Project Number 694

January 28, 2005

WOG-05-36

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Westinghouse Owners Group
Request for Enforcement Discretion For Reactor Coolant Pump Seal Performance Findings in Triennial Fire Protection Inspections (TAC No. MC3865) – Submittal of Information

Reference: (1) NRC Letter, Herbert N. Berkow to Frederick P Schiffley, II, "NRC Staff Review of the Westinghouse Owners Group Request for Enforcement Discretion For Reactor Coolant Pump Seal Performance Findings in Triennial Fire Protection Inspections (TAC No. MC3865)" dated November 12, 2004.

The purpose of this letter is to provide a summary of analysis results regarding Reactor Coolant Pump (RCP) seal performance under loss of seal cooling scenarios. The analysis results are provided in the enclosed report, WCAP-16396-NP, "Westinghouse Owners Group Reactor Coolant Pump Seal Performance for Appendix R Assessments".

The NRC granted enforcement discretion for RCP seal performance findings in triennial fire protection inspections while the WOG completed analyses on loss of RCP seal cooling (Reference 1). The enclosed report is provided to close a commitment by the WOG to provide the analyses to the NRC by January 28, 2005.

The enclosed report is provided for use by the NRC. The WOG does not request NRC approval of the report and therefore does not expect review fees.

D048

The WOG appreciates the process the NRC has adopted to resolve the issues associated with RCP seal performance findings in the triennial fire inspections. The WOG is pleased to meet with the NRC in the near future to discuss the contents of the report. If you have any questions or comments, please contact Mr. Paul Hijeck at (860) 731-6240.

Very truly yours,

Frederick P. "Ted" Schiffley, II, Chairman
Westinghouse Owners Group

FPS:PJH:las

Enclosure

cc: Steering Committee
Management Committee
Licensing Subcommittee
Analysis Subcommittee
Risk Management Subcommittee
H. Berkow, NRC
G. Shukla, NRC (3 copies)
S. Weerakkody, NRC
G. Bischoff, W
J. Gresham, W

Westinghouse Non-Proprietary Class 3

WCAP-16396-NP
Revision 0

January 2005

Westinghouse Owners Group Reactor Coolant Pump Seal Performance for Appendix R Assessments

Westinghouse

Westinghouse Non-Proprietary Class 3

WCAP-16396-NP

Revision 0

Westinghouse Owners Group
Reactor Coolant Pump Seal Performance for Appendix R Assessments

Prepared for
Westinghouse Owners Group

Developed under WOG Program
PA-RMSC-0181

January 2005

Author: Electronically Approved in EDMS 2000*
Robert J. Lutz, Jr.
Reliability and Risk Assessment

Author: Electronically Approved in EDMS 2000*
J. Gene Zottola
Pump and Motor Engineering

Approved: Electronically Approved in EDMS 2000*
Gary Ament, Manager
Reliability and Risk Assessment

Approved: Electronically Approved in EDMS 2000*
Jai Moxie, Acting Manager
Pump and Motor Engineering

*** Official record electronically approved in EDMS 2000**

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

LEGAL NOTICE

COPYRIGHT NOTICE

Acknowledgments

The authors would like to thank Bill Mendez and Eric Colvin of the Pump and Motor Operations group for their valuable assistance in assessing the data on pump performance under loss of seal cooling and restoration of seal cooling events. Also the contribution of Chris Cayer in tracking down operational events and data is gratefully acknowledged.

Substantial history of the development of the Westinghouse (WCAP-10541) and NRC (Rhodes Model) seal LOCA PRA models was provided by Tom Timmons. Other history and seal performance assessments have been provided by Dennis Salak and David Janocko of Curtiss Wright EMD.

The assistance of Ching Guey of Florida Power and Light and Steve Rosenau of Duke Energy in helping to obtain information related to the March 2001 loss of all AC power event at Maanshan is also gratefully acknowledged.

The thermal hydraulic analyses in Section 6 were completed by Josh Hartz of the Licensing Integrated Services group, for which the authors are quite grateful.

The authors would also like to acknowledge the assistance of Dan McLaughlin of the Reliability and Risk group for his assistance in completing the risk sensitivity assessments presented in this report.

Table of Contents

List of tables and figures

List of Acronyms

AC	Alternating Current
CCW	Component Cooling Water
CFR	Code of Federal Regulations
EdF	Electricite' de France
ERG	Emergency Response Guideline
GSI	Generic Safety Issue
LOASC	Loss of All Seal Cooling
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SDP	Significance Determination Process
SE(R)	Safety Evaluation (Report)
SG	Steam Generator
SW	Service Water
WOG	Westinghouse Owners Group

1. INTRODUCTION

1.1 OBJECTIVE

The objective of this report is to provide relevant information, from both old and new investigations, testing results and field data to support the Westinghouse position that the reactor coolant pump (RCP) seals are expected to remain functional in the event of a loss of all seal cooling event and that the leakage is expected to be in the range of a nominal 21 gpm per pump for the duration of the event. Regulatory compliance assessments that use deterministic assumptions regarding RCP seal performance under a loss of all seal cooling should be based on a 21 gpm per pump leak rate. This position applies to all loss of seal cooling events, regardless of whether seal injection cooling is restored after hot RCS fluid begins to heat the seal assembly, as long as the RCPs are stopped before the seals begin to heatup.

The information provided in this report does not change any previous Westinghouse or Westinghouse Owners Group (WOG) recommendations related to reactor coolant pumps seals for a loss of all seal cooling event. While changes to the acceptable Probabilistic Risk Assessment (PRA) model for RCP seal leakage may be appropriate as a result of the information in this report, the potential changes to the PRA model are not discussed in this report.

1.2 BACKGROUND

Nuclear Regulatory Commission (NRC) inspectors have identified findings during recent Triennial Fire Inspections at several Westinghouse Nuclear Steam Supply System (NSSS) plants related to RCP seal performance following a loss of all seal cooling (see for example Reference 1). In assessing fire coping strategies for compliance with Appendix R requirements, the NRC has assumed large leak rates (e.g., 182 gpm per pump) when the seal cooling is not restored before hot reactor coolant system (RCS) fluid contacts the seals and when seal cooling is restored after hot RCS fluid contacts the seals. These large pump seal leakage rates are taken from a recently approved PRA model (Reference 2) and are being used in deterministic regulatory compliance assessments associated with 10 CFR 50 Appendix R. Specifically, the NRC has assigned a high likelihood of a leakage rate of 182 gpm based on a single set of investigations, completed in the 1980's by their contractors, of the performance of the No. 2 reactor coolant pump seal following a loss of all seal cooling. This model is not supported by prototypical field experience or test observations. The use of the 182 gpm leakage rate in assessing plant specific Appendix R coping strategies has resulted in findings at several plants that the coping strategies are not capable of maintaining pressurizer level above the minimum measured span, as required by Appendix R. The NRC has subsequently used the same PRA-based leakage rate model in their Significance Determination Process (SDP) to determine the safety significance of the finding.

Westinghouse believes that the NRC is improperly using probabilistic assumptions and models in design basis regulatory compliance assessments. Based on prototypical testing and field experience, the RCP seals do not exhibit the large leakage rates assumed in the probabilistic models following a loss of all RCP seal cooling. Some of these same tests and filed experience also show that the restoration of seal cooling after a prolonged period without seal cooling does not lead to large seal leakage rates. Note that Westinghouse does not take issue with the use of the NRC's approved PRA model for RCP seal behavior in risk applications, including the SDP process.

NRC inspectors have identified similar findings at a number of plants that use Westinghouse reactor coolant pumps and seals and may have similar findings at other plants to be inspected in the near future. Rather than respond to the same issue in response to inspection findings at each plant, the WOG requested a meeting with the NRC to explore the possibility of reaching a generic resolution of the issue. The WOG expressed this opinion at a meeting with the NRC in June of 2004 (Reference 3) and requested that the NRC exercise enforcement discretion in resolving inspection findings until a generic resolution of the issue could be concluded. The WOG formally requested the enforcement discretion in a July 2004 letter to the NRC (Reference 4). Westinghouse subsequently issued a Technical Bulletin (Reference 5) to advise all utilities using Westinghouse reactor coolant pumps and seals of the NRC potential inspection findings at each plant. The NRC agreed to exercise enforcement discretion and seek generic resolution of the issue in a November 12, 2005 letter to the WOG (Reference 6).

The basis for the WOG request and NRC agreement for enforcement discretion was new information being developed by WOG investigations as outlined in the July 20, 2004 WOG letter and the November, 2004 Westinghouse Technical Bulletin. The investigations included:

- Obtaining and assessing existing test data from Electricite' de France (EdF) on the performance of RCP seals when seal injection is restored after a prolonged period without cooling,
- Developing a technical basis for exemption requests from the safe shutdown requirements in Appendix R (III.G.1.A and III.L.2.b) based on the level of conservatism in these requirements, and
- Quantifying the conservatism in the PRA model for RCP seal performance that is being applied in fire protection inspections.

Also, additional information was obtained for two important losses of seal cooling events that had not been previously documented. Finally, upon review of existing data presented in previous Westinghouse documents, it was determined that additional clarification of this

information would be beneficial in determining the appropriate RCP seal performance for evaluating the impact of loss of seal cooling events in regulatory compliance space.

This report documents the results of the WOG investigations. This new information, in combination with existing testing and field data, supports the Westinghouse position that the RCP seals will not exhibit the large leakage rates assumed in the probabilistic models, after either a loss of all RCP seal cooling or restoration of seal injection. Westinghouse believes that the likelihood of any leakage beyond 21 gpm per pump is remote enough that it should not be used in deterministic compliance assessments.

2. Relevant Background on RCP Seal Performance

In the 1970's and early 1980's a number of RCP seal failures occurred that could be categorized as small loss of coolant accidents. The significant seal failures that have occurred at plants with Westinghouse RCPs and seals are summarized in Table 2-1. These seal failures occurred during plant operation and are not a result of a loss of seal cooling. It is also noteworthy that the causes of the significant seal failures for Westinghouse plants have been addressed through design and operational changes and, as discussed later, are no longer considered to be credible failure mechanisms. These operational occurrences led the NRC to classify RCP seal performance as a generic issue in 1980 and was designated as Generic Safety Issue (GSI) 23, "Reactor Coolant Pump Seal Failure". While the original scope of GSI-23 was RCP seal failures during normal operation, it was expanded to include RCP seal failures caused by station blackout (SBO). The scope was later expanded again to include seal failures caused by the loss of component cooling water (CCW), which was originally designated GSI-65, "Probability of Core-Melt Due to Component Cooling Water System Failures", and the loss of service water (SW), which was originally designated GSI-1531 "Loss of Essential Service Water In LWRs". Thus the scope of GSI-23 included the loss of all seal cooling from normal operation, SBO, loss of CCW, and loss of SW.

Table 2-1 Significant Westinghouse RCP Seal Failures

Plant	Date	Estimated Leak Rate	Water in Containment
H. B. Robinson	May, 1975	300 – 500 gpm	132,500 gallons
Indian Point Unit 2	July 1977	75 gpm	90,000 gallons

The 1975 Robinson event was initiated by rubbing contact of the No. 1 seal faces. The pump was initially secured with leakage controlled by the No. 2 seal. When the pump was restarted, the debris from the rubbing of the No. 1 seal faces destroyed the No. 2 seal and resulted in a large quantity of reactor coolant being discharged directly to the containment. The Indian Point event occurred when the No. 1 and No. 2 seals suddenly experienced increased leakage. Post-incident inspection found foreign particles and No. 1 seal face chipping. It is postulated that the No. 1 seal face rubbed as a result of foreign particles clogging the seal or spalling of the No. 1 seal face. The No. 2 seal was damaged as a result of this debris.

The initial work to identify a resolution for this GSI included research to identify RCP seal failure mechanisms. A large body of work was completed by the RCP vendors, as well as the NRC staff and their contractors. This research addressed the degradation of polymer seals, conditions under which polymer seals would experience extrusion, and the effects of loss of cooling conditions on the primary hydraulic seals, including the conditions under which hydraulic seals may become unstable. The Westinghouse research was both analytical, to understand and predict seal behavior under loss of seal cooling conditions, and

experimental, to test the analytical model under prototypical conditions. A limited amount of operational experience was also used to validate the analytical and experimental research. A discussion of the Westinghouse research and research findings is documented in WCAP-10541, Rev. 2 (Reference 7).

As a result of operational experience and the Westinghouse research and development, a number of changes were made to the Westinghouse RCP seal package. The key changes included the materials used for the No. 1 seal faceplates, the use of O-rings qualified to survive predicted loss of seal cooling temperatures and differential pressures, operational limits for seal performance during normal operation and improved inspection processes. For example, subsequent to the H. B. Robinson and Indian Point events, Westinghouse changed the face material of the No. 1 seal from an aluminum oxide compound to a silicon nitride compound that is more resistant to rubbing damage. The new No. 1 seals began to be installed in W-NSSS plants in late 1980s. Also as a result of the Robinson incident, Westinghouse recommended that the seal return line from the No. 1 seal remain unisolated in the event of excessive leakage through the No. 1 seal to prevent debris from damaging the No. 2 seal. The value of these changes were validated by an event at Braidwood in October 2001 when the No. 1 seal was rubbed at startup, but no damage to the No. 2 seal occurred. Very little foreign material was generated and the No. 2 seal leakage remained at its nominal value. Subsequent to these changes, there have been no instances of RCP seal failures either during normal operation or during actual loss of all seal cooling events.

In the 1980's, the NRC promulgated new regulations related to station blackout (10 CFR Part 50.63) and fire protection (10 CFR Part 50.49 and 10 CFR 50, Appendix R). The institution of these new regulations resulted in the requirement for plants to backfit "coping strategies" to achieve and maintain the safe stable conditions required by the regulations. These strategies and the associated operator actions and equipment are beyond the plants' original design basis. However, since they are required by these newer regulations, the coping strategies and associated features are part of the plant licensing basis. These coping strategies were reviewed and approved by the NRC as a result of licensee submittals showing compliance with the new regulations. The RCP seal performance during the postulated station blackout and fire scenarios required by the regulations includes a loss of all seal cooling. The coping strategies have typically been developed based on RCP seal performance models that the seals will leak at approximately 21 gpm per pump, as documented in WCAP-10541, Rev. 2. These coping strategies have been included in plant specific emergency and off-normal operating procedures. Plant personnel have been trained in the appropriate responses for each condition. NRC inspections have found these coping strategies to be acceptable for over a decade.

In 1996, the WOG approved a recommended change (Reference 8) to the generic emergency response guidelines (ERGs) for the response to the loss of all AC power event. The change recommended that the plant operators use natural circulation cooldown of the

RCS to cool the RCP seals following restoration of AC power, rather than restoring seal injection or thermal barrier cooling. The recommended change indicated that the potential for loss of integrity of the CCW system due to water hammer, if thermal barrier cooling is restored, had not been generically analyzed and therefore this mode of RCP seal cooling was not generically recommended. The change also indicated that the restoration of seal injection could result in RCP shaft warping, which would result in a significant financial impact. Further, the potential for seal damage caused by thermal shock if seal injection is restored, while not expected, had not been assessed in detail. The change was applicable to all Westinghouse plants regardless of the O-ring material. Note that the field experience summarized in Section 4 of this report indicates that neither damage to the CCW system nor damage to the seals has ever occurred when seal cooling was re-established. On the other hand, pump shaft damage has been noted in a number of these instances which, while not a safety issue, has significant financial implications.

Based on the work by Westinghouse (Reference 7) and the NRC's own contractors, the NRC closed Generic Safety Issue 23 in 1999 (Reference 9) based on their conclusion that further generic regulatory actions would not lead to an improvement in safety. In closing this generic safety issue, the NRC also stated that they could not approve WCAP-10541 and that an RCP seal PRA model developed by their contractor, called the Rhodes model (Reference 10) would continue to be used by the NRC staff in probabilistic risk assessments.

In the year 2000, for the expressed purpose of expediting NRC review of risk informed applications, the WOG developed an RCP seal PRA model (Reference 2) that attempted to remove some of the conservatisms in the NRC's Rhodes model. During their review, the NRC requested substantial additional information to justify portions of the WOG model. Based on PRA analyses documented in Appendix A of Reference 11, the WOG concluded that the more conservative NRC PRA models would not significantly change the internal events at-power PRA results (i.e., core damage frequencies), compared to the more realistic model proposed by the WOG in Reference 2. As a result, the WOG chose not to expend the substantial additional resources necessary to further develop a justification for those aspects of the models in WCAP-15603 that the NRC staff continued to question. Two significant improvements in the WCAP-15603 model over the Rhodes model were:

- For plants that have installed the new high temperature O-rings, the model assumption that the O-rings would fail at 2 hours after a loss of all seal cooling and create a 300 gpm per pump leak rate was no longer valid. The new O-rings do not exhibit high temperature degradation and therefore no increased leakage due to high temperature degradation needs to be modeled.
- The probability of a popping and binding failure of the No. 1 seal, leading to a 480 gpm per pump leak rate was reduced by a factor of 2. This reduction was based on the resistance to temperature degradation of the new O-rings, with their inherent stability

and resistance to axial seal friction, would further reduce the possibility of binding of the No. 1 seal. Binding of the No. 1 seal is the dominant failure mode for this seal stage.

NRC approved the new model (see the NRC safety evaluation (SE) in Reference 2) for RCP seal leakage following a loss of seal cooling *for use in probabilistic risk assessments* for plants with high temperature O-rings. The NRC approval indicated that plants not employing high temperature O-rings would be restricted to the use of the Rhodes model. The new NRC approved model included several assumptions that represent additional conservatism beyond that already included in the previous industry PRA seal models (e.g., WCAP-10541). Specifically, the approved model differed from previous models that may have been the basis for plant specific Appendix R coping strategies in two key aspects:

1. The new PRA model (as well as the Rhodes model) conservatively assumes that any increased leakage from the RCP seals as a result of "popping or binding" of the No. 2 seal would occur at 13 minutes after the loss of all seal cooling. The 13-minute time frame is taken from Reference 7 as the approximate time at which pump internal volume will be completely purged and the seal area water temperature will be approaching the reactor coolant system temperature. The WCAP-10541 PRA model assumed that increased leakage might occur at 30 to 60 minutes after the loss of all seal cooling based on deterioration of the O-rings in the seal package. This failure mode is no longer credible for plants that have switched to the new high temperature O-rings. The No. 2 seal popping and binding failure was not the dominant failure mode considered in the WCAP-10541 model.

2. The new PRA model (and the Rhodes model) conservatively assumes that a popping or binding failure of the No. 2 seal (at 13 minutes) will result in a leakage rate of 182 gpm with an associated probability of occurrence of 0.20. The 182 gpm is based on the No. 2 seal opening to the limit of its travel with a subsequent failure of the No. 3 seal due to increased pressure in the chamber between the No. 2 and 3 seals. The old model assumed that a leakage of 300 gpm would occur in the 30 to 60 minute time frame after loss of all seal cooling.

The basis for concerns with seals "popping open" originates from scaled tests (2 7/8" and 4 3/4" test pieces) performed by NRC's contractor, AECL, and documented in NUREG/CR-4077 (Reference 12). AECL used these tests to develop an analytical model to predict seal instability for the Westinghouse seal package as documented in NUREG/CR-4821 (Reference 13). The results of this work have been used by the NRC to justify a potential instability of the Westinghouse seals under a loss of seal cooling condition. As a result, the NRC's PRA model for the No. 2 seal behavior assigns a relatively high likelihood to failure of the seal due to "popping open" and staying in the open position. All of the field experience and test observations to date indicate that the No. 2 seal remains stable and the

total leakage from the seal package is in the range of the 21 gpm value reported in WCAP-10541.

These two PRA model changes are significant if the RCP seal leakage model approved for PRA analyses is used for evaluating compliance with applicable deterministic regulatory requirements. Deterministic coping strategies developed based on the deterministic WCAP-10541 seal leakage model may not be compatible with deterministic compliance evaluations using the approved PRA model in Reference 2.

With respect to the two PRA model assumptions described above, it is Westinghouse's conclusion that the probability of 0.20 for the No. 2 seal to pop open is very conservative. The expected long term leakage rate following a loss of all seal cooling is nominally 21 gpm. This is supported by analytical, test and field experience where seal failures and high leak rates have not been observed.

The WOG recognized (see Appendix B of WCAP-16141) that the perceived effectiveness of some design basis coping strategies may be reduced if they are measured with the new PRA model in the NRC's SE in Reference 2. The WOG took the position in WCAP-16141, and still maintains that the approved PRA model is very conservative and should not be used to judge the effectiveness of coping strategies for the loss of RCP seal cooling. For example, the use of overly conservative assumptions and models can result in coping strategies that may increase risk by suggesting improper prioritization and sequencing of operator actions compared to a more realistic case (see Section 7 of this report). The same conclusion (risk may be increased by the use of conservative models) can be drawn for assessments of deterministic coping strategies using the Rhodes PRA model for plants without the high temperature O-rings.

3. OVERVIEW OF RCP SEAL OPERATION

This section provides an overview of the operation of the 3-stage seals manufactured by Westinghouse for use in Westinghouse reactor coolant pumps. The first subsection provides a brief overview of the design and operation of the RCP and 3-stage seal assembly. A more complete description can be found in WCAP-10541. The second subsection deals with seal operation under normal operating conditions. The third subsection deals with seal operation under a loss of all RCP seal cooling scenario.

3.1 RCP AND SEALS

The RCP hydraulic section consists of an impeller, diffuser, casing, thermal barrier heat exchanger, lower radial bearing, main flange and pump shaft. The seal assembly consists of three seals located in series along the pump shaft just above the main flange. Cooling of pump components above the impeller is provided by the thermal barrier heat exchanger and a seal water injection system. The thermal barrier heat exchanger is located between the impeller and the lower radial pump bearing while the seal injection is introduced between the thermal barrier heat exchanger cooler and the lower radial bearing. Either cooling mode, on its own, is capable of maintaining the pump components, including the bearings and seals, in an acceptable temperature range to prevent damage during operation or beyond design basis accident conditions.

The thermal barrier heat exchanger uses plant component cooling water to limit heat transfer between the hot system water and components above the impeller. The seal injection flow is provided by the charging pumps and enters the pump below the lower radial bearing. Seal injection flow is designed to exceed the flow through the No. 1 seal during normal operation. Part of the seal injection flow (equal to the flow through the No. 1 seal) goes up the shaft to the seal area while the remainder of the flow goes down the shaft and past the thermal barrier heat exchanger where it acts as a buffer between the hot reactor coolant and the remainder of the pump.

The No. 1 seal is the main seal of the pump. It is a controlled leakage (hydrostatic) film riding, radial taper face seal. The primary components are a runner that rotates with the shaft and a non-rotating seal ring that is attached to the lower seal housing. The seal ring can move axially to accommodate axial motions of the shaft. The pressure at the inlet to the seal is reactor coolant system pressure while the outlet pressure is nominally 30 psig. The pressure on the downstream side of the seal is controlled by the flow resistances in the No. 1 seal leakoff line and the volume control tank pressure. The gap between the two seal faces is held a constant distance by the design of the seals where the closing forces (which tend to close the gap between the runner and the ring) are equal to the opening forces during normal operating conditions. The sealing function is pressure dependent and does

not change with shaft rotation (rotation is not required to establish the film of hydrostatic seals). The flow through the No. 1 seal gap, except for the small amount of No. 2 seal leakage, is returned to the volume control tank and the suction of the charging pumps.

The No. 2 seal is a rubbing face seal. This seal directs the majority of the leakage from the No. 1 seal to volume control tank via the No. 1 seal leakoff line. The normal leakage through the No. 2 seal is on the order of 3 gallons per hour and the leakoff is diverted to the reactor coolant drain tank (RCDT).

In the normal mode of operation, the No. 2 seal operates with a differential pressure of approximately 30 psi across the face. The inlet pressure on the seal is normally determined by the volume control tank pressure and the backpressure is determined by the head of water maintained above the No. 2 seal leak off connection and/or pressure in the RCDT. When functioning as designed, the No. 2 seal will typically leak at a negligible rate (e.g., 3 gallons per hour) and any flow through the seal passes to the RCDT. As a result of this very low leak rate through the No. 2 seal, the indicated leakage on the No. 1 seal measured in the No. 1 seal leak-off line typically represents the total flow passing through the No. 1 seal.

On rare occasions, the No. 2 seal will "hang" open following RCP shutdown, as was reported by Ginna (Reference 26) following the August 2001 Northeast Blackout. "Hang" open is very different from the "popping and binding" open and is discussed in this report only to put "hang" open in its proper context. "Hang" open is diagnosed by zero flow in the No. 1 seal leak-off line and an increased flow in the No. 2 seal leak-off line. When a No. 2 seal is described as 'hung' open following RCP shutdown, the term means that the resistance to flow offered by the No. 2 seal is less than the resistance offered by the No. 1 seal leak-off line; it does not indicate that the No. 2 seal offers 'zero' flow resistance. The flow through the No. 1 seal is not significantly changed from its previous value. In these rare instances, the vendor's recommendation has typically been to re-start the RCP (assuming prerequisites for starting the RCP have been satisfied). This causes the No. 2 seal to become fully seated. The mechanical vibration associated with operating the pump and the modest change in the fluid pressure condition at the seal usually are sufficient to overcome the forces which were keeping the seal faces from being in contact. The causes of a "hang" open seal are different from those that are postulated to cause "popping and binding" failures and therefore, the two are not related.

