed a bypass line around a check valve with a normally closed valve in it to reduce the amount of time to align the FPS system for injection into the reactor for core cooling. The other licensee installed a fast

FPS system and incorporated it into its plant emergency instructions (PEI) improvement plan.

- None of the BWR 5/6 plants identify the impact of modifications on the original total and DHR CDF in the IPE.

- In general, improvements were made on a plant-specific basis and comparable modifications were not made among the various plants.

3.2.2 Review of PWR Submittals

The primary method for removing decay heat in PWRs is through the steam generators. This is the preferred method of DHR until shutdown cooling entry conditions are reached and the shutdown-cooling system is placed in service. Effective heat removal using the steam generators requires circulation of the primary coolant through the core with energy removal in the steam generators by use of steam release and makeup. Although it is preferred to utilize the main condenser as the heat sink, PWRs also have the capability to directly release steam to the atmosphere via the steam dump valves and other manually aligned pathways. There are two mechanisms available for steam generator makeup: the AFW (or, at some plants, EFW) and low pressure feed (LPF) using the condensate pumps. The normal method is using the AFW. In the case of a loss of all main and AFW, the backup method of steam generator makeup is LPF. This requires the depressurization of the available steam generators to below the maximum discharge pressure of the condensate pumps. This places reliance on the atmospheric dump valves' (ADVs) availability, as no other system is readily available to depressurize the steam generators to enable condensate pump flow.

PWRs also have the capability to provide DHR by HPI-cooling. HPI-cooling is a feed and bleed operation in which core decay heat is removed by injecting coolant with a HPI pump and discharging coolant through the power operated relief valves (PORV) on the pressurizer. This operation effectively transfers core heat to the containment atmosphere. In order to achieve a long-term stable state, heat must be removed from the containment via the containment heat removal systems (e.g., containment spray, RHR, or containment fan coolers).

HPI-cooling is a method of core cooling not generally analyzed in the Safety Analysis Report (SAR). It is typical for PRAs to credit this method, and other core-cooling methods not analyzed in the SAR, because the analyses are typically intended to reflect actual risk based on current plant capability. The risk results are therefore influenced by core-cooling methods that have not undergone regulatory review. Nevertheless, as discussed below, these methods play a significant role in DHR capability.

In summary, the primary front-line systems supporting DHR at PWRs are the MFW (and condensate), AFW (or EFW) and the HPI-cooling systems of HPI and LPI. Any modifications done to these front-line systems are considered in the resolution of USI A-45.

3.2.2.1 Combustion Engineering (CE) Plant Response

The CE plant group includes 15 plant units represented by 10 IPE submittals. As part of the NRC's program to assess regulatory effectiveness of USI A-45, the following plants were analyzed:

1. ANO-2
2. San Onofre 2 & 3
3. Calvert Cliffs 1 & 2
4. Millstone 2
5. Waterford 3
6. Fort Calhoun 1
7. Palisades
8. St. Lucie 1 & 2
9. Palo Verde 1, 2, & 3

The DHR evaluation for the CE plants yields the following results [Refs. 49–59]:

- Most CE plants have multiple sources of water as suction supplies to the AFW pumps. Only two plants, San Onofre and Palo Verde do not have alternate AFW suction pump sources. Under normal operating conditions, the AFW pumps take suction from the CST. However, in an emergency or when the CST is depleted, the remaining plants have different alternate water supplies to the AFW suction pumps: Millstone 2 and Palisades align the fire water and FPS as a backup suction source to the AFW pumps; Fort Calhoun uses the diesel driven fire pump for long-term makeup to the AFW system; ANO-1 uses SW as a backup to the EFW suction; Waterford uses two wet cooling tower basins and non-seismically qualified CST and its transfer pump and during SBO or total loss of feedwater it uses the FPS to provide flow to the steam generators; Calvert Cliffs can crosstie flow from the motor-driven pump with the other AFW unit from the other plant.

- Most CE plants take credit for feed and bleed. The ability to feed and bleed varies among the plants in this group. Three plants (San Onofre, Palo Verde and Waterford) do not have PORVs and therefore do not have the capability to feed and bleed. The success criterion for feed and bleed varies among the plants in this group. Three plants (Millstone 2, Calvert Cliffs and St. Lucie Unit 1) require two PORVs and one HPI pump for successful feed and bleed. All other plants require only one PORV and one HPI pump.

  One plant, ANO-2 has the capability to feed and bleed without any PORVs. The IPE states that once-through cooling can be achieved by opening the pressurizer ECCS vent valves or the low temperature overpressure protection (LTOP) valves and inject coolant via the HPSI pumps. The ANO-2 IPE analysis also states that without credit for feed and bleed cooling, the CDF would increase by 536 percent (from 3.4E-05/RY to about 2.1E-04/RY).

- Only six of the nine CE plants (San Onofre, Millstone 2, Palisades, Palo Verde, ANO-2 and Waterford 3) take credit for steam generator depressurization and the use of condensate as heat removal in the event that both MFW and AFW are not available.

- Only seven plants (San Onofre, Palisades, Palo Verde, ANO-2, Waterford, St. Lucie, and Calvert Cliffs) identify the DHR CDF in their IPE. Since all plants do not report the DHR CDF, the number of plants falling into a particular DHR risk category cannot be determined. For this purpose a query was developed using the IPE database to determine the "IPE DATABASE DHR CDF" for all the plants and categorize them into one of the three DHR risk categories. The results of this query are provided in Section 4, Table 5.

According to the query results obtained from the IPE database six plants (San Onofre, Millstone 2, St. Lucie, Fort Calhoun, ANO-2 and Waterford 3) have a DHR CDF that falls into Category 1 (less than or equal to 3E-05/RY) of the NRC's vulnerability classification scheme. The DHR CDF for the other three plants (Palisades, Palo Verde and Calvert Cliffs) falls into Category 2 (greater than 3E-05/RY but less than 3E-04/RY) of the NRC's vulnerability classification scheme. Only one plant, Calvert Cliffs, has a total CDF of 2.4E-04/RY that does not meet the staff's objective of an overall CDF of 1E-04/RY.

For two plants, Palisades and Calvert Cliffs, there is a discrepancy in the results obtained from the IPE and the IPE database. Calvert Cliffs reports a DHR CDF of 1.4E-04/RY in its IPE. However, the "IPE DATABASE DHR CDF" for Calvert Cliffs is 2.61E-05/RY. One important point to note is that some of the sequences in the IPE database are combined into one aggregate sequence that is simply identified as the "remainder," which were excluded from consideration. It is possible that for Calvert Cliffs some of the DHR contributions are in the remainder and could be identified if more information were available about the "remainder." On the other hand, Palisades identifies a DHR CDF of 3E-05/RY in its IPE. The "IPE DATABASE DHR CDF" for Palisades is 3.77E-05/RY. Since the IPEs do not give details on individual contributions to DHR CDF, the question as to why this discrepancy exists cannot be resolved.

Only one CE plant, Calvert Cliffs, identifies two DHR vulnerabilities in its IPE. The CDF due to DHR failure identified in the IPE is 1.4E-04/RY [Ref. 55].

The first vulnerability identified in the IPE is the failure of AFW with a 34 percent contribution to the DHR CDF. This is influenced by valve or human failures in hand operated valves needed for long term supply of water for AFW after the CST is depleted.

In the process of modeling the functions associated with the AFW, the likelihood that these valves would function was also evaluated. The review of the testing and maintenance practices on these valves found no surveillance test procedures, no performance evaluations and no preventive maintenance procedures. The IPE identified that the lack of such practices would significantly increase the failure likelihood of these valves to function on demand [Ref. 55]. As a result, the IPE identified a potential enhancement such as improved surveillance on AFW condensate-related manual valves. According to the IPE the surveillance would cycle these valves approximately every 6 months to ensure they remain functional and was implemented on July 1,1994. This improvement was incorporated into the original model as well as credited in the original IPE model. In response to an RAI, the licensee stated that the removal of the new surveillance would result in a 2.3 percent increase in the overall CDF to 3.2E-04/RY [Ref. 57].

The second vulnerability identified in the IPE are the failures associated with inadvertent actuation of the engineered safety features actuation system (ESFAS) and auxiliary feedwater actuation system (AFAS) with an 18 percent contribution to the DHR CDF.

The IPE identified that all operating crews will be trained on the consequences and recovery of inadvertent ESFAS/RPS/AFAS. Classroom instruction will be used for awareness training and the CCNPP's plant-specific simulator will be used to provide practical experience with the mitigation strategy for this issue. This training was completed during the 1994 operator requalification training cycle [Ref. 55]. It was estimated in the IPE that a 1 percent reduction in CDF could be achieved by improving the likelihood of success for operator actions associated with this issue.

- Only one CE plant, Calvert Cliffs, identifies two DHR modifications in its IPE that would improve the reliability of the DHR function.

The Calvert Cliffs IPE identified that changes in plant procedures and operator training were the most cost-beneficial improvements for the vulnerabilities described above. Some examples of other modifications made by plants in this group include the installation of an AFW pump, addition of accumulators for AFW regulating valves, and the revision of procedures using PORVs to depressurize the RCS.

The Palo Verde IPE identifies a DHR CDF of 8.1E-05/RY (approximately 85 percent) of the overall CDF of 9E-05/RY. The IPE evaluated the means for accomplishing DHR. The Palo Verde IPE takes credit for two secondary systems that provide secondary cooling: AFW system and alternate feedwater system (plant-specific name for LPF) [Ref. 53]. The IPE also considered additional means of improving DHR reliability by adding a gas turbine generator. This would provide additional power sources for feedwater recovery in the case of SBO. In addition, the IPE also considered the following modifications: PORVs to facilitate feed and bleed, procedure changes to improve the ability of downcomer feedwater isolation valve (FWIV) and feedwater control valves (FWCVs) to remain operable following loss of instrument air (IA).

After a cost benefit analysis, Palo Verde installed two gas turbine generators in response to the SBO rule resulting in a CDF of 2.7E-05/RY. No credit was taken for the installation in the IPE analysis.

- All CE plants do not identify the impact of modifications on the original total and DHR CDF in the IPE. Three plants (San Onofre, Waterford 3 and Fort Calhoun) do not identify any DHR modifications and therefore, the revised total and DHR CDF cannot be computed. Two plants, ANO-2 and Palisades do not give the impact of the modifications on the original total and DHR CDF. Millstone 2 gives the impact of the modification only on the total plant CDF. The St. Lucie IPE gives the total and DHR CDF after taking credit for modifications. Therefore, the impact of the modifications on the original total and DHR CDF for St. Lucie could not be computed.

