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October 6, 2011

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

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-11340

Subject: MHI's Revised Amended Response to US-APWR DCD RAI No.399-2992 Revision 0 (SRP 16)

References:
1) "Request For Additional Information No. 399-2992 Revision 0, SRP Section: 16 - Technical Specifications, Application Section: TS Section 3.4", dated June 18, 2009.
2) "MHI's Response to US-APWR DCD RAI No.399-2992 Revision 0," MHI letter UAP-HF-09377, July 13, 2009. (ML092010095)
3) "MHI's Revised Response to US-APWR DCD RAI No.399-2992 Revision 0", MHI letter UAP-HF-10434, dated December 22, 2010. (ML103620635)
4) "MHI's Amended Response to US-APWR DCD RAI No.399-2992 Revision 0", MHI letter UAP-HF-11160, dated May 30, 2011. (ML11152A238)

Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's Revised Amended Response to US-APWR DCD RAI No.399-2992 Revision 0 (SRP 16)". The original response (Reference 2) to the NRC's RAI (Reference 1) was discussed during a teleconference on December 2, 2010 and revised accordingly (Reference 3). The NRC subsequently requested that MHI revise the response to RAI 16-298 of Reference 3 during a teleconference on February 15, 2011. In Reference 4, MHI provided a second revised response to the NRC's RAI Question 16-298 of Reference 1. MHI has revised the "Impact on DCD" portion of the response to Question 16-298 provided in Reference 4 to include a revision to a Tier 1 table that was inadvertently missed in the previous response (Reference 4). The change to the Tier 1 table is being made to ensure consistency between the information contained in the Tier 1 and Tier 2 documents. This change does not include new information and therefore should have no impact on the DCD review schedule. This version of the response to Question 16-298 supersedes all prior versions of the response (Refs. 2, 3, and 4) in their entirety.

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc., if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

D081
NRO

Sincerely,

Y. Ogata

Yoshiki Ogata
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, Ltd.

Enclosures:

1. MHI's Revised Amended Response to US-APWR DCD RAI No.399-2992 Revision 0 (SRP 16) (non-proprietary)

CC: J. A. Ciocco
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ENCLOSURE 1

UAP-HF-11340
Docket No. 52-021

MHI's Revised Amended Response to US-APWR DCD
RAI No.399-2992 Revision 0
(SRP 16)

October 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/06/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 399-2992 REVISION 0**

SRP SECTION: **16 - TECHNICAL SPECIFICATIONS**

APPLICATION SECTION: **16**

DATE OF RAI ISSUE: **6/18/2009**

QUESTION NO.: 16-298

TS 3.4.9, Pressurizer.

Provide justification for the selected pressurizer water level limit of 92% specified in LCO 3.4.9.a and SR 3.4.9.1. Revise LCO 3.4.9.a, SR 3.4.9.1 and the associated TS bases, as appropriate.

In the APWR DCD Section 15.2, the initial conditions of the pressurizer water volume assumed in all heatup transients are below 1500 cu-ft. (see Figures 15.2.1-4, 15.2.7.4 and 15.2.8-3) which is less than 50% of the total pressurizer water volume (3000 cu-ft). A heatup transient or accident initiated at 92% pressurizer water level constitutes an unanalyzed event which could lead to a water solid pressurizer and liquid release through the pressurizer safety valves.

This information is needed to ensure adequacy and completeness of LCO 3.4.9 requirements.

ANSWER:

MHI will revise the TS 3.4.9 LCO such that the operability limit for pressurizer water level will be 60% span for MODE 1 and 92% span for MODES 2 and 3, in order that the most limiting anticipated operational occurrence (AOO) event will not result in pressurizer overfill nor water or two-phase relief through the pressurizer safety valves assuming that the initial pressurizer water level is at the LCO. This change ensures compliance with SRP 15.0 Section I.2.A.iii which requires that an AOO should not generate a postulated accident (PA) without other faults occurring independently.

Note that the pressurizer will not overfill in MODES 2 and 3 even if the initial pressurizer water level is 92% plus uncertainty. The potential heatup of the core is small because the maximum reactor power is less than 5% in MODES 2 and 3; therefore, the potential heatup and decay heat can be sufficiently removed by the secondary system and there is no risk of pressurizer overfill or water or two-phase relief. For this reason the LCO is maintained at 92% in MODES 2 and 3.

<u>Basis of the Revised Pressurizer Water Level LCO</u>

Each of the AOOs in Chapter 15 is considered in order to determine the limiting AOO for the determination of the pressurizer water level LCO. Each AOO event in Chapter 15 can be categorized relative to the protection function against pressurizer overfill and water or two-phase relief as follows:

Category-1	Events in which the increase in pressurizer water level can be mitigated by reactor trip
Category-2	Events in which the increase in pressurizer water level can continue after the reactor trip
Category-3	Events in which the increase in pressurizer water level can be terminated by isolating the charging inventory to the primary system
Category-4	Events in which the pressurizer water level decreases

For Category-1 events, the RCS fluid would shrink resulting in a decrease in pressurizer water level following the reduction in power and RCS temperature following the reactor trip. No further water level increase will occur due to sufficient decay heat removal by the secondary system. Therefore, the high pressurizer water level reactor trip will protect against pressurizer overfill and water or two-phase relief even if the initial water level is higher than the operating band, and hence these events are not taken into consideration when determining the value of the pressurizer water level LCO. Note that these events are implicitly considered when determining the high pressurizer water level reactor trip setpoint.

For Category-2 events, if the heat removal capacity of the secondary system is decreased when the reactor trips, the RCS temperature and pressure increase due to decay heat and result in an increase in pressurizer water level. Although such events will eventually be cooled down by the addition of emergency feedwater, the pressurizer water level will continue to increase until the cooling capacity of the secondary system is recovered and exceeds the decay heat generation. Therefore, if the initial water level is higher than the operating band, the pressurizer may be filled and water or two-phase relief may occur. For this reason, the Category-2 events should be taken into consideration when determining the value of the pressurizer water level LCO.

For Category-3 events, the increase of the pressurizer water level is terminated by automatic or manual operator actions to isolate charging flow. In these events, the pressurizer water level does not increase considerably after the isolation of the charging inventory into the primary system and therefore pressurizer overfill will not occur regardless of the initial pressurizer water level.

For Category-4 events, the pressurizer water level decreases and therefore it is not necessary to consider these events when determining the value of the pressurizer water level LCO.

Based on the definitions above, each AOO event in Chapter 15 can be categorized as described below.

1) <u>SRP 15.1 AOOs</u>
   The SRP 15.1 AOOs are AOOs that result in an increase in heat removal by the secondary system. As a result of the increased heat removal of the secondary side, these events all result in a cooldown of the RCS. The RCS cooldown causes the reactor coolant to shrink, which results in a decrease in pressurizer water level. The maximum pressurizer water level for all of these AOOs during the period analyzed is the initial water level. Therefore, these events are categorized as Category-4.

2) <u>SRP 15.2 AOOs</u>
   The SRP 15.2 AOOs are AOOs that result in a decrease in heat removal by the secondary system.

   For the loss of non-emergency AC power to the station auxiliaries (15.2.6) and loss of normal feedwater flow (15.2.7), the SG water level is reduced leading to the low SG water level reactor trip. Although the pressurizer water level temporarily decreases due to the reactor trip, it begins to increase again until the SG water level has recovered due to emergency feedwater and the cooling capacity of the SGs exceeds the decay heat generation. Therefore, these events are categorized as Category-2.

For loss of external load (15.2.1), just after the reactor trip, the RCS begins to cool down and the pressurizer water level decreases since there is a sufficient amount of SG secondary water and steam flow through the main steam safety valves. Therefore, this event is categorized as Category-1.