The No. 3 seal is also a rubbing face seal. This seal directs the leakage from the No. 2 seal to the No. 2 seal leakoff line. The normal leakage through the No. 3 seal is on the order of 400 to 600 cc per hour and the leakoff is diverted to the radwaste system.

Within the seal assembly, O-rings provide sealing between components that are in static contact while the channel seal is used for locations where dynamic contact occurs, such as between the No. 1 seal ring and the seal housing insert. The channel seal permits the seal

ring to move axially to follow shaft motions during reactor coolant pump operation and maintain a constant gap as reactor coolant system pressure and temperature change.

3.2 NORMAL OPERATION

During normal operation, the seal injection system provides cooling to the RCP seals and provides a buffer from the high temperature reactor coolant fluid. The thermal barrier heat exchanger limits heat transfer from the hot RCS fluid to pump components above the impeller and also provides a backup if seal injection cooling is lost.

The seal injection is provided by the charging pumps that are used to maintain RCS inventory during plant operation. Seal injection is typically at a pressure of about 2250 psig and near 130 degrees F. The total seal injection flow to each pump is typically 8 gpm. The amount flowing up and down depends on the specific conditions of the installed seal. Nominal flow through the No. 1 seal is 3 gpm and the normal operating range is 1 to 5 gpm.

3.3 LOSS OF ALL SEAL COOLING

3.3.1 NO. 1 SEAL BEHAVIOR

Shortly after a loss of all seal cooling, the RCP will be tripped and will begin a coast down. They will typically be stopped in two minutes. Water passing through the No. 1 seal would initially be the "clean/cool" seal injection water that was in the shaft annulus above the thermal barrier heat exchanger just prior to the loss of all seal cooling.

The time between initiation of the event and the time at which the No. 1 seal is exposed to RCS fluid at cold leg temperatures depends upon the volume of clean/cool water in the shaft annulus and the No. 1 seal leak rate during normal operation. This leak rate does not change as a result of the pump trip and coast down since the gap is determined by the pressures acting on the seal. WCAP-10541 states that the lower internal water volume would begin to be purged within approximately 10 minutes and would be followed by an increase in seal temperature due to the in-surge of high temperature reactor coolant. Approximately 13 minutes following a loss of all seal cooling, the lower pump internals volume would be completely purged and the seal area water temperature would be approaching the 550 degree F reactor coolant system fluid temperature.

As the seal inlet fluid approaches RCS cold leg temperature, an increase in the seal leakage rate is predicted to occur as a result of:

- A change in the viscosity of the fluid due to the temperature change of the fluid,
- A change in turbulent flow characteristics (seal flow is initially laminar) due to the change in fluid density as a result of the temperature change,

- The effect of two phase flow between the seal faces which results from flashing of the high temperature fluid between the seal faces,
- The effect of transient thermal gradients in the seal assembly components, and
- The effect of a change in the pressure gradient across the seal components.

The information presented in WCAP-10541 concludes that the expected seal leakage during a loss of all seal cooling after the hot RCS fluid reaches the RCP seal area is 21 gpm. The calculations supporting this value (with all 3 seals functioning) were independently checked by an NRC consultant, ETEC (Reference 14), and were determined to be conservative. This expected seal performance is based on analytical, test and field service evidence. As discussed further in Section 4, there is no field or prototypical experimental evidence to suggest that the steady state RCP seal leakage following a loss of all seal cooling will be significantly greater than this value.

3.3.2 No. 2 Seal Behavior

The No. 2 seals of the Westinghouse RCP package normally operate in a rubbing face or boundary lubricated mode. However, the No. 2 seal has been deliberately designed to enter a film riding mode when exposed to higher differential pressures, such as the predicted 800 psi differential for a loss of all seal cooling event (see Table 3-1). The No. 2 seal achieves film riding by using pressure induced mechanical deflections to cause convergence of the normally flat parallel faces of the No. 2 seal ring and runner. This results in an increased lifting force that causes the ring to separate from the runner. The leak rate is greater in the film-riding mode because the separation between the sealing surfaces, as well as the differential pressure across the seal face, is greater. Detailed thermal hydraulic and thermal stress analyses (Reference 7) show that during a loss of all seal cooling event, the pressure induced converging angle between the faceplates is reversed by imposing a large thermal gradient across the seal. The sharp thermal gradients cause thermal rotation of the seal faces counter to the rotation mechanically designed into the seal. This thermal gradient is enhanced by the initial presence of subcooled water in the No. 1 seal cavity (the inlet to the No. 2 seal) with a phase change in the No. 2 seal leakoff. As the converging angle between the No. 2 seal faceplates is progressively reduced, the leakage rate is decreased. The reduction in leakage rate results in a lower No. 2 seal leakoff pressure and consequently a lower saturation temperature on the downstream side of the No. 2 seal. The increased thermal gradient causes an even greater reduction in angle between the ring and runner sealing surfaces, resulting in a diverging angle between the surfaces. The leakage through the No. 2 seal is predicted to stop. With the leakage through the No. 2 seal stopped (or at a very small value), the thermal gradient across the No. 2 seal is maintained.

In the event that a No. 2 seal is 'hung' open at the initiation of a loss of all seal cooling event, the expected response would be for the seal to close as discussed above. The

closing forces on the seal faces would be much higher than any friction forces 'hanging' the seal open.

The initial research on the behavior of the RCP seal package was carried out by NRC contractors, as documented in Reference 12. Two-phase blowdown tests using small scale RCP seals showed potential for seal instability and, as a result, more testing was recommended. The tests showed that the No. 1 seal popping or jamming open seems plausible. The potential for No. 2 also popping open was also identified as a major concern.

Further NRC analytical research was carried out and documented in Reference 13. The analytical model of the RCP seals was done using the contractors' existing 2-phase code. The conclusions of the investigation were that the seals would be expected to survive under 2-phase provided that the inlet pressure was sufficiently above saturation (at least 50 degrees subcooled), the backpressure sufficiently high (greater than 50% of inlet pressure), and the seal convergence was sufficiently low. Further, the research concluded that divergent seals can pop open if they leak enough, and that the transient behavior is important because a seal could pop open before thermal distortion takes place (i.e. before the seal rolls closed). The report indicated that seal binding due to friction should only have a minimal effect on performance. The report also was critical of French tests (see Section 4.4) reported in Reference 7, because the test used a 7-inch seal design rather than the 8-inch design used in Westinghouse seals, and No. 2 seals had no wear (even though their tests reported in Reference 12 were even smaller scale seals).

The conclusion of Reference 13 for the Westinghouse design was that saturated inlet conditions, low backpressure and low balance ratio make the second stage seal highly susceptible to popping open. Reference 13 concluded that the popping open behavior was likely to occur during a station blackout. The report concluded that a high seal face divergence will not prevent unstable behavior if the seal faces are worn or scratched enough to permit significant leakage through the seal. The minimum leakage required to cause seal instability varies with fluid conditions and seal face divergence, but there is a reasonable possibility this minimum leakage would be exceeded if used seals have poor surface conditions.

Westinghouse provided evidence (References 24 and 25) that a large degree of subcooling at inlet to seal package would be maintained early in the transient. Flashing downstream of the No. 2 seal and the lower heat capacity of the No. 2 seal results in a thermal gradient which causes the faces to become highly diverging early in the transient under loss of cooling conditions. Extremely large closing loads are predicted to occur in the highly divergent No. 2 seals. Further the Westinghouse analyses showed that the seal would roll divergent before subcooling was lost at the No. 2 seal inlet and therefore a popping open failure due to unstable transient behavior is not expected. Reference 24 noted that the No. 1 seal transient response to a loss of all AC power exposes the No. 2 seal to hot single

phase high pressure water (subcooled) for ten or more minutes. The No. 2 would be stable during such a heat up transient due to the initial low converging taper, low friction and significant submergence. In the French tests, the No. 2 face divergence was calculated (based on pressure and temperature inputs) to have exceeded approximately 1000 micro-radians within 2-minutes of the high temperature transient (~12 minutes into the French test). Two phase conditions in the No. 1 seal cavity were not established until 13.5 minutes after the start of the high temperature transient (approximately minute 26 into the French test).

In response to the AECL contention that damaged or scratched No. 2 seal faceplates would increase leakage through the seal and therefore decrease the thermal gradient, Westinghouse stated (Reference 24), that less than 3.8% of the Westinghouse No. 2 seals have been replaced for damaged or scratched surfaces and that less than 1.4% of the seals would have shown any leakage. Further, Westinghouse noted that the Reference 13 analysis requires significant leakage for a divergent No. 2 seal to open and that the critical balance ratio doesn't apply to diverging seal.

Finally, it is noted that the Reference 13 work also considered Bingham seals (for CE-NSSS plants). For the Bingham seals, Reference 13 reports disagreement between analysis predictions and seal suppliers' tests. The AECL researchers concluded that their analyses in Reference 13 were likely too conservative in nature. These same "conservative" analysis methods were used to predict that the Westinghouse No. 2 RCP seal was likely to pop open at the beginning of the heatup transient. Thus, the conclusions from Reference 13, even without the other analyses, tests and field experience to the contrary, may not represent the expected behavior of the Westinghouse seals.

It is therefore concluded that there are some theoretical conditions under which popping of the No. 2 seal could occur. However, detailed analyses by Westinghouse show that these conditions are not prototypical of a loss of seal cooling event and are considered to be very unlikely to occur. This conclusion is supported by evidence from tests or operational occurrences as described in Section 4 and 5 of this report.

3.3.3 No. 3 Seal Behavior

The behavior of the No. 3 RCP seal only becomes important if the No. 2 seal does not remain functional during the loss of all seal cooling event. An assessment of the performance of the cartridge No. 3 seal, for the case of a No. 2 seal that is partially damaged or is in a film riding mode prior to coming to thermal equilibrium, was carried out as documented in Section 4 of WCAP-10541. The assessment assumed conditions of 400 psig and 445 degrees F on the upstream side of the No. 3 seal. In this case, the high thermal gradient causes a divergent seal face to develop, similar to the No. 2 seal behavior,

which results in high closing loads. For this design, it was concluded that the cartridge No. 3 seal offers significant capability to limit leakage from the RCP seal package.

The performance of the bellows No. 3 seal design is not discussed in WCAP-10541. The bellows design cannot be relied upon to handle significant pressure differentials. Since the bellows seal design is much more common, the integrity of the No. 3 seal is not relied upon in further analyses and evaluations in this report.

3.4 RCP SEAL LEAKAGE ESTIMATES

Estimates of the RCP seal leakage rates for various failures of the RCP seals are discussed in Section 5 of WCAP-10541. These leakage estimates were developed based on a coupling of the following analyses to determine the overall response:

- Detailed thermal stress analyses of the seal response to the thermal and pressure conditions that are predicted during a loss of all seal cooling event to determine the mechanical deformations of the seal components under these extreme conditions,
- Detailed thermal hydraulic analyses of the RCS fluid flow through the torturous pathway through the seal assembly to determine the flow resistances and effects of two phase flow through the seal package.

The results of these analyses, as presented in Table 5-1 of WCAP-10541, are summarized in Table 3-1.

Table 3-1: RCP Seal Leakage for Various Seal Failure Modes

Case	Seal Status			Pressure Differential (psid)			Leakage Rate (gpm)
	#1	#2	#3	#1	#2	#3	
1	Functional	Functional	-	1440	792	-	21.1
2	Open	Functional	-	0	2185	-	75.6
3	Functional	Open	Closed	1656	0	555	56.7
4	Functional	Open	Open	1800	0	0	182.0
5	Open	Open	Open	0	0	0	480.0

For all of the cases with the seal "open", the analyses assumed no flow resistance for the fluid passing through the seal ring portion of the seal assembly. However, the contorted flow passages for the No. 1, 2 and 3 seals were considered in the flow resistance. The 56.7, 75.6, 182.0 and 480.0 gpm leakage estimates should be the maximum leakage rates under those conditions. While the flow resistance through an "open" seal may be minimal, inherent in the zero flow resistance assumption is that the seal fails in a "maximum open" position. However, as discussed in Section 7 of this report, reasonable variations in the actual flow rates would not impact conclusions related to system performance, Therefore,

any conservatisms or uncertainties in the derivation of these flow rates would not be significant in terms of their eventual usage.

3.5 LONGER TERM SEAL PERFORMANCE

In their SE on WCAP-15603, the NRC has accepted that failure of high temperature O-rings in the RCP seals is highly unlikely if the reactor coolant system pressure is reduced to less than 1710 psig within 2 hours of the loss of all seal cooling. The basis for the requirement to show that the RCS pressure is below 1710 psig within 2 hours is based on bounding all of the available test results for O-ring survivability reported in Reference 15.

The referenced tests were conducted by AECL for Westinghouse and are documented in Reference 15. One hundred and twenty samples of high temperature O-ring material from thirty batches were tested. Of these, 100 samples were tested at No. 2 seal conditions. The initial survivability tests were conducted at 550 degrees F and 1200 psi pressure. To simulate worst case conditions, the shaft and bore were assembled fully concentric leaving a crescent shaped gap with a maximum 0.031 inch gap. Also, in order to bound actual conditions, the channels in which the O-rings were fitted were oversized compared to nominal dimensions. Some of the tests were run for 18 hours and others for 168 hours. In all cases, there were no O-ring failures at those conditions. In order to determine the maximum capability of the O-rings, after each test, the pressure was increased until an O-ring extrusion failure occurred. Three different size O-rings were used for the No. 2 seal portion of this test: 0.21 inch, 0.275 inch and 0.139 inch initial cross section diameters. The majority of the O-rings withstood pressures greater than 2500 psig and only eight failed at pressures less than 2300 psig, which is about the maximum RCS pressure that would be experienced based on the pressurizer relief valve capabilities. Six of the eight occurred in the 0.139 inch diameter O-ring tests.

The O-ring thicknesses used in the tests described above were representative of those used in the No. 2 seal stage assembly and therefore would not be subject to full system pressure. As shown in Table 3-1, if the No. 1 seal stage is functioning, the maximum pressure expected across the No. 2 seal O-rings is 792 psid. This is much less than the 1200 psi pressure differential used in the 18 and 168 hour qualification tests. Therefore, no seal failures would be expected following a loss of all seal cooling, regardless of the long term RCS pressure.

3.6 RESTORATION OF SEAL COOLING

If the loss of all seal cooling event continues for any significant length of time (e.g., greater than about 1 hour) it is expected that the shaft seal package will have achieved thermal

equilibrium. At this point, if seal injection is re-established[1] with a total seal leak rate of 21 gpm per pump, the temperature of the No. 1 seal inlet fluid would be on the order of 370 degrees F. This is based on mixing of the colder seal injection water at 8 gpm (this is a nominal value and can vary from 6 to 12 gpm) with the hotter RCS fluid at 13 gpm to match the expected 21 gpm seal leakage. The restoration of seal injection is expected to have beneficial effects on the shaft sealing system. With the reduction in the fluid temperature, the seal leak rate is expected to decrease due to:

- A change in the viscosity of the fluid due to the temperature change of the fluid,
- A change in flow characteristics due to the change in fluid density as a result of the temperature change,
- Any effect of two phase flow between the seal faces may be eliminated due to the lower fluid temperatures at the seal faces,
- The effect of a change in the pressure gradient across the seal.

Following the re-initiation of seal cooling, the leakage rate trend is expected to continually decrease as the lower temperature water at the seal components will decrease the leakage which in turn results in a lower fluid temperature at the seals. The fluid temperature reduction is due to the reduction in RCS fluid flowing through the shaft annulus and seal; the seal injection rate of colder water remains constant and the total seal leakage rate decreases. Over an extended period of time, the seal leakage rate would be expected to return to the nominal 3 gpm value.

Note that the expected seal response for the case in which the RCP seal temperature is decreased by cooldown of the RCS fluid (e.g., as a result of dumping steam from the steam generators) would be a decrease in RCP seal leakage for the same reasons as discussed above. However, in this case, the seals are cooled by the RCS fluid which is decreasing in temperature at a rate in the range of a degree per second, rather than the more rapid cooldown by restoration of seal injection or thermal barrier cooling.

In developing the guidance in Reference 8, Westinghouse assessed the potential seal failure mechanisms and concluded that additional work would be required before restoration of seal injection could be recommended by the seal vendor as a recovery strategy. In developing this recommendation, Westinghouse relied upon over 30 years of RCP design and operating experience involving general reviews of RCP components during station blackout related testing, seal material response to thermal shock under unloaded conditions, inspection of RCP components and sub-components following loss of seal cooling, and

[1] The restoration of RCP seal injection after heatup of the seal package is not recommended in the plant emergency operating procedures for the Loss of All AC Power event (Reference 8). However, coping strategies for loss of all seal cooling events for some plants (e.g., for fires or loss of component cooling water) may include the use of seal injection for RCS makeup. The behavior of the seals following restoration of seal injection after seal heatup has begun is relevant for these plants.

extensive knowledge and understanding of RCP design bases and limitations. The prior operational and testing data (discussed further in Sections 4 and 5) provided useful insights regarding seal performance during loss of all seal cooling and showed that seal failure did not occur during loss of all seal cooling events. Major seal damage was observed only when the RCP was re-started, after seal cooling was restored during these events. However, Westinghouse concluded that, while the present data do not provide an operational or empirical basis for concluding that the seals will fail catastrophically or leak uncontrollably beyond the defined flow range of 3 to 21 gpm, the available information did not provide a formal and rigorous technical basis supporting restoration of seal injection. The additional research to provide the additional information was never initiated.

4. Loss of RCP Seal Cooling

Considering all of the years of RCP operating experience, the loss of all RCP seal cooling during actual plant operation has rarely been experienced. However, the loss of all seal cooling has occurred on a limited number of occasions during pre-operational hot functional testing when the RCS was at pressure and temperature conditions equal to normal operating conditions. A listing of the known loss of all seal cooling (LOASC) events, both during operation and in testing, is provided in Table 4-1. The plant events up to and including the Asco event were reported in WCAP-10541. There have been two loss of seal cooling tests and one significant loss of seal cooling operational event that were not described in detail in WCAP-10541: 1) an EdF test specifically designed to assess seal response to a loss of all seal cooling, 2) an inadvertent loss of all seal cooling during a pump test at the Weir pump test facility in England, and 3) the Maanshan Station Blackout in 2001. These events are discussed in detail in this report.

Table 4-1 Summary of Loss of All Seal Cooling Events for Westinghouse RCPs

Plant	Date	Duration of LOASC	Seal Performance	
			No. 1 Seal	No. 2 Seal
Haddam Neck	April 1968	10 minutes	Nominal	Nominal
R. E. Ginna	May 1969	30 minutes	Nominal	Nominal
Haddam Neck	July 1969	2 minutes	Nominal	Nominal
Indian Point Unit 2	January 1971	60 minutes	Nominal	Nominal
Indian Point Unit 2	January 1971	45 minutes	Nominal	Nominal
H. B. Robinson Unit 2	March 1971	"several minutes"	Nominal	Nominal
D. C. Cook Unit 2	September 1977	8 minutes	Nominal	Nominal
Asco Unit 1	January 1982	9 minutes	Nominal	Nominal
EdF Hot Thermal Shock Test	May 1985	65 minutes	Nominal	Nominal
EdF Cold Thermal Shock Test	1987	Greater than 1 hour	Nominal	Nominal
Weir Pump Facility	1991	~20 minutes	Nominal	Nominal
Beaver Valley Unit 2	July 1999	3 minutes	Nominal	Nominal
Maanshan	March 2001	Greater than 1 hour	Nominal	Nominal

Note: Nominal seal performance for No. 1 seal means that any measured flow in the leak-off line was within the predicted range for a loss of seal cooling event; nominal performance for No. 2 seal means that no significant increase in flow in the leak-off line was indicated.

4.1 Actual Plant Events

Included in Table 4-1 are a number of events that have occurred at commercial nuclear power stations using Westinghouse RCP seals. All of the plant events listed in Table 4-1 from the 60's, 70's and 80's occurred during pre-operational hot functional testing when all of the backup safety equipment was not operational because the core had not yet been loaded. However, the reactor coolant system was at full pressure and temperature for these events, and therefore, the RCP seal behavior is relevant to the seal behavior considerations under a loss of all seal cooling. In each instance recorded in Table 4-1, seal cooling was

restored to end the loss of seal cooling event. Thus the "duration" shown in Table 4-1 is the time at which seal cooling was restored after the loss of seal cooling. .A brief discussion of selected events that occurred during hot functionals, taken from Westinghouse field reports, is provided below.

Ginna (May 1969) – During hot functional testing with the RCS at full pressure and temperature, a loss of all station power occurred with both emergency diesels out of service. CCW was restored in 30 minutes and seal injection recovered in 45 minutes. The restoration of CCW before seal injection is expected to result in less severe thermal transient on the seals, compared to the restoration of seal injection first. There was no indication of abnormal pump seal leakage. The pumps were disassembled and damage to the pump bearings and pump shaft was found. The seals in one pump were damaged by bearing debris. However, no abnormal pump seal leakage occurred.

Indian Point Unit 2 (January 1971) - During hot functional testing with the RCS at full pressure and temperature, a loss of all station power occurred with the emergency diesels out of service. Two of the four RCPs were without seal cooling for 45 minutes; the other two were without seal cooling for 60 minutes. There is no information related to the manner in which seal cooling was restored. The two pumps with 45 minute exposures to high temperatures were operated for the 5 subsequent days of the hot functional tests. This was based on confirmation of pump and seal integrity (i.e., no abnormal pump vibrations and no abnormal seal leakage). Disassembly of the pumps showed that the two pumps with a 45 minute exposure to high temperatures had no seal deviations from normal. The two pumps with a 60 minute exposure duration showed some O-ring abnormalities, but no abnormal leakage past the O-rings. The O-rings in this case were the "old" O-rings that were subsequently phased out by the new high temperature O-rings discussed in Section 3 of this report.

D. C. Cook (September 1977) - During hot functional testing with the RCS at full pressure and temperature, a loss of all station power occurred with both emergency diesels out of service. All four RCPs stopped along with a loss of thermal barrier cooling and seal injection. Hot RCS fluid entered the seal area for 5 to 8 minutes. Seal cooling, via both CCW and seal injection, was restored when power was restored. There was no abnormal leakage although the shafts on all 4 RCPs were bent and had to be replaced.

Asco (January 1982) – The Asco plant is a 3-loop Westinghouse NSSS in Spain. During hot functional testing with the RCS at full pressure and temperature, a loss of all station power occurred with both emergency diesels out of service. All three RCPs stopped along with a loss of thermal barrier cooling and seal injection. The duration of the loss of seal cooling was 9 minutes. For the 93D pumps installed at Asco, when seal injection is restored after a loss of all cooling it will continue to force hot water that is already in the thermal barrier heat exchanger annulus into the shaft annulus and to the seals. The No. 1 seal leakoff

temperatures increased to the range of 160 to 180 degrees C (320 to 356 degrees F) as a result of the loss of seal cooling. Seal cooling (not specified whether CCW, seal injection or both) was restored when power was restored. There was no abnormal leakage although all 3 RCPs experienced high vibration levels following the event. Post event teardown and inspection revealed minor damage to the RCP seals even though no abnormal leakage occurred.

This field experience confirms that the expected RCP seal behavior for a loss of all seal cooling event is a leakage rate through the No. 1 seal that does not challenge the RCS integrity and minimal leakage through the No. 2 and 3 seal assemblies. There is no indication of popping and binding of either the No. 1 seal or the No. 2 seal assemblies from any of the operational experience.

4.2 SIZEWELL B RCP TEST

The Westinghouse Model 100 RCPs for the Sizewell B plant were tested at the Wier Pump Facility in Scotland in 1991 prior to their installation at the plant. During this normal production test a complete loss of seal cooling occurred as a result of process control problems (Reference 16). The RCP was undergoing a loss of seal injection test when thermal barrier cooling was also lost for a time during the test. The data from the loss of seal cooling portion of the test is presented in Figures 4-1 and 4-2. The loss of seal injection was intentionally initiated at approximately 10:30 AM on April 4, 1991. At approximately 12:32 PM, power was lost to the pump providing CCW to the RCP thermal barrier heat exchanger, resulting in a temporary loss of all seal cooling. Within 2 to 3 minutes, the RCP was tripped. Approximately 14 to 16 minutes after the loss of CCW, the No. 1 seal leakoff flow began increasing from a nominal flow of 3 to 5 gpm to a value of 12 gpm at 20 minutes after loss of CCW. As the seal leakoff flow increased, the seal leakoff temperatures also increased. The bearing temperature RTD located in the seal injection plenum area indicated an increasing temperature as the seal inlet plenum was heated by 550 degree F water from the pump. The bearing RTD reached the limit of its indicating range (~400 degrees F) at 18 minutes and remained off-scale high for 22 minutes (until after seal injection and CCW were restored for the first time). At 20 minutes, CCW was regained temporarily and the No. 1 seal leak off flow peaked at approximately 13.5 gpm at 24 minutes and began decreasing. At 22 to 26 minutes, a small amount of seal injection flow was indicated. At 26 minutes, CCW was again lost. At 30 minutes, seal injection was restored at a flow of approximately 8 to 9 gpm. CCW was again restored at 34 minutes. Seal injection was stopped at 38 minutes. The bearing RTD also returned to on-scale readings (i.e., less than 400 degrees F) at 38 minutes. CCW was lost again at 42 minutes and restored at 48 minutes. The bearing RTD decreased below 300 degrees F by 50 minutes and continued a slow decrease thereafter.