Only one plant, Palo Verde, characterizes the net CDF effect from specific enhancements.

The Palo Verde IPE identifies the impact of the three modifications on the total and DHR CDF [Ref. 53]. This is shown in Table 3 below:

Table 3 Effect of Proposed DHR Enhancements on the Original Plant and DHR CDF

Modification	DHR CDF*	Total CDF*
None	8.1E-05	9E-05
Procedure Change	6.4E-05	7.1E-05
PORVs	6E-05	6.8E-05
Gas Turbine Generators	5.6E-05	6.3E-05
PORVs + Procedure Change	5.6E-05	6.3E-05
Gas Turbine Generators + Procedure Change	4.3E-05	4.9E-05
PORVs + Gas Turbine Generators	3.7E-05	4.4E-05
PORVs + Gas Turbine Generators + Procedure Change	3.5E-05	4.1 E-05

* CDF is calculated in terms of a per year basis.

Based on the above results, the Palo Verde IPE concluded that there were no unique plant-specific vulnerabilities associated with the loss of DHR. The addition of PORVs could not be justified from a risk-reduction perspective given installation of the gas turbine generators. The cost associated with the installation of PORVs or other hardware did not result in further significant reduction of plant risk and was not found to be cost beneficial [Ref. 53]. The Palo Verde IPE did not identify any cost effective plant improvements that would further improve the reliability of the DHR systems.

For Calvert Cliffs, the impact of the two DHR modifications gives a total revised CDF of 2.4E-04/RY. The IPE does not give the impact of the two modifications on the DHR CDF. However, according to a conservative estimate, the revised DHR CDF would be less than 1E-04/RY.

3.2.2.2 Babcock and Wilcox Plant Response

The B&W plant group includes seven plant units represented by five IPE submittals. As part of the NRC's program to assess the regulatory effectiveness of USI A-45, the following plants were analyzed:

1. ANO-1
2. Oconee 1, 2 & 3
3. Davis Besse
4. Crystal River 3
5. Three Mile Island Nuclear Power Plant, Unit 1 (TMI-1)

The DHR evaluation for the B&W plants yields the following results [Ref. 60–64]:

- Most B&W plants have multiple sources of water as suction supplies to the EFW pumps. Crystal River has three primary sources of EFW: a dedicated EFW storage tank, the CST and the condenser hotwell. ANO-1 has three primary sources of EFW: a CST, a backup "non-Q CST" which is interconnected with the unit 2 CST, and the SW loop supply which is an alternate source of EFW suction when the CST is unavailable. Davis Besse has two CSTs. In the event the CSTs are unavailable, two low pressure switches, located upstream of each pump suction isolation valve, automatically open the respective SW supply valves thereby providing a backup suction source from the SW system.

- All B&W plants take credit for feed and bleed. Six B&W units have HPI pumps with shutoff heads above the SRV set point, and one plant (Davis Besse) has a shutoff head that does not challenge the SRV setpoint. The Davis Besse IPE submittal credits high-pressure makeup pumps for feed and bleed; as a result operator action will be necessary for cooling to succeed. For all B&W plants, with the exception of Davis-Besse, the success criterion for feed and bleed is one PORV and one HPI pump.

- None of the B&W plants take credit for condensate pumps to feed the steam generator. In the event that both main feedwater and EFW are not available, feed and bleed is initiated.

- Only three plants (Oconee, ANO-1 and Davis Besse) identify the DHR CDF in their IPE. Since all plants do not report the DHR CDF, the number of plants falling into a particular DHR risk category cannot be determined. For this purpose a query was developed using the IPE database to determine the "IPE DATABASE DHR CDF" for all the plants and categorize them into one of the three DHR risk categories. The results of this query are presented in Section 4, Table 5.

  According to the query results obtained from the IPE database four plants (Oconee 1, 2, & 3, ANO-1, Crystal River 3 and TMI-1) have a DHR CDF that falls into Category 1 (less than or equal to 3E-05/RY) of the NRC's vulnerability classification scheme. The DHR CDF for Davis-Besse falls into Category 2 (greater than 3E-05/RY but less than 3E-04/RY) of the NRC's vulnerability classification scheme. All the B&W plants have a total CDF less than 1E-04/RY that meets the staff's objective of an overall CDF of 1E-04/RY.

  For one plant, ANO-1, there is a discrepancy in the results obtained from the IPE and the IPE database. ANO-1 reports a DHR CDF of 4.7E-05/RY in its IPE. However, the "IPE DATABASE DHR CDF" for ANO-1 is 2.28E-05/RY. One important point to note is that some of the sequences in the IPE database are combined into one aggregate sequence that is simply identified as the "remainder," which were excluded from consideration. It is possible that for ANO-1 some of the DHR contributions are in the remainder and could be identified if more information were available about the "remainder."

- None of the B&W plants identify any DHR modifications or DHR vulnerabilities to resolve USI A-45 in their IPEs. The Crystal River IPE states that the sequence type, which represents the total loss of DHR function, is dominated by SBO sequences. The IPE states that there are no DHR sequences with a CDF greater than 1E-08/RY involving independent failures of system components, which are directly responsible for providing DHR, because these systems are both diverse and redundant.

  The ANO-1 IPE performed a sensitivity study to evaluate the impact of the new AC power source and found that the total CDF could be reduced by nearly 36 percent reducing the DHR contribution to roughly 3E-05/RY. However, since this modification was considered as part of the SBO rule, it is not included in the A-45 analyses.

3.2.2.3 Westinghouse 2-Loop Plant Response

The Westinghouse 2-Loop plant group includes six plant units represented by four IPE submittals. As part of the NRC's program to assess the regulatory effectiveness of USI A-45, the following plants were analyzed:

1. Ginna
2. Point Beach 1 & 2
3. Prairie Island 1 & 2
4. Kewaunee

The DHR evaluation for the Westinghouse 2-Loop plants yields the following results [Refs. 65–68]:

- All Westinghouse 2-Loop plants have multiple alternate sources of water available for the suction of AFW pumps such as the SW system, fire water system and the condenser hotwell.

  For Prairie Island, the AFW pumps can take suction from either the CST or the cooling water system. In the event the CST is not available, the pumps can be lined up to take suction from the unlimited supply of cooling water from the Mississippi River.

  Ginna has both the AFW system and the standby AFW (SAFW) system. The AFW pumps take suction from one of the two CSTs. The SW system provides a backup water supply to the AFW system. Feedwater can also be provided per plant procedures through the yard fire hydrant system, the condenser hotwells and the outside CST.

  In the event that the AFW is unavailable, the SAFW is manually actuated and aligned to provide backup feedwater. The normal water supply for the SAFW system is the SW system. The fire SW system can also be used if there is a total loss of SW by use of a fire hydrant connection located inside the SAFW building. The CST with a 10,000-gallon capacity is provided to store condensate quality water as a source of supply for periodic testing of the SAFW system. The SAFW system is only started if all other forms of feedwater are lost to the steam generators. If SAFW is required, the system is aligned such that each pump supplies one steam generator [Ref. 65].

  For Point Beach, the AFW pumps initially receive water from the CSTs and can also use Lake Michigan water pumped directly to the suction of the AFW pumps via the SW system. The CSTs can also be refilled if necessary from the condenser hotwells, water treatment system or with Lake Michigan via the diesel driven FW pumps. In extreme situations LM water can be provided to refill the CSTs via the fire water system by fire engine pumper trucks from the "local creeks fire station" [Ref. 66].

  Kewaunee has the SW system as a backup suction supply to the SFW pumps.

- All Westinghouse 2-Loop plants take credit for feed and bleed. The success criterion for feed and bleed varies among the plants in this group. Only Ginna requires two of two PORVs and one HPI pump for successful feed and bleed. All other plants in this group require one PORV and one HPI pump.

- Only two Westinghouse 2-Loop plants (Point Beach and Ginna) take credit for steam generator depressurization and the use of condensate as heat removal (LPF) in the event that both MFW and AFW are not available.

  Feedwater addition through the condensate pumps is not credited in two IPEs: the Prairie Island IPE and the Kewaunee IPE. The Prairie Island IPE states that the majority of the failures for feedwater also fail condensate and therefore, this method of feedwater does not significantly reduce the potential for loss of secondary cooling.

- Only two plants, Ginna and Point Beach, identify the DHR CDF in their IPE. Since all plants do not report the DHR CDF, the number of plants falling into a particular DHR risk category cannot be determined. For this purpose a query was developed using the IPE database to determine the "IPE DATABASE DHR CDF" for all the plants and categorize them into one of the three DHR risk categories. The results of this query are presented in Section 4, Table 5.

  According to the query results obtained from the IPE database only one plant, Ginna, has a DHR CDF that falls into Category 1 (less than or equal to 3E-05/RY) of the NRC's vulnerability classification scheme. The DHR CDF for the remaining plants (Point Beach, Kewaunee and Prairie Island) fall into Category 2 (greater than 3E-05/RY but less than 3E-04/RY) of the NRC's vulnerability classification scheme. All Westinghouse 2-Loop plants have a total CDF which is in agreement with the staff's objective of an overall CDF of 1E-04/RY.

  For one plant, Ginna, there is a discrepancy in the results obtained from the IPE and the IPE database. Ginna reports a DHR CDF of 3.7E-05/RY in its IPE. However, the "IPE DATABASE DHR CDF" for Ginna is 1.09E-05/RY. One important point to note is that some of the sequences in the IPE database are combined into one aggregate sequence that is simply identified as the "remainder," which were excluded from consideration. It is possible that for Ginna some of the DHR contributions are in the remainder and could be identified if more information were available about the "remainder."

- None of the Westinghouse 2-Loop plants identify any DHR vulnerabilities.

  One plant, Kewaunee, does not identify any DHR vulnerability in its USI A-45 evaluation. However, the IPE identifies a vulnerability in the plant's AFW system. The IPE analysis shows that the AFW system contributes 32 percent to the total CDF. A diversion path created as a result of the failure mode associated with the makeup valve MU3A to the condenser directs the condensate from the CST to the main condenser and, therefore, reduces the quantity available to the AFW pumps for secondary cooling. This valve fails open on loss of IA or control power. If the operator fails to isolate this line, the success of AFW in providing heat removal is adversely affected. Another vulnerability is associated with the IA system. Air compressors are subjected to frequent outages for corrective maintenance and therefore, make the system less reliable. However, the licensee does not identify any modifications to improve the reliability of the AFW system.