The other 15.2 AOOs such as turbine trip (15.2.2), loss of condenser vacuum (15.2.3), and closure of main steam isolation valve (15.2.4) are bounded by the loss of external load analysis (15.2.1) as described in the DCD. Steam pressure regulator failure (15.2.5) is not applicable to the US-APWR as stated in the DCD.

3) <u>SRP 15.3 AOOs</u>
The SRP 15.3 AOOs are AOOs that result in a decrease in reactor coolant system flow rate. Although the reduction in RCS flow results in an increase in RCS temperature that causes a slight pressurizer water level increase, it is quickly stabilized due to the sufficient removal of decay heat by the secondary system. Therefore, these events are categorized as Category-1.

4) <u>SRP 15.4 AOOs</u>
The SRP 15.4 AOOs are AOOs that result in reactivity and power distribution anomalies. Although some 15.4 AOOs can result in an increase in pressurizer water level due to the power increase caused by the reactivity insertion, the increase in power is terminated by the reactor trip and the pressurizer water level is quickly stabilized due to the sufficient removal of decay heat by the secondary system. Therefore, these events are categorized as Category-1.

5) <u>SRP 15.5 AOOs</u>
The only AOO applicable to the US-APWR in SRP 15.5 is the CVCS malfunction that increases reactor coolant inventory, which is evaluated in DCD Section 15.5.2. For this AOO, the increase in RCS inventory (increase in pressurizer water level) is the initiating event, which is caused by assuming the full-open failure of the charging flow control valve (maximum charging flow). However, if the pressurizer water level reaches 92% span, the high pressurizer water level reactor trip and the automatic closure of charging line is initiated. Thus, the increase in RCS inventory is automatically terminated and this event is categorized as Category-3. DCD Section 15.5.2 (and related references) will be revised to explain this design feature as described in the "Impact on DCD" section below.

6) <u>SRP 15.6 AOOs</u>
The only AOO applicable to the US-APWR in SRP 15.6 is the inadvertent opening of a PWR pressurizer pressure relief valve, which is evaluated in DCD Section 15.6.1. This event results in a depressurization of the RCS and hence the pressurizer water level decreases. The maximum pressurizer water level during the period analyzed is the initial water level. Therefore, this event is categorized as Category-4.

In summary, the following two events are candidates for the limiting event to be taken into consideration when determining the value of the pressurizer water level LCO.

- Loss of non-emergency AC power to the station auxiliaries (15.2.6)
- Loss of normal feedwater flow (15.2.7)

Based on the US-APWR DCD Chapter 15 results, the maximum pressurizer water volume increases for these two events are summarized in Table 16-298.1 below.

Table 16-298.1
Maximum Pressurizer Water Volume Increase for DCD Chapter 15 AOOs

Event (DCD Section)	Pressurizer Water Volume Increase (ft^3)
15.2.6	600
15.2.7	450

Note that even though the loss of external load event is categorized as Category-1, this event also results in an increase in pressurizer water level. However, based on US-APWR DCD Chapter 15 results, the pressurizer water volume increase for the loss of external load event in 15.2.1 is approximately 330 ft^3. Therefore, this event is much less limiting than the other two events in the table above and is not considered for further evaluation.

MHI has reanalyzed the DCD 15.2.6 and 15.2.7 events to include an additional pressurizer overfill case that uses the revised TS LCO 3.4.9 value of 60% as the initial pressurizer water level. The results for the two events are shown below in Figure 16-298.1 and Figure 16-298.2, respectively. In addition, the pressurizer volume figures include a line indicating the volume associated with the physical location of the safety valves. As indicated by the figures, the location of the valves is very near the top of the pressurizer and therefore the line lies just below the maximum pressurizer volume line. The 60% LCO value provides significant margin to either of these lines such that the small difference between the location of the valves and the top of the pressurizer is negligible. This confirms that the possibility of water or two-phase relief from these valves is precluded for these limiting Chapter 15 AOOs and that pressurizer overfill does not occur. The DCD sections for these two events will be revised to include the case starting from the TS LCO 3.4.9 as discussed in the "Impact on DCD" section below.

<u>Consistency of Revised TS LCO, DCD Subsection 5.4.10.1 and DCD Subsection 15.2.8</u>

As described above, the basis of the revised TS LCO is to prevent water or two-phase relief from the pressurizer safety valves for the limiting Chapter 15 AOOs. Although water relief is allowed for PAs like the feedwater line rupture, US-APWR DCD Subsection 5.4.10.1 provides the following design requirement as one of the design bases of the pressurizer:

- The steam volume is large enough to prevent water relief through the safety valves following a feedwater line rupture.

Due to this MHI design requirement, the US-APWR is designed to prevent water relief through the safety valves following a feedwater line rupture assuming the initial water level is less than or equal to the nominal level. The supporting analysis in Subsection 15.2.8 "Feedwater System Pipe Break Inside and Outside Containment" provides the basis for this design feature and therefore the initial water level in the analysis is the nominal programmed level (44.2%) plus the instrument uncertainty (3.4%).

On the other hand, if the initial water level is assumed to be 60% consistent with the revised TS LCO, the pressurizer may fill and relieve water through the safety valves. However, this case is not provided in Chapter 15 since the water relief will occur at a time well after reactor trip such that the core power will already be at decay heat levels and the amount of relieved water is bounded by the Loss of Coolant Accident analysis provided in Subsection 15.6.5.

In order to provide clarity of the assumptions regarding pressurizer water level, MHI will revise DCD Subsection 5.4.10.1 and Subsection 15.2.8.1 as discussed in the "Impact on DCD" section below to clearly indicate that the assumed pressurizer water level conditions of the feedwater line break analysis in Chapter 15 are consistent with and support the pressurizer design basis statement in Chapter 5.

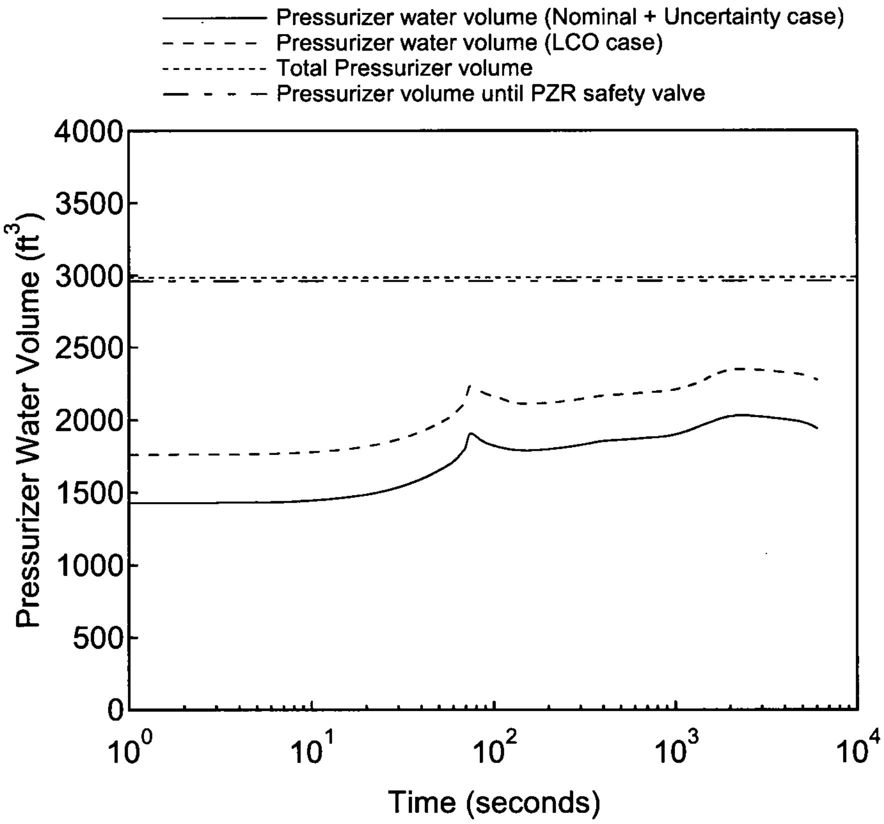

Figure 16.298-1 Pressurizer Water Volume vs. Time
Loss of Non-Emergency AC Power (DCD Subsection 15.2.6)