As the data in Figures 4-1 and 4-2 indicate, the loss of both seal injection and thermal barrier cooling for 20 minutes (the initial portion of the transient) resulted in a significant thermal transient in the pump and the seal package. Although the actual temperatures of the pump bearing and the seals are not known because the limits of the instrumentation were exceeded, it is evident that the temperatures were in the range where seal performance was affected. The seal leakage, as measured in the No. 1 seal leak-off line, remained below 14 gpm for the entire transient. The No. 2 seal exhibited a stable response with no significant leakage through the No. 2 seal. There was no damage to the seal package as a result of either the high temperatures or the cold thermal shock that followed re-initiation of seal cooling. Following this event, the loss of seal injection test was restarted and run for 50 hours with no adverse impact on the seal package.

This occurrence during the pump test demonstrates the reliability of the seal design with respect to rapid thermal transients. That is, the postulated popping and binding failures for the reactor coolant pump seals were not observed even though the seals were subjected to a thermal transient prototypical of a loss of all seal cooling event. Also, upon restoration of seal cooling, there was no damage to the seal package due to cold thermal shock.

Figure 4-1 Sizewell B Loss of Cooling Pump Test Hydraulic Conditions

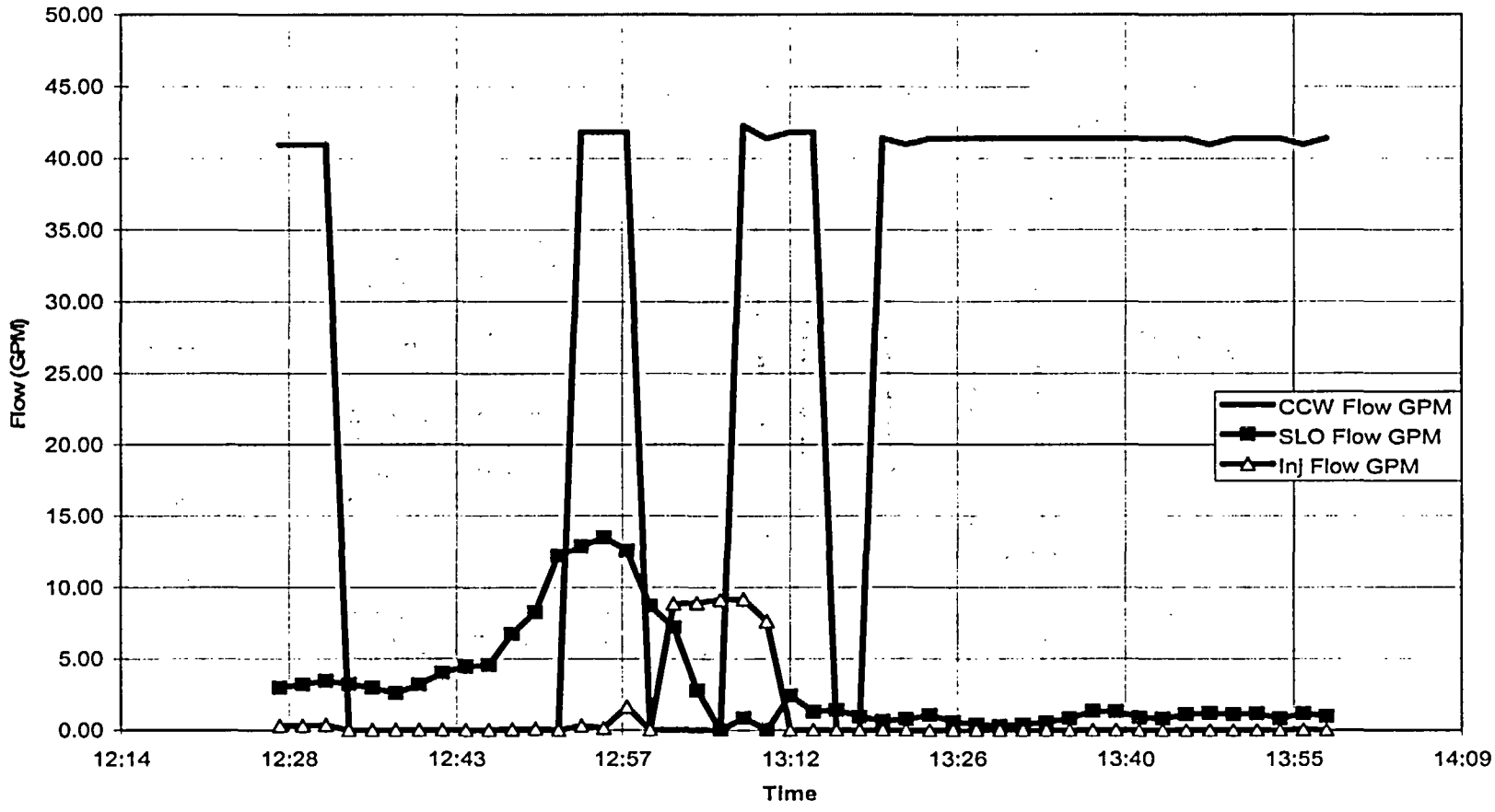
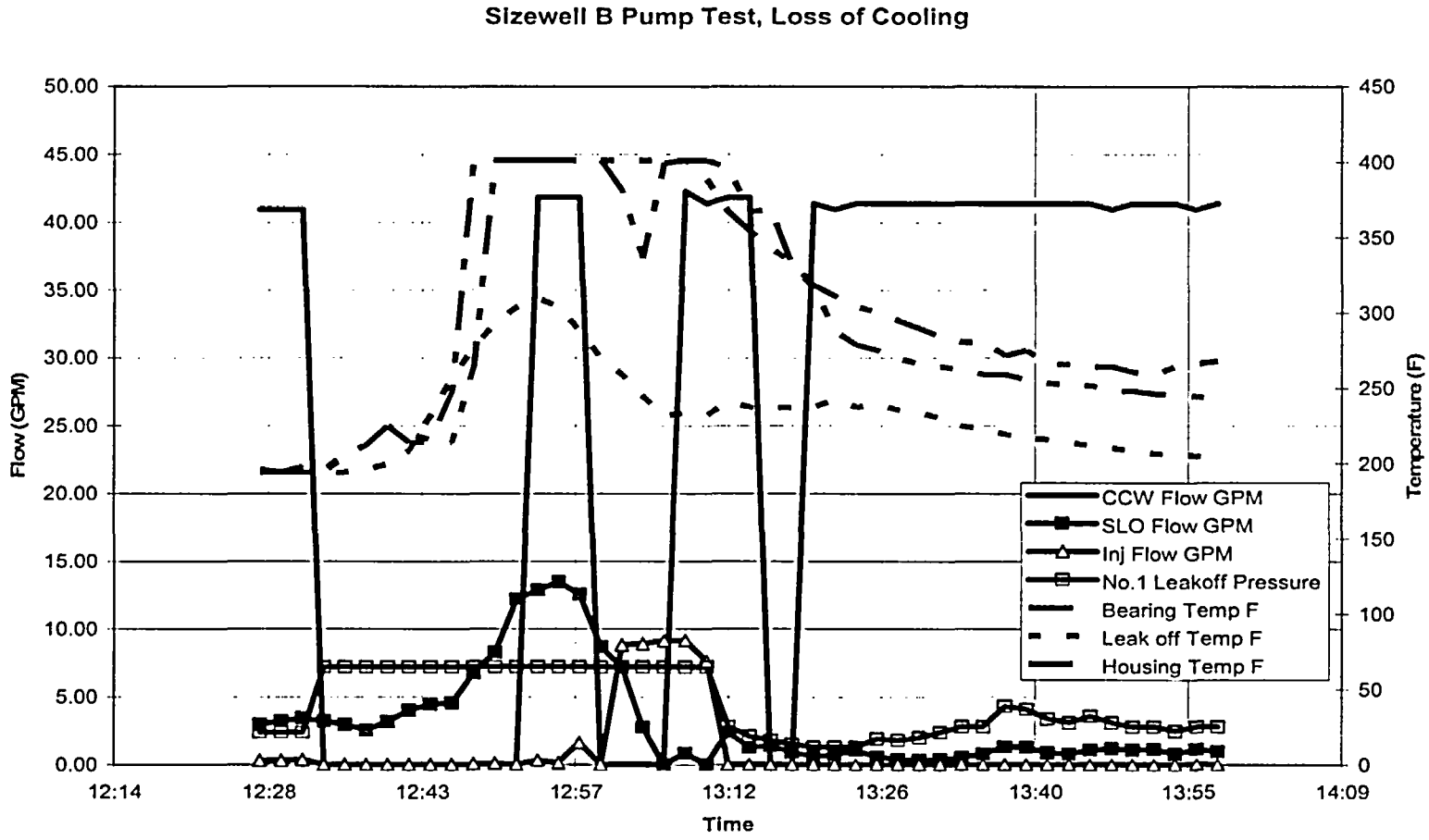


Figure 4-2 Sizewell B Loss of Cooling Pump Test Temperature Conditions



4.3 MAANSHAN STATION BLACKOUT

The Maanshan Nuclear Power Station experienced a station blackout event in March 2001. Maanshan is a Westinghouse 3-loop PWR with large dry containment located in Taiwan. It has a licensed full power reactor rating of 2775 MWth. The NSSS design is very similar to the Shearon Harris and V.C. Summer nuclear stations. The RCPs are Model 93A-1 pumps with cartridge seals.

There is not single reference from which all of the applicable information related to the performance of the RCP seals can be constructed. Therefore, three independent reference sources have been used to determine the sequence of events and the plant behavior. The first, by the National Science Commission in Taiwan (Reference 17), provides a detailed chronology of the event. Reference 17 also includes a plot of reactor coolant system pressures and temperatures during the event, as shown in Figure 4-3. The second is taken from a presentation by the MAAP Users Group to the NRC (Reference 18). The purpose of using this reference is that it contains very clear graphics of key plant variables as obtained from the Maanshan plant computer. Of lesser importance to the issue being addressed in this report, is the comparison of the actual reactor coolant system parameters with those predicted by two simulation codes. The third reference is a compilation of raw data from the event by Steve Rosenau of Duke Energy for the WOG RCP Working Group (Reference 19). This data was obtained through direct communication with the Manshaan engineering staff.

The event was initiated by a reactor scram that occurred on March 17, 2001 at 03:22 hours due to an unstable transmission line. The plant was placed in a hot standby condition. At 00:45 hours on March 18, 2001, with the reactor coolant system at nominal operating pressure and temperature, all offsite power was lost to the station due to further grid instabilities. At that time, both the "A" and "B" diesel generators failed to supply power to the plant emergency AC buses. This resulted in a classical station blackout which occurred at 21 hours 23 minutes after reactor scram. AC power to the station emergency bus was restored by a diesel generator at 02:51 hours on March 18th. The CCW pumps were started at 34 minutes after AC power was restored, but thermal barrier cooling was not restored. Makeup to the reactor coolant system using a charging pump was delayed another 10 minutes (44 minutes after AC power was restored), but seal injection was not restored. Thus, while the station blackout lasted about 2 hours 6 minutes, RCS makeup was not restored until 2 hours 50 minutes after the event initiation. The RCP seals were cooled by the RCS during the RCS cooldown that was initiated at about 45 minutes into the event. The use of RCS natural circulation to cool the RCP seals, rather than re-establishing seal cooling, is consistent with the guidance provided in Reference 8. The sequence of events, as provided in Reference 17, is shown in Table 4-2.

Reference 18 indicates that the SGs were used to cooldown and depressurize the RCS after the SBO initiation, after the turbine driven AFW pump was started. Reference 18 also indicates that accumulator injection began at 1 hour 45 minutes into the event, which would be at approximately 02:30 hours. As will be discussed later, this is consistent with the RCS pressure reduction reaching the nitrogen overpressure in the accumulators at this time.

Table 4-2 Maanshan Station Blackout Sequence of Events	
00:45:10	LOSS OF OFFSITE PWR SIGNAL TRAIN A
00:45:10	LOSS OF OFFSITE PWR SIGNAL TRAIN B
00:45:10	DIESEL GENERATOR A START
00:45:10	DIESEL GENERATOR B START
00:45:10	D/G A LATCH BLOCKED
00:45:10	D/G B LATCH BLOCKED
00:45:15	D/G TRAIN A GENERATOR GROUND DETECTION - GROUNDED
00:57:05	AFWS (TD) TRAIN A ACTUATION
00:57:15	STEAM TO AUX FW PUMP TURBINE OPEN
02:51:41	5TH D/G B BREAKER CLOSE
02:51:41	LOSS OF OFFSITE PWR SIGNAL TRAIN B NOT ACTUATED
02:51:41	AFWS (TD) TR B ACTUATION
03:13:59	NONSAFETY COOLING WATER PUMP START
03:26:34	CCW PUMP START
03:36:00	CENTRIFUGAL CHARGING PUMP B START

Reference 17 provides a plot of selected plant thermal hydraulic parameters during the event, which is reproduced in Figure 4-3.

Reference 18 provides a number of plots of the plant behavior as recorded on the plant computer, as well as simulations of the event using both the MAAP computer code by Fauske and Associates, Inc. and the MELCOR code by INER of Taiwan. Relevant plots from Reference 18 are provided as Figures 4-4 through 4-9 and are discussed below.

The steam generator pressure history is shown in Figures 4-4 and 4-5. The initial drop in the pressure of SG "B" is due to an operator action to admit steam to the turbine driven AFW pump. The operator action to intentionally depressurize the steam generators is indicated by the decrease in SG pressure which was initiated at about 45 minutes for SG "A" and about 1 hour for SG "B". Figure 4-5 shows the RCS pressure which initially decreased slightly due to starting the turbine driven AFW pump and then decreased more rapidly at about 1 hour in response to the SG depressurization. This plot shows that the RCS remained near the operating pressure for the first hour of the event before dropping to approximately 600 psig by the time AC power was restored. Figures 4-7 and 4-8 show the RCS hot and cold leg temperatures for the event. The RCS temperatures remained in the range of 550 degrees F (285 degrees C) for the first hour of the event. The temperatures show that natural circulation was maintained and slowly dropped to about 400 degrees F (200 degrees C) in response to the RCS cooldown with the steam generators. Finally, the pressurizer level history, as shown in Figure 4-9, shows that pressurizer level remained on-

scale for the first hour of the event. The pressurizer level was eventually lost due to RCS shrinkage when the RCS cooldown was initiated.

There is excellent agreement between the plant data reported in References 17 and 18. The MAAP and MELCOR analyses, that were developed to duplicate the event, modeled zero RCP seal leakage. The agreement between the two simulation codes and the actual event supports the independent observation from References 17 and 19 that the RCP seal leakage during the station blackout event, which resulted in the loss of all RCP seal cooling, was minimal.

There are several noteworthy features concerning the plant behavior in the actual event with respect to determining the RCP seal leakage:

- The seals were exposed reactor coolant fluids at nominal RCS operating conditions for the 45 minutes of the event before the RCS cooldown and depressurization was initiated. Even thought the loss of all seal cooling event occurred at hot shutdown conditions, the seal performance would be no different than if the loss of all seal cooling had occurred at full power. Therefore the RCP seal response at Maanshan during this event is applicable to the more generic determination of seal response under a loss of all seal cooling.

- The RCS pressure response is consistent with a low amount of inventory loss from the RCS. A large seal leak (e.g., 182 gpm per pump) would have resulted in an immediate decrease in RCS pressure to somewhere in the range of 1300 psig (9 MPa). Instead, the RCS pressure stayed in the range of 1900 to 2250 psig (13 to 16 MPa) until the operators began an RCS cooldown using the SGs. The pressurizer level also remained high until this time. Changes in RCS pressure and temperature, as well as pressurizer level, can be predicted by changes in SG pressure and do not reflect any appreciable inventory loss from the RCS. For example, the analyses presented in Section 6 of this report shows the loss of pressurizer level in less than 30 minutes and an RCS pressure if about 1300 psig at 30 minutes for a 182 gpm per pump leak. Pressurizer level was maintained for the Maanshan event and RCS pressure remained above 1900 psig until RCS cooldown was initiated at 45 minutes.

- Upon reaching the pressure of the nitrogen cover gas in the ECCS accumulators, the Reference 18 material indicates that accumulator injection began. However, the plot of RCS pressure would indicate that very little accumulator water was injected into the RCS as the RCS pressure stabilized and was maintained at the nominal accumulator cover gas pressure of 600 psig (4 MPa). It can be concluded that no appreciable RCS inventory loss had occurred up to this point and that accumulator injection did not play a role in maintaining the core mixture level.

Reference 19 contains information related to the RCP lower bearing temperatures and the No. 1 seal leak-off rate. This information is reproduced as Figure 4-10. For the Model 93A-1 pump, the bearing and seal inlet temperatures are measured from two redundant RTDs at the same elevation at the seal inlet. The bearing / seal inlet temperature and No. 1 seal leak-off began to rise at about 15 minutes after the loss of all AC power, as expected. The bearing temperature exceeded the instrumentation range of 329 degrees F for approximately 7 hours, which is well past the time at which RCS makeup was restored. From Figures 4-7 and 4-8, the RCS cold leg temperatures at the end of the reported data (3 hours) were about 365 degrees F (185 degrees C). This agrees with the reported RCP seal inlet temperature of greater than 329 degrees F in Figure 4-10. While the reported RCS temperature data does not go out to 7 hours, it is entirely conceivable that it would be in the same temperature range as the reported seal inlet temperatures from Figure 4-10. This gives another indication that seal cooling was not re-initiated when either CCW pump flow or charging pump flow was re-established at 03:26 hours and 03:36 hours respectively.

Figure 4-10 also shows that the No. 1 seal leak-off flow rate initially stayed at its nominal value of 1.1 gpm for 28 minutes before increasing to a value higher than the 6 gpm range of the leak-off measurement instrumentation and then quickly fell to a value of about 5 gpm within about 10 minutes. The short period of high RCS leakage is within the expected behavior of the RCP seals, as previously observed in the tests and operational experience. The long term leakage decreased to about 4 to 5 gpm as RCS pressure was reduced. At about 4 hours after the loss of all seal cooling the No. 1 seal leak-off rate was reduced to about 0.3 gpm. This change in seal leak-off rate is different than the pump bearing behavior where the temperature did not decrease until much later in time. Further investigations are continuing to understand these apparent discrepancies.

Additional information from Reference 19 indicates that neither seal injection nor thermal barrier heat exchanger cooling were re-established on March 21st following the recovery of the CCW pump at 03:26 hours and the recovery of the charging pump at 03:36 hours. The No. 1 seal leak-off flow (at an RCS pressure of 125 psig) for the three pumps was between 0.06 gpm and 0.18 gpm. This is well within the predicted leakage at that pressure for normally functioning RCP seals.

On March 23rd, the RCS was pressurized to 396 psig and each of the three RCPs was tested to determine if there were either shaft or seal irregularities. The test runs ranged from 30 seconds to 24 hours. In each case, the RCP seal leakage was in the range of 0.47 to 0.72 gpm, as measured at the No. 1 seal leak-off. This is within the normal operating range for this pressure and temperature. Also, there was no abnormal pump vibration or shaft run-out observed. For additional confirmation, the Loop "A" RCP seal was disassembled and inspected. The No. 1 seal faceplate showed no cracking or other visual damage. The O-rings were also determined to be unaffected by the high temperatures.

The overall conclusion from the reported plant behavior for the Maanshan station blackout is that there was no large, uncontrolled RCS leakage following the loss of all seal cooling initiated at nominal RCS operating conditions. The RCS pump seal behavior is consistent with previous analytical, test and operational experience as described elsewhere in this report.