- Most of the modifications done by the Westinghouse 2-Loop plants effect the safety injection or AFW systems. Examples of some modifications made by plants in this group include TS changes and EOP revisions. These modifications were generally plant specific and mostly dissimilar in nature.

- The impact of the modifications on the original total and DHR CDF is given in Table B-2. All Westinghouse 2-Loop plants do not identify the impact of modifications on the original total and DHR CDF in the IPE. Only one plant, Point Beach, gives both the original and the revised total and DHR plant CDF after implementing the modifications. The original total and DHR CDF for Prairie Island 1 & 2, Ginna and Kewaunee are not listed in the IPEs. The IPE for these plants give the revised total and DHR CDF after taking credit for modifications. Therefore, the impact of the modifications on the original total and DHR CDF cannot be determined.

3.2.2.4 Westinghouse 3-Loop Plant Response

The Westinghouse 3-Loop plant group consists of 13 plant units represented by 9 IPE submittals. As part of the NRC's program to assess the regulatory effectiveness of USI A-45, the following plants were analyzed:

1. Beaver Valley 1
2. Beaver Valley 2
3. Farley 1 & 2
4. H.B. Robinson 2
5. Summer
6. Turkey Point 3 & 4
7. North Anna 1 & 2
8. Shearon Harris 1
9. Surry 1 & 2

The DHR evaluation for the Westinghouse 3-Loop plants yields the following results [Refs. 69-76]:

- Plants in this group have multiple alternate sources of water available for the suction of the AFW pumps. Most of the plants in this group, such as Farley and Summer, use the SW as an alternate source to the suction of the AFW pump. Shearon Harris uses the ESW system as a backup source of raw water to the AFW pumps. North Anna uses the firemain system in addition to SW as an alternate water source. For H.B. Robinson, alternate supplies of water to the AFW include the SW system, firewater, and the deep well discharge header. Beaver Valley Unit 1 uses the plant demineralized water storage tank (PDWST) as the primary source of water. During a LOOP transient, the operators can provide water directly from the river water system to the AFW pumps.

  Turkey Point 3 & 4 has two CSTs and during a LOOP transient, makeup to the CSTs is provided from the water treatment system. In an emergency, makeup water can be provided by the primary water storage tank via the normal makeup line or from the dimeneralized water storage tank via normally removed spool piece to the AFW suction line at each CST. The plant also has two standby steam generator feedwater (SGFW) pumps that provide standby feedwater through the MFW bypass valves. These are independent of the SGFW pumps and the AFW pumps and are used to supply SGFW during startup, shutdown, hot standby and transient conditions [Ref. 74].

- All Westinghouse 3-Loop plants take credit for feed and bleed. The success criteria for feed and bleed varies among the plants in this group. For example, Summer requires two of three PORVs and one HPI pump. North Anna, Surry and Farley require one of two PORVs and one HPI pump.

- Most of the plants in this group take credit for steam generator depressurization and the use of condensate as heat removal. Only two plants, Shearon Harris and H.B. Robinson do not credit steam generator depressurization using condensate pumps. Beaver Valley 1 credits the use of a dedicated FW pump that tends to reduce the CDF for loss of FW events.

- Only two plants (Robinson and Shearon Harris) identify the DHR CDF in their IPEs. Since all plants do not report the DHR CDF, the number of plants falling into a particular DHR risk category cannot be determined. For this purpose, a query was developed using the IPE database to determine the 'IPE DATABASE DHR CDF" for all the plants and categorize them into one of the three DHR risk categories. The results of the query are presented in Section 4, Table 5.

  According to the query results obtained from the IPE database only one plant, Farley, has a DHR CDF that falls into Category 1 (less than or equal to 3E-05/RY) of the NRC's vulnerability classification scheme. The DHR CDF for the remaining plants falls into Category 2 (greater than 3E-05/RY but less than 3E-04/RY) of the NRC's vulnerability classification scheme. In addition, Shearon Harris, North Anna and Turkey Point have total CDF less than or equal to 1E-04/RY. All other Westinghouse 3-Loop have total plant CDF above this safety objective.

  For one plant, Shearon Harris, there is a discrepancy in the results obtained from the IPE and the IPE database. Shearon Harris identifies a DHR CDF of 7.7E-06/RY in its IPE. However, the "IPE DATABASE DHR CDF" for Shearon Harris is 4.49E-05/RY. Since the IPEs do not give details on individual contributions to DHR CDF, the question as to why this discrepancy exists cannot be resolved.

- None of the Westinghouse 3-Loop plants identify any DHR vulnerability.

  One plant, Turkey Point, identified a vulnerability in its IPE due to a transient-induced LOCA (TIL) contributing 60 percent to the total plant CDF. The plant installed a SW hose connection so that it could be operated independently of the CCW system to reduce the CDF from this accident. As a result, the total plant CDF reduced from 4E-04/RY to 1E-04/RY. However, since this modification was done in response to a RCP seal LOCA, it is not considered as part of the USI A-45 program.

- Modifications done by Westinghouse 3-Loop plants effect the safety injection systems, primary and secondary side depressurization, and containment cooling systems. Some of the modifications done by plants within this group include re-establishing the reactor building IA supply to allow opening of minimum of two PORVs during feed and bleed and revising test procedures to verify that the full flow recirculation valves are closed.

Four plants, Turkey Point, Surry, Shearon Harris 1 and Farley did not make any DHR modifications.

- All Westinghouse 3-Loop plants do not identify the impact of modifications on the original total and DHR CDF in their IPE. None of the plants, give both the original and revised total and DHR plant CDF after implementing the modifications. Therefore, the impact of the modifications on the original total and DHR CDF cannot be determined.

- It is important to note that in most cases the IPE submittals only report the CDF changes based on the overall proposed modifications. All Westinghouse 3-Loop plants do not give the net CDF DHR effect from specific enhancements in their IPE and therefore, the net DHR CDF from specific enhancements cannot be documented. Only North Anna gives the impact of the modifications on the net CDF effect, not the net DHR CDF effect. Therefore, the impact of the modifications on the DHR CDF cannot be easily determined.

3.2.2.5 Westinghouse 4-Loop Plants

The Westinghouse 4-Loop plant group consists of 32 plant units represented by 20 IPE submittals. As part of the NRC's program to assess the regulatory effectiveness of USI A-45, the following plants were analyzed:

1. Braidwood 1 & 2
2. Comanche Peak 1 & 2
3. Indian Point 2
4. Salem 1 & 2
5. Vogtle 1 & 2
6. Byron 1 & 2
7. DC Cook 1 & 2
8. Indian Point 3
9. Seabrook
10. Watts Bar 1
11. Callaway
12. Diablo Canyon 1 & 2
13. McGuire 1 & 2
14. Sequoyah 1 & 2
15. Wolf Creek
16. Catawba 1 & 2
17. Millstone 3
18. South Texas 1 & 2

The DHR evaluation for the Westinghouse 4-Loop plants yields the following results [Refs. 77-94]:

- All Westinghouse 4-Loop plants except Millstone 3 have alternate sources of water supply to the AFW pumps.

- All Westinghouse 4-Loop plants take credit for feed and bleed. The ability to feed and bleed varies among the plants in this group. Six Westinghouse 4-Loop plants (Braidwood, Comanche Peak, Vogtle, Byron, Indian Point 3, Salem) require one PORV and one HPI pump for successful feed and bleed. All the remaining Westinghouse 4-Loop plants require two PORVs and one HPI pump for successful feed and bleed.

- Only 12 plants take credit for steam generator depressurization and the use of condensate (LPF) as heat removal in the event that both MFW and AFW are not available.

- Only three plants (Comanche Peak, McGuire, and Catawba) identify the DHR CDF in their IPEs. Since all plants do not report the DHR CDF, the number of plants falling into a particular DHR risk category cannot be determined. For this purpose a query was developed using the IPE database to determine the "IPE DATABASE DHR CDF" for all the plants and categorize them into one of the three DHR risk categories. The results of the query are presented in Section 4, Table 5.

  According to the query results obtained from the IPE database only thirteen plants (Braidwood, Indian Point 2, Salem, Vogtle, Byron, DC Cook, Indian Point 3, Seabrook, McGuire, Sequoyah, Catawba, Millstone 3 and South Texas) have a DHR CDF that falls into Category 1 (less than or equal to 3E-05/RY) of the NRC's vulnerability classification scheme. The DHR CDF for the remaining plants falls into Category 2 (greater than 3E-05/RY but less than 3E-04/RY) of the NRC's vulnerability classification scheme. In addition, Watts Bar and Sequoyah have total CDF, which is greater than the staff's objective of 1E-04/RY. All other Westinghouse 4-Loop plants have total plant CDF that satisfies this safety goal.

  For one plant, Comanche Peak, there is a discrepancy in the results obtained from the IPE and the IPE database. Comanche Peak identifies a DHR CDF 1.6E-05/RY in its IPE. However, the "IPE DATABASE DHR CDF" for Comanche Peak is 3.25E-05/RY. Since the IPEs do not give details on individual contributions to DHR CDF, the question as to why this discrepancy exists cannot be resolved.

- None of the Westinghouse 4-Loop plants identify any DHR vulnerability in their IPEs.

- Most of the modifications done by the Westinghouse 4-Loop plants include changes in secondary side feed and heat removal systems. These modifications were generally plant specific and mostly dissimilar in nature.

- All Westinghouse 4-Loop plants do not identify the impact of modifications on the original total and DHR CDF in their IPE. None of the plants give both the original and revised total and DHR plant CDF after implementing the modifications. Therefore, impact of the modifications on the original total and DHR CDF cannot be determined.

- It is important to note that in most cases the IPE submittals only report the CDF changes based on the overall proposed modifications. All Westinghouse 4-Loop plants do not give the net CDF DHR effect from specific enhancements in their IPE and therefore, the net DHR CDF from specific enhancements cannot be determined.

3.2.3 Evaluation of USI A-45 in Response to Generic Letter 88-20, Supplement 4

NUREG-1742 provides an assessment of the findings and plant modifications associated with USI A-45 using the following criteria.

