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Behavior of PWR Reactor Coolant System Components, Other than Steam Generator Tubes, under Severe Accident Conditions - Phase I Final Report

Argonne National Laboratory

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by

Saurin Majumdar and William Shack

Abstract

A critical step in the assessment of risk of containment bypass attributable to pressure- and temperature-induced failures of steam generator (SG) tubes during severe accidents is the prediction of the sequence in which the SG tubes fail relative to other reactor coolant system (RCS) components. This report summarizes our current understanding of the behavior of coolant system components other than SG tubes during severe accidents in pressurized water reactors (PWRs). A detailed analysis of RCS components during severe accidents reported in NUREG-1570 predicted that the failure times of steam generator tubes, the pressurizer surge line, and hot leg piping are very close. However, the analyses conducted for predicting failure of RCS components were less rigorous and detailed than those for SG tubes. This report reviews the methods used in NUREG-1570 to analyze the behavior of RCS piping, power-operated relief valves, safety relief valves, and manway bolted connections during severe accidents and recommends future research and analyses that should be conducted to bring the failure prediction methodology on a par with that followed for SG tubes. This will make possible a more balanced assessment of the potential for containment bypass attributable to SG tube failures.

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Executive Summary

This report summarizes our current understanding of the behavior of coolant system components other than steam generator (SG) tubes during severe accidents in Pressurized Water Reactors (PWRs). The risk of containment bypass during such accidents will be significantly reduced if it can be demonstrated that the primary system is depressurized or that the primary-side pressure is significantly reduced prior to SG tube rupture. Such mitigating events may be caused by a breach in the passive components due to rupture of the hot leg piping, pressurizer surge line, SG manways, etc., or a failure of one or more of the active components such as the reactor coolant system (RCS) coolant pump, power-operated relief valves (PORVs) and pressurizer safety valves (PSVs). Of the two, the behavior of RCS coolant pump, PORVs and PSVs during severe accidents is more difficult to predict because these components are manufactured from complicated parts fitted together with tight tolerances and experimental data relevant to their behavior under severe accident conditions are not available.

A detailed analysis of RCS components during severe accidents reported in NUREG-1570 predicted that the failure times of SG tubes, the pressurizer surge line, and hot leg piping are very close. However, the analyses conducted for predicting failure of RCS components were less rigorous and detailed than those for SG tubes. This report reviews the methods used in NUREG-1570 to analyze the behavior of RCS piping, PORVs, PSVs, and manway bolted connections during severe accidents and recommends future research and analyses that should be conducted to bring the failure prediction methodology on a par with that followed for SG tubes.

To evaluate our current understanding of how the various active components (PORV, PSV, gasketed joints) of a PWR RCS might behave during severe accidents and what research should be done to help answer some of these questions, a workshop was held at Argonne National Laboratory (ANL). Representatives from valve and gasket manufacturers, industry experts, Electric Power Research Institute (EPRI), (U.S.) Nuclear Regulatory Commission (NRC) and ANL participated in the workshop. In general, the participants felt that, in the absence of test data, it will be difficult to analytically predict the behavior of the active components during severe accidents. Even if test data were available, it was suggested that analyses by NRC and industry, and experience have shown that relatively small changes in timing or effective flow areas of valves that are stuck open can lead to widely differing outcomes. Participants were more optimistic about analytical prediction of behavior of bolted connections during severe accidents.

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List of Acronyms

ANL	Argonne National Laboratory
ANO-2	Arkansas Nuclear One-Unit 2
ASME	American Society of Mechanical Engineers
CFD	Computational fluid dynamics
COA	Crack opening area
COD	Crack opening displacement
EPRI	Electric Power Research Institute
IPE	Individual plant examination
MTI	Materials Testing Institute
NRC	(U.S.) Nuclear Regulatory Commission
OEM	Original equipment manufacturer
PORV	Power operated relief valve
PSV	Pressurizer safety valve
PVRC	Pressure Vessels Research Council
PWR	Pressurized water reactor
RCPB	Reactor coolant pressure boundary
RCS	Reactor coolant system
SBO	Station blackout
SG	Steam generator
SGT	Steam generator tube
SV	Safety valve
WRC	Welding Research Council

List of Symbols

C constant for Larson Miller parameter

p_{lm} Larson Miller parameter

T temperature

t_R time to rupture

1 Introduction

During postulated pressurized water reactor (PWR) severe accidents, there is a concern that degraded core effluents may be allowed to bypass the containment if structural failures are experienced in the SG tubes. However, if other reactor coolant system (RCS) components (i.e., non-SG tubes) fail before the SG tubes, containment bypass may be averted. Prediction of RCS component failure will help determine the related RCS thermal hydraulic response and the relative order of the RCS failure sequences, the risk importance, and associated uncertainties.

The ultimate objective of this research is to improve our ability to predict the behavior (i.e., failure modes and locations and times-to-failure) of selected PWR RCS components (e.g., power-operated relief valves (PORVs), pressurizer safety valves (PSVs), surge lines, hot-leg piping, SG manways) under severe-accident conditions, and to compare the findings with the predicted failure modes and times-to-failure of SG tubes. To evaluate our current understanding of how the various active components (PORVs, PSVs, gasketed joints) of a PWR coolant system might behave during severe accidents and determine what research should be done to help answer some of the questions, a workshop was held at Argonne National Laboratory (ANL) with participants from valve and gasket manufacturers, industry experts, Electric Power Research Institute (EPRI), U.S. Nuclear Regulatory Commission (NRC), and ANL.

The present report reviews current methods and models used to predict the failure modes and times-to-failure of RCS components as described in NUREG-1570 and its associated supporting documents and data bases. It also discusses how the existing methods and models can be improved, including the identification of any needs to develop additional data bases (e.g., materials testing, etc.), and estimates of the associated level of effort that would be necessary to achieve these improvements.

The report also provides recommendations for additional work that is needed to meet the ultimate objectives of this research.

2 Background

Risk assessment of the severe-accident integrity of SG tubes has been studied in the past and reported in NUREG-0844 (Sept, 1988), NUREG/CR-4551 (Dec. 1990) and NUREG-1477 (June, 1993). The depth of the analyses of the thermal hydraulics of the primary/secondary system and rupture of tubes was later improved in NUREG-1570 (March, 1998), which also included some discussions of the behavior of PORVs, PSVs, manways, and bolted joints during severe accidents.

In the past few years, the NRC has been investigating the behavior of PWR SG tubes and other RCS components under severe-accident conditions. This research is documented in NUREG-1570, NUREG/CR-6575, NUREG/CR-6521, and NUREG/CR-6664. It has been concluded that the times-to-failure of the SG tubes and other critical RCS components are relatively close to each other; however, the evaluation of the SG tubes has been more rigorous and much better validated than the evaluation of other RCS components. Two analytical models were developed for predicting the failure behavior of SG tubes during severe accidents: one, based on flow stress theory, the other, based on creep rupture theory. The flow stress model was an extension of a widely used low-temperature failure model based on flow stress of the material in which the flow stress (average of yield and ultimate tensile strengths) was assumed to be a function of temperature. Because of the assumptions of the model, the failure pressure and temperature predicted by the flow stress model are independent of the ramp rate. The creep rupture model was based on a well-known linear damage rule, which uses the high temperature creep rupture curves as the basis for predicting failure. Because creep is a time-dependent phenomenon, the failure pressure and temperature predicted by the creep rupture model is dependent on the ramp rate. Tests were conducted on flawed and, unflawed Alloy 600 tubes subjected to simulated severe-accident transients. The tests showed that the predictions of the creep rupture model were more in accord with experimental results than those of the flow stress model. Although the prediction of rupture of hot-leg and surge line piping in NUREG-1570 was also based on a creep rupture model, the stress analysis of the piping was based on highly simplified models of the piping. In view of the closeness of the predicted rupture times for SG tubes, surge lines and hot leg piping reported in NUREG-1570, improved models for predicting the failure of non-SG tube RCS components are warranted.