Figure 4-3 Maanshan Station Blackout RCS Behavior

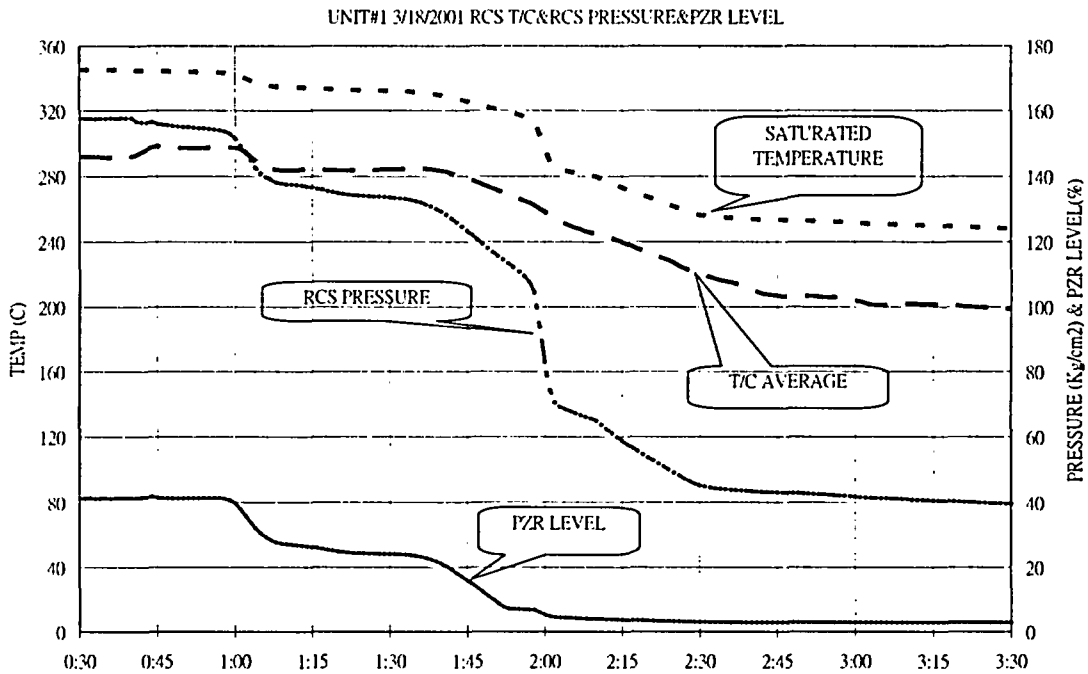


Figure 4-4 Maanshan SG "A" Pressure History

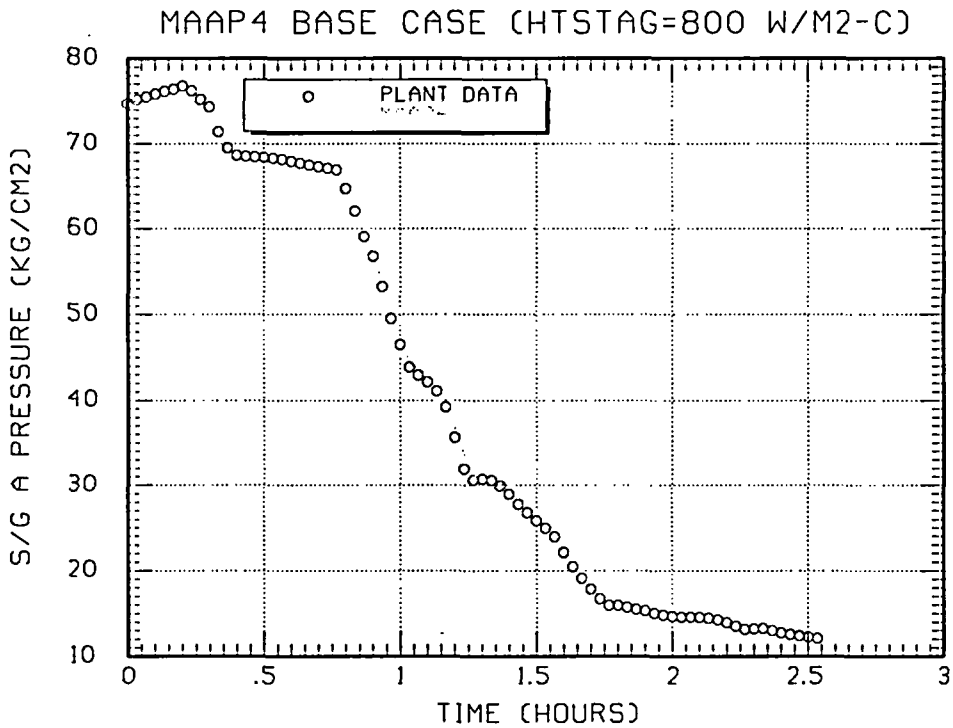


Figure 4-5 Maanshan SG "B" Pressure History

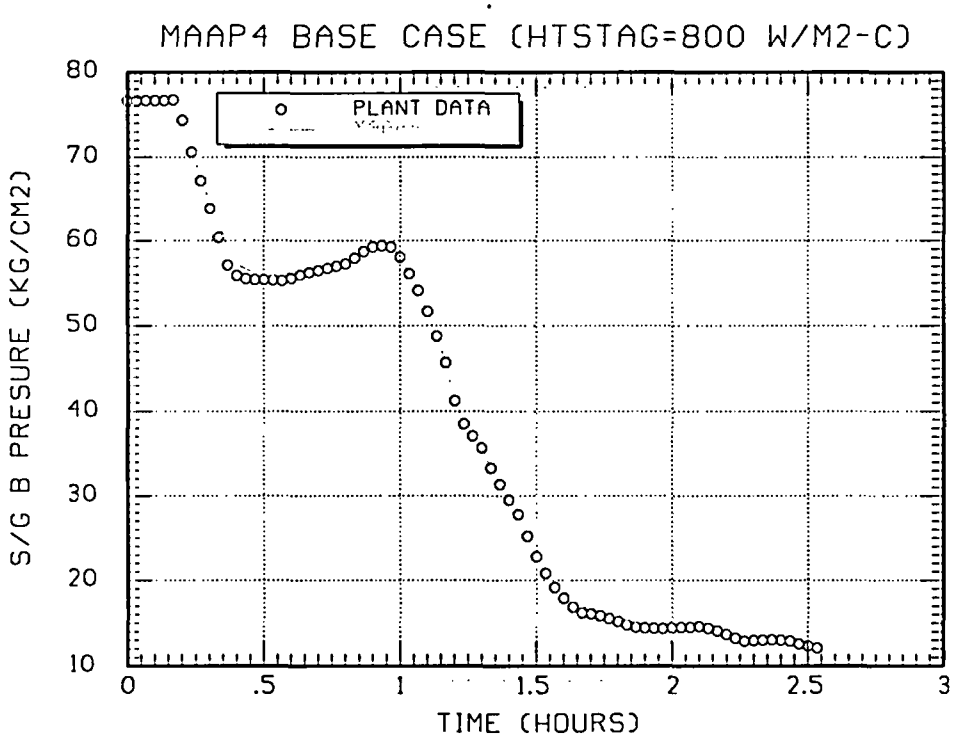


Figure 4-6 Maanshan Pressurizer Pressure History

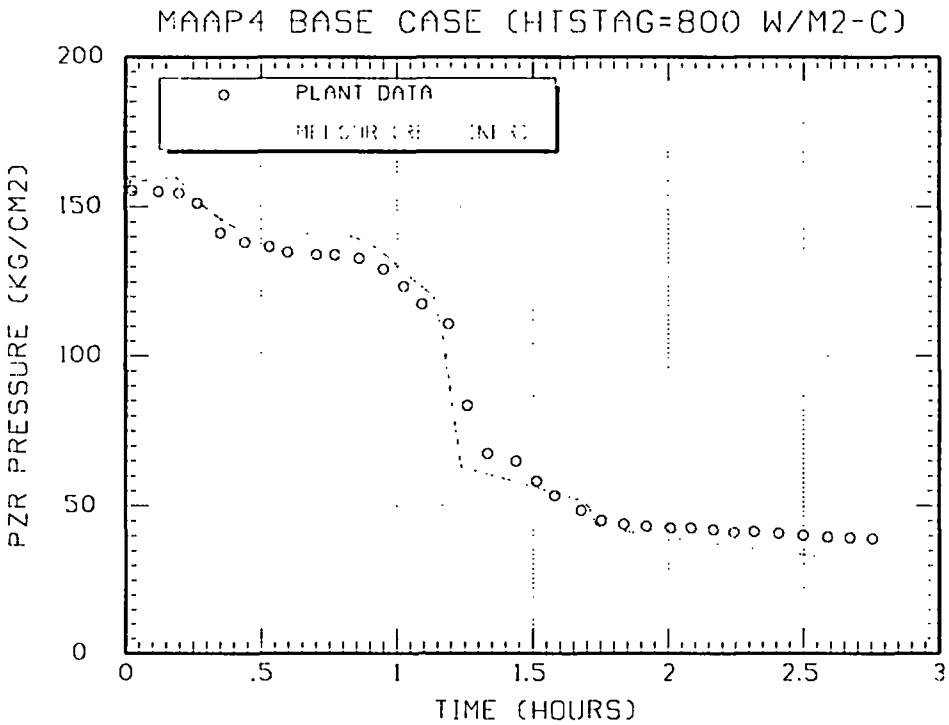


Figure 4-7 Maanshan Loop A Pressure and Temperature History

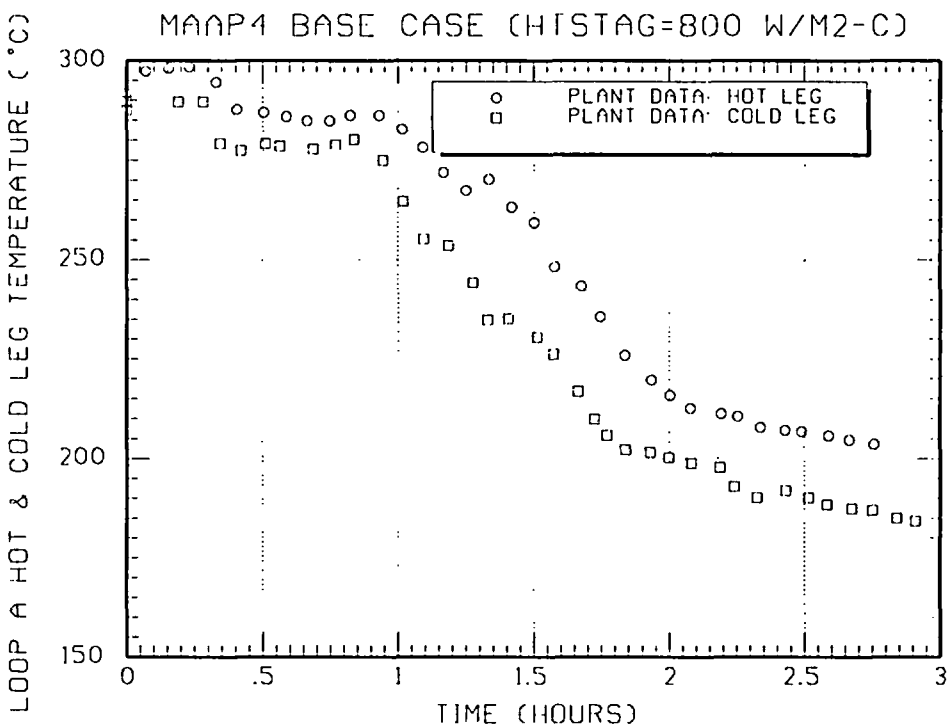


Figure 4-8 Maanshan Loop "B" Pressure and Temperature History

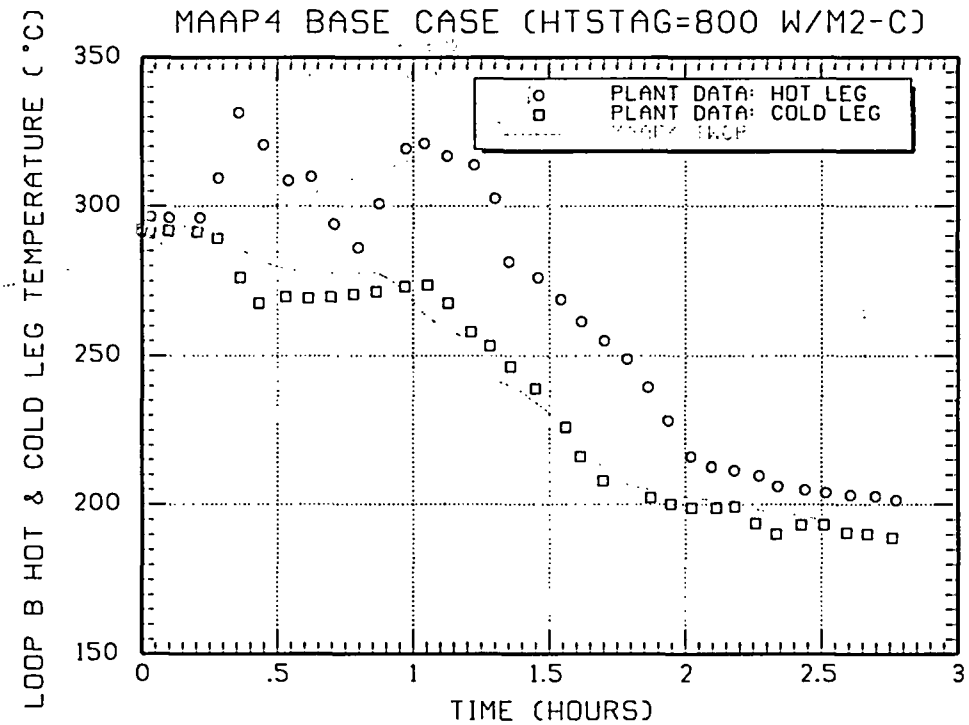


Figure 4-9 Maanshan Pressurizer Level History

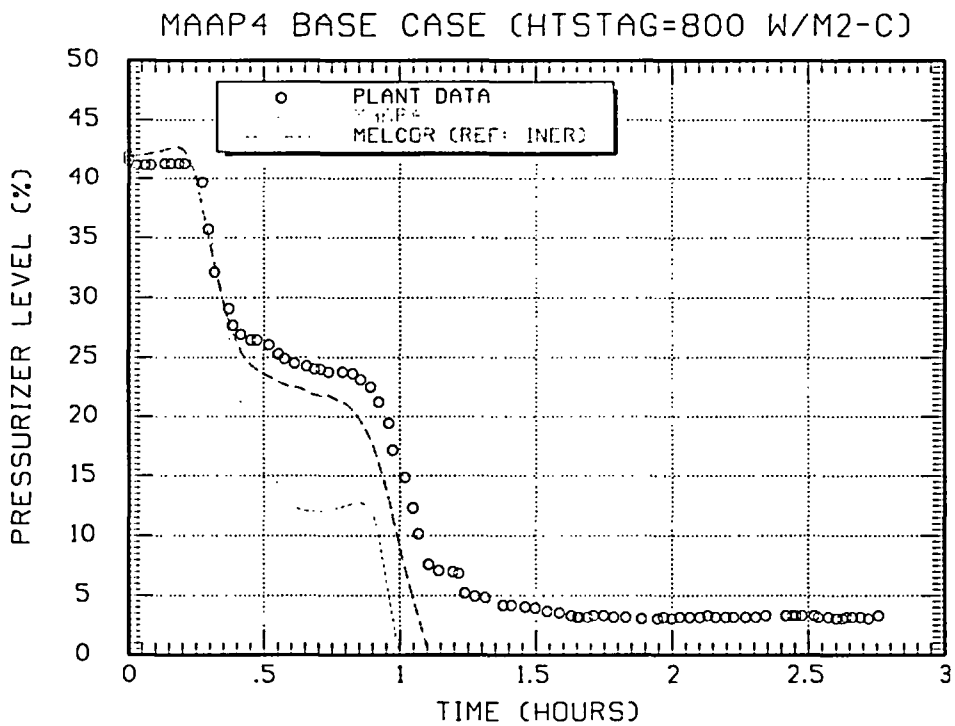
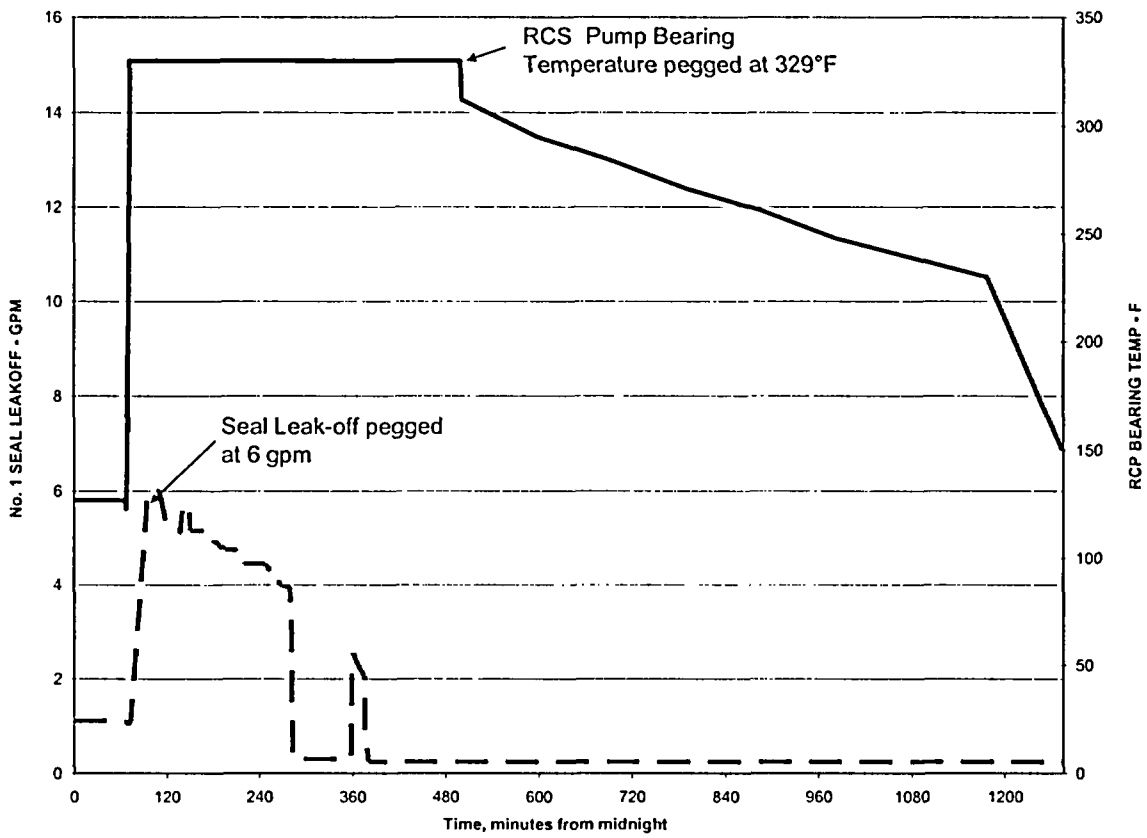


Figure 4-10 RCP Seal Leakage and Bearing Temperature



4.4 EDF HOT THERMAL SHOCK TEST

In 1985, a full scale test of a 7-inch Model 93D reactor coolant pump seal was performed in Montreau, France. The 7-inch RCP seal used in that test was very similar to the 8-inch RCP seal used by Westinghouse. The results of that test are reported in WCAP-10541. The seal was exposed to prototypical hot RCS conditions (pressure and temperature) for over 30 minutes. As reported in WCAP-10541, the leakage through the No. 1 seal increased from a value of 3 gpm before the hot fluid came in contact with the seal package to an equilibrium value in the neighborhood of 16 gpm. As the seal was first subjected to the high temperature fluid, the No. 1 seal it exhibited a transient flow response in which the leakage peaked near 60 gpm. This transient response was predicted by analyses based on changes in the fluid conditions and distortion of the seal faces. This transient leakage rate was of very short duration and decreased from 60 gpm to about 28 gpm. This decrease corresponds to the development of pressure in the No. 1 seal leak-off cavity before eventually decreasing to the 16 gpm steady state value. This is consistent with the expected steady state leak rate of 21 gpm for an 8-inch Westinghouse seal. There was no appreciable increase in the leak-off flow from the No. 2 seal during this test.

Although this test has previously been reviewed in great detail during the resolution of GSI-23, it is worth noting again in this new context. In the resolution of GSI-23, the focus of the investigations was on the performance of the No. 1 seal (i.e., the leakage rate through the No. 1 seal during a loss of all cooling event) and the subsequent behavior (i.e., survivability) of the O-rings in the seal package when exposed to hot RCS fluid. The O-rings were equivalent to the "old" O-rings originally used in the Westinghouse seal package before the new high temperature O-rings were introduced. In part, as a result of this test, the leakage rate of 21 gpm through the No. 1 seal was accepted as an expected result. It was also concluded that the old O-ring material did not degrade and result in increased seal leakage when exposed to hot RCS fluid. While it was concluded that these O-rings were capable of sustaining their sealing function during contact with high temperature fluid, "new" O-rings were developed to provide a high level of assurance that the seal integrity would be maintained for prolonged exposure to high temperature fluids. In the present context of the performance of the No. 2 seal during a loss of all seal cooling event where the No. 2 seal is exposed to hot RCS fluid, it is worthy of note that the No. 2 seal performed as expected and no seal popping or binding of the No. 2 seal occurred.

4.5 EDF COLD THERMAL SHOCK TEST

In 1987, another test was conducted at test was conducted in Montreau, France to assess the impact of a cold thermal shock to the Model 93D 7-inch shaft seals, following a loss of all seal cooling event. The main purpose of this test, as documented in Reference 20, was to observe the behavior of the RCP seal package under a simulated restoration of seal cooling

to a hot seal package. This is discussed further in Section 5 of this report. However, in order to achieve the hot seal conditions for the test, the seal package underwent a heatup transient that is similar to what would be expected for a loss of all seal cooling event.

As discussed in Section 5, the test stabilized the seals at the standard operating conditions with a reported No. 1 seal leak rate of approximately 3.2 gpm at 89.6 degrees F, a leak rate for the No. 2 seal of less than 0.26 gallons per hour, and essentially no leakage through the No. 3 seal.

Initiation of the hot thermal shock was then accomplished in two steps, as discussed in Section 5, which led to a flow of fluid at prototypical RCS conditions of approximately 21.5 gpm, which resulted in a thermal transient at the rate of 316 degrees F/min. The seal leakage initially peaked at approximately 75 gpm at about 1 minute after initiation of the transient, and then quickly decreased to 33.5 gpm. As the test conditions were stabilized at 536 degrees F and 2250 psig, the leak rate was observed to trend down to 28.5 gpm. During the transient and just prior to the 1 hour stabilization at 536 degrees F and 2250 psig, the No. 2 seal leak rate was calculated to be less than 0.2gpm.

This test further confirms that the expected RCP seal behavior for a loss of all seal cooling event is a long term leakage rate of approximately 21 gpm through the No. 1 seal and very low leakage through the No. 2 and 3 seal assemblies. As in all previous test and operational experience, there is no indication of popping and binding of either the No. 1 seal or No. 2 seal assemblies.

4.6 Loss of Seal Cooling Conclusions

The behavior of the Westinghouse reactor coolant pump seals following a loss of seal cooling event has been investigated through analytical, experimental and field experience. All of the available information shows that the expected RCP seal behavior for a loss of all seal cooling event is a long term leakage rate through the No. 1 seal of approximately 21 gpm and a very low level of leakage through the No. 2 and No. 3 seal assemblies. In all of the test and operational experience, there is no indication of popping and binding of either the No. 1 seal or No. 2 seal assemblies.

The loss of all seal cooling is beyond the original design basis for the reactor coolant pump seals. However, for reactor accidents that could result in a loss of all RCP seal cooling, each licensee is required to define and implement coping strategies to assure that the plant can achieve and maintain a safe stable state as part of the licensing basis for the plant. While some NRC studies suggest that certain seal failure modes cannot be definitively ruled out, operational and test data does not indicate that they are likely to occur. In other design and licensing basis areas, a single test is sometimes sufficient to provide reasonable assurance that a component will operate as designed. For example, in environmental

qualification testing, a single test of a component according to a rigorous procedure is sufficient to provide confidence of the components' operability under conditions enveloped by the qualification test. In the new performance based regulatory oversight environment, if there is no test or operational data to question the performance of a component under prototypical conditions, then it should be assumed that there is reasonable confidence that the component will perform as intended. It is reasonable to apply this rationale to the reactor coolant pump seal performance under a loss of all seal cooling. The test and operational experience with the RCP seals under loss of all seal cooling conditions supports the conclusion that the seals will perform as intended, with a controlled steady state leakage in the range of 21 gpm per pump. This leakage rate should be used in deterministic regulatory assessments where the performance of the RCP seals following a loss of all seal cooling is part of the overall assessment.

In both the French hot and cold thermal shock tests, a transient behavior of the RCP seals is noted where the leakage through the No. 1 seal increases to 60 and 75 gpm respectively before decreasing to a value in the range of 21 gpm. This transient behavior was predicted by Westinghouse as a result of the transient pressure and temperature response of the seal package. However, this transient response in the French hot thermal shock test was interpreted by some NRC contractors to be an indication of seal instability. They claimed that the leakoff only decreased because the test facility could not maintain the high flows through the seal. This same transient behavior is likely to have occurred at both the Sizewell pump test event and the Maanshan Station Blackout event, but the measurement capability of the leakoff flow was limited and was therefore unable to fully measure any transient behavior. In both of these cases, the capacity of the "reservoir of hot fluid" was large enough that significant RCP seal leakage rates could be maintained for a significant period of time that seal instabilities could have been maintained. However, the recorded seal leakage and observations from both Sizewell and Maanshan indicates that the leakage behavior was as expected and that no instabilities in seal behavior existed. Therefore, the instability assertion is further challenged by actual data.

5. RESTORATION OF RCP SEAL COOLING

This section discusses actual events and tests to illustrate the expected behavior of the RCP seals if cooling is restored to a hot seal package.

5.1 OPERATIONAL EXPERIENCE

As discussed in Section 4.1, a number of losses of all seal cooling events, followed by a restoration of seal cooling, occurred during pre-operational hot functional testing. In all cases the reactor coolant system was at full pressure and temperature. In each instance, seal cooling was restored to end the loss of seal cooling event.

The key events from Table 4-1 are the 1969 Ginna event, the 1971 Indian Point 2 event, the 1977 D. C. Cook event and the 1981 Asco event:

- The loss of all seal cooling at Ginna occurred during hot functional testing with the RCS at full pressure and temperature, CCW was restored in 30 minutes and seal injection recovered in 45 minutes. The restoration of CCW before seal injection most likely resulted in less severe thermal transient on the seals, compared to the restoration of seal injection first. The pumps were disassembled and damage to the pump bearings and pump shaft was found. The seals in one pump were damaged by bearing debris. However, no abnormal pump seal leakage occurred.
- The Indian Point Unit 2 event also occurred during hot functional testing with the RCS at full pressure and temperature. Two of the four RCPs were without seal cooling for 45 minutes; the other two were without seal cooling for 60 minutes. There is no information related to the manner in which seal cooling was restored. After confirmation of pump and seal integrity (i.e., no abnormal pump vibrations and no abnormal seal leakage), the two pumps with 45 minute exposures to high temperatures were operated for the 5 subsequent days of the hot functional tests. Disassembly of the pumps showed that the two pumps with a 45 minute exposure to high temperatures had no seal deviations from normal. The two pumps with a 60 minute exposure duration showed some O-ring abnormalities, but no abnormal leakage past the O-rings.
- The D. C. Cook event occurred during hot functional testing with the RCS at full pressure and temperature. Hot RCS fluid entered the seal area for 5 to 8 minutes before seal cooling, using both CCW and seal injection was restored. There was no abnormal leakage, although the shafts on all 4 RCPs were bent and had to be replaced.
- The Asco event occurred during hot functional testing with the RCS at full pressure and temperature. The duration of the loss of seal cooling was 9 minutes. For the 93D pumps installed at Asco, when seal injection is restored after a loss of all cooling it will continue to force hot water that is already in the thermal barrier heat exchanger annulus into the shaft annulus and to the seals. The No. 1 seal leakoff temperatures increased to the range of 160 to 180 degrees C (320 to 356 degrees F) as a result of the loss of

seal cooling. There was no abnormal leakage, although all 3 RCPs experienced high vibration levels following the event. Post-event teardown and inspection revealed minor damage to the RCP seals even though no abnormal leakage occurred.

Even though the duration of the loss of seal cooling for some of the other events listed in Table 4-1 is less than the time predicted for hot RCS fluid to reach the seals, the seals may have been exposed to hot RCS fluids for a limited period of time. When all seal cooling is lost, RCS fluid would begin to move past both the thermal barrier cooler and the seal injection point, and travel up the shaft to the seal area. When seal injection is restored, any hot RCS fluid above the point where seal injection enters the pump annulus would be "pushed" upward toward the seal area. The amount of hot RCS fluid above the seal injection entry point would depend on the pump model (the amount of water in the pump annulus above and below the seal injection entry point and the seal leakage rate before the loss of seal cooling. While some mixing of the hot and cold fluids would occur as they travel up the shaft, the fact remains that for all but the shortest duration loss of seal cooling events (i.e., the 1969 Haddam Neck event and the 1971 H. B. Robinson event), the seals may have been exposed to hot RCS fluids for a period of time. However, the thermal inertia of the seal components would limit their short term heatup and subsequent cold thermal shock response. In any case, none of the field reports indicate that excessive leakage occurred. In several instances, damage to the pump shaft and bearings was noted.