- The licensee's IPEEE covers the USI A-45 scope and is capable of identifying plant vulnerabilities related to this issue.
- The licensee's assessment demonstrates an in-depth knowledge of the external events aspects and plant characteristics relevant to USI A-45.
- The licensee's assessment results are reasonable for the design, location, features, and operating history of the plant.
- No potential vulnerabilities associated with USI A-45 were identified in the submittal or if potential vulnerabilities were identified, plant-specific improvements to eliminate or mitigate a potential vulnerability were implemented or planned.

A plant is considered verified for A-45 if the above criteria are met.

NUREG-1742 states, "Most licensees explicitly addressed USI A-45 in their IPEEE submittals. Those submittals that did not explicitly mention this issue implicitly addressed USI A-45 by providing adequate information on the potential loss of decay heat removal capability through the evaluation of seismic and fire events, which would ensure adequate decay heat removal under these conditions." It states that, "The NRC concludes that all plants have adequately addressed USI A-45. All plants have identified at least one method of removing decay heat for postulated fire events. While not all plants have an identified margin in excess of the needs for safe shutdown during an SSE (e.g., the reduced-scope plants) the NRC has determined that the IPEEEs have performed an adequate assessment to identify potential vulnerabilities in the decay heat removal systems consistent with the guidance in NUREG-1407, and no vulnerabilities were found."

NUREG-1742 concludes that USI A-45 was fully verified for all plants.

4 COMPARISON OF EXPECTATIONS AND OUTCOMES

The IPE database was used to determine the CDF contribution of DHR accident sequences to the total CDF based on NUREG-1289's definition of DHR. Specifically, the DHR function includes those components and systems required to maintain primary and secondary coolant inventory control and transfer heat from the RCS to an ultimate heat sink following shutdown of the reactor for normal events or abnormal transients such as the loss of main feedwater, LOOP and SBLOCAs.

Table 4, "DHR CDF for BWRs," and Table 5, "DHR CDF for PWRs," summarize the results of the DHR CDF reported in the IPE. It is important to note that some plants do not specify the DHR CDF in their IPEs. For such plants, the "IPE DHR CDF" is reported as being not available (N/A). Table 4 and Table 5 also list the DHR CDF including SBO, the non-SBO DHR CDF and the non-SBO, non-HPI DHR CDF obtained from querying the IPE database. These are labeled as "IPE DATABASE DHR CDF," "IPE DATABASE NON-SBO DHR CDF," and "IPE DATABASE NON-SBO, NON-HPI DHR CDF," respectively. The DHR risk categories for each of the plants are listed using the results obtained for the DHR CDF with SBO and the non-SBO DHR CDF based on the criteria defined in NUREG-1289. The DHR risk categories for the non-HPI, non-SBO DHR CDF are somewhat similar to the non-SBO DHR CDF. In conditions where under the non-HPI, non-SBO DHR CDF the plant (e.g., Surry 1 & 2) falls under Category 1 instead of Category 2, the entry is marked with a double asterisk (**).

For BWRs, a fix is considered related to USI A-45 if it addresses at least one of the following classes of accidents:

- Any small LOCA-initiated core damage sequence
- Any LOOP-initiated core damage sequence
- Any transient-initiated (non-ATWS) core damage sequence

The above initiators were the primary basis for determining DHR accident sequences. Those accident sequences with SBO as an attribute were removed, and the remaining DHR accident sequences were considered to contribute to "IPE DATABASE NON-SBO DHR CDF." The "IPE DATABASE NON-SBO, NON-HPI DHR CDF" was not computed for the BWRs because all DHR CDF is considered to involve loss of high-pressure makeup. The results are presented in Section 4, Table 4. These results are based on information available in the IPE database.

A fix for PWRs is considered related to USI A-45 if it addresses at least one of the following classes of accidents:

- Small LOCA-initiated sequences with loss of AFW or HPI capability
- SGTR-initiated sequences with loss of AFW or HPI capability
- LOOP-initiated sequences with loss of AFW or HPI capability
- Transient-initiated (non-ATWS) sequences with loss of AFW or HPI capability

Table 4 DHR CDF for BWRs

Plant Type	Name	Total IPE Database CDF*	IPE DHR CDF*	IPE Database DHR CDF*	IPE Database NON-SBO DHR CDF*	DHR Risk Category for IPE Database DHR CDF	DHR Risk Category for IPE Database Non-SBO DHR CDF
BWR Class 1/2/3	Millstone 1	1.10E-05	N/A	8.74E-06	2.20E-06	1	1
	Nine Mile Point 1	5.50E-06	3.5E-07	3.80E-06	6.43E-07	1	1
	Dresden 2 & 3	1.85E-05	1.4E-05	1.36E-05	1.29E-05	1	1
	Oyster Creek	3.69E-06	1.46E-07	2.70E-06	9.10E-07	1	1
BWR Class 3/4	Quad Cities 1 & 2	1.20E-06	3.6E-07	8.04E-07	2.33E-07	1	1
	Pilgrim 1	5.80E-05	8.8E-07	5.12E-05	5.12E-05	1	1
	Peach Bottom 2 & 3	5.53E-06	3E-06	3.13E-06	2.68E-06	1	1
	Brunswick 1 & 2	2.70E-05	6.75E-06	2.64E-05	8.33E-06	1	1
	Fermi 2	5.60E-06	3.6E-06	1.23E-06	1.14E-06	1	1
	Browns Ferry 2	4.80E-05	3E-05	2.75E-05	1.61E-05	1	1
	Vermont Yankee	4.30E-06	1E-07	2.82E-06	2.19E-06	1	1
	Hatch 1	2.23E-05	4.6E-06	1.08E-05	8.81E-06	1	1
	Hatch 2	2.36E-05	4.6E-06	1.15E-05	9.61E-06	1	1
	Monticello	2.60E-05	1.7E-05	2.17E-05	9.86E-06	1	1
	Duane Arnold	7.84E-06	5.9E-06	4.96E-06	3.21E-06	1	1
	Cooper	7.97E-05	7.74E-06	6.83E-05	4.12E-05	1	1
	Limerick 1 & 2	4.30E-06	6.9E-07	2.66E-06	2.60E-06	1	1
	Hope Creek	1.29E-05	5.45E-07	4.27E-05	7.88E-06	1	1
	Susquehanna 1 & 2	1.10E-07	N/A	N/A	N/A		
	Fitzpatrick	1.92E-06	N/A	1.85E-06	2.67E-07	1	1
BWR Class 5/6	Nine Mile Point 2	3.10E-05	9.1E-06	2.23E-05	1.79E-05	1	1
	LaSalle 1 & 2	4.74E-05	N/A	3.43E-05	7.38E-06	1	1
	WNP-2	1.75E-05	1.4E-06	1.66E-05	5.83E-06	1	1
	Clinton	2.60E-05	5.2E-06	2.65E-05	1.67E-05	1	1
	River Bend	1.55E-05	1.55E-05	1.60E-05	2.23E-06	1	1
	Perry 1	1.32E-05	5.1E-06	7.76E-06	4.95E-06	1	1
	Grand Gulf 1	1.72E-05	N/A	1.62E-05	8.86E-06	1	1

* CDF is calculated in terms of a per year basis.

Table 5 DHR CDF for PWRs

Plant Type	Name	Total IPE Database CDF*	IPE DHR CDF*	IPE Database DHR CDF*	IPE Database Non-SBO DHR CDF*	IPE Database Non-SBO, Non-HPI DHR CDF*	DHR Risk Category For IPE Database DHR CDF	DHR Risk Category for IPE Database Non-SBO DHR CDF
CE	San Onofre 2 & 3	3.00E-05	2.3E-05	1.52E-05	1.32E-05	1.15E-05	1	1
	Millstone 2	3.42E-05	N/A	1.54E-05	1.54E-05	1.53E-05	1	1
	St. Lucie 1	2.30E-05	<2E-05	6.32E-06	4.02E-06	8.18E-07	1	1
	St. Lucie 2	2.60E-05	<2E-05	5.95E-06	3.94E-06	1.18E-06	1	1
	Palisades	5.07E-05	3E-05	3.77E-05	2.57E-05	1.73E-05	2	1
	Fort Calhoun 1	1.36E-05	N/A	4.33E-06	4.33E-06	1.44E-06	1	1
	Palo Verde 1, 2 & 3	9.00E-05	5.6E-05	8.02E-05	6.12E-05	5.71E-05	2	2
	ANO-2	3.40E-05	3E-05	2.37E-05	2.31E-05	2.26E-05	1	1
	Waterford 3	1.70E-05	1.4E-05	7.69E-06	3.62E-06	4.84E-07	1	1
	Calvert Cliffs 1 & 2	2.40E-04	1.4E-04	2.61E-05	2.50E-05	1.11E-05	1	1
Babcock & Wilcox	Oconee 1, 2 & 3	2.30E-05	9E-06	8.35E-06	5.78E-06	5.46E-06	1	1
	ANO-1	4.67E-05	4.7E-05	2.28E-05	1.04E-05	4.00E-06	1	1
	Crystal River 3	1.53E-05	N/A	3.86E-06	6.48E-07	4.02E-07	1	1
	TMI-1	4.20E-05	N/A	1.54E-05	1.42E-05	2.55E-06	1	1
	Davis-Besse	6.60E-05	N/A	3.88E-05	3.88E-05	3.88E-05	2	2
Westinghouse 2-Loop	Ginna	8.74E-05	3.7E-05	1.09E-05	1.09E-05	5.82E-06	1	1
	Point Beach 1 & 2	1.15E-04	4.6E-05	6.30E-05	4.80E-05	3.67E-05	2	2
	Kewaunee	6.65E-05	N/A	3.76E-05	1.15E-05	9.72E-06	2	1
	Prairie Island 1 & 2	5.00E-05	N/A	3.19E-05	2.88E-05	2.06E-05	2	1
Westinghouse 3-Loop	Beaver Valley 1	2.14E-04	N/A	1.09E-04	8.74E-05	-	2	2
	Beaver Valley 2	1.92E-04	N/A	1.15E-04	7.27E-05	1.24E-05	2	2**
	H.B. Robinson 2	3.20E-04	9.7E-05	2.14E-04	1.88E-04	1.68E-04	2	2
	Turkey Point 3 & 4	4.62E-04	N/A	8.30E-05	6.21E-05	5.21E-05	2	2
	Shearon Harris 1	7.00E-05	7.7E-06	4.49E-05	2.69E-05	1.33E-05	2	1
	Farley 1 & 2	1.30E-04	N/A	2.14E-05	1.78E-05	1.78E-05	1	1
	Summer	2.00E-04	N/A	1.27E-04	7.87E-05	3.62E-06	2	2**
	North Anna 1 & 2	7.16E-05	N/A	3.15E-05	1.06E-05	4.24E-06	2	1
	Surry 1 & 2	7.40E-05	N/A	2.46E-04	8.03E-05	5.13E-06	2	2**

Table 5 DHR CDF for PWRs (Cont.)