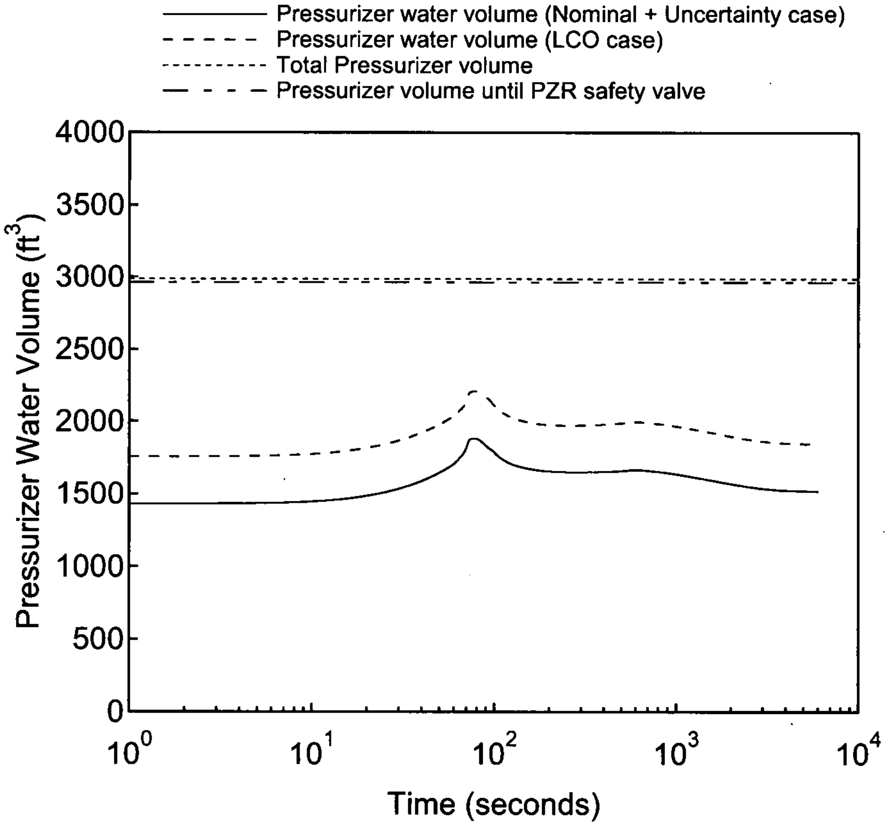

Figure 16-298.2 Pressurizer Water Volume vs. Time
Loss of Normal Feedwater Flow (DCD Subsection 15.2.7)

Impact on DCD

As described in this RAI response, DCD Section 15.5.2, "CVCS Malfunction that Increases Reactor Coolant Inventory", is revised to credit automatic isolation of CVCS with the occurrence of the high pressurizer water level signal for event termination. As a result of this change, DCD Tier 1 Table 2.5.4-3 and Tier 2 Table 7.5-5, and Section 15.0.0.6 are also revised in order to maintain consistency between the various places within the DCD that discuss manual operator actions. See Attachment 1 for the mark-ups of DCD Tier 1 Table 2.5.4-3, Table 7.5-5, DCD Section 15.0.0.6, and DCD Section 15.5.2 corresponding to this change.

DCD Chapter 16, GTS 3.4.9 and B3.4.9 BASES are amended to include a new requirement on pressurizer water level and the associated BASES changes. See Attachment 2 for the GTS and BASES mark-up.

The loss of non-emergency AC power to the station auxiliaries (15.2.6) and loss of normal feedwater flow (15.2.7) are both revised to include an additional pressurizer overfill case using the revised LCO 3.4.9 limit as the initial condition for pressurizer water level. See Attachment 3 for the mark-up of DCD Sections 15.2.6 and 15.2.7 associated with the revised analysis results.

As described in this RAI response, DCD Subsection 5.4.10.1, "Design Bases", is revised to clarify the pressurizer steam volume design basis and DCD Section 15.2.8, "Feedwater System Pipe Break Inside and Outside Containment" is revised to clearly indicate consistency with the revised pressurizer design basis. See Attachment 4 for the mark-up of DCD Subsection 5.4.10.1 and Section 15.2.8 corresponding to these clarifications.

Impact on R-COLA

There are impacts on R-COLA Part 4 to incorporate the changes to DCD Chapter 16, Technical Specifications.

Impact on S-COLA

There are impacts on S-COLA Part 4 to incorporate the changes to DCD Chapter 16, Technical Specifications.

Impact on PRA

There is no impact on the PRA.

This completes MHI's response to the NRC's question.

The following changes to DCD Tier 1 and Tier 2 Chapters 7 and 15 are necessary as a result of the change to DCD Section 15.5.2 to credit automatic CVCS isolation on high pressurizer water level as terminating the CVCS Malfunction that Increases Reactor Coolant Inventory event.

DCD Tier 1 Table 2.5.4-3 will be revised as follows:

Control Rod Insertion Limit Alarm
High Source Range Neutron Flux Alarm
~~High Pressurizer Water Level Alarm~~
Main Steam Line Radiation (N-16) Alarm
Low Pressurizer Water Level against Program Water Level Alarm
Containment High Range Area Radiation Alarm
Low Volume Control Tank Water Level Alarm

DCD Table 7.5-5 will be revised as follows:

Accident	Alarm	Credited Manual Action
Inadvertent Decrease in Boron Concentration in RCS (Subsection 15.4.6)	- Control Rod Insertion Limit Alarm - High Source Range Neutron Flux Alarm	Closure of Charging Flow Isolation Valve or Closure of Primary Makeup Water Control Valve or Stop of Primary Makeup Water Pump
~~CVCS Malfunction that Increases Reactor Coolant Inventory (Subsection 15.5.2)~~	~~High Pressurizer Water Level Alarm~~	~~Closure of Charging Line Isolation Valve or Charging Line Containment Isolation Valve~~
Radiological Consequences of a SG Tube Failure (Subsection 15.6.3)	- Main Steam Line Radiation (N-16) Alarm - Low Pressurizer Water Level against Programmed Water Level Alarm	- Manual reactor trip - Isolation of Affected SG - Cooldown of Primary Coolant System by using Main Steam Depressurization Valve - Equilibrium of Pressure between Primary and Secondary Coolant System by using Safety Depressurization Valve - Stop of Injection from ECCS
Rod Ejection Accidents (Subsection 15.4.8)	Containment High Range Area Radiation Alarm	-Manual C/V Spray System Operation -Manual Annulus Emergency Exhaust System Operation
Failure of Small Lines Carrying Primary Coolant Outside C/V (Subsection 15.6.2)	Low Volume Control Tank Water Level Alarm	RCS Sample Lines or CVCS Letdown Line Isolation

The fifth paragraph of DCD Section 15.0.0.6 will be revised as follows:

> Operator actions required to mitigate accidents are described in the individual event evaluation sections. The non-LOCA events whose analyses credit operator actions are inadvertent dilution of boron concentration in the RCS (Section 15.4.6)~~, CVCS malfunction that increases RCS inventory (Section 15.5.2),~~ and steam generator tube failure (Section 15.6.3). The radiological consequence events whose analyses credit operator actions are RCCA ejection (Section 15.4.8) and failure of small lines carrying primary coolant outside containment (Section 15.6.2). In addition, operator

actions are credited to prevent boric acid precipitation to assure post-LOCA long term cooling (Section 15.6.5).