The failure of reactor coolant boundary components will be strongly dependent on the thermal histories of the components. In NUREG-1570, for the thermal hydraulic analyses of a model of the Surry SG, SCDAP/RELAP 5 was used. The modeling of the coolant system was not sufficiently detailed in all cases to provide accurate axial, circumferential, and throughwall-thickness variations of temperature in the piping, and detailed temperature distributions in PORVs, PSVs or the various manways were not included in the analysis. The analyses of the failure of the piping for NUREG-1570 assumed that the pipes were infinitely long, at a uniform temperature, and with a creep behavior that can be described in terms of the Larson-Miller parameter. The creep damage was calculated by ignoring stress distributions within the pipe wall and around the circumference. No consideration was given to spatial temperature variations, structural boundary conditions (e.g., pipe supports and gap restraints), junctions between pipes, bends in the pipes, and dissimilar material joints. All of the calculations in NUREG-1570 were global, and local failures that may occur because of temperature spikes were not addressed.

The basic strategy followed in NUREG-1570 for assessing risk during severe accidents is the accident progression event tree (APET), which comprises of the following steps:

- estimating the frequency of the various primary/secondary-system conditions that could challenge the SG tubes,
- characterizing the core degradation processes and resultant pressure/temperature challenges to the RCS and SG tubes for each APET branch,
- determining the overall probability that SG tube integrity will be maintained over the range of tube challenges.

This assessment provides the overall estimate of the frequency of pressure- and temperature-induced failures of SG tubes and containment bypass attributable to severe accidents. NUREG-1570 concentrated on those branches of the APET that, during a station blackout (SBO) accident, lead to so called "high-dry" sequences, in which the primary side pressure remains high while the secondary side is dry and depressurized, because the challenge to SG tubes is highest during these sequences. A critical ingredient in this assessment process is the ability to predict the behavior of the RCS components other than the SG tubes during the progression of the severe accident, because their behavior will control the pressure and temperature history to which the SG tubes will be subjected. A major uncertainty in the flow path calculation is the maintenance of the loop seal, which effectively prevents full-loop circulation of the hot gas and keeps the SG tubes and the RCS pump at a much lower temperature than if the loop seal were lost. If the loop seal is maintained, a countercurrent circulation is set up in the hot leg and a major uncertainty in the flow path calculation is in the treatment of mixing of flow within the SG inlet plenum. Sensitivity analyses were conducted to determine the impact of various assumptions on the results.

The risk of containment bypass will be significantly reduced if it can be demonstrated that the primary-side is either fully or significantly depressurized prior to SG tube rupture. Such mitigating events may be caused by a breach in the passive components due to rupture of the hot-leg piping, pressurizer surge line, SG manways, etc., or a failure of one or more of the active components such as the RCS coolant pump, PORVs, and PSVs. Of the two causes, the behavior of RCS coolant pump, PORVs, and PSVs during severe accidents is more difficult to predict because these components are manufactured from complicated parts fitted together with tight tolerances and experimental data relevant to their behavior under severe-accident conditions are not available.

Hence, although the failure of the hot-leg piping and surge line was calculated by analytical techniques in NUREG-1570, failure (i.e, failure to close or stuck-open position) of the active components was treated probabilistically. For example. the probability of a stuck-open PORV was evaluated from the product of the demand rate times the probability of failure to close per demand. In the calculations, a generic value of probability of failure obtained from field data was used. The number of challenges to the pressurizer PORVs or SVs is a function of plant characteristics, including RCS volume, set point staging of multiple valves, flow capacity of the PSVs and valve operating characteristics. From a search of a database of 42 individual plant examinations (IPEs), the sum of the frequencies of sequences with high primary-system pressure and dry secondary conditions at the time of core damage for most plants was in the range from 2 x 10^{-6}/RY to 4 x 10^{-5}/RY. SCDAP/RELAP5 calculations for Surry predicted 30 steam cycles, followed by 59 liquid cycles (two-phase flow) before core uncovery for the case of one SG depressurized and 20 steam cycles, followed by 56 liquid cycles for the case with all SGs depressurized. MAAP, which is the industry code for severe-accident analysis, also predicts similar number of cycles. Although some limited test data on the behavior of PORVs during such cycling at relatively low temperatures are available, the same is not true at higher temperatures after core uncovery, when the primary side consists of superheated steam. Also, NUREG-1570 used an approximate model for predicting failure of bolted connections to analyze the behavior of SG manways during severe accidents. To evaluate the current understanding of how the active components and bolted connections may be expected to behave during severe accidents and what additional research is needed, a workshop with participants from valve and gasket manufacturers, industry, EPRI, NRC and ANL was held at ANL.

2.1 Workshop on Behavior of PORVs, PSVs, and Bolted Connections during Severe Accidents

A workshop on PORVs, PSVs, and bolted connections was held on November 14, 2001 at ANL to discuss the potential behavior of these components during severe accidents in PWRs. The agenda for the workshop is enclosed as Appendix A. Two representatives from EPRI, three from the NRC, two from ANL, and a consultant attended the workshop. In addition, four persons, representing EPRI, Dzurik/Copes-Vulcan (PORV manufacturer), Anderson-Greenwood-Crosby (PSV manufacturer), and Flexitallic (spiral gasket manufacturer), participated in the workshop by teleconference. A list of attendees is enclosed as Appendix B. A summary of the workshop is included in Appendix C. The workshop explicitly did not consider the failure of piping components, because it was felt that the factors responsible for uncertainties in those predictions were reasonably well understood, although additional analyses and testing would be needed to reduce the uncertainties and validate the results.

Before the workshop, review materials (i.e., selected parts of NUREG-1570 and questions) were provided to attendees for their review. The workshop started with presentations of background material by Joel Page and Steve Long of the NRC; these were followed by a round-table discussion. In the morning, the discussion concentrated on the behavior of relief valves (plus some bolted connections), whereas the afternoon session was devoted exclusively to discussions on the behavior of bolted connections. The major issues are discussed below as they related to the questions provided to the attendees.