This operational data suggests that the restoration of seal cooling for a hot seal assembly will not result in a significant increase in RCP seal leakage due to damage to the seals.

5.2 SIZEWELL B RCP TEST

As discussed in Section 4.2, the test of the Westinghouse RCP intended for the Sizewell B Station in England experienced an accidental loss of all cooling transient in which cooling was restored after the seal package had heated up. As indicated in Figures 4-1 and 4-2, when CCW was restored after a 20 minute loss of all seal cooling episode, the indicated bearing temperature was off-scale high at 400 degrees F. From this, it can be concluded that the seals were also heated to a high temperature. The cold shock thermal transient on the reactor coolant pump seals introduced by the first restoration of thermal barrier cooling was not as severe as might be predicted for a restoration of seal injection. However, the observed seal performance during the thermal transient, in which no significant increase in seal leakage occurred upon restoration of seal cooling, is another indication of the survivability of the Westinghouse seal package under thermal transient conditions.

5.3 EDF COLD THERMAL SHOCK TEST

In 1987, EdF conducted a test (Reference 20) to assess the impact of a cold thermal shock to the RCP seals, following a loss of all seal cooling event. Figure 5-1 depicts the schematic

for the testing system. The test loop utilized energy available from a 250 MW coal fired plant to develop the rapid temperature increase and decrease associated with the postulated event.

The test initially established standard operating conditions to verify normal functioning of the seal set. This was demonstrated with a reported No. 1 seal leak rate of approximately 3.2 gpm at 89.6 degrees F, a leak rate for the No. 2 seal of less than 0.26 gallon per hour and essentially no leakage of the No. 3 seal.

After allowing the seals to stabilize for approximately 1 hour, seal injection water temperature was increased to 149 degrees F over a period of 1 hr and then allowed to stabilize for approximately 15 minutes. Initiation of the hot thermal shock was then accomplished through two successive operations. In the first, the by-pass valve (flow from point 17 to 18 on Figure 5-1) was opened to permit an additional 5 gpm flow to pull enough hot water into the seal inlet to get the desired temperature transient. This resulted in an initial seal inlet temperature increase from 149 to 185 degrees F in approximately 90 seconds (22.5 degrees F per minute). The by-pass valve was then opened further to permit a flow of approximately 21.5 gpm, which resulted in a thermal transient at the rate of 316 degrees F per minute.

The shaft sealing system responded to the transient over a period of 22 minutes. The No. 1 seal leak rate peaked at approximately 75 gpm at about 1 minute after initiation of the transient, then quickly decreased to 33.5 gpm, followed by a second spike at approximately 40 gpm, and then a decrease to 25 gpm and finally stabilizing at 30 gpm. As the test conditions were stabilized at 536 degrees F and 2250 psig, the leak rate was observed to trend down to 28.5 gpm. During the transient, and just prior to the 1 hour stabilization at 536 degrees F and 2250 psig, the No. 2 seal leak rate was calculated to be less than 0.2 gpm.

Following stabilization of the seals at 536 degrees F and 2250 psig, a cold thermal shock event was initiated to simulate seal injection flow of 2 cubic meters per hour (8.8 gallons per minute). In parallel with this operation, the flow of the hot injection fluid was decreased by the same amount that the cold fluid was being injected.

Figure 5-2, reproduced from Annex 3 of the EdF report, shows a schematic (not to scale) of the time history of the pressure, temperature and injection flow during the entire test. This schematic is useful to understand the behavior of all of the key parameters over the entire test sequence. The circled numbers below the x-axis represent the various phases of the test and the time scale on the x-axis presents the duration of each phase. Note that the x-axis is not linear. The parameter values on the figure generally represent the value of the parameter at the beginning and the end of each phase of the test. In the case of seal injection flow, this does not accurately depict the actual flow history during the phase where injection was re-initiated. Figure 5-3, reproduced from Planches 8, 9 and 10 of the EdF

report, depicts the No. 1 seal upstream temperature, the No. 1 seal upstream pressure and the seal injection flow (at 86 degrees F). This plot shows the actual values of these three parameters as a function of time during the phase of the test when cold water was re-introduced to the seal.

From this plot, the following observations are made:

$t_0 - t_{2min}$ Injection Flow (DF) = 0 gpm
t_{2min} Injection Flow (DF) = 0.25 cubic meters per hour (1.1 gpm)
$t_{2min} - t_{2min45sec}$ Injection Flow (DF) = 0.25 cubic meters per hour (1.1 gpm)
$t_{2min45sec} - t_{3min}$ Injection Flow (DF) = increase to 0.5 cubic meters per hour (2.2 gpm)
$t_{3min} - t_{3min45sec}$ Injection Flow (DF) = decrease to 0.375 cubic meters per hour (1.65 gpm)
$t_{3min45sec} - t_{5min15sec}$ Injection Flow (DF) = increase to 2 cubic meters per hour (8.8 gpm)

These observations indicate that seal injection was re-established at both a prototypical flow (8.8 gpm) rate and temperature (86 degrees F) over a maximum of 3 minutes and 15 seconds. Discounting the small transients in the injection flow prior to $t_{3min45sec}$, the flow can be reasonably considered to have been re-established over a 90 second period. The rate at which the seal injection flow was re-established and the temperature of the seal injection water are to be representative of expected conditions in a plant which has experienced a loss of all seal cooling event.

This conclusion is further supported based on a comparison of the actual and predicted temperatures at the seal following restoration of seal injection. A simple energy balance of an 8.8 gpm water stream at 86 degrees F mixing with a 15 gpm flow at 550 degrees F shows the expected fluid temperature would be 373 degrees F. The flow rates of 8.8 gpm and 15 gpm were taken from the test data. This agrees closely with the measured seal inlet temperature of 370.4 degrees F. Thus it is concluded that the cold thermal shock was representative of restoration of seal cooling after a loss of all seal cooling.

Figure 5-1 Model 93D RCP Thermal Shock Test, Process and Instrumentation Diagram

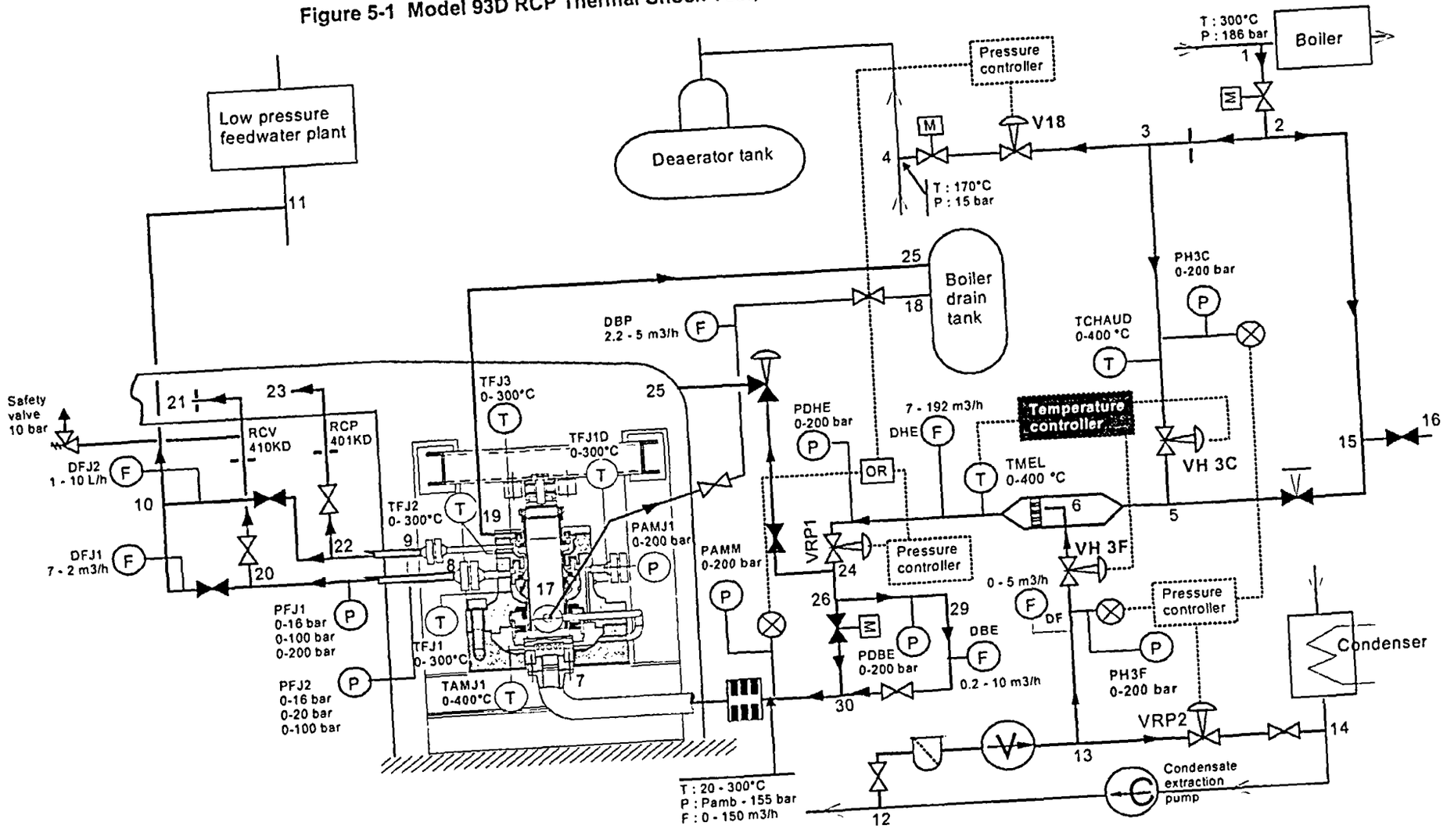


Figure 5-2 Representation of Key Parameters from Thermal Shock Test

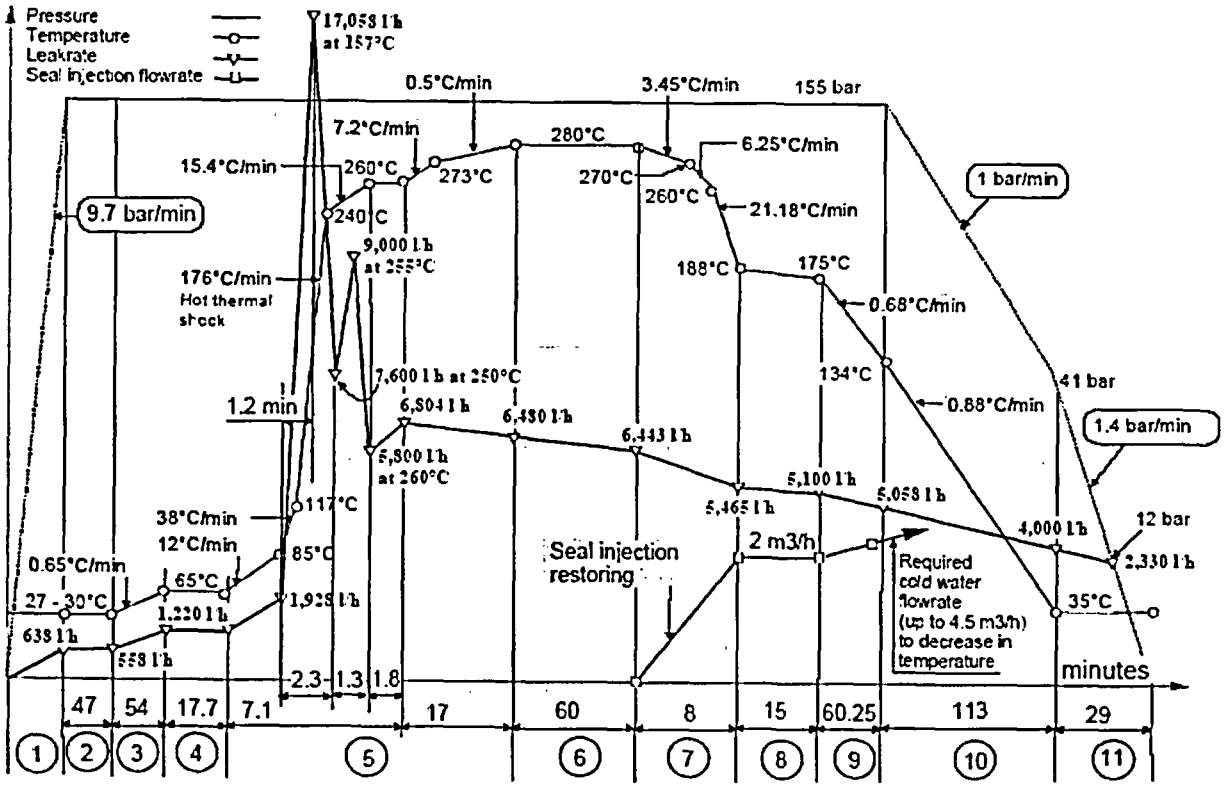
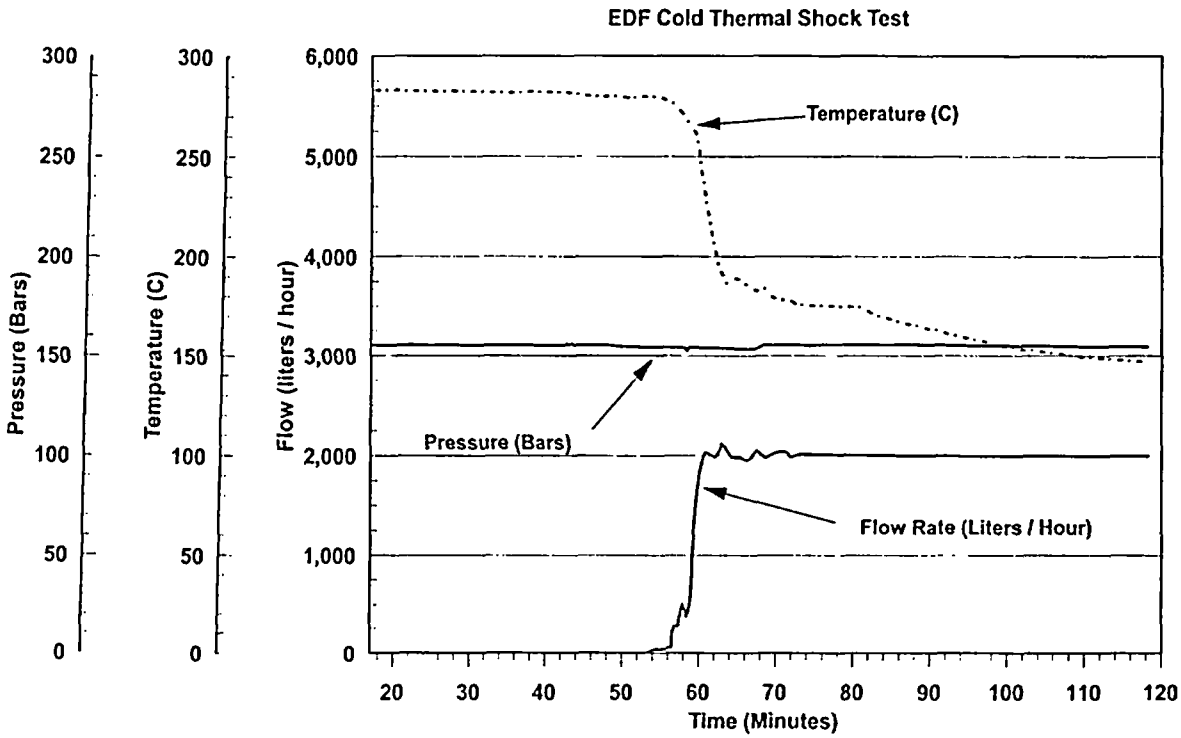


Figure 5-3 Plot of Key Parameters from Thermal Shock Test



5.4 RESTORATION OF SEAL COOLING CONCLUSIONS

The behavior of the Westinghouse reactor coolant pump seals following restoration of seal cooling has been investigated through analytical, test and field experience. All of the test and field experience shows that the no significant increase in leakage through the seal package occurs after the restoration of seal cooling. While some damage to the pump shaft and seals has been observed following these events, there has been no evidence of increased seal leakage. However, the analytical basis to support this conclusion has not been thoroughly developed.

The guidance provided by the WOG for the Station Blackout event (Reference 8) recommends that the plant operators use natural circulation cooldown of the RCS to cool the RCP seals following restoration of AC power, rather than restoring seal injection or thermal barrier cooling. The recommended change was specifically directed to the station blackout event where restoration of AC power would permit injection to the RCS using the safety injection pumps. Therefore, it would not be required to re-establish seal injection in order to achieve a safe stable state. The recommendation also considered the potential damage to the RCP shaft and the extended down-time for repairs that might result if seal injection is restored to a hot seal package. Further, Westinghouse explained in Reference 8 that it is not expected that any increase in leakage would occur if seal injection was restored to a hot seal package.

However, the loss of seal cooling and the subsequent restoration of seal cooling is beyond the design basis of the seal package. Design basis type analyses have not been performed to provide a robust assurance of acceptable seal performance under these conditions. However, the test and field experience do not provide an operational or empirical basis for concluding that upon seal cooling restoration, the seals will fail catastrophically or leak uncontrollably beyond the defined flow range of 3 to 21 gpm.

6. Safety Assessment of a Loss of Seal Cooling

A safety assessment was undertaken by the WOG to determine the impact of large leakage rates (e.g., 182 gpm per pump) from RCP seals on core coolability when RCS makeup is delayed, as in an Appendix R fire scenario. As previously discussed, many of the fire coping strategies were developed meet Appendix R criteria based on the 21 gpm per pump leak rate documented in WCAP-10541. When a larger pump seal leak rate is postulated, the coping strategies may not be able to prevent the loss of pressurizer level. Even though the expected RCP seal performance is a 21 gpm per pump leak, an investigation was undertaken to determine the core response to such an event where makeup to the RCS may be delayed for up to one hour. The analyses show that, even though the very stringent Appendix R criterion for hot standby cannot be met with the 182 gpm per pump leak, there is a large margin to core uncovery and long term core cooling can be maintained. This provides additional evidence that the health and safety of the public are protected even when highly unlikely RCP seal leakages are assumed. Therefore uncertainties in the seal model are adequately addressed in the deterministic safety assessment of Appendix R coping strategies.

6.1 Reactor Coolant System Response

The assessment of the thermal hydraulic response for a generic reference Westinghouse PWR is detailed in Reference 21. The purpose of these analyses was to document the core and RCS behavior following a loss of all RCP seal cooling that result in a 182 gpm leak in each RCP pump seal with the longest time delay to recovery of RCS makeup using an SI pump.

The analyses were performed with the NOTRUMP computer code for the 3-loop design, using conservative analysis assumptions regarding heat removal capability. The base case used a 25% degradation in "nominal" SI flow capability. Based on experience, the 3-loop NSSS is typically the most challenging from a small break LOCA standpoint, and as such is used here for representative cases. The 3-loop design, in combination with the conservative analysis assumptions for heat removal and the 25% degradation of injection flow, provides results which are representative of the thermal hydraulic response for W NSSS plants in general.

The analyses used similar initial conditions and assumptions. These are summarized below.

1. *Initial Conditions (General)*
   a. Operational parameters (initial flow rates, temperatures, etc.) based on plant operation at 100% of Licensed Power Level
   b. RCS leakage of 12 gpm per Tech Specs (taken to be the same location as the RCP seal)
   c. Letdown based on representative value for plant class
   d. Plant in steady state operation

e. Seal leakage location is taken at top of cold leg consistent with the location of the seals in relation to the cold legs

2. *Post-Trip Conditions (General)*
   a. Initial RCP seal leakage of 21 gpm per pump
   b. Decay heat based on nominal 1979 ANS standard
   c. Break flow is subcooled critical flow based on Henry-Fauske Critical Flow Model
   d. All rods assumed to insert on reactor trip signal (RTS)
   e. No operator actions for 30 minutes

3. *Post Trip Conditions (Event Specific)*
   a. Simultaneous loss of RCP seal cooling and RCS injection flow
   b. RCP seal leakage increases to 182 gpm per pump at 13 minutes after loss of seal cooling.
   c. RCS cooldown is limited to natural circulation conditions with the secondary side at the MSSV setpoint pressure. No operator actions to influence cooldown are credited.
   d. Auxiliary feedwater is provided by turbine driven auxiliary feedwater pumps. One turbine driven AFW pump provides feedwater to two of the three SGs. This is conservative since RCS heat removal occurs via only two SGs, thereby degrading the core heat removal capability via the SG secondary sides.

Selected plots for the base case are provided in Figures 6-1 through 6-3. For this case, the pressurizer empties at about 1500 seconds and the vessel starts to develop a vapor space in the upper plenum at about 1700 seconds. Up to this time, all the mass lost from the primary system (through the RCP seals) comes from the pressurizer. The change in RCS inventory is quite rapid during this time frame since the primary system undergoes a subcooled decompression with the higher density for the break flow. Note that the RCS remains at natural circulation cooldown conditions during this time frame. At about 2000 seconds, the vessel mixture level reaches the top of the hot legs where it starts to stabilize due to mass contribution from the upper head, SG tubes and the RCS loops. Most of this mass is from Loop 3 where the lack of feedwater to the Loop 3 SG causes a significant boil-off in this SG. The resultant reduced heat transfer causes natural circulation to break down in this loop which leads to a drain-down of the tubes.

RCS makeup is initiated at 3600 seconds. From 3600 to 14,000 seconds, the vessel mixture level remains essentially constant due to the loss of RCS fluid through RCP seal leakage being made up by the ECCS injection flow. As ECCS inventory is introduced during this time frame, the RCS starts to re-pressurize because make-up flow is greater than break flow. With the 25% degradation of the ECCS flow, the pressure never fully recovers to nominal hot standby conditions. Since no credit for operator action to depressurize is assumed in the analysis, RCS pressure remains essentially constant in the long term.

Based on the results of the analysis, it is concluded that a stable condition is reached with no core uncovery occurring. Accordingly, the RCS is amenable to cooldown and depressurization using the SGs and/or other steps necessary to reach shutdown cooling entry conditions without the risk of core uncovery.

Case 2 has the same initial conditions and assumptions as Case 1, except for the RCS injection flow. Case 2 assumes full 100% ECCS flow instead of the 25% degradation assumed in Case 1. Selected plots for Case 2 are provided in Figures 6-4 through 6-6. The analysis indicates that although the mixture level decreases below the top elevation of the hot leg as in Case 1, after about 7000 seconds the RCS level is regained due to sustained ECCS flow that is not degraded. Pressurizer mixture level is also regained beyond this time frame due to the increased ECCS flow. As the pressurizer fills, the RCS pressure also increases and levels out at about 1700 psia. The larger ECCS flow in relation to the break flow between about 3600 seconds and 12000 seconds substantiates results in the regaining of the vessel and pressurizer mixture level and increasing the RCS pressure. The equalization of the break flow and the ECCS flow beyond the 12000 second time frame suggests stabilization of the levels and pressure. Another difference between Case 1 and Case 2 is the regaining of natural circulation flow conditions in Loop 3 for Case 2 unlike that seen for Case 1. The increased ECCS flow for Case 2 provides adequate cooling in Loop 3 resulting in the re-establishment of natural circulation conditions. For Case 2, the RCS mass is also regained as it nearly reaches the original mass beyond the 12000 second time frame. Again, this occurrence can be traced to the increased ECCS flow.

Selected plots for a third case (Case 3) are provided in Figures 6-7 through 6-9. This case has similar conditions as for the second case, but has a larger ECCS flow injection delay of 5000 seconds (~83 minutes), instead of the 1 hour delay that is assumed for the first case. The most significant difference between the results for this case and Cases 1 and 2 is an episode of minor uncovery of the core (mixture level below the elevation of the top of active fuel). It shows that the vessel mixture level decreases very rapidly initially resulting in uncovering the top of the hot leg at about 2500 seconds. This is followed by minor core uncovery at about 15000 seconds. The cause of the uncovery is the increased delay in ECCS flow injection. The delayed ECCS injection causes further depletion of the RCS inventory in relation to the makeup flow (compared to Cases 1and 2), resulting in the minor uncovery of the core. Further delays would likely result in degradation of natural circulation flow and more severe subsequent core uncovery. The results for this case show a stability trend in the 5 to 6 hour time frame which indicates the core is not in imminent danger of severe uncovery and that RCS cooldown and depressurization actions would likely place the plant in a stable, cold shutdown condition.