DCD Section 15.5.2 will be revised as follows:

15.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

The increase in reactor coolant inventory due to the addition of borated water to the RCS by the chemical and volume control system (CVCS) is terminated by the automatic CVCS isolation function described in Subsection 7.3.1.5.11. After the high pressurizer water level setpoint for CVCS isolation is reached, the charging line isolation valves will close and no further increase in PZR water level will occur. However, this section evaluates the time available after the high pressurizer water level alarm for the operator to perform actions to end the transient before the pressurizer fills assuming the automatic CVCS isolation is hypothetically ignored. ~~This section describes the analysis of the increase in reactor coolant inventory due to the addition of borated water to the RCS by the chemical and volume control system (CVCS).~~ Subsection 15.4.6 analyzes the reactivity aspects of a boron dilution due to the addition of unborated water to the RCS by the CVCS.

15.5.2.1 Identification of Causes and Frequency Classification

A CVCS malfunction that increases RCS inventory can be caused by an operator error, a test sequence error, or an electrical malfunction. The CVCS normally operates with one charging pump running and a constant letdown flow through the letdown path. The increase of RCS inventory may be caused by an increase in charging flow with letdown operating or by isolation of the letdown path (letdown line and excess letdown line). If the CVCS boron concentration is larger than the RCS boron concentration, the reactor may experience a negative reactivity insertion resulting in a decrease in reactor power and subsequent coolant shrinkage.

This event is classified as an anticipated operational occurrence (AOO). Historically, this event has been classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.5-1). Event frequency conditions are described in Section 15.0.0.1.

15.5.2.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the CVCS malfunction that increases RCS inventory event is described in the results section.

Three cases are considered for this event. The CVCS normally operates with one charging pump running and a constant letdown flow through the letdown path. The increase of RCS inventory may be caused by the full-open failure of the charging flow control valve with one pump running, the spurious startup of a non-operating charging pump, or by the closure of the letdown path (letdown line and excess letdown line). Of these cases, the continuation of ~~a~~ the full-open failure of the charging flow control valve with one pump running has been shown to result in a slightly larger net CVCS flow addition, so only this case is described and analyzed in this section.

The full-open failure of the charging flow control valve causes a net increase in CVCS borated water flow from the volume control tank (VCT) into an RCS cold leg. This results both in a net increase in coolant mass to the RCS and, if the VCT boron concentration is larger than that in the RCS, an increase in RCS boron concentration. During the initial phase of the transient, the boration can cause an insertion of negative reactivity, which in turn can result in a power and RCS pressure decrease. Because the power decrease and pressure decrease have opposite effects on the DNBR, there is very little sensitivity to the difference in boron concentration between the VCT and the RCS. As a result, the CVCS boron is assumed to be injected at the RCS boron concentration, and the event is analyzed for pressurizer overfill only.

The net addition of mass to the RCS by the CVCS will result in an increase in pressurizer level. The pressurizer high level alarm is set 15% above the normal programmed level and will alarm in the control room to alert the operator that a level increase is in progress. If left unmitigated, the reactor will trip on a high pressurizer water level signal. In addition to reactor trip, the high pressurizer water level signal will also result in the automatic closure of the charging line isolation valves. The automatic closure of these valves will terminate this event eliminating the potential for filling the pressurizer. Although this event is mitigated by the automatic CVCS isolation, the amount of time available before the pressurizer fills can be determined by ignoring this design feature. In this case~~After the reactor trip~~, the CVCS charging pumps are assumed to continue to inject water, causing the potential for filling the pressurizer. The Barrier Performance evaluation addresses the maximum pressurizer level encountered during this transient and the time available for operator action to isolate the CVCS flow; however, no specific operator actions are assumed in the analysis.

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could, in turn, cause a loss of offsite power, which could, in turn, cause an RCP coastdown. As discussed in Subsection 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable.~~ Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.~~ However, the case without offsite power results in a slightly shorter time to pressurizer overfill. Therefore, a LOOP is considered to determine the minimum time available for operator action.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

The following automatic reactor trip signals are assumed to be available to provide protection from this transient:

- High pressurizer pressure
- High pressurizer water level

The automatic CVCS isolation on high pressurizer water level is also assumed to be available to provide protection from this transient.

The availability and adequacy of instrumentation and control is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event as discussed in Section 15.0.0.5.

15.5.2.3 Core and System Performance

This event is not limiting with respect to fuel damage limits. As a result, DNBR and related fuel parameters (e.g., heat fluxes, and RCS temperatures) are not presented. A single case is analyzed to evaluate peak pressurizer water volume ~~crediting operator actions to isolate CVCS~~ as described in Section 15.5.2.4.

15.5.2.4 Barrier Performance

15.5.2.4.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of reactor power, RCS pressure, and reactor coolant temperature following a CVCS malfunction that increases RCS inventory. The evaluation model also includes pressurizer spray and RCS safety valves. This evaluation model is described in Section 15.0.2.2.1. Additional details on the MARVEL-M code are provided in Reference 15.5-2.

15.5.2.4.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative results for a CVCS malfunction that increases RCS inventory:

- The initial power level is taken as 102 percent of the licensed core thermal power level. The nominal value of core power condition is described in Table 15.0-3.
- The initial reactor coolant temperature is 4°F below the nominal value and the initial pressurizer pressure is 30 psi above the nominal value. This combination of initial uncertainties minimizes the coolant shrinkage after reactor trip. The nominal values of reactor coolant temperature and pressure conditions are described in Table 15.0-3.
- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4 corresponding to beginning of fuel cycle conditions. The Doppler power coefficient is assumed to be the minimum feedback limit shown in Figure 15.0-2. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.
- The reactor is assumed to be automatically tripped by the high pressurizer water level signal. Table 15.0-4 summarizes the trip setpoint and signal delay time used in the analysis.
- Automatic CVCS isolation is not assumed to occur during the analysis.
- The analysis setpoint for the pressurizer high level alarm is conservatively assumed 20% above the normal programmed level.
- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. RCCA insertion characteristics assumed in the analysis are described in Section 15.0.0.2.5.
- The plant is assumed to be operating in manual rod control.
- The pressurizer heaters and pressurizer spray are assumed to operate as designed. This will minimize the time available for operator action (i.e., period of time between the high pressurizer level alarm and when the pressurizer fills).
- The pressurizer safety valves are modeled for this event. They are assumed to open at 2525 psia and be fully open at 2575 psia.
- Borated water from the volume control tank is assumed to be at the same concentration as the RCS.
- CVCS flow is conservatively assumed to be injected into the RCS cold legs by one charging pump from full power conditions at a constant 310 gpm. Letdown is assumed to be isolated.

15.5.2.4.3 Results

A single limiting case is analyzed to evaluate pressurizer overfill and the associated time available for manual actions to isolate the CVCS flow. The sequence and timing of major events for the CVCS malfunction that increases RCS inventory event is shown in Table 15.5.2-1.

Figures 15.5.2-1 through 15.5.2-5 are plots of the transient response of system parameters for the Barrier Performance Evaluation case.

In the evaluated case, the full-open failure of the charging flow control valve with one pump running leads to an addition of mass to the RCS resulting in an increase in the pressurizer water volume. Table 15.5.2-1 shows that the high pressurizer water level alarm occurs 404 seconds after the CVCS malfunction occurs. Table 15.5.2-1 shows that the reactor trips at 1062 seconds, as indicated by the distinctive drop in reactor power and RCS temperature at this time in Figures 15.5.2-1 and 15.5.2-4. The RCPs coast down at the same time due to the assumed LOOP. The CVCS charging pump continues to inject water until the pressurizer fills, which occurs at ~~1176~~ 1146 seconds per Figure 15.5.2-3. Thus, there are ~~12.8~~ 12.3 minutes available after the high pressurizer level alarm for the operator to perform actions to end the transient before the pressurizer fills. This case demonstrates that a reasonable amount of time exists for the operators to terminate this event if operator action were necessary. However, the event is actually terminated by the automatic CVCS isolation function described in Subsection 7.3.1.5.11. After the high pressurizer water level setpoint for CVCS isolation is reached, the charging line isolation valves will close and no further increase in PZR level will occur. Therefore, this event does not result in pressurizer overfill.