2.1.1 Anticipated Behavior of PORVs and PSVs During Severe Accidents

In response to the questions posed before the workshop about whether there are design differences between PORVs and PSVs designed for PWR RCS service versus those designed for higher temperature service, e.g., that seen in fossil plants, participants seemed to agree that, although the valve body materials may differ (e.g., stainless steel for PWR RCS PORVs and Cr-Mo-V steel for high-temperature fossil application), the design of safety valves (SVs) for low-temperature (nuclear) applications is basically the same as that for high-temperature (fossil) applications. Fossil experience with relief valves should be useful for predicting severe-accident behavior of PWR SG relief valves, because clearances in fossil valves and PORVs are quite similar. Some PORVs, with actuator (spring return), diaphragm, and solenoid valve, are made of 316 stainless steel. The cage is in the valve body and both the cage (17-4 PH steel) and cage spacer (316 SS) are captured by the bonnet. Fossil plant valves operate at 593°C (1100°F).

In response to questions about the behavior of valves during cyclic operation, the participants from the valve manufacturers and a representative from EPRI agreed that sustained flow of water at the expected subcooling during discharge is not expected to damage the PORVs and PSVs. Water passage while the valve is opening will help push the valve disk open, but the representatives from valve manufacturers did not foresee much damage from that. However, some of the industry experts and NRC representatives were not fully convinced of this, because earlier EPRI experiments on PSVs indicated chattering during two-phase flow. During the experiments, tests were halted when chattering occurred but no significant damage was detected in the valves. Also, if the valve is open, the very high-temperature discharge could cause damage to the actuator. Leakage can be expected from high-nickel alloy on stainless steel seating surfaces, but less from Stellite on Stellite seating surfaces. There was a feeling that high-nickel alloys should be able to tolerate the cyclic operation during severe accidents better than stainless steel. Thus, the possibility that a valve would stick open during severe accidents will depend on the behavior of the valve material when exposed to high temperature. The high-temperature fluid that must pass through the valve during discharge produces another effect. Because the PORV cage is a relatively thin member when compared with the valve body, it will heat up first and will try to grow in the axial and radial directions. However, because it is captured between the body and the bonnet face by the body-bonnet studs, potential damage to the cage or cage spacer (excessive bending) or additional stress on the body-bonnet studs could result.

EPRI has been running tests on non-nuclear safety valves to explore the effect of ring settings with 25°C (40°F) subcooled water (17 MPa [2400 psi]) for 15 minutes, during which the valves experience two-phase flow. Tests on safety valves were conducted by Combustion Engineering in Windsor with their high-temperature supercritical boiler at 593°C (1100°F) and 21 MPa (3000 psi). Valves showed stable behavior with little chattering.

The consensus was that the materials of construction for the PORVs should survive the high temperature transients that are expected during severe accidents. The body-bonnet stud material ratings in the Code, however, only permit their use in applications to 538°C (1000°F). The valve internals are mostly metallic and would not be expected to degrade significantly (lose their general shape or function) at 704°C (1300°F). The PORV air actuator diaphragm is buna-N rubber and could be damaged at temperatures that exceed 93°C (200°F). The diaphragm is necessary to open the valve but is not necessary or it to go to the fail-safe position, which is closed. One participant guessed that the valve should fail closed during severe accidents. However, the diaphragm chamber must be vented for the PORV to close; this requires that the solenoid valve be operational. These solenoids typically contain ethylene propylene seals, which could be damaged at temperatures above 120°C (250°F).

When addressing failure mechanisms, participants felt that bearing surface galling and set-point drift are potential problems for valves during severe accidents. However, participants were not optimistic that the information needed (e.g., clearances and detailed design information) to perform analyses of potential relief valve sticking would be easily obtained from the manufacturers. PORVs now allow more clearance between the packing gland and the stem, mainly because of thermal growth and the potential for galling of the stem. The PORVs would be susceptible to failure if the ambient temperatures were too high. They contain elastomeric materials (air diaphragms, O-rings, etc.) that cannot tolerate high temperatures for extended periods. If the ambient temperatures would not be excessive (below 93°C [200°F]) in the pressurizer doghouse (i.e., the structure above the pressurizer enclosing the PORVs) during this transient, PORVs should survive severe-accident transients.

The consensus on set point drift during severe accidents was that PSV discharge during severe accidents will cause the valve temperature to rise, which will cause the nozzle to expand diametrically, leading to heating of the disc, which could drop the set point significantly. On the other hand, axial thermal expansion may have an opposite effect. Recent tests by EPRI show that ambient temperature has more effect on set-point drift than the fluid temperature under normal operating conditions. Under normal operating conditions, the environment around PORVs is air at ≈ 55°C (130°F). However, during severe accident conditions, the environment around PORVs can be ≈ 180°C (350°F), 0.4 MPa (60 psi) steam. Tests have shown that a drop of as much as 6-7% can occur in the set-point setting in such an environment. During severe accidents, this could lead to a longer blow-down phase because the valve will close at a lower pressure. However, it was also pointed out that set-point drift should affect the opening load and closing load equally, so the opening time may not be significantly affected. Furthermore, it is expected that the high temperature of the PSV during discharge will affect the spring elasticity as well as yield stress. Change of set point because of loss of spring stiffness is a potential problem.

The cage material of PORVs can be 17-4 PH steel, condition H1100, which means that the cage was heat treated and tempered at 593°C (1100°F). Obviously, if the temperature of the cage exceeds 593°C (1100°F), the cage will lose all of the mechanical properties that were obtained from heat treating. Aside from thermal growth, the stresses in the cage should not be too high. However, the high temperature, combined with the loss in some of the mechanical properties can increase the galling potential between the plug and cage. Should galling occur, the valve would not be operational. Thermal binding of the disk and guide is possible and should be evaluated via finite-element modeling. However, because the transient involves repeated operation with a gradually increasing fluid temperature, the heating of the disk and guide is expected to be fairly uniform, i.e., they will heat up together. Detailed finite-element analyses should be used to estimate the various clearances during temperature transients. Several participants thought that binding should not be a problem for PORVs or PSVs because the temperature differential within the valve body should not be large owing to the stepwise increase of temperature with each opening and closing during severe accidents. Rapid heat up is of more concern because it may lead to large temperature differentials.

2.1.2 Anticipated Behavior of Bolted Connections during Severe Accidents

According to a gasket manufacturer's representative, various sizes, materials, and styles of spiral-wound gaskets are in use in existing manway/RCS systems. Most gaskets are the spiral-wound type, with stainless steel or nickel alloy winding that contains a flexible graphite filler. However, in his view the response to severe accidents should be somewhat similar for all gaskets that are currently in use. No gasket will be able to hold the pressure in case the clamp force (bolt load) is lost because of high temperature etc. (insulated studs will reach the internal high temperature quickly). Almost all gaskets are fit in a groove (captured in the cavity) with a metal-to-metal contact. This will reduce the possibility of a gasket blowout. The flexible graphite used as the filler material in spiral-wound gaskets is capable of surviving the predicted high temperatures for a short time (a few hours). However, a gross steam leak can wash out this soft material.

Although many details for differing manways are similar at first glance, material and geometrical differences would significantly affect response and time to leak. A heavy cover flange stud is bolted to a "boss" on the pressure vessel of most manways. However, bolt material, number of bolts, cover thickness as well as diameter, type, and design of external insulation differ from original equipment manufacturer (OEM) to OEM and over the years. It is also possible that the OEM bolted-joint design has been modified with "improvements," e.g., replacing B7 studs with B660 studs.