Plant Type	Name	Total IPE Database CDF*	IPE DHR CDF*	IPE Database DHR CDF*	IPE Database Non-SBO DHR CDF*	IPE Database Non-SBO, Non-HPI DHR CDF*	DHR Risk Category For IPE Database DHR CDF	DHR Risk Category for IPE Database Non-SBO DHR CDF
Westinghouse 4-Loop	Braidwood 1 & 2	2.74E-05	N/A	2.50E-05	1.93E-05	1.91E-05	1	1
	Comanche Peak 1 & 2	5.72E-05	1.6E-05	3.25E-05	1.71E-05	1.47E-05	2	1
	Indian Point 2	3.13E-05	N/A	1.34E-05	1.25E-05	1.03E-05	1	1
	Salem 1	6.25E-05	N/A	3.00E-05	8.85E-06	8.32E-06	1	1
	Salem 2	6.35E-05	N/A	2.61E-05	9.66E-06	9.46E-06	1	1
	Vogtle 1 & 2	4.90E-05	N/A	2.97E-05	8.27E-06	5.97E-06	1	1
	Byron 1 & 2		N/A	2.76E-05	2.39E-05	2.29E-05	1	1
	DC Cook 1 & 2	6.26E-05	N/A	1.62E-05	1.52E-05	2.85E-06	1	1
	Indian Point 3	4.40E-05	N/A	2.46E-05	1.49E-05	1.45E-05	1	1
	Seabrook	6.05E-05	N/A	1.04E-05	8.68E-06	8.68E-06	1	1
	Watts Bar 1	3.30E-04	N/A	9.59E-05	7.84E-05	8.44E-06	2	2**
	Callaway	5.85E-05	N/A	4.18E-05	1.70E-05	2.08E-06	2	1
	Diablo Canyon 1 & 2	8.80E-05	N/A	3.56E-05	3.15E-05	5.89E-06	2	2**
	Mcguire 1 & 2	4.00E-05	1.6E-05	1.41E-05	9.26E-06	1.58E-06	1	1
	Sequoyah 1 & 2	1.70E-04	N/A	2.81E-05	2.64E-05	8.47E-06	1	1
	Wolf Creek	4.20E-05	N/A	3.30E-05	1.81E-05	1.10E-05	2	1
	Catawba 1 & 2	5.80E-05	7.3E-06	7.70E-07	7.70E-07	7.70E-07	1	1
	Millstone 3	5.61E-05	N/A	2.25E-05	1.84E-05	1.69E-05	1	1
	South Texas 1 & 2	4.27E-05	N/A	2.15E-05	7.45E-06	6.96E-06	1	1

* CDF is calculated in terms of a per year basis.

For PWRs, SGTR-initiated sequences are included due to the similarity of these sequences to other small LOCA-initiated sequences. With the above-mentioned initiators, accident sequences that involve loss of AFW capability (identified as a secondary side make-up accident sequence) or loss of HPI were considered to contribute to "IPE DATABASE DHR CDF." As was done for BWRs, those accident sequences with SBO as an attribute were removed from this set, and the remaining DHR accident sequences were considered to contribute to a "IPE DATABASE NON-SBO DHR CDF." Finally, those accident sequences with loss of HPI, but not loss of AFW capability were removed. The remaining non-SBO DHR accident sequences were considered to contribute to "IPE DATABASE NON-SBO, NON-HPI DHR CDF." The results are presented in Section 4, Table 5. Again, these results are based on the information available in the IPE database.

In addition, many plants have accident sequences that have frequencies below a screening threshold. In the IPE database, these sequences are combined into one aggregate sequence that is simply identified as the "remainder." For the purposes of this analysis, remainder sequences were excluded from consideration. It is possible that additional contributions to the above DHR CDF categories would be identified if more information were available about the "remainder."

To provide some comparison to the results in Table 5, the DHR evaluation results discussed in the San Onofre 2 & 3 (SONGS 2/3) IPE submittal are presented. The DHR CDF in the SONGS 2/3 IPE submittal is 2.3E-05/RY. This compares to the result in Table 5 for SONGS 2/3 of 1.52E-05/RY for all DHR accident sequences (including those due to SBO and HPI failure).

The SONGS 2/3 IPE discussion states that the staff definition of DHR in USI A-45 is an expanded version of the functional definition of DHR in a PWR. This expansion results in the inclusion of inventory makeup systems in addition to the function of DHR. Therefore, in discussions of contributors to the CDF in the SONGS 2/3 IPE, it is important to ensure that the definition of DHR being used is that of the NRC in USI A-45 rather than the definition traditionally assigned to DHR function in PRAs.

In this context, the following functional classes from the SONGS 2/3 IPE submittal were considered as part of the USI A-45 definition of DHR:

- Accident sequences involving loss of both primary and secondary coolant makeup in the injection phase
- Accident sequences involving loss of both primary and secondary coolant makeup in the recirculation phase
- Accident sequences involving loss of both primary and secondary coolant makeup due to station blackout
- Accident sequences involving an induced small LOCA with loss of primary coolant makeup or loss of adequate heat removal in the injection phase
- Accident sequences involving an induced small LOCA with loss of primary coolant makeup or loss of adequate heat removal in the recirculation phase

- Accident sequences initiated by a small LOCA with loss of primary coolant makeup or loss of adequate heat removal in the injection phase

- Accident sequences initiated by a small LOCA with loss of primary coolant makeup or loss of adequate heat removal in the recirculation phase

- Accident sequences initiated by a medium or large LOCA with loss of primary coolant makeup or loss of adequate heat removal in the recirculation phase

From the SONGS 2/3 IPE discussion, it appears that the most significant contributor to the difference between the SONGS IPE result and the results presented in Table 5 is the inclusion of the medium and large LOCA initiators. According to the SONGS 2/3 IPE, the inclusion of medium and large LOCA initiators gives a CDF contribution of 4.6E-06/RY. The remaining difference between the Table 5 result and the result in the SONGS 2/3 IPE submittal appears to be due to differences in accounting for accident sequences associated with the loss of primary coolant makeup during the recirculation phase.

Another important point to note is the discrepancy that exists in the DHR CDF obtained for six plants (Calvert Cliffs, Palisades, ANO-1, Ginna, Shearon Harris and Comanche Peak). For Calvert Cliffs, ANO-1 and Ginna, the DHR CDF obtained from the IPE database ("IPE DATABASE DHR CDF") is much lower than the DHR CDF reported in the IPE ("IPE DHR CDF"). The "IPE DATABASE DHR CDF" puts the plants into Category 1 instead of Category 2. As mentioned earlier, in the IPE database many plants have sequences that are combined into one aggregate sequence that is simply identified as the "remainder." Since the remainder sequences were excluded from consideration, it is possible that some of the DHR contributions are in the remainder and could be identified if more information were available.

On the other hand, for the remaining plants, the "IPE DATABASE DHR CDF" puts the plants into Category 2 instead of Category 1. Since, all plants used a different definition of DHR, it is not possible to resolve this discrepancy at this time.

Due to the above-mentioned discrepancies, to assess the effectiveness of USI A-45, the IPE database was used to determine the DHR CDF for all of the plants. The contribution of DHR accident sequences to the total CDF was based on NUREG-1289's definition of DHR.

4.1 USI A-45 Expectations and Outcomes

Table 6, "Summary of USI A-45 Expectations and Outcomes," summarizes the USI A-45 expectations derived from NUREG-1289 and reports the results obtained from the IPE database in the areas of Total CDF, DHR risk categories and DHR vulnerability.

In NUREG-1289, the staff concluded that little, if any, cost-beneficial modifications would be warranted if the DHR CDF for the plants is less than or equal to 3E-05/RY. Also, Alternative 2 "limited scope PRA" selected for the evaluation of USI A-45 asked the licensees to include LOOP transients in their PRAs. The alternative defined in NUREG-1289 did not explicitly state that LOOP transients (non-SBO) should be excluded from the A-45 analyses. This is why all plants in their DHR evaluation also included the CDF due to SBO failures.

Table 6 Summary of USI A-45 Expectations and Outcomes

USI A-45 EXPECTATIONS			ACTUAL OUTCOMES		OBSERVATIONS
AREA	**EXPECTED RESULTS**		**BWRs***	**PWRs***	
TOTAL CDF	**<1E-04**		Average: 1.98E-05 BWR 1/2/3: 5.72E-05 BWR 3/4: 1.90E-05 BWR 5/6: 2.69E-05	Average: 8.99E-05 W2-Loop: 6.81E-05 W3-Loop: 1.34E-04 W4-Loop: 6.59E-05 CE: 6.60E-05 B&W: 3.19E-05	Expectations mostly met. Ten PWRs have total CDF greater than 1E-04. All BWRs have total CDF less than or equal to 1E-04.
DHR RISK CATEGORIES C1: Less than 3.0E-05 C2: Less than 3.0E-04 but greater than 3.0E-05 C3: Greater than 3.0E-04	C1: Less than or equal to 3.0E-05	**IPE DATABASE DHR CDF**	Average BWR: 1.72E-05 BWR 1/2/3: 8.49E-06 BWR 3/4: 1.76E-05 BWR 5/6: 2.17E-05	Average PWR: 3.61E-05 W2-Loop: 3.97E-05 W3-Loop: 7.39E-05 W4-Loop: 2.62E-05 CE: 3.03E-05 B&W: 1.51E-05	Average PWR DHR CDF well above C1 and dominated by W2Loop and W3Loop plants. Out of 47 PWRs, 19 fall into DHR risk Category 2. All BWRs fall into DHR risk Category 1. None of the plants fall into DHR risk Category 3.
		IPE DATABASE NON-SBO DHR CDF	Average BWR: 8.44E-06 BWR 1/2/3: 5.91E-06 BWR 3/4: 8.99E-06 BWR 5/6: 6.97E-05	Average PWR: 2.55E-05 W2-Loop: 2.93E-05 W3-Loop: 5.06E-05 W4-Loop: 1.76E-05 CE: 2.43E-05 B&W: 1.16E-05	Expectations mostly met. Out of 47 PWRs, 11 fall into DHR risk Category 2. All BWRs fall into DHR risk Category 1. None of the plants fall into DHR risk Category 3.
		IPE DATABASE NON-SBO, NON-HPI DHR CDF	Not Computed for BWRs	Average PWR: 1.61E-05 W2-Loop: 2.17E-05 W3-Loop: 2.97E-05 W4-Loop: 9.24E-06 CE: 1.97E-05 B&W: 8.88E-06	Expectations exceeded. Out of 47 PWRs, 5 fall into DHR risk Category 2. None of the plants fall into DHR risk Category 3.
DHR VULNERABILITY IDENTIFICATION	Not Indicated GL 88-20 did not identify "what constitutes a vulnerability".		None of the BWRs analyzed identified any DHR vulnerabilities.	Only one plant, Calvert Cliffs, identified two DHR vulnerabilities and made modifications. Kewaunee identified vulnerability in its AFW system.	Even after making the modifications, the total plant CDF remains well above 1E-04. Kewaunee did not make any DHR improvements.