The CVCS malfunction that increases RCS inventory event does not result in exceeding any reactor coolant pressure boundary or containment volume fission product barrier design limits. The results of the pressurizer water volume case demonstrate that the RCS pressure and main steam system pressure remain well below 110% of their respective system design pressures. Therefore, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

15.5.2.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.5.2.6 Conclusions

The chemical and volume control system malfunction that increases reactor coolant system inventory event does not challenge the DNBR 95/95 limit, and no fuel failures are predicted.

Automatic CVCS isolation ~~Sufficient time exists to enable operator action to~~ prevents the pressurizer from filling and the RCS pressure and main steam system pressure remain well below 110% of their respective system design pressures, so the integrity of the reactor coolant pressure boundary and main steam system are maintained.

This event does not lead to a more serious fault condition.

DCD Table 15.5.2-1 will be revised as follows:

Event	Time (sec)
CVCS malfunction that increases RCS inventory	0.0
High pressurizer water level alarm	404
High pressurizer water level reactor trip analytical limit reached	1062
Reactor trip initiated (rod motion begins)	1064
RCP coastdown begins	1064
Peak pressurizer water volume occurs	~~1176~~ 1146

DCD Figures 15.5.2-1 through 15.5.2-5 will be revised as follows to reflect the analysis results consistent with the above changes.