In general, ASME Code, Section III is used to design bolted connections. Within this section of the Code, bolted joints may be designed by rules within the code or they may be designed "by analysis" as set forth by the Code. In either of these cases, various geometries are possible within the rules. Because there is incentive to optimize component weight and thickness, and because other components are often subjected to considerable analysis, it is likely in most cases that the major joints are designed "by analysis." Thus, the joints will differ in design details such as component dimensions and number of bolts. In this manner, weight savings are achieved but standardization is compromised. For smaller joints (such as PORVs and PSVs), it is more likely that similar spiral-wound gaskets for standard (ASME B16.5) raised facings are used. In this case the gaskets would have inner and outer compression stop rings that provide confinement. However, it is quite possible that other types of gaskets have been used or tried.

Qualification tests on manways and RCS gaskets include mechanical tests (full-scale compression and leak tests). The production compression tests are performed during manufacturing. No data base is available for high temperature testing of these gaskets, However, the Pressure Vessels research council (PVRC) has published some high temperature and room temperature data for small size (15 cm [6 in.]) gaskets: WRC Bulletins 292, 294, and 309. The PVRC and the Materials Testing Institute (MTI) have sponsored a test program to characterize and qualify gaskets in elevated temperature service.

A gasket manufacturer's workshop participant indicated that their field experience shows that, because of their large size, gaskets in the bolted reactor coolant pump attachment joints may exhibit higher than normal susceptibility to damage at high temperature. This susceptibility may have a bearing on the loop seal clearing scenario, because, for the countercurrent flow case with the loop seal blocked, the temperature of the reactor coolant pump seal remains low and failure of the joint is unlikely. On the other hand, if during loop seal clearing sequences it can be shown that there is a high likelihood of pump gasket failure, it would be useful to know if RCS blowdown occurs; this is important because cases with full loop circulation during severe accidents are particularly damaging to the SG tubing and are very likely to lead to bypass of containment.

2.1.3 Conclusions from the Workshop

Most participants suggested that finite element analyses be conducted to determine how the various clearances within the valves change during the course of a severe accident. However, participants were not optimistic that the information (e.g., clearances and detailed design information) needed to perform analyses of potential relief valve sticking would be easily obtained from the manufacturers. Even if the thermal analyses suggest that the valve will stick, it will be difficult to draw any conclusion as to whether it will stick in the open, closed, or somewhere in-between position, without recourse to testing. In general, the participants felt that it will be difficult to analytically predict the behavior of the active components during severe accidents in the absence of test data. Even if test data were available, it was suggested that analyses by NRC and industry, and experience have shown that relatively small changes in timing or effective flow areas of stuck-open valves can lead to widely differing outcomes.

Participants were more optimistic about analytically predicting the behavior of bolted connections during severe accidents. The critical failure modes, such as loss of bolt tension due to thermal mismatch, loss of elastic modulus, or thermal creep, are addressable by finite-element analysis. It may be necessary to generate some high-temperature mechanical properties data to perform the analyses, which should be carried out for bolted connections of several sizes.

3 Evaluation of Current Analysis Methods

During severe accidents, components can be heated to a much higher temperature than that encountered under normal operation. Components can be heated to such an extent that their structural capacity is reduced and rupture may be induced by the combined effects of temperature, pressure, and dead load. The general philosophy of the severe-accident thermal hydraulic and rupture analyses in NUREG-1570 was to use a best-estimate approach within the limitations of available methods, with appropriate consideration of uncertainties.

A fundamental step in the analysis to estimate the potential for containment bypass via SG tube rupture during severe accidents is the calculation of failure probability of SG tubes relative to failure probabilities of other reactor coolant pressure boundary (RCPB) components. NUREG-1570 reported quantitative analysis results for the SG tubes, hot leg, pressurizer surge line, and reactor vessel. Other RCPB components will also be challenged during severe accidents; these include bolted joints, relief valves, and manways. Although potential failure modes of these components were discussed in NUREG-1570, they were not analyzed to the same level of quantitative detail as other components.

3.1 Current Models of Thermal Boundary Conditions for Mechanical Components

The thermal hydraulic analyses needed for NUREG-1570 were performed with the SCDAP/RELAP 5 code, which is a detailed, mechanistic severe-accident code (but is less rigorous than a computational fluid dynamics [CFD] code). SCDAP/RELAP 5 has been used to analyze high-pressure natural-circulation sequences in PWRs in numerous other studies. The code divides the RCS into several control volumes and satisfies continuity and energy conservation across boundaries among the volumes. To provide a range of predicted thermal-hydraulic conditions to accommodate the APET and to provide insights into the impact of design differences, several analyses were performed for both the Surry and Arkansas Nuclear One-Unit 2 (ANO-2) designs. To perform the analyses, values are needed for inlet plenum mixing parameters, such as mixing fraction, hot-leg-to-tube-bundle flow ratio, and the fraction of forward flow or hot tubes. These parameters cannot be calculated by SCDAP/RELAP 5, but rather are entered into it. The basis for the mixing model is the experimental result from the Westinghouse $1/7^{th}$ scale transient test. Sensitivity analyses showed that these parameters have a modest effect (20°C [36°F]) on the peak average tube temperatures.

Generally, the nodalization used in the NUREG-1570 thermal hydraulic analyses was relatively coarse. A limited sensitivity study of the countercurrent natural circulation with more refined nodalization of the inlet volumes showed that the temperatures remained within 15°C (27°F) of the initial results. Subsequent sensitivity studies used an updated version of SCDAP/RELAP 5 to more appropriately treat the combined effect of forced and free convection heat transfer in vertical pipes. Again, the results were not significantly changed from the original. When fluid-to-fluid heat transfer and circumferential-pipe-wall-conduction were modeled, the temperature difference between the upper hot vapor and the cooler fluid below was reduced, relative to the original calculations. This lowered the tube temperatures and delayed failure of the SG tubes by ≈ 5 minutes.

From the complete set of calculations, the NRC staff and its contractor drew the overall conclusion that the uncertainties in the thermal hydraulic results are modest and the SG tube temperature at the time of surge line failure is ≈ 687 ± 20°C (1268 ± 36°F), while the surge line temperature is typically several hundred degrees F hotter. The RCS pressure at failure was assumed to be ≈ 17 MPa (2400 psi).

The industry has analyzed a similar set of severe accidents with MAAP. Two differences in the phenomenological models used in MAAP and SCDAP/RELAP 5 may have a significant effect on the results. The first is that SCDAP/RELAP 5 ignores, but MAAP includes, radiative heat transfer between the hot gases, the RCS piping, and other structures. Inclusion of radiative heat transfer could reduce the hot-leg temperature significantly. Second, MAAP contains a specialized model for the perforated plate that covers the entrance to the pressurizer surge line in most PWRs. The presence of the cover reduces the countercurrent drainage of water from the pressurizer to the core after the RCS becomes voided and the core is uncovered. This, in turn, means that the pressurizer contains water until after creep rupture of the hot-leg pipe has occurred. The presence of the water that covers the exit of the surge line would protect the pressurizer PORV, PSV, and manway from the hot gas environment during the time of interest of the accident. This could mitigate or minimize the problem of high-temperature-induced leakage in any of these components. The effect of the perforated plate cover at the pressurizer surge line entrance may not have been accounted for in the SCDAP/RELAP 5 calculations. Industry participants at the workshop felt that the effect of the modeling differences had not been fully assessed.