The results of the analyses lead to the following conclusions for W NSSS plants:

- A loss of pressurizer level would occur at about 25 minutes for Appendix R fire scenarios with increased RCP seal leakage modeled in conformance with the NRC "approved" PRA model for RCP seals (~ 182 gpm per pump at 13 minutes).
- The core remains covered and natural circulation is maintained for Appendix R fire scenarios with a 182 gpm per pump seal leak at 13 minutes if makeup to the RCS is restored within one hour and a secondary side heat sink is available.
- Even though the mixture level in the reactor vessel drops below the top of the hot leg, natural circulation cooling in the active steam generator loops.
- The RCS make-up flow capability (typically one SI pump) for the various Westinghouse NSSS designs that is bounded by this analysis is shown in Table 6-1.

Table 6-1 RCS Makeup Flow Rates for Short Term and Long Term Analyses

Pressure (PSIA)	2 Loop Plant Makeup (lbm/s)	3 Loop Plant Makeup (lbm/s)	4 Loop Plant Makeup (lbm/s)
1500	36.8	36.8	56.3
1400	38.9	38.9	59.5
1300	40.0	40.0	61.2
1200	41.6	41.6	63.6
1100	43.1	43.1	65.9
1000	44.2	44.2	67.6

- Minor core uncovery may occur for Appendix R fire scenarios with increased RCP seal leakage (~182 gpm at 13 minutes) and delays in restoring RCS makeup that are greater than about 80 minutes.
- For Appendix R scenarios with increased RCP seal leakage (182 gpm at 13 minutes), operator actions to cooldown and depressurize the RCS using the steam generators would increase the available time window for successfully restoring RCS makeup prior to core uncovery.

These analyses demonstrate that, even with an assumption of an initial 182 gpm leak rate and the use of conservative licensing basis codes, continued long term core cooling can be maintained, if makeup to the reactor coolant system restored within about 80 minutes after the loss of all seal cooling. At the termination of these analyses, the plant is at a stable condition and subsequent operator actions, per the plant emergency operating procedures, can be implemented to bring the plant to a cold shutdown.

While the expected RCP seal response is a 21 gpm per pump leak, these analyses provide another layer of assurance that the health and safety of the public would not be compromised if the leakage was 182 gpm per pump, as assumed in the NRC assessment in Reference 1.