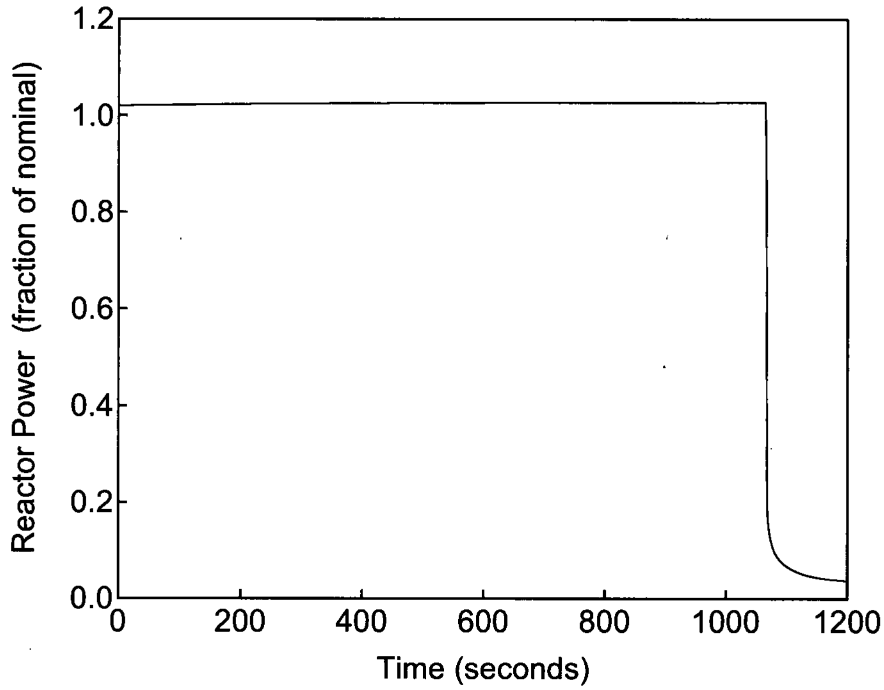

Figure 15.5.2-1 **Reactor Power versus Time**

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

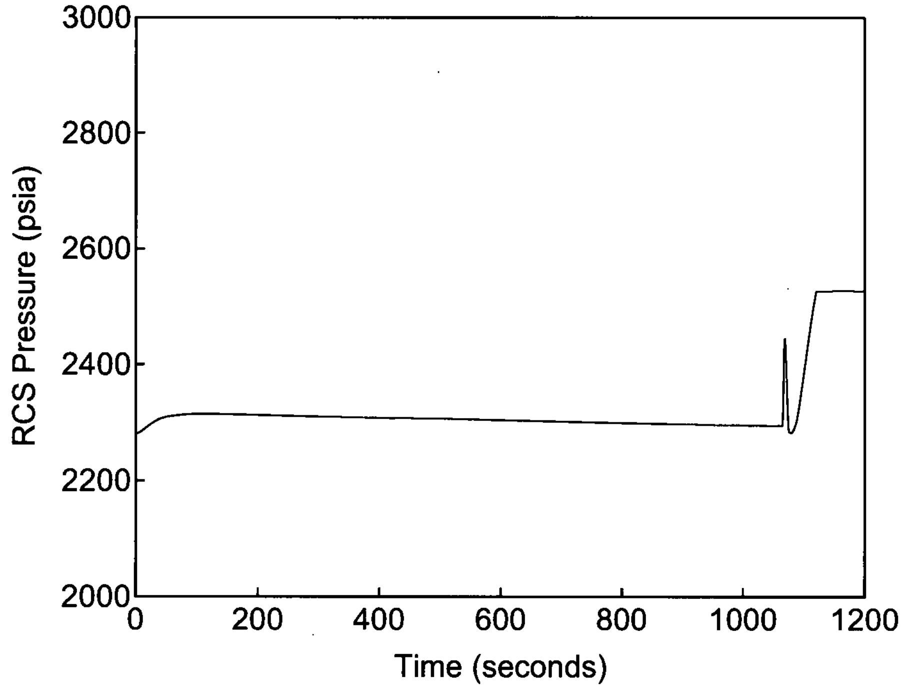

Figure 15.5.2-2 RCS Pressure versus Time

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

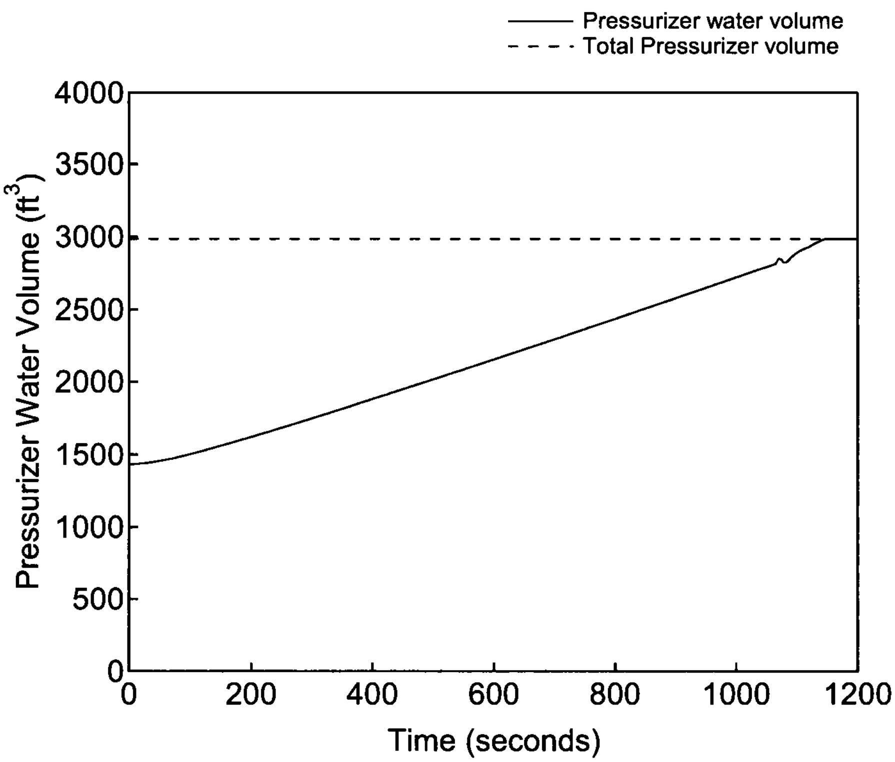

Figure 15.5.2-3 Pressurizer Water Volume versus Time

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

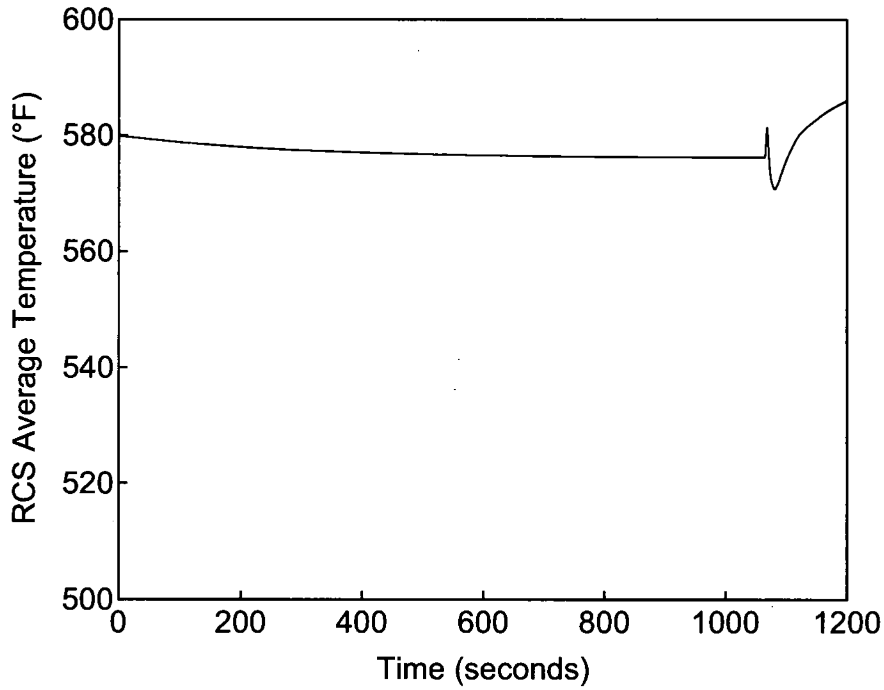

Figure 15.5.2-4 **RCS Average Temperature versus Time**

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

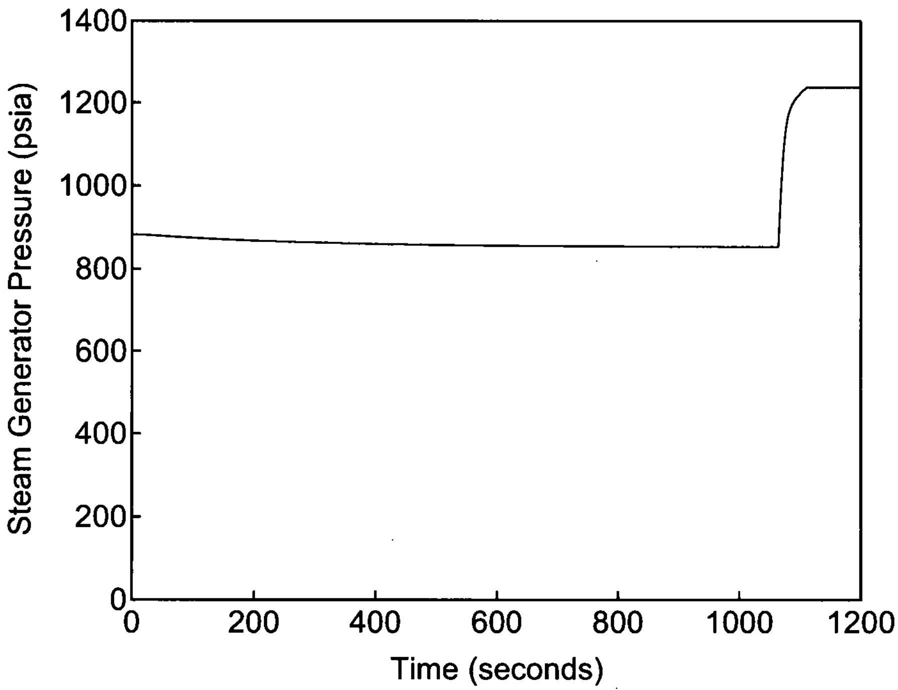

Figure 15.5.2-5 **Steam Generator Pressure versus Time**

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 60% (MODE 1)
Pressurizer water level ≤ 92% (MODES 2 and 3) and

b. Three groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 120 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Pressurizer water level not within limit.	A.1	Be in MODE 3.	6 hours
	AND		
	A.2	Fully insert all rods.	6 hours
	AND		
	A.3	Place Rod Control System in a condition incapable of rod withdrawal.	6 hours
	AND		
	A.4	Be in MODE 4.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One required group of pressurizer heaters inoperable.	B.1	Restore required group of pressurizer heaters to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ 60% in MODE 1, or ≤ 92% in MODES 2 and 3.	[12 hours OR In accordance with the Surveillance Frequency Control Program]
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is ≥ 120 kW.	[24 months OR In accordance with the Surveillance Frequency Control Program]

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls and emergency power supplies. Pressurizer safety valves and safety depressurization valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Safety Depressurization Valves (SDVs)," respectively.

The intent of the LCO is to ensure that a liquid-vapor interface ~~steam bubble~~ exists in the pressurizer ~~prior to power operation to minimize the consequences of potential overpressure transients~~ to permit effective RCS pressure control during normal operation and assure the pressurizer continues to provide proper pressure control response for Anticipated Operational Occurrences (AOOs). The presence of an adequate steam ~~bubble~~ volume is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control. The maximum water level defined by the LCO preserves the steam space necessary for pressure control, prevents overfilling the pressurizer, and ensures that two-phase or water relief does not lead to a more severe accident in accordance with the requirements of SRP 15.0 (Ref. 2). Note that the LCO is not intended to be interpreted as an operating band. The actual operating band for pressurizer water level is controlled more tightly than this LCO and is defined by the deviation alarms associated with the programmed level which is a function of power.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation

BASES

BACKGROUND (continued)

temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for an adequate steam volume ~~bubble~~ is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam ~~bubble~~ volume and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses do~~es~~ not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure (Ref. 1).

The maximum pressurizer water level limit, which ensures that a steam volume ~~bubble~~ exists in the pressurizer and prevents two-phase or water relief and pressurizer overfill, satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Some Chapter 15 AOOs result in an increase in RCS temperature and resultant increase in pressurizer level. For many of these events, the decrease in reactor power following reactor trip effectively terminates this increase in RCS temperature and leads to a stabilization or decrease in pressurizer level. Therefore, such events are protected from pressurizer overfill and water or two-phase relief by the high

BASES

APPLICABLE SAFETY ANALYSES (continued)

pressurizer water level reactor trip, specified in Table 3.3.1-1 of TS 3.3.1 and Table 7.2-3 of Section 7.2 of Chapter 7. This is also true for all Chapter 15 AOOs that begin from MODES 2 and 3 because the potential heatup of the core is limited by the low (≤ 5%) or zero power in those MODES.

However, certain Chapter 15 AOOs beginning from MODE 1, such as the loss of non-emergency AC power to the station auxiliaries (Ref. 3) and the loss of normal feedwater flow (Ref. 4), result in a continued increase in pressurizer water level even after reactor trip, mainly due to the presence of decay heat and reduced secondary heat sink capability. In these events, the initial steam volume needs to be sufficient to accommodate the increase in pressurizer water level without leading to overfill and two-phase or water relief. The basis for the pressurizer water level LCO value in MODE 1 is that the safety analysis of these limiting Chapter 15 AOO have sufficient margin to pressurizer overfill and two-phase or water relief when initiated from the LCO value.

Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. ~~2~~5), is the reason for providing an LCO.