An important conclusion that can be drawn from the current analyses is that the PORVs are at a significantly higher temperature than the PSVs because they cycle many times during the severe accident transient passing very hot primary fluid during discharge; whereas, the PSVs only cycle a few times early and pass relatively cooler two phase flow. However, even if the uncertainties in the overall analyses are small, most of these analyses were conducted with a relatively coarse nodalization scheme, so detailed temperature variations within the inlet plenum, valves, or manways are not available.

3.2 Stress and Rupture Analyses

3.2.1 Passive Components – Piping and Flanges

SCDAP/RELAP 5 can provide creep rupture life estimates of unflawed piping subjected to hoop stress due to internal pressure, based on Larson-Miller parameter representation of creep properties. However, the analyses of the piping conducted to date have ignored the effects of the geometrical complexity of the piping, boundary conditions (e.g., hangers, snubbers, and stops), thermal stress due to temperature gradient across the cross section of the hot-leg pipe with countercurrent circulation, variation of stress through the wall thickness of the piping, presence of welds, residual stresses, and flaws. Inclusion of these effects in the analysis may significantly influence the creep rupture time of the hot-leg piping.

The results of some detailed modeling of bolted connections (with gaskets) was reported in NUREG-1570. This work sought to determine whether sufficient yielding of bolt material would occur to produce joint separation and thus significant leakage. Specifically, the bolted joints of the primary-side SG and pressurizer manways, and the bonnet flanges of the loop isolation valves were modeled. A detailed finite-element analysis was not performed; instead, the bolted joints were analyzed for a given bolt preload by solving simultaneous semielastic characteristic equations for the structurally coupled bolts and gasket, in equilibrium with the system fluid pressure. Thermal expansion of the individual components was considered. Constitutive equations were linearized over a narrow range and a computer code was written to compute the results. A gap was predicted to occur in the joint whenever the gasket was unloaded.

In reality, leakage should occur long before such a condition is reached because the hydrodynamic forces of the leaking fluid were not taken into account. Erosion of the flange faces may also increase leakage. At high temperatures during severe accidents, both creep and loss of strength of the bolt material with temperature will tend to unload the gasket. Either or both phenomena may control the response of the joint, depending on the temperature and time parameters. Larson-Miller parameter methodology was used to calculate creep deformation. Mechanical properties of bolting material were obtained from the ASME Boiler and Pressure Vessels Code and Air Force Materials Handbook.

The stainless-steel-clad, carbon steel SG manway with 16 4.78-cm (1.88-in.)-diameter AISI 4140 bolts was analyzed with the above model. Using a cover thickness of 16.5 cm (6.5 in.) (nominal manway diameter is 46 cm [18 in.]); an Inconel spiral-wound gasket 0.6 x 5 cm wide (0.25 thick x 2 in. wide); 25% gasket springback; and a bolt preload of 3.44 x 10^5 N (77,330 lbf) per bolt, researchers estimated that the joint would separate under a fluid pressure of 17 MPa (2400 psi) at 627°C (1160°F). Since the thermal hydraulic model for the inlet plenum did not contain a node near the manway, the temperature at the bottom of the hot leg was taken as representative of that of the manway. It was estimated that the gasket on the SG manway would very likely fail before the pressurizer manway.

3.2.2 Active Components – PORVs and PSVs

One obvious effect of high temperatures during a severe accident is a change in the set points of the PSVs. In NUREG-1570, the change in the set point of the PSV was estimated from the loss of elastic modulus of the spring with temperature. The set point was reduced from 17.1 to 15.7 MPa (2485 to 2270 psi) when the valve body temperature reached 427°C (800°F), as predicted by the SCDAP/RELAP 5 analysis, which does not consider local temperature variations in detail. Because the escaping gases are much hotter than 427°C (800°F), it was speculated that leakage through the PSV would raise the temperature of the valve body further, which would reduce the spring stiffness, reduce the set point, and lead to more leakage, and so on. Clearly, a more detailed transient thermal and structural analysis, including potential creep deformation of the spring, is needed to predict the behavior of valves during severe accidents.

EPRI has been developing a finite element model to evaluate the thermomechanical behavior, including the forces on the components, of relief valves. However, these analyses are at relatively low temperatures, where creep effects are negligible. A similar finite-element-based elastic-plastic-creep analysis of PORVs and PSVs, with detailed geometric modeling of the body, bonnet, spring, and the gap between various sliding parts (e.g., valve stem and bonnet guide), will need to be analyzed for their behavior during severe accidents. Because of the large number of cycles involved, close valve tolerances, and possible differential thermal expansion between the stainless steel bonnet guide and the 17-4 PH stem material, PORVs are likely to stick during severe accidents. Galling between the stainless steel bonnet guide and 17-4 PH valve stem has been reported to occur occasionally, even under normal operating conditions, and it is much more likely that

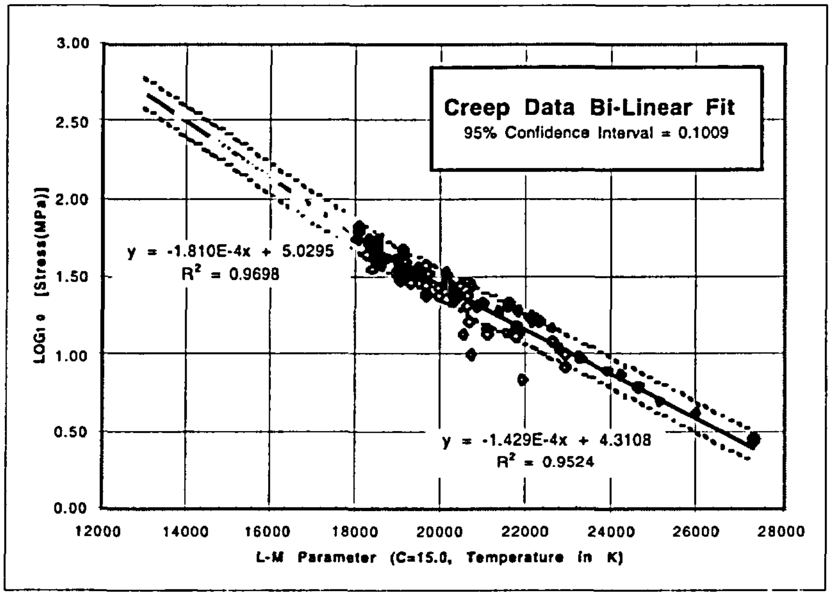

Figure 1. Bilinear fit to existing creep rupture data for Alloy 600. Three low outliers were not included in the analysis.

galling will occur with the reduced clearances that exist under severe-accident conditions. However, analytical prediction of when galling will occur is difficult, and would require development of a high-temperature material galling database. The temperatures during severe accidents are well outside the normal design range for the materials involved; hence, relevant data are limited in some cases.