Figure 6-1: Case 1 Vessel Mixture Level

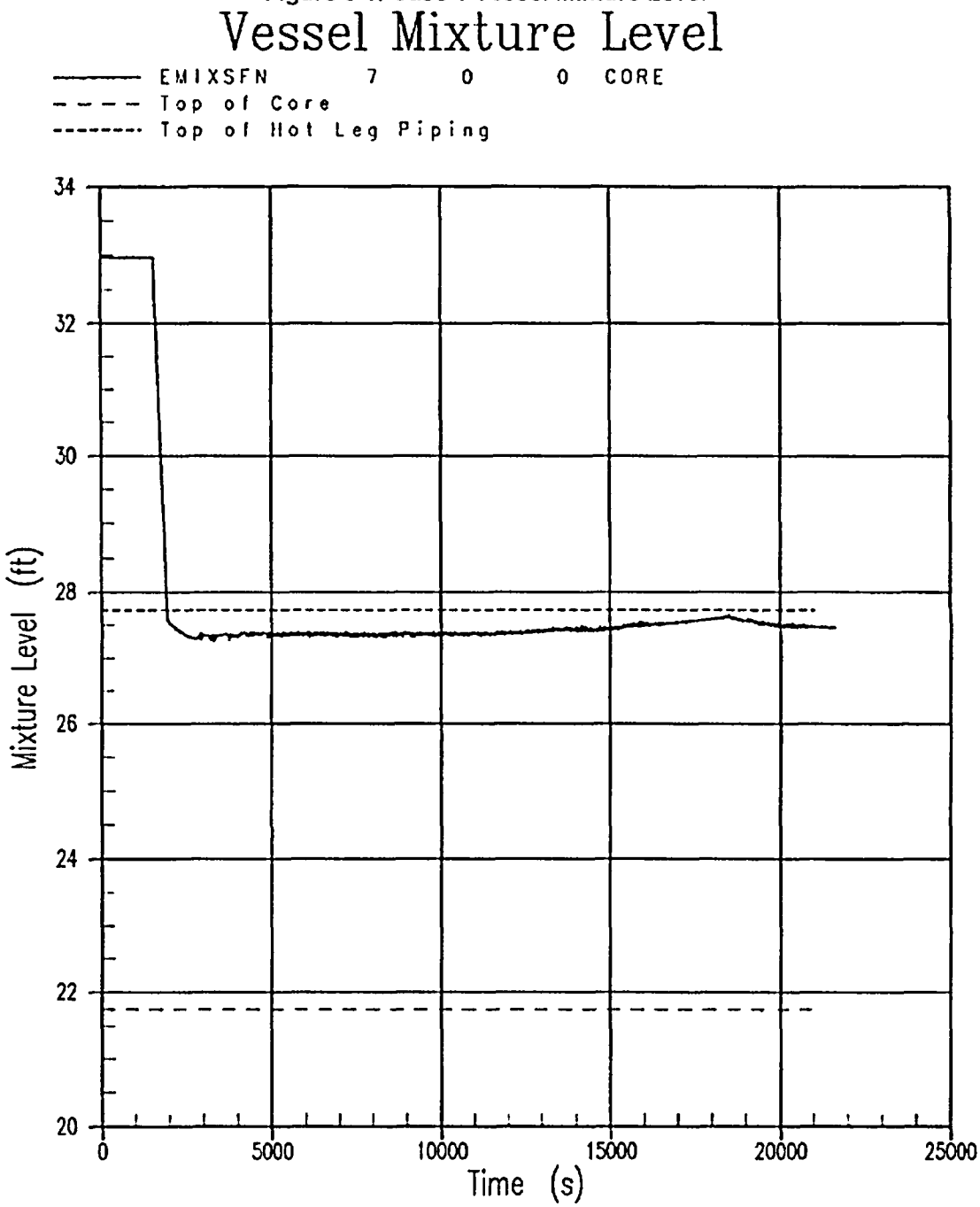


Figure 6-2: Case 1 Pressurizer Pressure

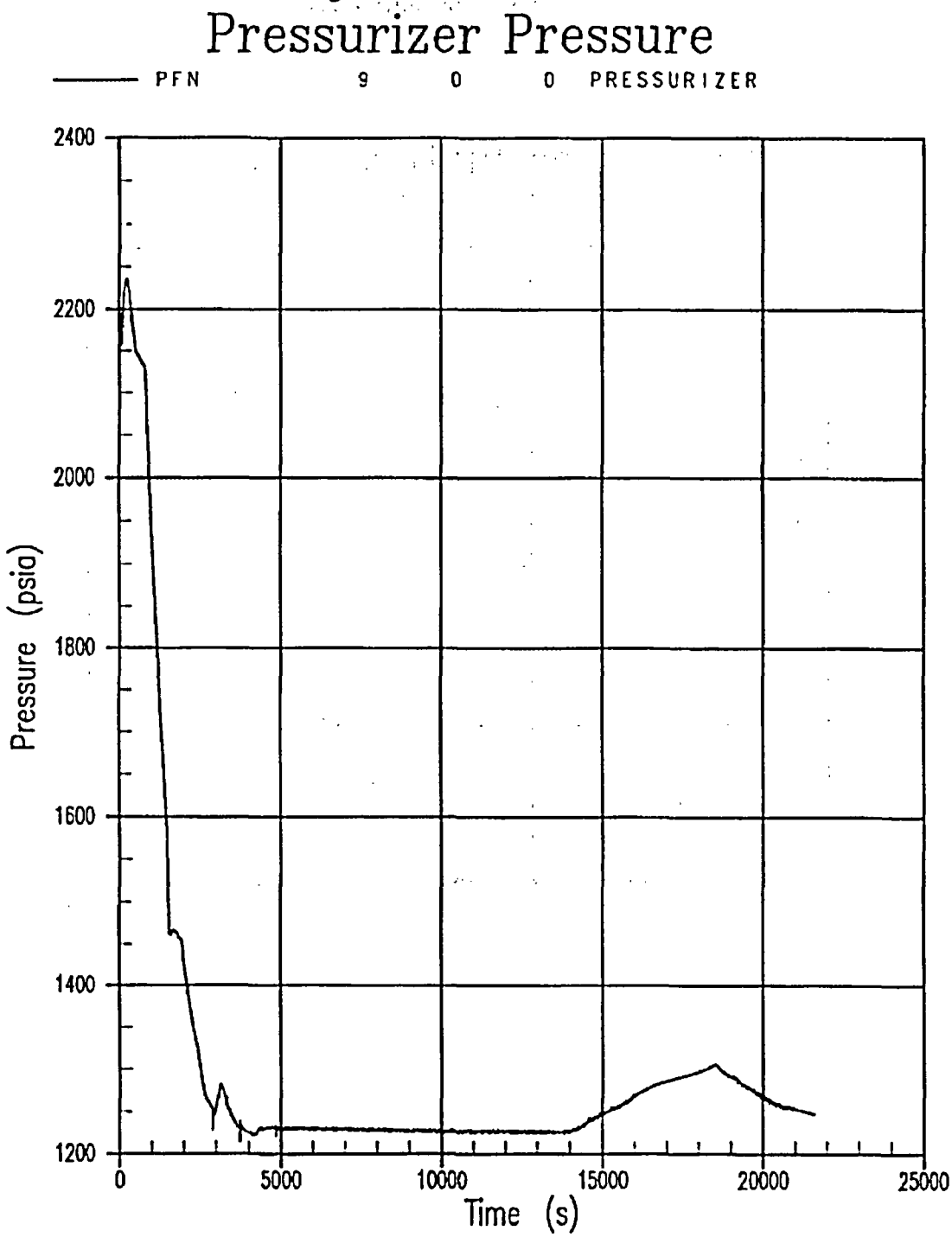


Figure 6-3: Case 1 Pressurizer Mixture Level

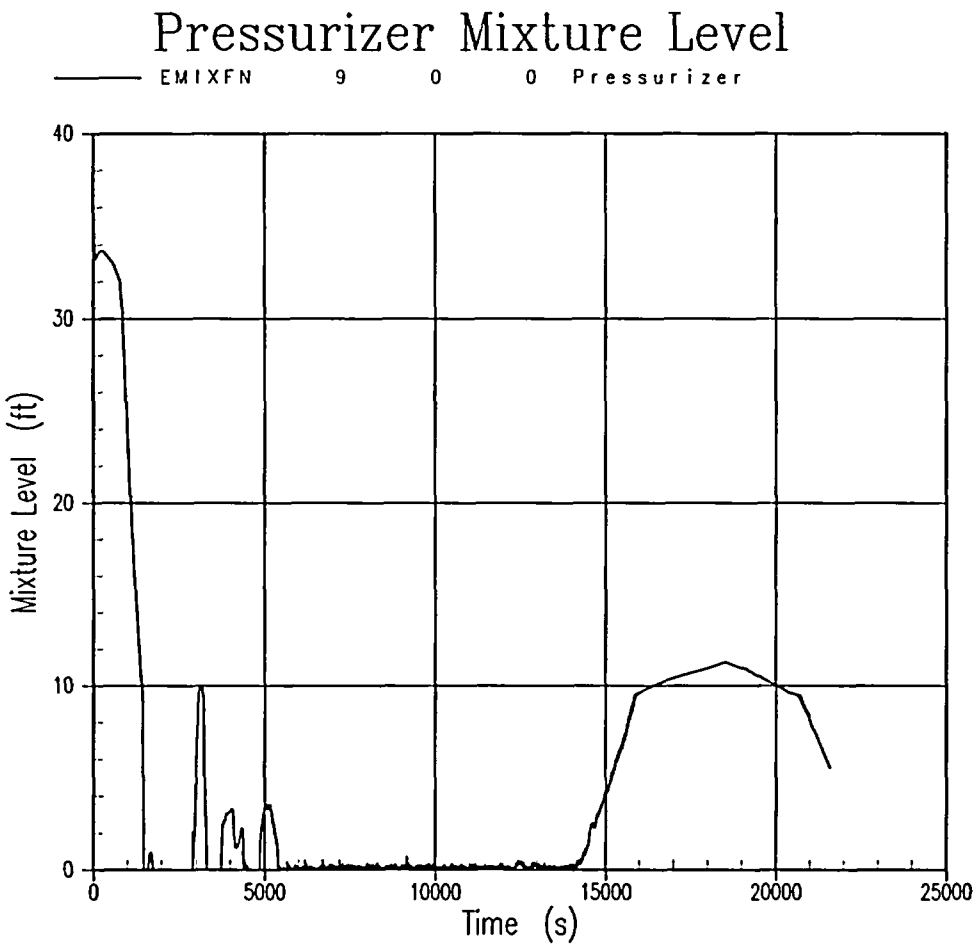


Figure 6-4: Case 2 Vessel Mixture Level

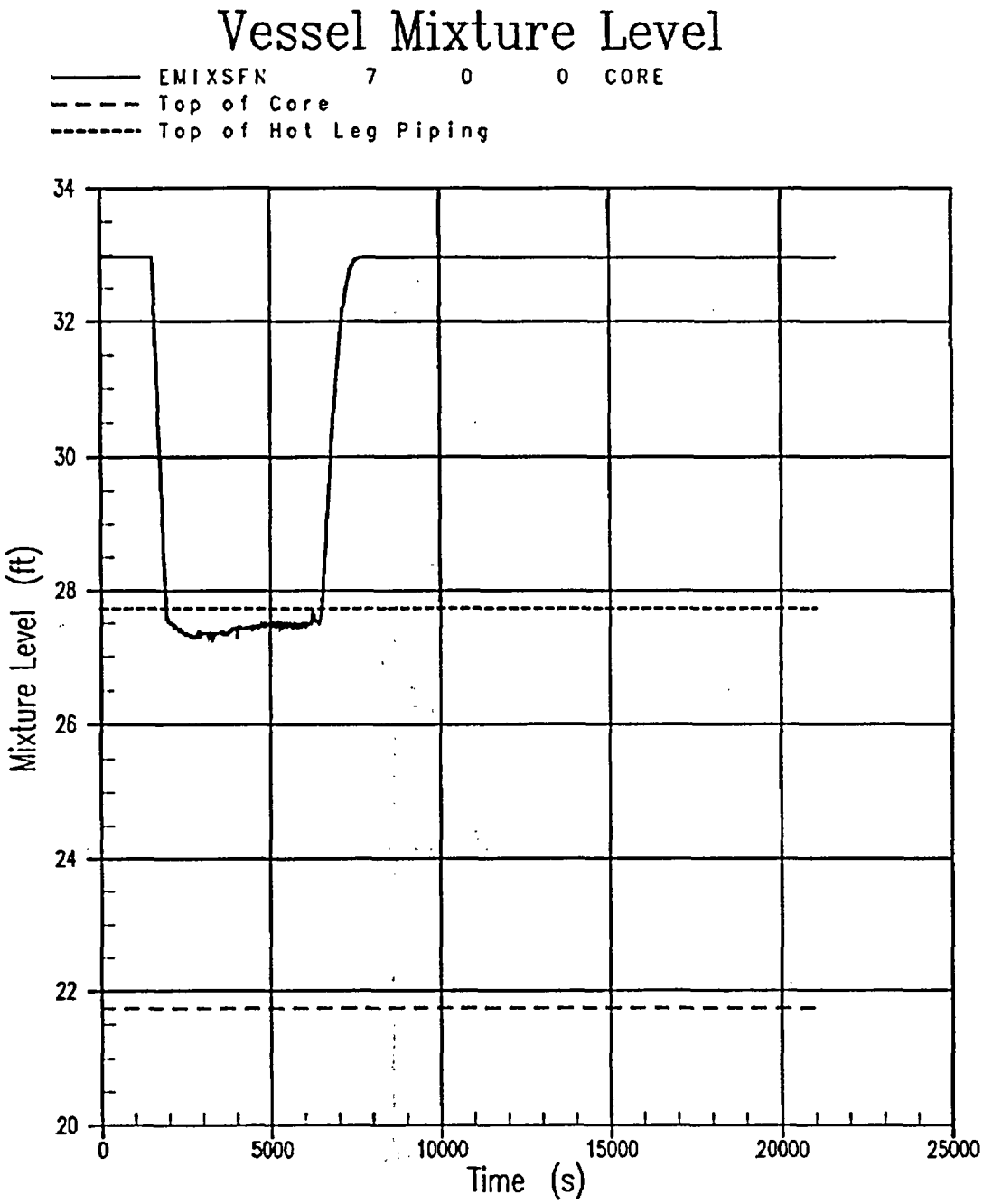


Figure 6-5: Case 2 Pressurizer Pressure

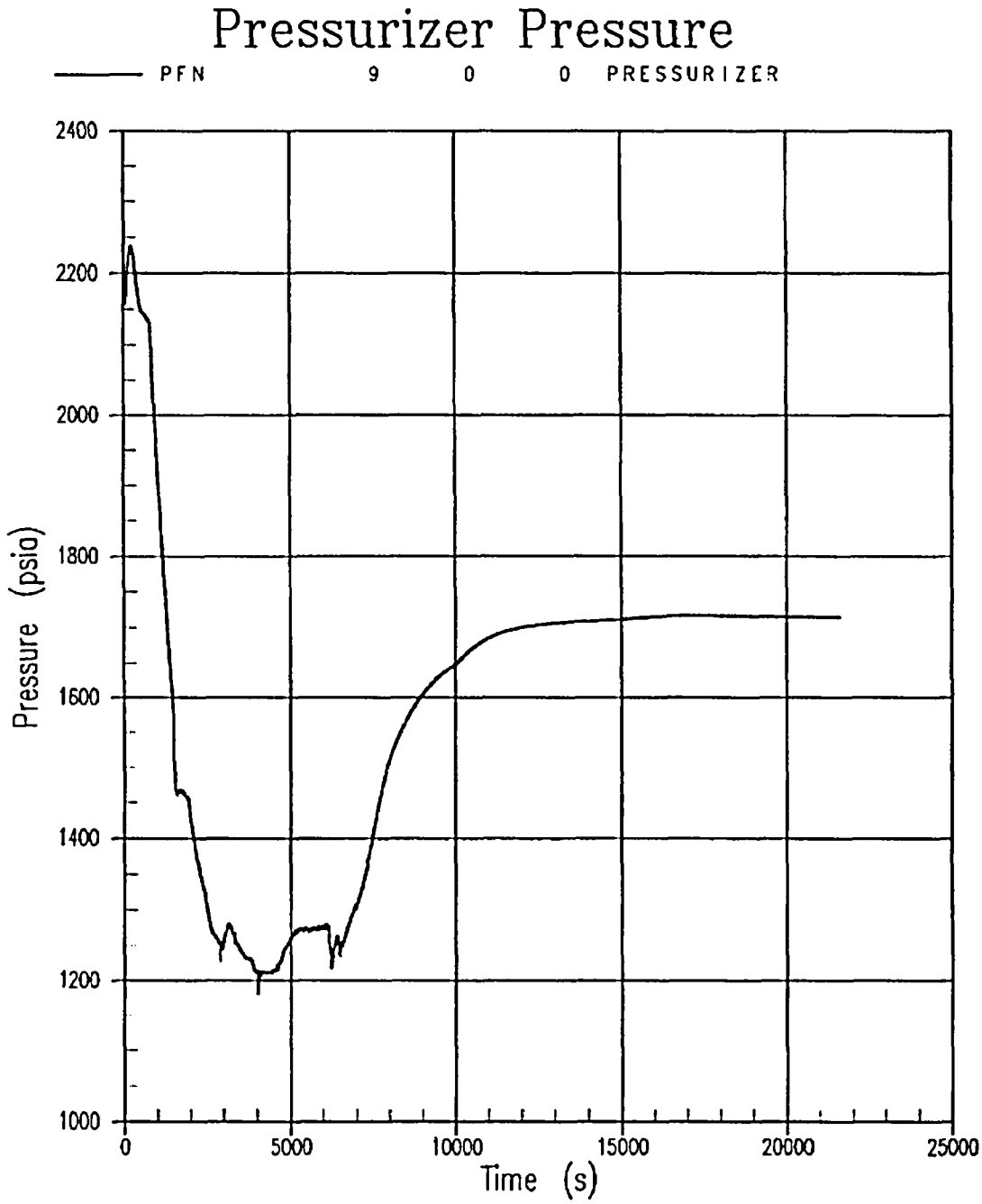


Figure 6-6: Case 2 Pressurizer Mixture Level

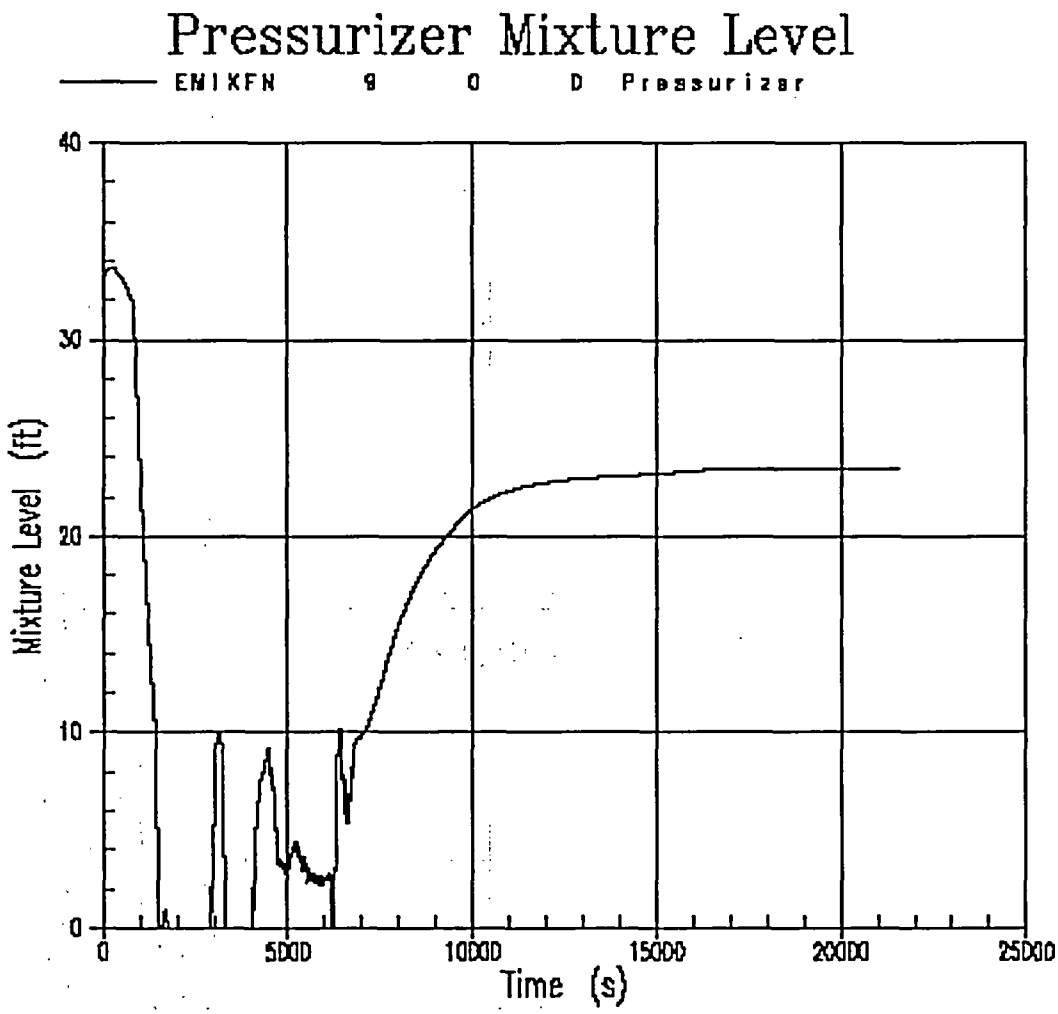


Figure 6-7: Case 3 Vessel Mixture Level

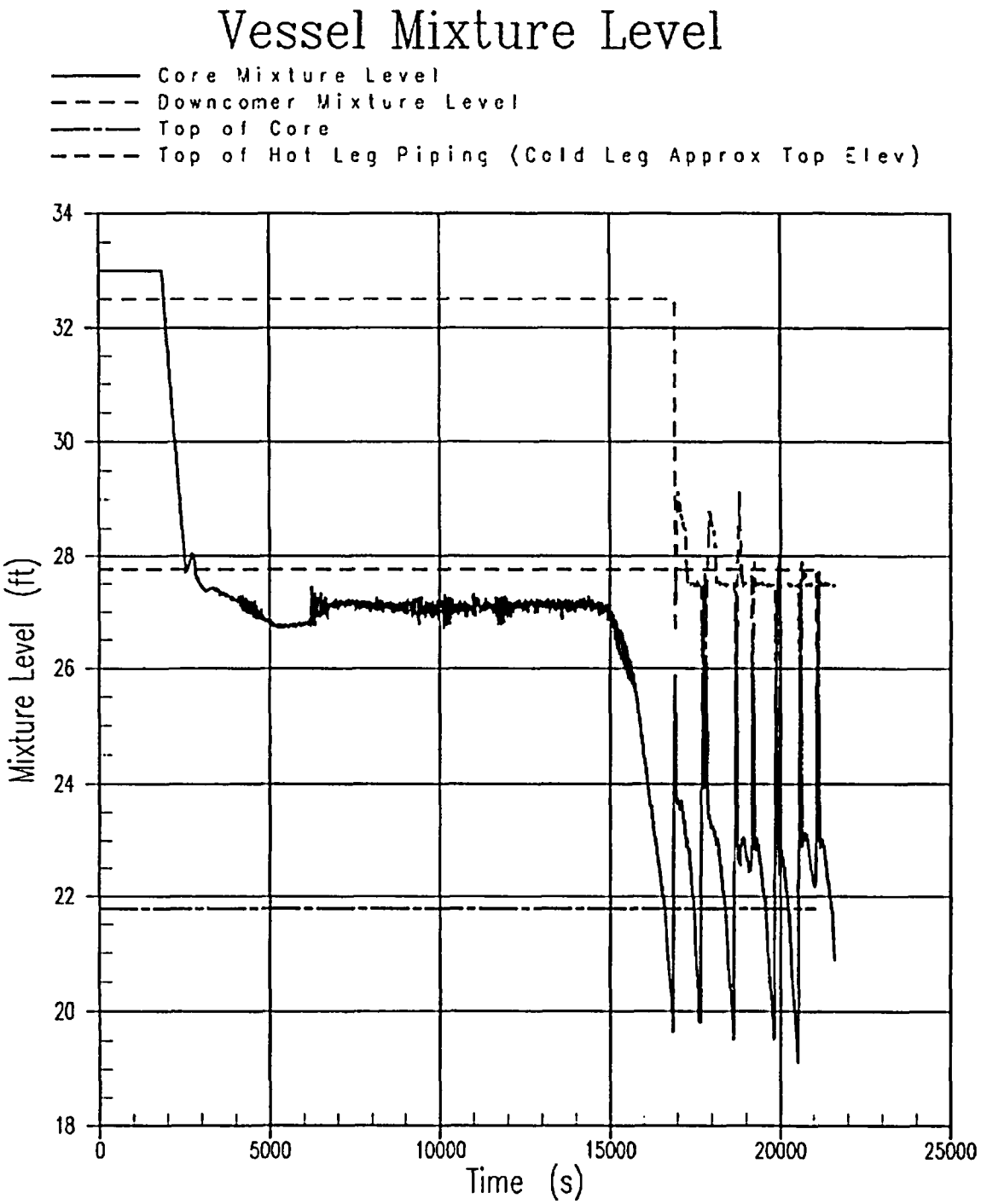


Figure 6-8: Case 3 Pressurizer Pressure

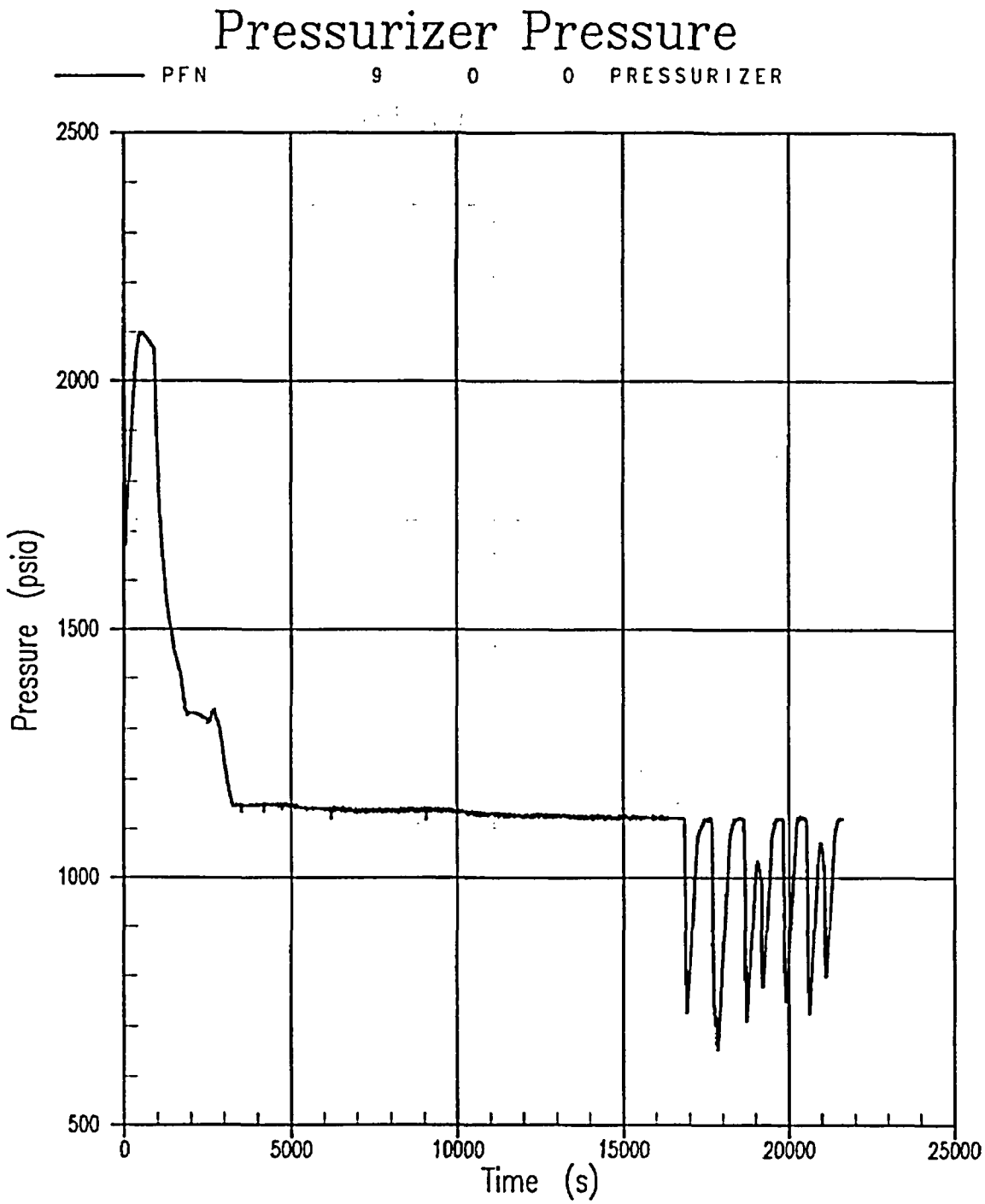
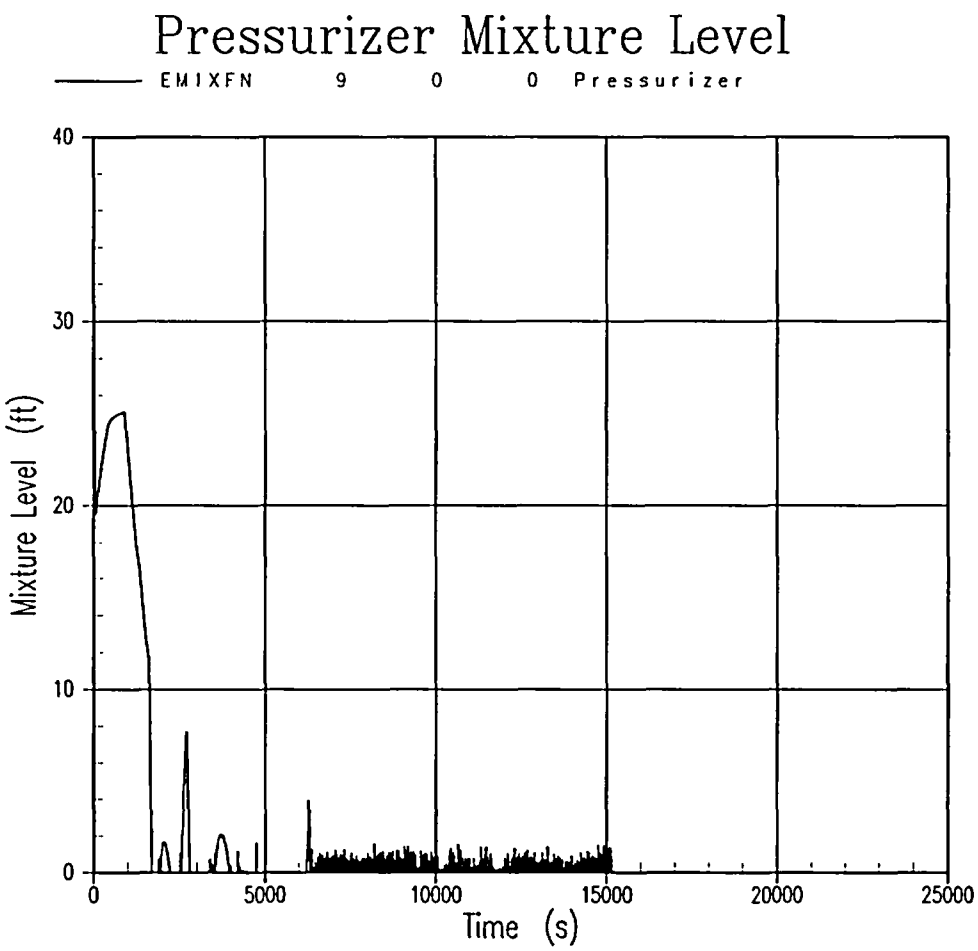


Figure 6-9: Case 3 Pressurizer Mixture Level



6.2 Long Term Cold Shutdown Capability

The Appendix R coping strategies cover a wide range of postulated fire scenarios, including fires that require evacuation of the main control room and subsequent control of the plant from the alternate shutdown panel (also referred to as the remote shutdown panel). While the alternate shutdown panel is central to the control of the plant for these scenarios, some operator actions and process monitoring may occur in other remote locations (e.g., local instrumentation and control). Each plant has developed a set of procedures for use in fire scenarios that are typically part of the plant's off-normal (also called abnormal) operating procedures. Typical off-normal procedures for control room evacuation fire scenarios prescribe a limited number of actions to be taken prior to evacuation of the control room (e.g., isolating power to vital equipment to prevent spurious operation) and specific procedures for each operator after evacuation from the control room. While there is no plant-to-plant consistency in these procedures, one operator will typically be assigned to the alternate shutdown panel and the others to remote location(s). Control of the plant depends on actions that can be taken from the alternate shutdown panel and from remote locations, coordinated through a dedicated communications system.

Plants may have different licensing requirements for the instrumentation and control capability to achieve a shutdown condition. PWRs would typically require the following functional groups to achieve hot and cold shutdown: reactivity control, reactor coolant inventory control and decay heat removal. As an example of the instrumentation and control systems that might be specified for bringing the plant to a stable hot shutdown state for a typical Westinghouse PWR can be found in the background for the Westinghouse NSSS Improved Technical Specifications (Reference 22):

Source range neutron flux	Boric Acid Storage Tank Level
Extended range neutron flux	Condensate Storage Tank Level
RCS Cold Leg Temperature	Auxiliary Feedwater Flow Rate
RCS Hot Leg Temperature	Steam Generator Pressure
Core Exit Thermocouple Temperature	Control Rod Drive Control
RCS Pressure	Pressurizer PORV Control
SG Level	Auxiliary Feedwater Control
Pressurizer Level	Steam Generator Atmospheric Relief Control
RWST Level	RCS Charging Pump Control

The safe stable state for fires requiring control room evacuation is typically Mode 3. For a typical W-NSSS, to reach Mode 3 for scenarios with no significant RCS inventory losses, the operator must assure that the reactor is subcritical (neutron flux monitoring), that heat is being removed to the steam generators via natural circulation (RCS hot and cold leg temperatures) and from the steam generators by dumping steam (SG level, and control of the SG atmospheric

relief valves). Also RCS makeup (pressurizer level and control of charging pumps) and boration capability (control of boration from the boric acid blender) using the normal charging pathway would be required to achieve a Mode 3 safe stable state. To achieve cold shutdown (Mode 5) as required by Appendix R, no additional capability is required, except for control of the RHR isolation valves and the cooling water to the RHR heat exchanger.

In the event of a fire scenario that results in control room evacuation and significant loss of RCS inventory (e.g., a 182 gpm per pump RCP seal leak), the operators of a typical W-NSSS would also need the ability to align and control a safety injection pump drawing from the RWST to restore inventory to the RCS and thereby retain core cooling capability. This event would be no worse than a fire initiated spurious opening of a pressurizer PORV in terms of RCS inventory loss rate. However, in this case, pressurizer level is lost for a significant portion of the transient as shown in the analysis in Section 6.1 of this report. Without pressurizer level, the operators would continue to use the available safety injection pump drawing from the RWST for RCS inventory control. They would then begin a cooldown of the RCS using the SG atmospheric relief valves to minimize RCS inventory losses and to begin to approach the entry conditions for normal RHR cooling. When pressurizer level is re-established as a result of the RCS cooldown and inventory loss reduction, the operators would be required to transfer from RCS makeup via a safety injection pump to makeup via a charging pump in the normal charging mode. Prior to SI termination, the other SI termination criteria of RCS subcooling would also have to be met. This information is available via the RCS pressure and temperature indications at the alternate shutdown panel. Also required to get to RHR entry conditions would be isolation of the SI accumulators; this would likely be done at a remote location once the operators were confident that they would not be needed. A final RCS pressure reduction using the pressurizer PORV may be required to complete the depressurization of the RCS to typical RHR entry conditions of 400 psig and 350 degrees F. From this point, as long as cooling water to the RHR heat exchanger can be controlled, the plant can be brought to Mode 5 cold shutdown conditions. It is noted that all of these actions to achieve cold shutdown are identical to those required for the case without significant RCS inventory losses, except for the transfer from RCS makeup via SI to RCS makeup via normal charging.

Thus, plants can typically achieve a cold shutdown condition without first establishing the hot shutdown state assumed in the Appendix R regulatory requirements. Instrumentation and control systems to achieve a cold shutdown state with a significant RCS inventory loss rate are generally identical to those required where there is no significant inventory loss, except for the transfer from SI to normal charging for RCS makeup control. These actions are already likely to be part of the fire coping capability needed to meet the Appendix R requirement for cold shutdown. Therefore the analyses presented in Section 6.1 should be applicable to the fire scenarios with control room evacuation.

7. Risk Assessment of a Loss of RCP Seal Cooling

The RCP seal model, and its associated probabilities, documented in WCAP-15603 were specifically developed to expedite NRC review of risk-informed applications submitted in accordance with Regulatory Guide 1.174. It was recognized by the WOG that many of the assumptions in the model were not best estimate expectations, but rather conservative estimates to account for uncertainties to assure that the model would not be likely to under-predict the consequences of a loss of all seal cooling. The WOG validated, in WCAP-16141, that these conservatisms were unlikely to affect any conclusions drawn from PRA models for internal events from at-power conditions. This includes both the overall plant core damage frequency, the contribution to core damage from station blackout scenarios and the risk importance of various components modeled in the PRA. Since the conservatisms were not predicted to impact any results or near term uses of the internal initiating events PRA, it was determined that it would not be cost beneficial at that time to pursue the additional activities that would have been required to obtain NRC staff concurrence to remove identified conservatisms from the model. However, when conservatisms can impact decisions, then the impact of the conservatisms needs to be examined more closely. This section presents the results of such an examination.

Specifically, the objective of this risk assessment is to provide a basis for understanding the impact of alternate fire coping strategies that might be considered to maintain pressurizer level on-scale as opposed to only preventing core uncovery. First the impact of any conservatisms and uncertainties in the RCP seal leakage model need to be understood. These include the time at which increased RCP leakage is assumed to occur, the number of RCPs that simultaneously experience increased leakage, the assumed leakage rates for the postulated failure modes and the failure mode probabilities. In addition, the assessment will consider the impact of restoration of RCP seal injection on RCP seal integrity. The uncertainties in the models and assumptions used in the risk assessment will be addressed through sensitivity studies to identify any important uncertainties that could impact the result and conclusions of the risk assessment.

7.1 Simultaneous Leakage from All RCPs

The NRC RCP seal leakage model (Rhodes Model) postulates that if a leakage scenario occurs, all RCP pumps with the same seal material in a given unit would respond with the same leakage. This assumption is likely to be conservative. However, the WOG determined that addressing this issue rigorously would make the WOG 2000 model very complicated. Therefore, in order to maintain the simplicity of the model, this issue was recognized as a potential conservatism and was not addressed in WCAP-15603. Consequently, in their SE on WCAP-15603, the NRC agreed with this assessment noting the analysis would need to encompass a "greater level of complexity, including the need to address the potential common cause conditions ... and common cause failure modes between RCPs".

The goal of this evaluation is to illustrate the impact of a more realistic model with respect to the number of RCPs with increased leakage. The NRC approved model considers 21 gpm/pump and 182 gpm/pump leaks as the most likely scenarios. As shown in WCAP-16141, a typical Westinghouse NSSS plant is likely to be able to cope with a 21 gpm/pump leak and therefore, this scenario is a negligible contributor to risk. Table 7-1 illustrates this (Case 1) for a typical 3-loop Westinghouse NSSS plant, where time to core uncovery due to a 21 gpm leak in each of three RCPs is greater than 20 hours after the accident.

Using the NRC approved model, the most risk significant scenario is that involving a 182 gpm/pump leak at 13 minutes after the loss of all seal cooling. For the same 3-loop Westinghouse NSSS plant, the time to core uncovery is predicted to be 5.2 hours, if no operator action for RCS cooldown and depressurization is taken, and 17.1 hours if operator actions are assumed to cooldown and depressurize the RCS beginning at 30 minutes after the loss of all seal cooling. This is shown as Case 4 in Table 7-1.

Given the field and test experience for which increases in steady state RCP leakage above the range of 21 gpm have not been observed, it is reasonable to expect that the popping and binding failure might not occur simultaneously in all RCPs with 100% certainty. Some degree of randomness in the RCP popping and binding failures is a reasonable assumption. Cases 2 and 3 were designed to illustrate the scenarios with only one RCP seal experiencing increased leakage (Case 2) and with only two RCP seals experiencing increased leakage (Case 3). The time to core uncovery for Case 2 and 3 were interpolated based analyses documented in WCAP-16141. Two sequences are evaluated for each leakage rate: RCP seal leakage with RCS cooldown and depressurization, and RCP seal leakage with TDAFW availability only. The interpolation of the analysis results was not straight forward based on the proportional leakage rate in all cases. The analyses in WCAP-16141 included the assumption that auxiliary feedwater to the steam generators was terminated when the condensate storage tank (CST) was emptied (i.e., no refill assumed). For the three loop case with RCS cooldown and depressurization, the CST emptied at about 15 hours. For the case of no RCS cooldown and depressurization, the CST does not empty until 17 hours into the event. After the CST is emptied, the SG boils dry in approximately 45 minutes and then the RCS pressure increases to the pressurizer PORV setpoint. The subsequent pressurizer PORV opening results in a high RCS inventory loss rate compared to the pump seal leakage which takes these scenarios to core uncovery in a very short period of time after CST depletion.

Table 7-1: Time to Core Uncovery For Various RCP Leakage Rates

Sensitivity Case	Number of RCPs With Seal Leaks		Total RCS Inventory Loss Rate After 13 Minutes (gpm)	Time to Loss of Pressurizer Level (hrs)	Time to Core Uncovery (hrs)	
	21 gpm	182 gpm			with RCS C/D	with TDAFW only
Case 1	3	0	63	1.15	20.8	20.4
Case 2	2	1	224	1.05	19.7	18.2
Case 3	1	2	385	0.95	18.5	8.1
Case 4	0	3	546	0.85	17.1	5.2

As expected, for the sequences with RCS cooldown and depressurization, the time to core uncovery is long and is greater than 17 hours for each case. Since Appendix R coping strategies to implement RCS makeup are assumed to be completed well before this time, the variation in time across the 4 cases is not significant with respect to the ability to successfully implement Appendix R coping strategies. Even considering any uncertainties in the time at which these coping strategies would be successfully implemented, these variations are not significant.

Without RCS cooldown and depressurization, the time to core uncovery is greater than 5 hours in all cases. Again, this is significantly longer than the time for Appendix R coping strategies to be put into place. In the extreme case where implementation of backup coping strategies are significantly delayed (e.g., greater than 4 to 5 hours), the number of pumps assumed to experience the increased leakage may impact strategy selection. However, this is an extremely unlikely event that is outside the considerations for preplanned coping strategies and is therefore not considered further. Note that this discussion assumes that establishing safety injection prior to core uncovery would prevent core damage (e.g., sustained high core temperatures). This is based on typical PRA success criteria for W-NSSS plants for station blackout power recovery.

Table 7-1 also shows the time to loss of pressurizer level. This shows that pressurizer level is lost between 50 and 70 minutes (0.85 and 1.15 hours) following a loss of all seal cooling with no RCS makeup. By contrast, the NOTRUMP analyses for the case of three RCPs leaking at 182 gpm shows a loss of pressurizer level at about 25 minutes (Case 1 in Section 6.1). No information is available for NOTRUMP analyses with a long term 21 gpm leak in all pumps, but the loss of pressurizer level is assumed to be over 1 hour based on the acceptability of current Appendix R coping strategies where increased leakage is not assumed to occur for over one hour. The MAAP analyses assumed increased leakage began at 30 minutes as opposed to 15 minutes in the NOTRUMP analyses. Based on the WCAP-16141 assessment of the impact of the difference in timing for increased leakage at 30 vs. 13 minutes, the loss of pressurizer level for Cases 2 through 4 of Table 7-1 would be up to 17 minutes sooner. Thus, for Case 4 (all pumps leaking at 182 gpm, the time of emptying the pressurizer would be about 33 minutes, which is in closer agreement with the NOTRUMP prediction. Other code differences, such as

extraction steam for the turbine driven auxiliary feedwater pump, could account for the remaining differences.

Thus it is concluded that the assumption of the number of RCPs developing the 182 gpm leakage is not significant in terms of preventing core uncovery, but is significant in terms of assessing coping strategies to meet the current Appendix R requirement that pressurizer level remain on-scale during the simulated transient. That is, the conditional core damage probability for this event is not impacted by the number of pumps developing a 182 gpm leak. However, the acceptability of current coping strategies, in terms of compliance with the Appendix R requirement for pressurizer level is impacted by the number of pumps assumed to leak at 182 gpm. Thus, the acceptability of Appendix R coping strategies, while not safety significant, is impacted by the level of uncertainty regarding the number of RCP seals experiencing popping and binding seal failures.

7.2 RCP LEAKAGE RATES

As discussed in Section 3.4 of this report, except for the 21 gpm leakage rate, the RCP seal leakage rates discussed in WCAP-10541 and subsequently used in the PRA model in WCAP-15603 are based on analytical assessments of the resistances provided by the torturous flow path. For the case of the 182 gpm leak rate, the estimate is based on the No. 2 and No. 3 seals open at the maximum extent of their travel. Based on the popping and binding failure mode, other lesser leakage rates can be postulated. The range of total RCS inventory loss rates discussed in the previous section can serve as a foundation for understanding the impact of leakage rates less than 182 gpm per pump in the event of a failure of the No. 2 seal due to popping and binding. The range of possible leakage ranges from something greater than 21 gpm per pump (63 gpm total for a 3-loop plant) to the maximum of 182 gpm per pump (546 gpm total for a 3-loop plant). The conclusions from the previous section are directly applicable:

- The assumed leakage rate for a popping and binding failure of the No. 2 RCP seal is not significant in terms of preventing core uncovery if safety injection is restored.
- The assumed leakage rate is significant in terms of assessing current coping strategies to meet the Appendix R requirement that pressurizer level remain on-scale during the simulated transient.
- The acceptability of Appendix R coping strategies, while not safety significant, is impacted by the level of uncertainty regarding the leakage rate for RCP seals experiencing popping and binding seal failures.

7.3 TIME OF INCREASED RCP SEAL LEAKAGE

The time at which RCP seal leakage increases from its nominal value during operation to a higher value can vary both with the seal behavior prior to and following a loss of all seal cooling. WCAP-10541 and WCAP-15603 use a time of 13 minutes after the loss of all seal cooling as

the time at which increased RCP seal leakage would be expected to occur. This is based on an estimate of the time at which hot RCS fluid would reach the No. 1 RCP seal, based on the No. 1 seal leakoff prior to the loss of seal cooling and the volume of water in the pump casing between the thermal barrier cooler and the No. 1 seal that must be purged through the No. 1 seal before hot RCS fluid will be in contact with the No. 1 seal. When the hot RCS fluid reaches the No. 1 seal, the thermal growth of the seal components, the thermal stress and the thermal hydraulic phenomena result in an increase in No. 1 seal leak-off rate to about 21 gpm when all of the seal components are functioning as designed (i.e., no popping or binding of seals and no O-ring failures). There is a very short period of time (e.g., a few minutes) when the hot RCS fluid first reaches the No. 1 seal in which the No. 1 seal leakage rate increases from its pre-loss of cooling value (e.g., 3 gpm) to a much larger rate (e.g., 60 gpm for approximately 1 minute) before decreasing to its long term value of 21 gpm. This represents the period of thermal growth of the seal components and the re-balancing of forces across the seal faces.