* CDF is calculated in terms of a per year basis.

On the basis of the DHR CDF with SBO for all plants in Table 4 and Table 5, 19 plants have DHR CDF greater than 3E-05/RY and fall into Category 2 of the NRC's vulnerability classification scheme. There are some plants, for example, Palisades, Kewaunee, Prairie Island 1 & 2, Shearon Harris 1, North Anna 1 & 2, Comanche Peak 1 & 2, Callaway, Wolf Creek, that fall into the Category 2 of the NRC's classification scheme. However, if SBO is excluded from the DHR CDF all these plants fall in the Category 1 of the NRC's classification scheme. Therefore, if LOOP including SBO are excluded from the A-45 analyses the total number of plants that fall into Category 2 are reduced to 11. Finally, removing accident sequences with loss of HPI, but not loss of AFW capability reduces the number of plants that fall into Category 2 to 5.

The average DHR CDF for BWRs is well below the Category 1 of the NRC's classification scheme with the inclusion of SBO. Therefore, removing SBO from the accident sequences results in a much lower average. On the other hand, the average DHR CDF for PWRs varies depending on whether the DHR CDF is computed including SBO or without SBO. The average DHR CDF with SBO for the PWRs is 3.61E-05/RY, which is well above the NRC's vulnerability classification scheme and is dominated by Westinghouse 2-Loop and Westinghouse 3-Loop plants. However, if SBO is excluded from the A-45 analyses, the average CDF is reduced to 2.55E-05/RY. Finally, removing accident sequences with loss of HPI, but not loss of AFW capability results in an average DHR CDF of 1.61E-05/RY.

Unlike the SBO rule, which stated that the implementation of the SBO rule would result in an industry risk reduction of 2.6E-05/RY, the expectations resulting from USI A-45 were not explicitly defined. The expectations listed in Table 6 are based on the QDOs in NUREG-1289.

4.2 Sample Assessments of Averted Health Risk from Changes in CDF Due to A-45 Initiatives

4.2.1 General Discussion

There are a number of possible ways of addressing averted health risks resulting from changes to structures, systems, and components (SSCs) and/or procedures at a nuclear power plant. We will consider two of them here in a general sense and then consider the application to changes in CDF due to A-45 initiatives. Both approaches start with an assumed understanding of the changes in CDF as a function of SSCs and/or procedures.

Only changes in health risk that are driven by changes in CDF will be considered here. The other component that impacts changes in health risk is that which is driven by changes in containment performance (e.g., changes in containment isolation frequency) with no associated change in CDF.

4.2.1.1 "Changes in LERF" as a Metric

Large Early Release Frequency (LERF) is used together with CDF in risk-informed regulation as a measure of health risk to the public. Applications of Regulatory Guide 1.174 (and its derivative regulatory guides) are examples. Important attributes of LERF are:

- LERF correlates with early health effects

- LERF is easier to assess than a full Level 3 PRA

- LERF should be used with its companion metric, namely CDF. In an approximate way, CDF acts as a surrogate for many consequence types that are not addressed by LERF, e.g., late containment failure

- The simplified models generally used for LERF, such as the one used below, are approximate even for LERF purposes

In NUREG-1765, "Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP)," dated December 2002 [Ref. 95], a methodology is presented for determining the LERF-importance (and thereby averted health risk importance) of core damage sequences as a function of reactor and containment type. The study has determined weight factors "F," defined by the relationship:

$$\Delta LERF = \text{"F"} x(\Delta CDF \text{ affecting LERF sequences})$$

"F" is a function of sequence type and containment type. Thus, if a ΔCDF is known for a given sequence, a change in LERF can be determined and thereby a measure can also be determined of the averted risk importance of the thing that caused the ΔCDF. Many of the "Fs" are zero, that is, for many sequences, a change in CDF results in very little change in LERF.

For sequences of interest to A-45, that is, transients and small break LOCAs (excluding SBO sequences), the following statement can be made:

> If ΔLERF is used as a risk metric for PWRs, according to the referenced NUREG/CR, no reductions in CDF resulting from A-45 initiatives will result in reducing early-fatality risk to the public.

However, if SGTR core damage sequences are included in the family of sequences of interest to A-45, the following statement can also be made:

> If ΔLERF is used as a risk metric for PWRs, according to the referenced NUREG/CR, reductions in CDF resulting from A-45 SGTR initiatives will result in proportionately reducing early-fatality risk to the public, i.e., ΔCDF is equivalent to ΔLERF; the value for "F" is 1.0.

If ΔLERF is used as a risk metric for BWRs, the situation is quite different. For BWRs for sequences of interest, the values for "F" are as follows:

Sequences	Containment Type	"F" Factor	Comments
All transients & SBLOCAs involving high RCS pressure	BWR Mark I	1.0	ΔCDF is equivalent to ΔLERF
	BWR Mark II	0.3	
	BWR Mark III	0.2	
All transients involving low RCS pressure & dry drywell floor at vessel breach	BWR Mark I	0.5	
	BWR Mark II	0.0	No LERF impact
	BWR Mark III	0.0	No LERF impact

Thus, for A-45 applications, a rough LERF measure of averted risk for BWRs can be obtained once the ΔCDF and the sequence bin (all high RCS pressure sequences or all low RCS pressure sequences with a dry drywell floor) is known.

Clearly, by using LERF as a measure, plants with Mark I containments can claim the greatest averted early-fatality health risk for a given A-45 ΔCDF.

4.2.1.2 "Changes in Person-Rem" as a Metric

In the Value-Impact part of Regulatory Analysis (Note NUREG/BR-0058 and NUREG/BR-0184), averted health risk is assessed by calculating the change in person-rem/year at 50 miles as a function of the change in CDF. (Often, the person-rem/year value is converted into dollars for comparison to the impacts (or costs) associated with a regulatory action.)

This metric, as compared to ΔLERF, includes consideration of late and early releases from the containment. Even more importantly, it considers latent health effects, as well as early health effects, in contrast to the LERF metric, which considers only early health effects.

The results using this metric can be quite different relative to using the ΔLERF metric. For example plant-specific analyses have shown relatively large averted risk values for "transient" sequences in PWRs while, as noted above, the averted risk would be zero using the ΔLERF metric. Thus, it is worth considering both metrics.

The difficulty with applying the person-rem measure, as contrasted to applying the ΔLERF measure, is that it is more difficult to quantify. One must either perform a plant-specific Level-3 PRA or use existing information regarding plant damage states, containment matrices, release categories, and site characteristics.

Averted person-rem as a health benefit measure for a variety of LWRs is documented in NRC regulatory analyses and in licensee Severe Accident Mitigation Alternative (SAMA) reports, submitted as part of the environmental reviews for license renewal. The SAMA studies consider candidate hardware and procedural fixes that will reduce public health risks (as measured by person-rem at 50 miles), including DHR fixes. About ten SAMA reviews have been conducted to date by the NRC. An example of the impact of decay heat removal SAMAs for the Surry plant is presented in Table 7. The maximum health-effect risk reduction possible is 18 person-rem/year. Each fix has a health-effect benefit that is a percentage of that value. The total benefits (which include the dollar equivalent of the health-effect benefits) range from very small to about $250K.

Since the SAMAs are typically more costly than the benefits, the SAMAs are usually deemed not cost beneficial.

4.2.2 Application to USI A-45

In general, there is very little information regarding the *change in CDF* as a function of the plant modifications for DHR in response to A-45. One exception is that for Brunswick Units 1 & 2, as noted in Section 3.2.1.2. The reduction in CDF due to A-45 modifications is 5E-06/RY. Using the methodology described in NUREG-1765 [Ref. 92], the reduction in LERF due to A-45 modifications is also 5E-06/RY, if the sequences involved are "transients & SBLOCAs involving high RCS pressure." Further investigation would be needed to determine the contribution from other sequences (e.g., those transients involving low RCS pressure & dry drywell floor at vessel breach) and the overall impact on LERF.

Another exception is that for Palo Verde Units 1, 2, & 3, as noted in Section 3.2.2.1. The reduction in CDF due to A-45 modifications (PORVs + Gas Turbine Generators + Procedural Change) is 5E-05/RY. Using the methodology described in NUREG-1765 [Ref. 95], the reduction in LERF due to A-45 modifications would be very small unless some of these modifications reduce the frequency of SGTRs. (It should be noted that PORVs have not been backfitted into the Palo Verde units since this modification was deemed not cost beneficial.)

Since neither the Brunswick licensee nor the Palo Verde licensee has submitted a license-renewal SAMA analysis, a direct estimate of the reduction in person-rem (50 miles) from DHR can not be made. A "like-plant" analysis could be performed, but would be beyond the scope of this activity.

The important point to note is that the averted health risk from changes in CDF due to A-45 initiatives can vary depending on the characterization of the contributing sequences, the containment type, and the regional demographics. For the same ΔCDF, the averted risks can vary (whether using ΔLERF or averted person-rem/year) from quite small to substantial.