3.3 Larson-Miller Parameter

In creep rupture evaluations, the method proposed by Larson and Miller ("A Time-Temperature Relationship for Rupture and Creep," Trans. ASME, July 1952, pp. 765-775) has been widely used to interpolate and extrapolate creep rupture data in the time-temperature domain. Although derived empirically, Larson-Miller creep rupture relationships take the form of Arrhenius type equations. The Larson Miller parameter is defined as :

$$P_{lm} = T[\log_{10}(t_R) + C] \quad (1)$$

where T is temperature in K, t_R is time to rupture in h, and C is a fitting constant. The Larson-Miller parameter for any material is a function of stress. The available creep rupture data for Alloy 600 can be accurately described in terms of the Larson-Miller parameter by assuming that the Larson-Miller parameter is a bilinear function of stress as shown, in Fig. 1 (from

NUREG/CR-6575). A similar correlation is used in SCDAP/RELAP 5, but with a linear dependence on stress (Fig. 2).

The Larson-Miller parameter has been used to represent creep rupture data of carbon steels and stainless steels in many studies (e.g., see NUREG/CR-5642). Although other empirical parameters for fitting creep rupture data are available (e.g., Manson-Haferd parameter), the Larson-Miller parameter (Fig. 1) generally fits creep data well, and, as noted, was used successfully in NUREG/CR-6575 to predict rupture times of both flawed and unflawed SG tubes subjected to temperature and pressure ramps. Being a one-parameter correlation, it is better suited than other multiparameter correlations to represent creep rupture properties as functions of stress and temperature based on limited data.

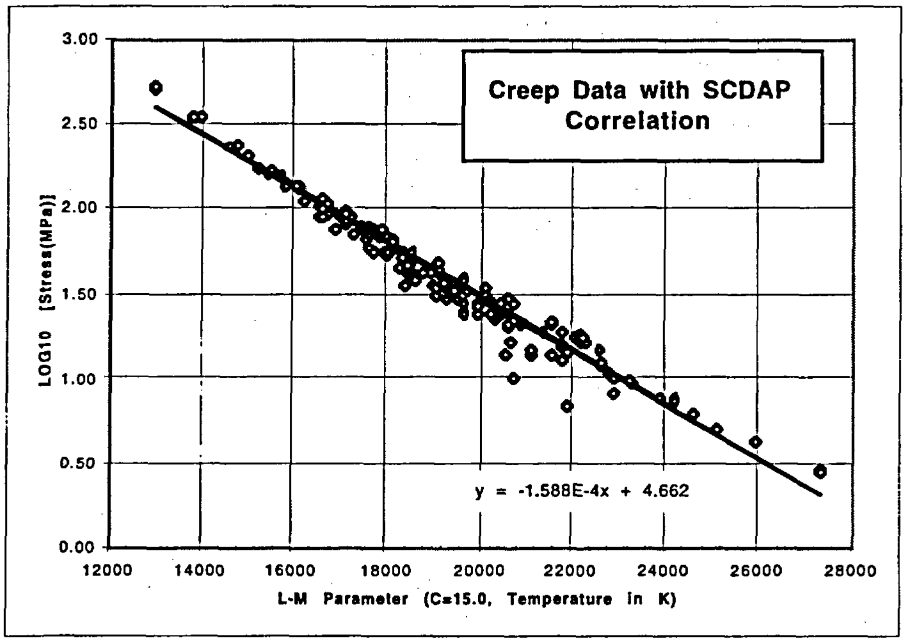

Figure 2. Correlation currently used in SCDAP/RELAP 5 and existing data for Alloy 600.

4 Conclusions and Recommendations

The thermal hydraulic results reported in NUREG-1570 for the Surry and ANO-2 plants (or the Zion plant, if available) should be used to carry out a more detailed finite-element based stress and creep rupture analysis of the hot-leg piping, surge line, and nozzles. In some cases, a few additional runs with SCDAP/RELAP 5 with additional elements may be needed to better define the temperature distribution in the critical regions. Because some modeling differences exist between SCDAP/RELAP 5 and MAAP, it would be useful if results from similar runs with the industry code MAAP can be obtained to assess the potential impact on failure predictions. High-temperature yield, ultimate, and creep rupture data for stainless steels in the needed temperature range are available. Similar data were developed for pressure vessel steels in the NRC light water reactor lower head failure analysis program and can be used in the analysis. Although the primary loading on the piping is due to pressure, thermal expansion of the piping system could produce additional loading that should be addressed. A realistic layout of the piping, together with the various supports and boundary conditions, is needed to determine these loads. However, if such piping layout details are not available, the analyses can be conducted parametrically. In any case, the parametric studies will be needed to get a more generic assessment for a range of geometries.

A simplified model for analyzing bolted connections that was developed to support the studies in NUREG-1570 should be validated and extended with finite-element analyses. The hydrodynamic forces due to escaping fluid after leakage starts should be estimated and included in the analysis. High-temperature properties of the gasket, including the effect of temperature on elastic springback, will be needed.

The SCDAP/RELAP 5 results reported in NUREG-1570 may not be sufficient for the analysis of the transient behavior of PORVs and PSVs. We can develop a detailed computational thermal model of a valve, or several valves that can be used to calculate rapid heatup of the valve subsequent to initiation of high-temperature flow. For initial assessments, however, it may not be necessary to calculate the detailed fluid flow field; instead, high-temperature boundary conditions can be suddenly applied and the heatup calculated. The aim would be to assess whether nonuniform heatup and consequent nonuniform thermal expansion can lead to deformation of the valve and subsequent sticking of valve components or to impairment of function that could affect opening, closing, or sealing. However, detailed design information about the valve, including the tolerances between tightly fitting parts will be needed to carry out such analyses, and it is not clear whether valve manufacturers would be willing to make such information available. In addition, although the analysis will determine when the gaps will close, it will still be difficult to predict whether galling will actually occur and whether the valve will fail in the stuck-open, closed, or in-between position. However it is expected that limited materials and performance information can be obtained for the seating surfaces. Therefore, it is recommended that initial investigations of PORVs and PSVs be limited to the high temperature impact behavior between the sealing surfaces.

Appendix A PORV/PSV and Bolted Connections Workshop Agenda

PORV/PSV and Bolted Connections Workshop (November 14, 2001)

AGENDA (8.30 am to 5 pm)

1. Welcome – W. J. Shack

2. Welcome & Workshop Overview - J. Page

    A. Attendee Introductions

    B. Meeting Agenda Review

    C. Overview of NRC's program addressing RCS components other than SG tubes

3. NUREG-1570 Overview (as it applies to Workshop Objective) - S. Long

    A. Pressure - Temperature transient applicable to relief valves and flanges

    B Additional complexities introduced early in transient (i.e., steam flow, water flow / slugs, very hot steam flow, and, at last, superheated gas)

4. Relief Valve, Bolted connection and Gasket Workshop - W. Shack and S. Majumdar

    A. Workshop Structure and objectives

    B. Written feedback from attendees

    C. PORVs and PSVs Question and Answer Session

    D. Bolted connections and Gaskets Question and Answer Session

5. Closing comments - J. Page

6. Adjourn

Appendix B PORV/PSV and Bolted Connections Workshop Attendees

PORV/PSV and Bolted Connections Workshop (November 14, 2001)