LCO

The LCO requirement in MODE 1 for the pressurizer to be OPERABLE with a water volume ≤ ~~2668~~1757 cubic feet, which is equivalent to ~~92~~60%, ensures that a sufficient steam volume ~~bubble~~ exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam ~~bubble~~ volume is also consistent with analytical assumptions. This LCO requirement further ensures that the limiting AOO that results in an increase in pressurizer level that cannot be terminated by prior operator action will not result in pressurizer overfill and water or two-phase relief even if the event initiates from the LCO. The limiting AOOs for this LCO is the loss of non-emergency AC power to the station auxiliaries (Ref. 3) and loss of normal feedwater flow (Ref. 4).

BASES

LCO (continued)

The LCO requirement in MODES 2 and 3 for the pressurizer to be OPERABLE with a water volume ≤ 2600 cubic feet, which is equivalent to 92%, is provided for the same reasons as in MODE 1. However, the LCO value is higher due to the reduced risk of pressurizer overfill or water or two-phase relief since the initial power level is low (≤ 5%).

Note that these LCO requirements are not intended to define the operating band of the pressurizer water level. The operating band is controlled more tightly than these LCO requirements and is defined by the deviation alarms associated with the programmed level which is a function of power.

The LCO requires three groups of OPERABLE pressurizer heaters, each with a capacity ≥ 120 kW, capable of being powered from either the offsite power source or the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of 120 kW is derived from the use of three heaters rated at 46.8 kW each. The amount needed to maintain pressure is dependent on the heat losses.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODE~~S~~ 1~~ and 2~~. MODE 1 is the condition that provides minimum margin to pressurizer overfill and two-phase or water relief for AOOs that result in a net integrated pressurizer insurge. MODE 2 is applicable for the same reasons, although the LCO value is increased due to the lower initial power level of ≤ 5%. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters, capable of being powered from an emergency

BASES

APPLICABILITY (continued)

power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1, A.2, A.3, and A.4

~~Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level - High Trip.~~

If the pressurizer water level is not within the limit, action must be taken to bring the plant to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3 with all rods fully inserted and incapable of withdrawal. Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station powered heaters.

BASES

ACTIONS (continued)

C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below ~~the nominal upper limit to provide a minimum space for a steam bubble~~ 60% for MODE 1 and 92% for MODES 2 and 3. The Surveillance is performed by observing the indicated level. [The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumption of ensuring that a sufficient steam ~~bubble~~ volume exists in the pressurizer. Alarms are also available for early detection of abnormal level indications. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. [The Frequency of 24 months is considered adequate to detect heater degradation and based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle length. This equipment is not at risk of imminent damage as it is designed to

BASES

SURVEILLANCE REQUIREMENTS (continued)

remain functional and in good condition while in operation, thus significant degradation due to a longer surveillance interval should not be of major concern. The design reliability is, therefore, maintained by taking these considerations based on sound engineering judgment. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

REFERENCES

1. Subsection 15.0.0.2.2.
2. Standard Review Plan (SRP) Section 15.0 "Introduction – Transient and Accident Analyses".
3. Subsection 15.2.6
4. Subsection 15.2.7
5. NUREG-0737, November 1980.

The following additional description is added to Section 15.2.6.4.1, Evaluation Model:

The MARVEL-M plant transient analysis code is used to calculate transient responses of reactor power, RCS pressure, and reactor coolant temperature following a loss of non-emergency ac power. This evaluation model is described in Section 15.0.2.2.1. Additional details on the MARVEL-M code are provided in Reference 15.2-4.

Two analysis cases to maximize peak pressurizer water volume are performed as part of the barrier performance evaluation. These two cases are identical except for differences in the initial pressurizer water level condition assumptions. The case initiating from the nominal level plus uncertainty is consistent with the design basis of the pressurizer described in Section 5.4.10.1 and the case initiating from the maximum level allowed in LCO 3.4.9 demonstrates that there is no possibility for this AOO to lead to a more severe accident.

The first sentence of Section 15.2.6.4.2, Input Parameters and Initial Conditions, is revised as follows:

The following assumptions are utilized in order to calculate conservative pressurizer water volume transient results for a loss of non-emergency ac power event that begins from a nominal plus uncertainty pressurizer water level condition.

Add a new bullet item 2 in Section 15.2.6.4.2, Input Parameters and Initial Conditions, as follows:

- The initial pressurizer water level is assumed to be the nominal level plus uncertainty.

The following additional description is added to the end of Section 15.2.6.4.2, Input Parameters and Initial Conditions.

The same input parameters are used for the second loss of non-emergency ac power event pressurizer water volume case, with the exception of the initial pressurizer water level condition, which is described below.

- The initial pressurizer water level is assumed at the maximum pressurizer water level allowed in Technical Specification LCO 3.4.9.

The second paragraph of Section 15.2.6.4.3, Results, is revised as indicated below:

Figures 15.2.6-1 to 15.2.6-10 are plots of system parameters versus time for the Barrier Performance Evaluation base (nominal level plus uncertainty) case that demonstrates that natural circulation flow is established and is adequate to remove long-term decay heat following the event.

The last paragraph of Section 15.2.6.4.3, Results, is revised as indicated below:

Figure 15.2.6-4 shows that the maximum pressurizer water volume remains well below the pressurizer capacity throughout the transient regardless of whether the initial pressurizer water level is at the nominal plus uncertainty or LCO water level. Therefore, ~~all pressurizer safety valve flow is steam since~~ the pressurizer does not fill with water and water relief through the pressurizer safety valves is precluded.

Table 15.2.6-1 is revised as follows:

Table 15.2.6-1
Time Sequence of Events for Loss of Non-Emergency AC Power to the Station Auxiliaries - Pressurizer Water Volume Analysis

Event	Nominal Plus Uncertainty Case Time (sec)	LCO Case Time (Sec)
Main feedwater flow stops	0	0
Low steam generator water level analytical limit reached	66	66
Reactor trip initiated (rod motion begins), ac power is lost, reactor coolant pumps begin to coastdown	68	68
Pressurizer safety valves open	70	69
Maximum RCS pressure occurs	71	71
Main steam safety valves open	75	75
Emergency feedwater initiated	208	208
Core decay heat decreases to emergency feedwater system heat removal capacity	2319	2311
Maximum pressurizer water volume occurs	2319	2311

Figure 15.2.6-4 is revised to include the pressurizer water level transient for the new LCO case, in addition to the existing nominal plus uncertainty level case.

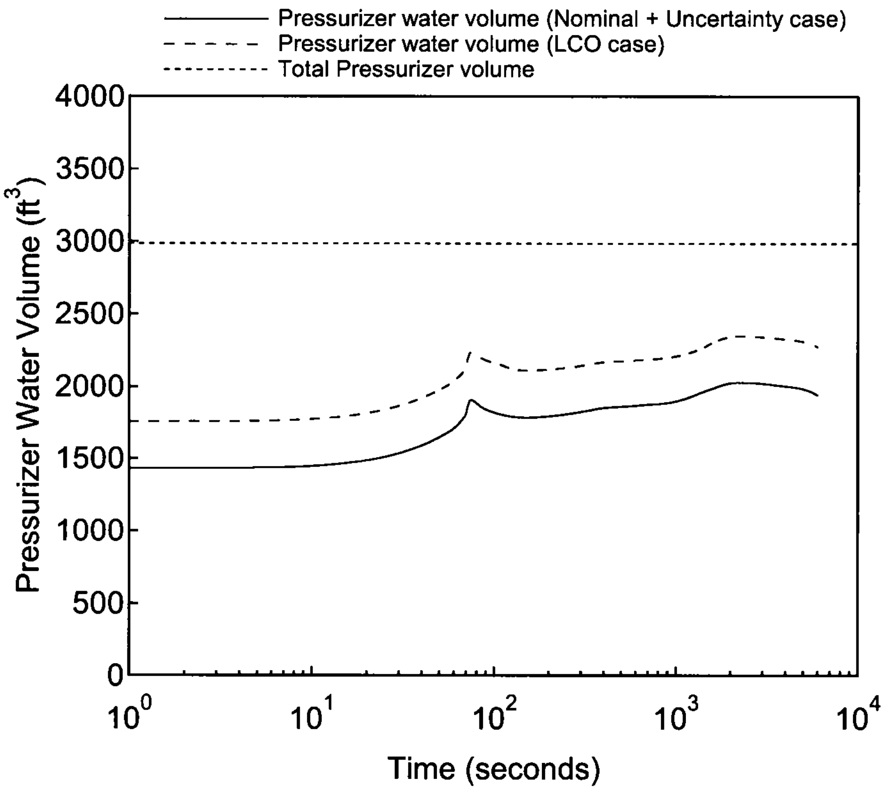

Figure 15.2.6-4 **Pressurizer Water Volume versus Time**

Loss of Non-Emergency AC Power to the Station Auxiliaries

The following additional description is added to the second paragraph of Section 15.2.7.4.1, Evaluation Model:

> Three analysis cases are performed as part of the barrier performance analyses, one to maximize peak RCS pressure and the other two cases to maximize peak pressurizer water volume using different initial pressurizer water level conditions. These cases are identical to the core and system evaluation cases for limiting DNBR, except for differences in initial conditions and control system operation assumptions indicated in Section 15.2.7.4.2 below. For the peak pressurizer water volume cases, the case initiating from the nominal level plus uncertainty is consistent with the design basis of the pressurizer described in Section 5.4.10.1 and the case initiating from the maximum level allowed in LCO 3.4.9 demonstrates that there is no possibility for this AOO to lead to a more severe accident.

Add a new bullet item to the description of the case that maximizes the peak pressurizer water volume in Section 15.2.7.4.2, Input Parameters and Initial Conditions:

- The initial pressurizer water level is assumed to be the nominal level plus uncertainty.

The following additional description is added to the end of Section 15.2.7.4.2, Input Parameters and Initial Conditions.

> The same input parameters are used for the second loss of normal feedwater flow event pressurizer water volume case, with the exception of the initial pressurizer water level condition, which is described below.
>
> - The initial pressurizer water level is assumed at the maximum pressurizer water level allowed in Technical Specification LCO 3.4.9.

The third paragraph of Section 15.2.7.4.3, Results, is revised as indicated below:

> Figure 15.2.7-11 shows that under conditions that maximize pressurizer water volume, the maximum pressurizer water volume remains well below the pressurizer capacity throughout the transient regardless of whether the initial pressurizer water level is at the nominal plus uncertainty or LCO water level. Therefore, the pressurizer does not fill with water and water relief through pressurizer safety valves is precluded. EFW is sufficient to provide decay heat removal from the steam generators following reactor trip until the RHR system can be used. No other plots are provided for the peak pressurizer water volume cases since the other parameters of interest are enveloped by the base barrier performance analysis case for peak RCS pressure.

DCD Figure 15.2.7-11 is revised to include the pressurizer water level transient for the new LCO case, in addition to the existing nominal plus uncertainty level case.

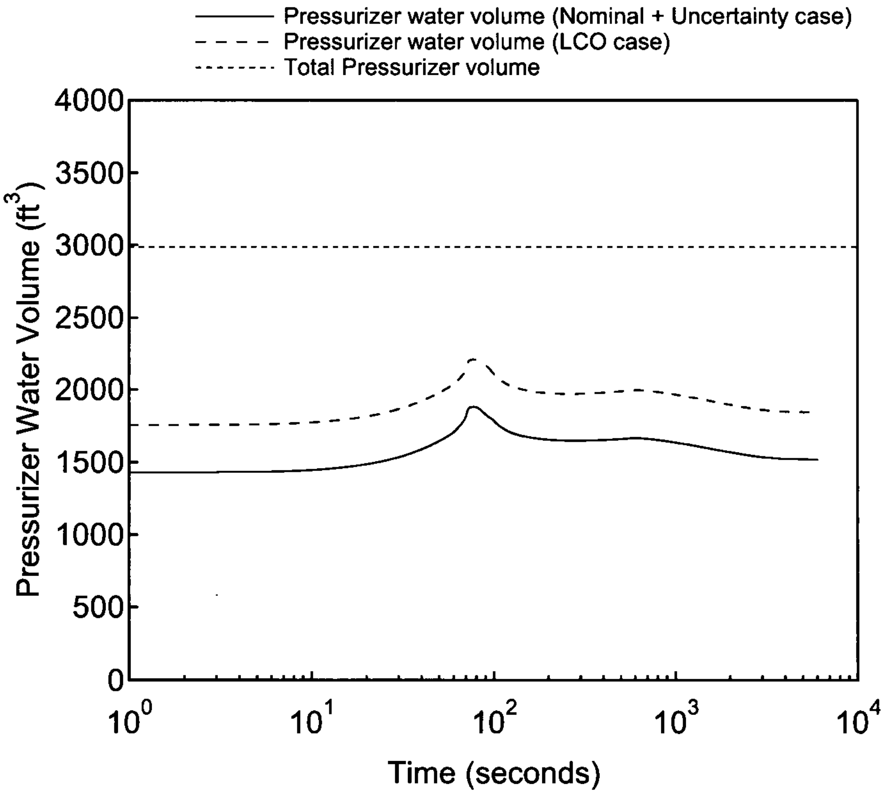

Figure 15.2.7-11 **Pressurizer Water Volume versus Time**

Loss of Normal Feedwater Flow
- Pressurizer Water Volume Analysis

Attachment 4

The fourth bullet of the first paragraph of DCD Subsection 5.4.10.1, Design Bases, will be revised as described below to clarify the pressurizer steam volume design basis:

- The steam volume is large enough to prevent water relief through the safety valves following a feedwater line rupture initiated from an initial pressurizer water level that is less than or equal to the nominal level plus instrument uncertainty.

The last paragraph of DCD Subsection 15.2.8.1, Identification of Causes and Frequency of Classification, will be revised as indicated below.

> Event frequency conditions are described in Section15.0.0.1. In addition to the general AOO and PA acceptance criteria described in Section 15.0.0.1, MHI conservatively adopts two additional acceptance criteria: (1) to evaluate hot leg boiling and (2) to not allow the pressurizer to overfill when the initial pressurizer water level is less than or equal to the nominal level plus instrument uncertainty.

The sixth paragraph of Subsection 15.2.8.2, Sequence of Events and Systems Operation, will be revised as follows:

> Each emergency feedwater pump supplies emergency feedwater independently to each steam generator taking water from the emergency feedwater pits. The EFWS is sized to have the capability of supplying sufficient EFW to preclude the pressurizer filling with water during a postulated feedwater system pipe break initiated from the nominal pressurizer water level plus instrument uncertainty, assuming a single failure ~~in loss~~ of one EFWS train. The protective actions are automated for the US-APWR.

The last paragraph of DCD Subsection 15.2.8.4.1, Evaluation Model, will be revised to include a new sentence as indentified below:

> Three cases are analyzed for the feedwater line break event. Each case is designed to evaluate the worst case conditions for (1) peak RCS pressure, (2) hot leg boiling, or (3) pressurizer water volume. The pressurizer water volume case provides the design basis of the pressurizer steam volume described in Subsection 5.4.10.1.

The 8th bullet in Subsection 15.2.8.4.2, Input Parameters and Initial Conditions, will be revised as indicated below:

- The event is analyzed assuming steam relief from the pressurizer safety valves. One of the cases verifies that the pressurizer does not over fill for this event when initiated from an initial pressurizer water level that is less than or equal to the nominal level plus instrument uncertainty, consistent with the pressurizer design basis provide in Subsection 5.4.10.1.