Table 7 Improvements Related to Decay Heat Removal Capability for Surry

Analysis Case and Applicable SAMAs (the bold acronym indicates the analysis case) (the number refers to the SAMA ID)	Analysis Assumption	Risk Reduction		Total Benefit
		CDF	Person-Rem	
Improvements Related to Decay Heat Removal Capability				
DHR 35-install a filtered containment vent to remove decay heat 36-install an unfiltered containment vent to remove decay heat	Replace event tree functional equations related to containment heat removal with an event that has an unavailability of zero.	4.9% 4.9%	5.5% 1.6%	$67K $45K
FWS 111-install accumulators for turbine driven AFW pump flow control valves 115-provide portable generators to be hooked in to the TDAFW after battery depletion	Modify event tree functional equations related to AFW in an SBO to use a basic event whose unavailability is zero.	0.1%	0.04%	$2K
FDW 122-create passive secondary side coolers	Modify event tree functional equations related to MFW or AFW to use a basic event whose unavailability is zero.	12.8%	17.2%	$245K
SGP 123-automate air bottle swap for SG PORVs	Set basic event REC-INAIR-LOCAL to zero.	0.0%	0.03%	<$1K
SLB 158-install secondary side guard pipes up to the MSIVs	Set the MSLB initiating event frequencies to zero.	0.0%	0.0%	$0
CND 124-utilize bypass around the main steam trip valves to use condenser dump after safety injection	Remove house event XHOS-NO-CND-DUMP from five fault trees and gates.	2.2%	0.01%	$17K

4.3 Effect of Feed and Bleed on Reducing Plant Core Damage Frequency

As an exercise, an additional analysis was performed to evaluate the effect feed and bleed capability had on reducing plant CDF. This analysis was performed using four PWR Standardized Plant Analysis Risk (SPAR) Revision 3i models [Ref. 96]. The four plant models analyzed were Braidwood 1 & 2, Fort Calhoun, Millstone 2, and Sequoyah 1 & 2. The effect of feed and bleed was quantified by removing the bleed capability from the success criteria in the model. Specifically, the operator action necessary to initiate bleed was changed from a probability value to a failed event. Then, the resulting CDF was compared to the model's baseline CDF. The change in CDF ranged from 2.20E-05/RY to 8.60E-05/RY for the four plant models. The specific changes in CDF for each model are shown in Table 8.

Table 8 Impact of Feed and Bleed Capability on CDF

SPAR Model	SPAR Model Baseline CDF*	SPAR Model CDF without Bleed Capability*	Delta CDF due to Bleed Capability*
Braidwood 1 & 2	6.67E-05	1.53E-04	8.60E-05
Fort Calhoun	1.83E-05	7.06E-05	5.23E-05
Millstone 2	2.78E-05	8.57E-05	5.79E-05
Sequoyah 1 & 2	3.27E-05	5.47E-05	2.20E-05

* CDF is calculated in terms of a per year basis. A year is defined as 7000 critical hours.

These results are consistent with NUREG/CR-5230 [Ref. 17], and suggest that feed and bleed capability provides a risk reduction greater than the expected risk reduction from the implementation of the SBO rule (2.6E-05/RY).

4.4 Effect of Other Regulatory Initiatives

Between March 1981, when the Commission identified "Shutdown Decay Heat Removal Requirements" as USI A-45, and the submittal of the IPE and the IPEEE summary reports, which occurred in the early and mid 1990's, nearly a decade of regulatory activity had transpired. During that period, several significant regulatory initiatives resulted in plant improvements, many directly associated with capability to remove decay heat. These initiatives include:

USI A-44	Station Blackout
USI A-46	Seismic Qualification of Equipment in Operating Plants
GI 70	Power-Operated Relief Valve and Block Valve Reliability
GI 124	Auxiliary Feedwater System Reliability
GL 89-16	Installation of a Hardened Wetwell Vent

The impact of the actions that resulted from the above initiatives is generally reflected in the IPE and IPEEE summary reports. NUREG-1560 evaluated the impact of the SBO rule on the reported IPE core damage frequencies. A total of 54 plants credited the SBO rule; 10 plant submittals (15 plants) provided information on the reduction in CDF. NUREG-1560 states that the impact of implementing the SBO rule is "a measurable reduction in total CDF (an estimated mean of 2E-5/RY), a significant portion of which is due to the reduction in SBO CDF." Improvements in the mitigation of the loss of offsite power, improved reliability of AFW, and the availability of a hardened wetwell vent generally were not identified in the IPEs as DHR improvements.

Many of the support system failures that were identified by the A-45 case studies are associated with the reliability of onsite emergency AC power. NUREG-1289 Alternative 3, Application of Specified Systems Modifications to All Plants, provides a list of modifications that could reduce the decay heat vulnerabilities identified in the case studies. A review of these modifications shows a focus on emergency AC power and battery depletion. NUREG-1289 states in its assessment of other generic issues that "USI-A44 would achieve about 40 to 70 percent of the estimated reduction in core damage frequency associated with the modifications considered in USI A-45."

NUREG/CR-5230 states that external events contributed over 50 percent of the core melt probability in each of the case studies with fire and seismic being the most important. Although the scope of the original IPE Program was limited to internal events, the resolution of USI A-46 and the plant-specific IPEEE reviews were effective in reducing the fire and seismic contributions to the loss of DHR. NUREG-1742 found that seventy percent of the plants proposed improvements as a result of their seismic IPEEE analyses and that over sixty percent identified and /or implemented improvements to reduce fire risk. Although these improvements are not explicitly identified as DHR improvements, many result in improved DHR capability.

Generic Issue 70 resulted in Generic Letter 90-06, "Resolution of Generic issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Pressure Overpressure Protection for Light-Water Reactors." This letter requested enhanced quality controls for the PORVs and block valves.

Generic Issue 124 was concerned with the high rate of AFW system failure seen in operating experience and indicated by industry studies. NUREG-0933, Revision 2, states that because of the significance of the AFW system in reducing core-melt frequency, all PWRs should meet the reliability criterion specified in SRP Section 10.4.9. This criterion was not previously applied to operating reactors. In addition, some licensees were required to upgrade their AFW systems to safety-related standards. The staff also concluded that several two-pump AFW systems required upgrading.

Generic Letter 89-16 requested licensees with Mark I Containments to provide its plans for the installation of a hardened wetwell vent. The letter states "implementation of reliable venting capability and procedures can reduce the likelihood of core melt from accident sequences involving loss of long-term decay heat removal by about a factor of 10."

4.5 Related Generic Issues

Two other generic issues credit, in part, their resolution to USI A-45: Generic Issue 70: PORV and Block Valve Reliability and Generic Issue 125: Davis-Besse Loss of all Feedwater Event of June 9, 1985 – Long Term Actions. GI-70 addresses the issue of non-safety grade PORVs used to mitigate a design basis accident. Although the resolution of GI-70 included the issuance of Generic Letter 90-06 requesting enhanced quality controls, it also deferred the need to improve PORV/block-valve reliability related to specific needs of feed and bleed to USI A-45. Two actions associated with GI-125 were also resolved by crediting USI A-45. The first issue is concerned with the qualification of the PORVs. The GI-125 Task Action Plant states "The temperature, pressure, and moisture conditions of the containment environment can create a differential thermal expansion of the value disc and body and may cause the PORV to stick, failing open or closed, or the PORV can close shortly after beginning feed-and-bleed because of short circuits." The resolution for this issue states that the use of feed and bleed and the need for environmental qualification of the PORV for DHR is included in the USI A-45 program. The second issue addresses the generic implementations of the "very limited capability of the Davis-Besse plant to remove decay heat using feed-and-bleed techniques." Possible solutions included increasing the pressure relief capacity or increasing the makeup capacity. This action was dropped under GI-125, but continued under USI A-45 as one option in the resolution process.

5 CONCLUSIONS

This evaluation included performing a regulatory effectiveness review of licensees' actions to resolve safety issues regarding USI A-45, "Shutdown Decay Heat Removal Requirements." As part of this assessment, plant modifications made at 26 BWR plants and 47 PWR plants were reviewed.

5.1 Key Findings

- Significant reduction in DHR-related risk was achieved as a result of plant changes from the implementation of regulatory initiatives such as USI A-44: Station Blackout, USI A-46: Seismic Qualification of Equipment in Operating Plants, GI 124: Auxiliary Feedwater System Reliability and GL 89-16: Installation of a Hardened Wetwell Vent, as well as other regulatory initiatives. As a result, fewer DHR-related vulnerabilities remained to be found through the plant-specific IPEs and IPEEEs. However, the plant-specific IPEs and IPEEEs were effective at verifying that the plant changes resulting from these initiatives and the improvements that were identified by these studies addressed the DHR-related risk.

- The resolution of USI A-45 through the IPE and IPEEE programs was based primarily on examination of plant behavior in sequences initiated at full power. Shutdown events were not considered in most cases.

- In general, the IPE and IPEEE submittals did not include an evaluation of change in CDF based on the DHR enhancements. The changes in CDF were only reported for the totality of the enhancements proposed by the utility to reduce severe accident frequency or ameliorate their consequences.

5.2 BWR Findings

- The CDF due to the failure of DHR systems for all BWRs fall into Category 1, CDF due to DHR < 3E-05/RY, of the NRC's vulnerability classification scheme.

- For BWRs, in most cases the DHR evaluation was restricted to the final heat sink options: RHR, PCS, or containment venting.

- No "vulnerabilities" associated with DHR function were identified at any BWRs. DHR enhancements were identified in some of the plants. Enhancements were performed to increase the DHR capabilities and reliability. For example, at one plant, a procedural enhancement for alignment of low pressure coolant injection (LPCI) or containment spray (CS) pumps to condensate storage tank (CST), when suppression pool cooling (SPC) cannot be established, provided a five-fold reduction in total CDF and reduced the contribution of long-term failure of SPC from about 69 percent to a small fraction (8.2 percent) of CDF.

- Even within a given plant type, modifications were mostly dissimilar between plants. An exception implemented across the BWR 1/2/3 and BWR 3/4 plants that have Mark I containments is the installation of the hardened wetwell vent. However, the installation of the hardened wetwell vent was the result of the issuance of GL 89-16. Enhancements related to GL 89-16 were addressed in all BWR IPEs having Mark I containments.

- Of the 17 BWR plants analyzed having Mark I containments, five plants, (Dresden, Millstone 1, Cooper, Monticello and Browns Ferry) did not take credit for the hardened wetwell vent in their IPE. Millstone's IPE analysis showed that less than 1 percent reduction in CDF would be achieved from the proposed hardened wetwell vent (less than 1 E-07/RY). However, since GL 89-16 required licensees having Mark I containments to install the hardened vent under 10 CFR 50.59, all the plants that did not install the vent during the IPEs, installed it after the completion of the IPE program.