LIST OF ATTENDEES

Name	Affiliaion	Email address	Participated by
1) John Hosler	EPRI	jhosler@epri.com	by telecon.
2) Mati Merilo	EPRI	MMERILO@epri.com	in person
3) Marc Kenton	Creare, Inc.	mak@creare.com	in person
4) Jim Payne	JPAC, Inc.	jimpayne@att.net	in person
5) Ebadollah Jamalyaria	Flexitallic	jamal_jamalyaria @flexitallic.com	by telecon.
6) Allan K. Shea	DeZurik/ Copes-Vulcan	allan shea @dezurik.spx.com	by telecon.
7) Dave Thibault	Anderson-Greenwood-Crosby	dthibault@tycoepg.com	by telecon
8) Steve Long	NRC/NRR	sml@nrc.gov	in person
9) Charles Hammer	NRC/NRR	CGH@nrc.gov	in person
10) Joel Page	NRC/RES	jdp2@nrc.gov	in person
11) Bill Shack	ANL/ET	wjshack@anl.gov	in person
12) Saurin Majumdar	ANL/ET	majumdar@anl.gov	in person

Appendix C PORV/PSV and Bolted Connections Workshop Summary

PORV/PSV and Bolted Connections Workshop (November 14, 2001)

Summary

A workshop on power-operated relief valves (PORVs), pressurizer safety valves (PSVs), and bolted connections was held on the 14th of November at Argonne National Laboratory (ANL) to discuss the potential behavior of these components during severe accidents in PWRs. The agenda for the workshop is enclosed as Attachment 1. Two representatives from EPRI, three from NRC, two from ANL, and a consultant attended the workshop. Additionally, four persons representing EPRI, Dzurik/Copes-Vulcan, Anderson-Greenwood-Crosby, and Flexitallic, participated in the workshop by teleconference. A list of attendees is enclosed as Attachment 2.

Prior to the workshop review materials (i.e., selected parts of NUREG-1570 and questions) were provided to attendees for their review. The workshop started out with presentations of background material by Joel Page and Steve Long of the NRC, which were followed by a round table discussion. The discussion concentrated on behavior of relief valves (plus some bolted connections) in the morning, while the afternoon session was devoted exclusively to discussions on behavior of flanged and bolted connections. The major issues are discussed below as they related to the questions provided to attendees.

The following is a summary of the discussions held at the workshop, together with follow-up information provided by the attendees.

I. PORVs and PSVs

1. What are the design differences between PORVs and PSVs designed for PWR RCS service versus those designed for higher temperature service, such as that seen in fossil plants?

Participants seemed to agree that although the materials are different, the design of safety valves for low temperature (nuclear) applications is basically the same as that for high temperature (fossil) applications.. Fossil experience with relief valves should be useful for predicting severe accident behavior of PWR Pressurizer PORVs because clearances in fossil valves and PORVs are quite similar. Some are 316 stainless steel valves using actuator (spring return), diaphragm, and solenoid valve. The cage is in the valve body and both the cage (17-4 ph steel) and cage spacer (316 SS) are captured by the bonnet. The valves operate at 593°C (1100°F).

EPRI is developing a finite-element relief valve model to evaluate the thermo-mechanical behavior, including the forces on the components, of relief valves. They advised that a presentation on this subject will be made at the upcoming Pressure Relief Users Group (PRDUG) meeting in Orlando, FL, in late January 2002.

As shown in the background materials, the PORVs and PSVs are predicted to operate many times during conditions that start out as 600°F steam and progress in about 2 hours to 1400°F gas.

2. How long would the PORV or PSV be expected to stay open on each stroke? Would these valves be expected to operate somewhat perfectly during these conditions?

All valve experts agreed that sustained flow of water at the expected subcooling during discharge is not expected to damage the pressurizer safety valves. If this is true, then the possibility that such a valve would stick open will depend on the behavior of the valve material when exposed to high temperature. Basically, the materials of construction for the PORV should survive the high temperature transient that is expected during severe accidents. The body-bonnet stud material ratings in the Code however only permit use in applications to 1000°F. The valve internals are mostly metallic and would not be expected to degrade significantly (lose their general shape or function) at 1300°F. The diaphragms in the actuator air operators on PORVs are made of buna-N rubber and could be damaged at temperatures exceeding 200°F. The diaphragm is necessary to open the valve but is not necessary to go to the fail-safe position, which is closed. One participant guessed that the valve should fail closed during severe accidents. However, the diaphragm chamber must be vented for the PORV to go closed. This requires the solenoid valve to be operational. These solenoids typically have ethylene propylene seals, which could be damaged at temperatures above 250°F.

3. What types of failures might the valves experience and when?

Bearing surface galling and set point drift could be potential problems for valves during severe accidents. PORVs now allow more clearance between the packing gland and the stem, mainly because of thermal growth and the potential for galling of the stem.

The actuator air operators in PORVs would be susceptible to failures if the ambient temperatures were too high. They have elastomeric materials (air diaphragms, O-rings etc) that cannot take high temperature for extended periods. If the ambient temperatures would not be excessive (below 200°F) in the pressurizer doghouse during this transient, PORV should survive the severe accident transients.

The diaphragm in PORVs will fail above 93°C (200°F). Stellite should survive high temperature. The high temperature fluid must pass through the valve during discharge. Since the cage is a relatively thin member when compared to the valve body, it will heat up first and will try to grow in the axial and radial directions. However, it is captured between the body web and the bonnet face, and retained by the body-bonnet studs. Potential damage to the cage or cage spacer (excessive bending) could occur or additional stress on the body-bonnet studs could result.

EPRI has been running tests on safety valves to explore effect on ring setting with 22°C (40°F) subcooled water (16.5 MPa [2400 psi]) for 15 minutes during which the valves see two-phase flow. Tests on safety valves were conducted by Combustion Engineering in Windsor using their high temperature supercritical boiler at 593°C (1100°F) and 21 MPa (3000 psi). Preliminary results showed stable behavior with little chattering.

Water passage during valve open will push the plug open, but the experts did not foresee a lot of damage from that. However, if the valve is open it could cause damage to the actuator. Leakage can be expected from inconel on stainless steel joints (seat damage), less from stellite on stellite joints. There was a feeling that inconel should be able to take the cyclic operation during severe accidents better than stainless steel.

At least one participant thought that we will not find a valve failure mode that we can depend on with high confidence.

4. Would set point drift be expected to take place?

PSV discharge during severe accidents will raise the valve temperature. It is expected that high temperature will cause the nozzle to expand diametrically leading to heating of the disc which could drop the set point significantly. Axial expansion may have an opposite effect.

Tests show that ambient temperature has more effect on set point drift than the fluid temperature under normal operating conditions. Under normal operating conditions, the environment on PORVs is air at ≈ 55°C (130°F). However, during accident conditions, the environment on PORVs can be ≈ 180°C (350°F), 0.4 MPa (60 psi) steam. Tests have shown that there may be as much as 6-7% drop in the set point setting. During severe accidents, this could lead to longer blow down phase because the valve will close at a lower pressure. However, it was also pointed out that set point drift should affect the opening load and closing load equally, so that the opening time may not be significantly affected.

5. Would spring relaxation or Young's Modulus be affected?

It is expected that high temperature of the PSV during discharge will affect the spring elasticity as well as yield stress. Loss of spring stiffness could lead to lowering of set point.