The 13 minutes may be equated to a water volume of 39 gallons and a nominal seal leak-off rate of 3 gpm. Thus, the 13 minutes is a representative value. An acceptable No. 1 seal leak-off rate specified by the pump vendor (Westinghouse) is between 1 and 5 gpm. Thus, the actual time at which the increased leakage might occur, based on a 39 gallon water volume in the casing, could vary from a little over 8 minutes to about 40 minutes. There is also some variability in the volume of water in the pump casing between the thermal barrier cooler and the seal package for the different RCP models manufactured by Westinghouse. Also, although the thermal hydraulic analyses assumed 21 gpm leakage per RCP *starting at reactor trip*, any actual increased leakage does not start until the RCP seal inlet water volume has been purged of cooler water by the incoming hot RCS fluid. Finally, the thermal capacity of the thermal barrier cooler and the pump internals above the thermal barrier cooler could also delay the onset of heatup of the seal components. On the other hand, some mixing of the hot RCS fluid and the cold water in the pump casing would occur at the interface between the two fluids (i.e., there is not a distinct slug of hot RCS fluid pushing the cold water through the No. 1 seal). Therefore, the time at which the RCP seal package would begin to exhibit increased leakage from 21 to 182 gpm depending on assumptions for seal operation could vary from as little as 8 minutes to over 40 minutes.

WCAP-16141 provides a qualitative evaluation of the PRA model of the impact of the time until increased RCP seal leakage on the PRA model. Examination of the results for the base case analyses (in WCAP-16141) provides some insights that permit a qualitative assessment of the sensitivity of the results to the time at which the seal leakage rate changes:

- For all the 21/480 gpm RCP seal leakage scenarios and for the 21/182 gpm RCP seal leakage scenario with AFW available but no RCS cooldown and depressurization, there should be a direct relationship between the time of increased RCP seal leakage and the time at which core damage occurs. Decreasing the time at which increased RCP seal leakage occurs will result in a proportional change in the time of core damage. That is, if the

increased RCP seal leakage is assumed to occur at 13 minutes after the event initiation (as opposed to the 30 minute start time assumed as a baseline), then the time of core damage would be about 17 minutes earlier than that predicted in the base case analyses.

- For all the 21/21 gpm RCP seal leakage scenarios and the 21/182 gpm RCP seal leakage scenario with AFW available and successful RCS cooldown and depressurization, there is no relationship between the time of increased RCP seal leakage and the time at which core damage occurs. Decreasing the time at which increased RCP seal leakage occurs will result in no significant change in the time of core damage.
- For the 21/182 gpm RCP seal leakage scenario with no AFW available, there is some relationship between the time of increased RCP seal leakage and the time at which core damage occurs. Decreasing the time at which the increased RCP seal leakage occurs will result in a change in the time of core damage, but not by a proportional amount. That is, if the increased RCP seal leakage is assumed to occur at 13 minutes after the event initiation (as opposed to a 30 minute start time), then the time of core damage would be less than 17 minutes earlier compared to that predicted in the base case analyses.

Overall, the variability in the assumed time at which RCP leakage increases (in terms of the ability to prevent core damage) is not significant. At most, the minimum time available to implement recovery strategies may change by about 15 minutes in either direction. The non-recovery probabilities are not sensitive to a 15 minute change when the recovery time is beyond an hour. Thus, the impact on core damage is negligible. However, the impact on the time that pressurizer level is lost can be significant in relation to the time at which strategies can be implemented to prevent the loss of pressurizer level, as required by Appendix R. Since the time to loss of pressurizer level is in the 30 to 60 minute time frame, some current coping strategies may no longer be acceptable within the range of uncertainties studied in this investigation.

The assessment of the variability in the time at which the increased leakage may be initiated supports the conclusions from the previous assessments:

- The assumed time at which a popping and binding failure of the No. 2 RCP seal is assumed to occur is not significant in terms of preventing core uncovery if safety injection is restored.
- The assumed time that popping and binding failures of the No. 2 seal are assumed to occur is significant in terms of assessing coping strategies to meet the Appendix R requirement that pressurizer level remain on-scale during the simulated transient.
- The acceptability of Appendix R coping strategies, while not safety significant, is impacted by the level of uncertainty in the time at which popping and binding failures of the No. 2 seal are assumed to occur.

7.4 RCP Seal Leakage Probabilities

As shown in Table 3-1, five unique classes of RCP seal leakage scenarios can be postulated, based on the performance of the three seal stages. A basis for the failure probabilities for each

of the three seal stages is documented in WCAP-15603. The probabilities in WCAP-15603 were developed to account for the installation of high temperature O-rings in the RCP seal packages, thus eliminating or reducing the probability of failure modes associated with degradation of the O-rings in earlier NRC models. Section 2.2.1.1 of the BNL report (Reference 23) states that the binding failure mode is driven by premature extrusion failure of the O-rings or channel seal elastomers that make up secondary seals and that the binding failures dominate the failure mode for the first and third seal stages. Since the earlier models did not distinguish between the performance of the old and new elastomer materials in the O-rings, WCAP-15603 argued that the failure probabilities for stages 1 and 3 should be reduced by 50% from the values in the previous models. These changes resulted in a recommended probability of each unique leakage category as follows:

Table 7-2: WCAP-15603 RCP Seal Leak Probabilities

RCP Seal Long Term Leakage Rate (gpm)	Probability
21	79%
57	14.42%
76	1%
182	5.33%
480	0.25%

Note that the failure probabilities in the BNL report were used as the basis for the probabilities recommended in WCAP-15603. All of the relevant test results and operational experience do not support any leakage greater than the 21 gpm category. Thus, the probabilities are subject to some uncertainties. In fact, the BNL guidance states that the relatively high probability of the second stage seal failure of 20% due to popping open of the seal "... was chosen to account for uncertainties in the processes related to the popping open failure mechanism". The impact of these uncertainties and conservatisms are discussed below.

In the SE of WCAP-15603, the NRC assumes that the "failure mode for the third seal stage be set to unity, considering that it will be subjected to a pressure differential greater than the normal operation pressure differential following failure of the second seal stage." Further, since there is no data to support the integrity of the No. 3 seal; therefore, when exposed to (near) full system pressure, the No. 3 seal is expected to always fail. This assumption results in a 19.75% probability of a 182 gpm/pump leak (with a zero probability of a 57 gpm leak). The NRC's SE on WCAP-15603 did not accept the argument that the third stage would not automatically fail upon popping open of the second stage, in spite of the results presented in WCAP-10541 that show that the No. 3 cartridge seal is predicted to be exposed to a pressure differential of about 555 psig under these conditions. Thus, the third stage cartridge seal is not expected to fail. As discussed in Section 3, this is not true for the No. 3 seal bellows design which has little pressure retention capability.

Thus, the relatively high probability of 19.75% associated with the 182 gpm leakage value is based on conservatisms in the probabilities of failures of both the No. 2 and No. 3 seals. The popping and binding of the No. 2 seal, which leads to the 182 gpm leakage rate, is not supported by any relevant test results or operational experience; it is supported only by theoretical concerns raised by NRC contractors during their assessments of the behavior of the RCP seals under a loss of all cooling situation. This relatively high assumed probability of a 182 gpm leak may have contributed to its use in assessing the adequacy of Appendix R coping strategies.

The overall result of the variability in the assumed probability for various RCP leakage rates in terms of the ability to prevent core damage is not significant. For most scenarios, a change in leakage rate from 57 to 182 gpm changes the time at which the recovery strategies must be implemented by several hours. However, the non-recovery probabilities are not particularly sensitive to these changes when the shortest required recovery time is beyond about one hour. Thus, the impact on core damage is negligible. However, the impact on the time that pressurizer level is lost can be significant in relation to the time at which strategies can be implemented to prevent the loss of pressurizer level, as required by Appendix R. The time to loss of pressurizer level is in the 30 to 60 minute time frame for the 182 gpm leak and well beyond one hour for a 57 gpm per pump leak. Coping strategies designed for compliance with Appendix R with the smaller leakage rates would not be acceptable for the 182 gpm assumption.

The assessment of the variability in the assumed leak rate probabilities supports the conclusions from the previous assessments:

- The assumed probability of a popping and binding failure of the No. 2 RCP seal is not significant in terms of preventing core uncovery if safety injection is restored.
- The assumed probability of a popping and binding failure is significant in terms of assessing coping strategies to meet the current Appendix R requirement that pressurizer level remain on-scale during the simulated transient.
- The acceptability of Appendix R coping strategies, while not safety significant, is impacted by the level of uncertainty regarding the probabilities for RCP seals experiencing popping and binding seal failures.

7.5 Impact of Restoration of RCP Seal Cooling

In their evaluation of the Appendix R coping strategies, the NRC has assumed that if seal cooling via seal injection is re-initiated after temporary loss of seal cooling, the RCP seals would be susceptible to thermal shock damage which could result in large uncontrolled leakages. The basis for the NRC's assumption was WOG ERG maintenance item DW-94-11 which recommends, for the WOG Loss of All AC Power Emergency Response Guideline (ECA-0.0),

that if RCP seal cooling is lost for an appreciable period of time, then RCP seal cooling should be re-established by means of a controlled natural circulation cooldown rather than restoration of the normal RCP seal cooling systems (e.g., seal injection). However, DW-94-11 does not prohibit restoration of seal cooling by seal injection and/or CCW if circumstances require, but it cautions operators of risks associated with the various options. As discussed previously in Section 5, restoration of seal cooling to a hot seal package is beyond the design basis of the seal assembly and was not extensively analyzed and tested to validate its survivability. However, there is no operational or empirical basis for concluding that the seals will fail catastrophically or leak uncontrollably beyond the defined flow range of 3 to 21 gpm upon seal cooling restoration.

For Appendix R coping strategies that use safety injection pumps, seal injection would not be restored in most cases because the safety injection pathway bypasses the seal injection pathway. Therefore, the issue of restoration of seal injection is not an issue for these plants. For Appendix R coping strategies that use normal charging pumps to maintain RCS inventory, there is test and operational data to conclude that an increase in seal leakage (e.g., to 182 gpm) would not occur as a result of cold thermal shock to the seals. Therefore, pressurizer level should not be lost any more quickly that that originally assumed in developing the coping strategy. If the assumption is made that restoration of seal cooling results in the creation of a large RCP leak, the result would likely be a loss of pressurizer level. Thus, with a large RCP seal leak, the coping strategy would not support compliance with the Appendix R requirement to maintain pressurizer level. However, this scenario would only lead to core damage if a safety injection pump could not be started within 4 to 5 hours after the loss of all seal cooling, as discussed in section 7.1. Of course this discussion does not apply to plants that restore seal cooling prior to the time at which hot RCS fluid comes in contact with the seals.

Therefore, it is concluded that the assumed probability of seal leakage due to a cold thermal shock failure of the No. 2 RCP seal is not significant in terms of preventing core uncovery. However, the probability of increased seal leakage due to cold thermal shock at re-initiation of seal injection to a hot seal package is significant in terms of assessing coping strategies to meet the current Appendix R requirement that pressurizer level remain on-scale during the simulated transient. If increased seal leakage due to cold thermal shock is assumed, the coping strategies that are found to be not risk significant will be found to be inadequate with respect to the Appendix R criterion for maintaining pressurizer level on-scale.

7.6 IMPACT OF NEW PRESSURIZER LEVEL CONTROL FEATURES AND PROCEDURES

As a result of the NRC's use of the 182 gpm per pump RCP seal leakage assumption in Appendix R inspections, several plants are studying new measures to enable coping strategies to be in compliance with the Appendix R criterion that the analyzed pressurizer level remains within the measurable range. Other plants are investigating the potential for either procedural

changes or new systems to cope with the 182 gpm seal leakage scenario being used by NRC in fire inspections.

From a risk perspective, it has previously been shown that there is no significant risk benefit associated with new procedures or equipment whose purpose is to control pressurizer level. Considering uncertainties in the models and assumptions for RCP seal leakage and recovery features in the previous assessments, the new procedures or equipment, even if perfectly implemented, would provide a negligible improvement in risk.

The only piece of the model that has not been considered in previous risk assessments in this report is the potential detriment to risk associated with new procedures to implement Appendix R coping strategies on an accelerated time scale to prevent pressurizer level from falling off-scale. While a quantitative generic assessment cannot be completed due to the plant specific nature of such procedures, it needs to be recognized that such procedures could distract operator attention from other critical response measures and thereby lead to an increase in risk.

The conclusion of the previous sections of this report is that there is no substantive risk improvement as a result of using pressurizer level instead of core uncovery as a metric for judging the acceptability of Appendix R coping strategies. Therefore, the net balance in risk as a result of the most recent direction in NRC assessment of coping strategies and the associated need to revise the coping strategies is risk/safety neutral at best and possibly risk adverse. In any event, it is likely that a cost benefit assessment could not justify the changes required to meet the use of the new models and assumptions by the NRC in judging the adequacy of Appendix R coping strategies.

7.7 Risk Assessment Conclusions

Based on the qualitative assessments provided in this section, it has been shown that the conditional core damage probability is relatively insensitive to the choice of assumptions in the RCP leakage model when considering core uncovery as the acceptance criteria. Specifically, the changes in RCP seal leakage rates, timing of the initiation of leakage and seal failure probabilities were not predicted to result in significant changes in conditional core damage probability for Appendix R fire initiating events.

However, the choice of assumptions in the RCP seal leakage model is significant in terms of assessing coping strategies to meet the current Appendix R requirement that pressurizer level remain on-scale during the simulated transient. Specifically, the acceptability of Appendix R coping strategies, while not safety significant, is impacted by the level of uncertainty or conservatism inherent in the leakage rate assumptions for RCP seals. New procedures to implement Appendix R coping strategies on an accelerated time scale to prevent pressurizer level from falling off-scale may result in an increase in risk if they distract operator attention from

other critical response measures. Additionally, the implementation of new plant features to maintain pressurizer level on-scale for these larger RCP seal leakage rates cannot be justified on cost benefit basis and do not represent an increase in safety.

Therefore, it is concluded that the conservatisms in the RCP seal leakage model are impacting the acceptability of Appendix R coping strategies and the re-examination of those assumptions and/or the Appendix R requirements is appropriate.

8. SUMMARY AND CONCLUSION

8.1 SUMMARY

This report presents significant information relative to the appropriate reactor coolant pump seal models to be used in assessing the adequacy of design basis and/or licensing basis features relative to compliance with regulatory requirements. Specifically, the issue was raised in regards to the appropriate RCP seal model for use in Triennial Fire Inspections during which the adequacy of fire coping is assessed relative to compliance with the Appendix R requirements. The change in NRC assumptions used in these assessments, based on their approval of a PRA model, has resulted in inspection findings for utilities. Once identified, these findings must be resolved through design and/or operational changes. The use of overly conservative assumptions can lead to the expenditure of resources for no real change in plant safety, or in the extreme, a decrease in plant safety.

The information presented in this report is a combination of information previously published in reports reviewed by the NRC staff and new information that heretofore was not widely available. In the case of information that is contained in reports that has been previously reviewed by the NRC, the information is provided here to:

- Assure that it is considered in the context of the issue of the appropriate RCP seal assumptions and models for regulatory compliance issues, and
- Improve readability of this report and comprehension of the technical bases for the position presented in this report by providing a single source of relevant information.

The overall conclusion of this report is that design/licensing basis regulatory compliance assessments, such as the Triennial Fire Inspections, should use the expected RCP seal performance of a 21 gpm per pump long term leakage with no additional increase in leakage due to postulated seal failures when hot RCS fluid contacts the seal assembly, no additional increase in leakage upon re-initiation of seal injection for either a hot or a cold seal assembly and no increase in leakage as a result of O-ring failures at prolonged high RCS pressures. This conclusion is supported by the relevant analytical, test and field experience to date. The probabilistic 182 gpm leak rate postulated as a result of the No. 2 seal popping and binding failure mode that is currently being used in these regulatory compliance assessments is not appropriate. This assumption may be appropriate for a PRA model in which all of the physically possible "what-ifs" are admitted into the assessment and assigned appropriate probability values.

This conclusion is supported by analytical assessments that are validated by all of the available test and operational experience. Specifically:

- Detailed analytical thermal stress and thermal hydraulic assessments of the performance of the RCP seals under loss of all seal cooling conditions that support the expected seal performance to be a long term nominal leakage rate in the range of 21 gpm (as measured at full system pressure and temperature).
- Specifically designed prototypical large and small scale tests of the performance of RCP seals under both the loss of RCP seal cooling and the restoration of RCP seal cooling after the seal package has been in contact with hot fluids for a substantial period of time, where no seal leakage beyond the expected range of 21 gpm was observed.
- Operational experience in which a total loss of RCP seal cooling has been experienced as a result of operational occurrences, including a number of occasions for which RCP seal cooling was subsequently restored to a hot seal package, where no seal leakage beyond the expected range of 21 gpm was observed.

The deterministic analyses provided in this report also demonstrate that even with an assumption of an initial 182 gpm leak rate and use of conservative licensing basis codes, continued long term core cooling can be maintained, with makeup to the reactor coolant system restored within 80 minutes after the loss of all seal cooling. The current Appendix R coping strategies for restoration of RCS makeup using SI pumps are designed to be implemented within this time frame. For plants that show compliance with regulatory requirements by having the capability to restore RCP seal cooling before the seal package is exposed to hot RCS fluids (e.g., within 13 minutes of the loss of seal cooling), the increase in leakage to 182 gpm (and therefore this assessment) does not apply. This provides another layer of assurance that the health and safety of the public would not be compromised by the removal of the excess conservatisms in the regulatory oversight process represented by the use of the PRA seal model assumption of 182 gpm per pump leakage in conjunction with the requirement to maintain pressurizer level on-scale.

The plant conditions at the end of the deterministic analyses are amenable to subsequent operator actions to initiate a plant cooldown to normal RHR conditions and eventual cold shutdown. An assessment is also provided to show that operator actions to achieve a cold shutdown state can be achieved from the alternate shutdown panel. This provides additional assurance that the plant can be brought to a safe shutdown state using the current Appendix R coping strategies for Appendix R fires.

Finally, an assessment is provided using best estimate tools typically found in PRA analyses to show that the uncertainties in the RCP seal performance under a loss of all seal cooling conditions does not increase plant risk as measured by conditional core damage probability. Specifically, the changes in RCP seal leakage rates, timing of the initiation of leakage and seal failure probabilities were not predicted to result in significant changes in conditional core

damage probability for Appendix R fire initiating events. The risk assessment also shows that changes to comply with the Appendix R pressurizer level requirement with an assumed 182 gpm per pump RCP seal leak, may be risk adverse and, in any event are not cost beneficial in terms of increasing safety.

It is also noted that the station blackout coping strategies required to meet 10 CFR 50.63 requirements only have to prevent core uncovery, as opposed to maintaining the pressurizer level on-scale for Appendix R coping strategies. While part of the basis for the more restrictive Appendix R criterion is the need to perform a plant cooldown from outside the control room, it is noted that the conservative analyses in Section 6 show that if a large RCP seal leakage occurs that causes a loss of pressurizer level, the pressurizer level can be restored by initiation of safety injection prior to the time that RCS cooldown to cold shutdown would be initiated. Additionally, actions to bring the plant to a cold shutdown state can be performed from outside the control room even in the event that pressurizer level is not available. Therefore, the Appendix requirement for maintaining pressurizer on-scale is a conservatism that does not significantly improve overall plant safety.

8.2 CONCLUSIONS

The overall conclusion of this report is that the RCP seal behavior for use in regulatory compliance space should be 21 gpm per pump. This conclusion applies to all loss of seal cooling events, regardless of whether seal injection cooling is restored after hot RCS fluid begins to heat the seal assembly, as long as the RCPs are tripped before the seals begin to heatup, This conclusion is supported by robust deterministic analyses, testing and operational experience, and risk analyses. The combination of the pressurizer level requirement in Appendix R and the PRA seal model of 182 gpm per pump leak rate is an overly conservative method for assuring fire safety and that conservatism may result in utilities implementing features that either have no impact on risk or actually increase risk and therefore have no safety benefit.

The information provided in this report does not change any previous Westinghouse or Westinghouse Owners Group recommendations related to reactor coolant pumps seals for a loss of all seal cooling event. For example, the WOG and Westinghouse guidance for restoration of seal cooling following a prolonged loss of AC power, as documented in Reference 8, remains unchanged. Additionally, while a change to the acceptable PRA model for RCP seal leakage may be appropriate as a result of the information in this report, the potential changes and attendant assessments to support those changes to the PRA model are not discussed in this report.

9. REFERENCES

1. "Final Significance Determination For White Findings And Notice Of Violation", (NRC Inspection Report No. 05000280/2004008 And 05000281/2004008 Surry Power Station), U.S. Nuclear Regulatory Commission, September 15, 2004.
2. "WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs", WCAP-15603, Rev. 1-A, Westinghouse Electric Co., June 2003.
3. "Trip Report: NRC Meeting on RCP Seal Performance in Triennial Fire Inspections (PA-WOG-0107, Rev. 1)", Letter WOG-04-341, July 1, 2004.
4. "Request for Enforcement Discretion for Reactor Coolant Pump Seal Performance Findings in Triennial Fire Protection Inspections (PA-WOG-0107, Rev. 1.), Letter WOG-04-365, July 1, 2004.
5. Reactor Coolant Pump Seal Performance and Appendix R Compliance, TB-04-22, Westinghouse Electric Company, LLC, November 17, 2004.
6. NRC letter from Herbert N. Berkow, to Frederick P. Schiffley, II, "Subject: NRC Staff Review of the Westinghouse Owners Group request for Enforcement Discretion for Reactor Coolant Pump Seal Performance Findings in Triennial Fire Protection Inspections (TAC NO. MC3865)", Accession No.: ML043170324; November 12, 2004.
7. "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power", WCAP-10541, Rev. 2, Westinghouse Electric Corp., November 1986 (Westinghouse Proprietary).
8. "Transmittal of 1996 ERG Maintenance Program Update", Letter WOG-97-034, Westinghouse Owners Group, February 28, 1997 (see also Direct Work Number DW-94-011, Response letter ERG-96-019, November 15, 1996).
9. "Closeout Of Generic Safety Issue 23, "Reactor Coolant Pump Seal Failure", internal NRC Memorandum from Thadani to Travers, November 8, 1999.
10. "Cost Benefit Analysis for Generic Issue 23; Reactor Coolant Pump Seal Failure", NUREG/CR-5167, April 1991.
11. "RCP Seal Leakage PRA Model Implementation Guidelines for Westinghouse PWRs", WCAP-16141, Westinghouse Electric Co., August 2003.
12. "Reactor Coolant Pump Shaft Seal Behavior During Station Blackout", NUREG/CR-4077, AECL, April, 1985.
13. "Reactor Coolant Pump Shaft Seal Stability During Station Blackout", NUREG/CR-4821, May 1987
14. "Leak Rate Analysis of the Westinghouse Reactor Coolant Pump", NUREG/CR-4294, July 1985
15. "High Temperature Extrusion Qualification Testing of Seals Eastern 7228A O-ring Compound", Supplement 1 to WCAP-10541, Revision 2, Westinghouse Electric Corp., 1988.
16. "Westinghouse Owners Group Transmittal of Information on Westinghouse Reactor Coolant Pump Seal Performance (MUHP 1063)", WOG Letter OG-99-086 from Liberatori (WOG) to Messrs. Bahadur and Wermiel (NRC), September 17, 1999.

17. "Nuclear power Plant 3 Loss of In/Out AC Current Incident Investigation Report", Executive Branch National Science Commission, April 9, 2001.
18. "Presentation at the MAAP Users Group Meeting with NRC", September 22, 2004.
19. Presentation by Steve Rosenau of Duke Energy to the WOG RCP Working Group, Revision 1, December 16, 2004.
20. "900 Mwe PWR Units Cold Thermal Shock Test on Type 93D Reactor Coolant Pump Shaft Seals at Shutdown", L. Randrianarivo, Electricite de France, January 1987.
21. Technical Basis and Guidance for Appendix R Pressurizer Level Exception", WCAP-16352-P, January, 2005.
22. "Standard Technical Specifications Westinghouse Plants," NUREG-1431, Rev. 2.
23. "Guidance Document for Modeling of RCP Seal Failures", BNL Technical Report W6211-08/99, Brookhaven National Laboratory, August 1999.
24. "RCP Seal Integrity Generic Issue B-23", Westinghouse/WOG Presentation to the NRC on July 15, 1987.
25. "Analysis and Transient Behavior of the Westinghouse 8-Inch No. 2 RCP Seal During a Loss of All Seal Cooling Event Representative of a Loss of All AC Power", Supplement 2 to WCAP-10541, Revision 2, Westinghouse Electric Corp., 1988.
26. "Major Power Grid Disturbance Causes Loss of Electrical Load and Reactor Trip R. E. Ginna Nuclear Power Station", LER 2003-002, October 9, 2003.