- Two BWR 6 plants, Perry and River Bend, evaluated the possibility of installing a passive containment vent and hardened vent, respectively. The Perry IPE analysis identified that the addition of a passive containment vent path independent of AC power would reduce the CDF and also result in reduction of offsite releases. However, the vent was not installed. The River Bend IPE estimated that a hardened wetwell vent would reduce CDF by about 1 E-07/RY (less than 1 percent reduction) and, therefore, did not find this modification to be cost effective.

- Other operational and procedural changes analyzed at plants having Mark I containments were to enhance their depressurization capability. For example, Vermont Yankee enhanced the depressurization capability by allowing use of an auxiliary diesel generator to power the AC operated valves needed to accomplish injection. Nine-Mile Point Unit 1 emphasized depressurizing the RPV in time.

- Some plants having Mark III containments also emphasized the importance of RPV depressurization. For example, Clinton emphasized the importance of both manual and automatic depressurization in training and evaluated the appropriateness of making permanent changes to the training program.

- Many licensees made modifications to reduce CDF from transients to resolve USI A-45. Some of these modifications included establishing a crosstie to RHR, either from service water (SW) or firewater.

- IPE modeling differences made it more difficult to derive general insights than would be the case if more consistent methodologies had been used. For example, the Cooper study suggested that containment venting, which is required several hours into a loss of DHR event, is relatively unimportant because it affects long-term, slowly-developing accident sequences in which there is time available for the plant staff to adopt other restoration strategies. Conversely, results from the Brunswick IPE lead to a different conclusion, namely, that containment venting is more important in loss of DHR events because less reliance is placed on recovery actions.

- Another important insight in the USI A-45 analyses was that BWR plants credited: 1) alternate injection sources such as firewater and SW, and 2) containment venting, in the IPEs. Some IPEs also credited the use of the hardened wetwell vent. All BWR plants, except Browns Ferry, took credit for at least one of the two strategies. Such credit reduced the transient CDF in those IPEs.

- Important elements of the overall DHR function at BWRs were addressed through specific functional requirements adopted before completion of the IPE program, such as hardened vents, improvements to depressurization, and SBO-related enhancements. Other plant-specific improvements were made to DHR in the course of the IPE studies, but these varied considerably in character. It is unlikely that a general requirement would have had the same result, apart from modifications related to SBO and GL 89-16, because the plant-specific improvements exhibited such variety.

5.3 PWR Findings

- The CDF due to the failure of DHR of over 50 percent of all PWR units fall into Category 1, CDF due to DHR < 3E-05/RY, of the NRC's vulnerability classification scheme. Approximately 15 percent of the PWR units have a total internal-events CDF that exceeds 1E-04. These results are summarized below in Table 9.

- In most cases the DHR evaluation at PWRs was restricted to 1) heat removal from the RCS using steam generators (primary to secondary power conversion) using three different steam generator feedwater supply systems: main feedwater (MFW), emergency feedwater (EFW) and auxiliary feedwater (AFW) and 2) heat removal from the RCS using power operated relief valves (PORVs) while replacing inventory using high pressure injection (HPI) (feed and bleed with the RCS intact).

- Feed and bleed cooling was found to be an important backup for transient sequences with loss of steam generator heat removal. Most PWR plants took credit for feed and bleed in their IPE. Some Combustion Engineering (CE) plants do not have PORVs. Only three CE plants (San Onofre, Palo Verde, Waterford 3) did not credit feed and bleed in their IPE analysis. The inability to use feed and bleed cooling for these CE plants was generally offset by the ability to depressurize the steam generator and use condensate for cooling.

Table 9 DHR Vulnerability Classification (Based on the IPE Database)

Type	Category 1 CDF Due to DHR < 3E-05	Category 2 CDF Due to DHR < 3E-04, but > 3E-05	Total CDF Exceeds 1E-04
CE	6 sites (8 units)	3 sites (6 units)	1 site (2 units)
B&W	4 sites (6 units)	1 site (1 unit)	None
Westinghouse 2-Loop	1 site (1 unit)	3 sites (5 units)	None
Westinghouse 3-Loop	1 site (2 units)	7 sites (11 units)	5 sites (8 units)
Westinghouse 4-Loop	13 sites (22 units)	5 sites (7 units)	2 sites (3 units)
Total	25 sites (39 units)	19 sites (30 units)	8 sites (13 units)

- One CE plant, Arkansas Nuclear One, Unit 2 (ANO-2), took credit for feed and bleed without any PORVs. At this plant, once-through cooling can be achieved by opening the pressurizer emergency core cooling system (ECCS) vent valves or the low temperature overpressure protection (LTOP) valves and injecting coolant via the high pressure safety injection (HPSI) pumps. The ANO-2 IPE analysis also showed that without credit for feed and bleed cooling, the CDF would increase by 536 percent (from 3.4E-05/RY to about 2.1 E-04/RY).

- Most PWR plants, except the Babcock & Wilcox (B&W) plants, credited steam generator depressurization and use of condensate as a successful means of heat removal.

- Only one PWR plant, Calvert Cliffs, explicitly stated that it had DHR vulnerabilities. DHR modifications were identified in some of the other PWR plants and modifications were made to increase the DHR capabilities and reliability.

- Calvert Cliffs 1 & 2 identified two "DHR vulnerabilities." The IPE reported a DHR CDF of 1.4E-04/RY. The first vulnerability identified in the IPE is the failure of AFW with a 34 percent contribution to the DHR CDF. This was influenced by failures in hand operated valves needed for long term supply of water for AFW after the CST is depleted. The second vulnerability identified in the IPE involved failures associated with inadvertent actuation of engineered safety features actuation system (ESFAS) and auxiliary feedwater actuation system (AFAS) with an 18 percent contribution to the DHR CDF. The licensee made two modifications to improve DHR reliability. One was to provide surveillance on AFW condensate-related manual valves to ensure a supply of water for AFW if the CST is lost or is depleted, and the second was to train all operating crews on the consequences and recovery of inadvertent ESFAS/RPS/AFAS actuation.

- One Westinghouse-2 Loop plant, Kewaunee, identified a vulnerability in its AFW system. The IPE analysis showed that the AFW system contributed 32 percent to the total CDF. Approximately 21 percent was found to be directly related to the reliability of the turbine-driven AFW pump. At the time of the IPE, evaluations were performed to improve the reliability of the turbine-driven AFW pump. However, no modifications were made to improve the reliability of the AFW system.

- Modifications were mostly dissimilar between plants within given PWR groups. Most modifications were done to either improve the LPI systems, increase reliability of the AFW systems, or increase feed and bleed capability. The B&W plant group did not make any DHR-related modifications.

- IPE modeling differences made it more difficult to derive general insights than would be the case if more consistent methodologies had been used.

- All plants took credit for at least one of two alternate DHR strategies: 1) feed and bleed and 2) steam generator depressurization using condensate pumps.

- For PWRs, expectations for DHR-related contributions to CDF were met, based to some extent on credit for feed and bleed and steam generator depressurization using condensate pumps (LPF).

5.4 Overall Conclusions

From the above discussion, it appears that significant reduction in DHR-related risk was achieved at BWRs as a result of generic requirements (USI A-44, USI A-46, and GL 89-16) imposed before the IPEs and IPEEEs were completed. Partly as a result of these earlier requirements, no DHR "vulnerabilities" were identified at BWRs in the IPE or IPEEE programs, although many minor modifications were made.

For PWRs, significant reduction in DHR-related risk was also achieved as a result of generic requirements (USI A-44, USI A-46, GI 70 and GI 124). Under the IPE and IPEEE programs, licensees took credit for alternate DHR strategies that were not required by regulation. Analyses supporting these alternate strategies were not reviewed and approved by the NRC. Many licensees took credit for the strategies of feed and bleed, and steam generator depressurization using condensate, for heat removal. No licensees credited feed and bleed, or enhancements to current feed and bleed capability, to fix a "vulnerability." Note that when the IPE program was initiated, crediting the strategies of feed and bleed, and steam generator depressurization, in PRAs had become commonplace. The DHR-related CDFs would have been higher and the "vulnerability" discussions might have been different if licensees had not taken credit for these two strategies.

The USI A-45 resolution differed from the SBO and ATWS resolutions in several ways. SBO and ATWS were addressed through definitive requirements, in that specific capability needs were stated and required by regulation. The resolution of USI A-45 was through generic communications which did not have or establish definitive requirements. The generic communications requested that an IPE and IPEEE be performed to identify and address any DHR "vulnerabilities." A definition of DHR "vulnerability" was not provided. Performance criteria analogous to those for ATWS and SBO were not imposed.

The USI A-45 program expectation regarding DHR-related contribution to CDF was generally met without the imposition of generic hardware fixes expressly for USI A-45. In addition, it is likely that licensees' awareness of the significance of DHR strategies was fostered through the conduct of the IPEs and IPEEEs. Therefore, the USI A-45 resolution approach appears effective in achieving its goals.

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11. ABSTRACT *(200 words or less)*

As part of the U.S. Nuclear Regulatory Commission's program to assess regulatory effectiveness, the Office of Nuclear Regulatory Research has examined regulations such as the station blackout rule, and anticipated transient without scram rule. As part of this program, the Office of Nuclear Regulatory Research is also reviewing the effectiveness of generic safety issue resolution. One such issue currently being reviewed is Unresolved Safety Issue (USI)A-45 to determine if the requirements are achieving the desired outcomes. It is anticipated that the results of these reviews can be used to improve the effectiveness of NRC requirements and guidance, staff inspection guidance, and oversight decision processes for NRC licensee performance. This report evaluates the effectiveness of the USI A-45 resolution by comparing USI A-45 expectations to outcomes. A set of baseline expectations was established from NUREG-1289, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," November 1988, and NUREG/CR-5230, "Shutdown Decay Heat Removal Analysis: Plant Case Studies and Special Issues," April 1989, and the actual outcomes were obtained from the individual plant examinations and individual plant examination of external events in the areas of total core damage frequency, decay heat removal risk categories and decay heat removal vulnerability. The report concludes that the USI A-45 program expectation regarding decay heat removal-related contribution to core damage frequency was generally met without the imposition of generic hardware fixes expressly for USI A-45. In addition, it is likely that licensees' awareness of the significance of decay heat removal strategies was fostered through the conduct of the individual plant examinations and individual plant examination of external events. Therefore, the USI A-45 resolution approach appears effective in achieving its goals.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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