6. Would binding between the cage and plug, or other subcomponents, be expected?

<u>PORVs</u>
The cage material of PORV can be 17-4 ph steel, condition H1100. This means that the cage was heat treated and tempered at 593°C (1100°F). Obviously, if the temperature of the cage exceeds 593°C (1100°F), the cage will lose all of the mechanical properties that were obtained from heat treating. Aside from thermal growth, the stresses in the cage should not be too high. However, the high temperature, combined with the loss in some of the mechanical properties can increase the galling potential between the plug and cage. Should galling occur, the valve would not be operational. Thermal binding of the disk and guide is possible and should be evaluated via finite element modeling. However, since the transient involves repeated operation with a gradually increasing fluid temperature, the heating of the disk and guide is expected to be fairly uniform, i.e., they will heat up together.

<u>PSVs</u>

Sticking will be an issue with PSVs for severe accidents. Detailed finite element analyses should be used to estimate the various clearances during temperature transients.

Several participants thought that binding should not be a problem for PORV or PSV. Temperature differential within the valve body should not be large because temperature should increase stepwise with each opening and closing during severe accidents. Rapid heat up is of more concern because it may lead to large temperature differentials.

II. Bolted (Flanged/Gasketed) Connections

1. Can PWR RCS gasketed joints (PORV, PSV, SG manways, pressurizer manways), including their gaskets, be considered as somewhat generically the same?

A representative from a gasket manufacturer stated that there are different sizes, materials, and styles of spiral wound gaskets in use in existing manway/RCP systems. Most gaskets are the spiral wound type with stainless steel or Inconel winding with a flexible graphite filler. However, in his view the response to severe accidents should be somewhat similar for all gaskets in use. No gasket will be able to hold the pressure in case the clamp force (bolt load) is lost due to the high temperature, etc. (insulated studs will reach the internal high temperature quickly). Almost all gaskets are fit in a groove (captured in the cavity) with a metal-to-metal contact. This will reduce the possibility of gasket blowout. The flexible graphite used as the filler material in spiral-wound gaskets is capable of surviving the predicted high temperatures for a short time (few hours), however, a gross steam leak can wash out this soft material.

While many details of the manways are typically similar at first glance, there could be material and geometrical differences that would significantly affect response and time to leak. Most manways are a heavy cover flange stud bolted to a "boss" on the pressure vessel. However there are likely differences in bolt material, number of bolts, cover thickness and diameter and type and design of external insulation from OEM (original equipment manufacturer) to OEM and over the years. It is also possible that the OEM flanged joint design has been modified with "improvements"; for example, replacing B7 studs with B660 studs.

For large joints, such as the manways, the gasket is confined. That is, it is positioned in a groove so that at full compression there is metal-to-metal contact. However there are gasket diameter and width differences to be found. In some designs by one OEM, double concentric grooves are used where leakage may be bled from the confined annulus. In general the large joints have been designed "by analysis" and therefore will differ as to component dimensions and number of bolts for example. In this manner weight savings are achieved but standardization is compromised. For smaller joints (e.g., PORV's and PSV's), it is more likely that similar spiral wound gaskets for standard (ASME B16.5) raised facings are used. In this case the gaskets would have inner and outer compression stop rings providing confinement. It is quite possible that other types of gaskets have been used or tried.

Although the RCS coolant pump flanges could be challenged, it was pointed out that the pump flanges would remain relatively cool because of the remaining water level in the RCS.

Other flanged connections? Identify which ones.

Most other joints would be various smaller piping or instrument connections probably with standard ASME B16.5 class 1500 piping type joints. While it was pointed out that this feature is no longer used it is interesting to consider whether or not the connection still exists in some plants, and if it does, is it significant. On-site "walk-thrus" of several plants by different OEM's would be useful in this regard as would a paper survey of all plants. It could be worthwhile to review several typical Piping & Instrument drawings as well to get a better feel for the total number of connections that might be flanged.

2. What design codes or standards have typically been used to design the flanged connections/gaskets in PWR RCSs?

   In general the ASME Code, Section III is used. Within this bolted joints may be designed by rules within the code or they may be done "by analysis" as set forth by the code. In either of these various geometries are possible within the rules. Since there is incentive to optimize component weights and thickness, and since considerable analysis is performed for other components, it is likely in most cases that the major joints are designed "by analysis." It may be worthwhile to survey knowledgeable persons within the OEMs to gain more detail on past bolted joint design practice. As above, smaller piping type connections are likely to be dimensioned according to ASME B16.5.

3. Is there an available data base that contains information regarding which types of gaskets/flanges are used in various specific PWR RCSs?

   Not to anyone's knowledge.

4. If the answer to 3 is "No," would the creation of such a data base significantly help the research project?

   The creation of such a data base, carefully constructed as to detail, would significantly help the research project. A database of only manways, for example, would find those designs that are likely most vulnerable to severe accident conditions by virtue of component materials, insulation detail and rigidity.

5. Are there available data bases regarding high temperature performance characteristics of flanges and gaskets? Are these data bases pertinent to our research or are the designs of those components different enough to not be applicable?

   Qualification tests done on manway, and RCS gaskets include mechanical tests (full scale compression and leak tests). The production compression tests are performed during manufacturing. A list of engineers from Westinghouse, CE, B&W and Flowserve familiar with these tests was provided by the representative from the gasket manufacturer.

   TTRL of Ecole Polytechnique University, Montreal, Canada, and CETIM lab in France are the two major sources used in testing gaskets. No data base is available for high temperature testing of these gaskets, however, there are some high temp., and room temp. data available for small size (15 cm [6 in.]) gaskets published by PVRC (Pressure Vessel Research Council). Reference: WRC Bulletins 292, 294, & 309.

There have been papers presented over the years since the '40's regarding high temperature performance characteristics of flanges and gaskets. Several of these contain experimental high temperature bolted joint relaxation data as well as theoretical considerations. The PVRC and the MTI have sponsored a test program concerning the characterization and qualification of gaskets in elevated temperature service. A number of references were provided for results on graphite filled spiral wound gaskets subjected to 649°C (1200°F).

6. Should additional high temperature data bases be created for flanged connections in severe accidents?

   A comprehensive literature review and the creation of a reference data base pertinent to Phase II severe accident research should be helpful to the project.

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10. SUPPLEMENTARY NOTES

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11. ABSTRACT (200 words or less)

A critical step in the assessment of risk of containment bypass attributable to pressure- and temperature-induced failures of SG tubes during severe accidents is the prediction of the sequence in which the SG tubes fail relative to other RCS components. This report summarizes our current state of understanding of behavior of PWR coolant system components other than steam generator tubes during severe accidents. A detailed analysis of RCS components during severe accidents reported in NUREG-1570 predicted the failure times of steam generator tubes, the pressurizer surge line, and hot leg piping to be very close. However, the analyses conducted for predicting failure of RCS components were less rigorous and detailed than those for SG tubes. This report reviews the analysis methods followed in NUREG-1570 regarding behavior of RCS piping, PORVs, SRVs, and manway bolted connections during severe accidents and recommends future research and analyses that should be conducted to bring the failure prediction methodology on a par with that followed for SG tubes. This will enable a more balanced assessment of the potential for containment bypass attributable to SG tube failures.

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

Steam Generator
Pressurized Water Reactor
Reactor Coolant System Components
Severe Accidents

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