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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-10343

Subject: MHI's Revised Response to US-APWR DCD RAI No.399-2992 Revision 0 (SRP 16 TS Section 3.4)

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)

Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's Revised Response to US-APWR DCD RAI No. 399-2992 Revision 0". The original response (Reference 2) to the NRC's RAI (Reference 1) was discussed during a teleconference on December 2, 2010. The NRC requested that MHI revise the response to RAI 16-298 of Reference 2 during the teleconference. The enclosed material provides MHI's revised response to the NRC's RAI 16-298 of Reference 1.

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.

Sincerely,

Y. Ogata

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

Enclosure:

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

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NRO

CC: J. A. Ciocco
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ENCLOSURE 1

UAP-HF-10343
Docket No. 52-021

MHI's Revised Response to US-APWR DCD RAI No.399-2992
Revision 0

December 2010

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/22/2010

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 75% 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 water or two-phase relief through the pressurizer safety or relief 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 should be considered in order to determine the limiting AOO for the LCO. Each AOO event can be categorized according 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 after 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. Note that the US-APWR is designed to not overfill the pressurizer (i.e., become water solid) for such events if the event initiates from the nominal water level.

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 to 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 to 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 to 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) SRP 15.3 AOOs
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 to Category-1.

4) SRP 15.4 AOOs
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 to Category-1.

5) SRP 15.5 AOOs
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 to Category-3.

Note that this automatic isolation of the charging line is conservatively ignored in the analysis provided in DCD Section 15.5.2. The purpose of the analysis described in DCD Section 15.5.2 is to determine if there is sufficient time available for the operator to terminate the CVCS malfunction before the pressurizer fills with water. As described in DCD Table 7.5-5, the analysis credits the high pressurizer water level alarm for notifying the operator of the event; the occurrence of this alarm starts the clock for determining the time margin. The results of the analysis show that there is sufficient time (greater than 10 minutes) from the occurrence of the high pressurizer water level alarm for the operator to take manual actions to terminate this event before the pressurizer is filled.

6) SRP 15.6 AOOs
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 to Category-4.

In summary, the following 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)

The maximum pressurizer water volume increases for these US-APWR DCD Chapter 15 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

Based on the table above, the most limiting AOO for the LCO is the loss of offsite power described in DCD Section 15.2.6. The corresponding maximum increase in pressurizer water volume, as shown in Table 16-298.1 above, is 600 ft^3. Assuming that the initial pressurizer water level is at 75% span plus uncertainty (3.4%), the maximum pressurizer water volume during the transient is estimated to be 95.1%, which does not fill the pressurizer.

Note that the increase in pressurizer water volume is the integrated effect of the pressurizer insurge and outsurge during the transient, independent of initial level. Although the pressurizer water level analysis provided in DCD Section 15.2.6 starts from the nominal water level (44.2% span) plus uncertainty, the expansion of RCS coolant is virtually the same if the transient starts from 75% span plus uncertainty. Therefore, the safety function of the revised LCO that the limiting AOO does not result in pressurizer overfill or water or two-phase relief is sufficiently assured by the current Ch.15 analysis.

Sensitivity Analysis

To confirm the validity of this estimation, MHI performed a sensitivity analysis of the loss of non-emergency AC power to the station auxiliaries assuming that the initial pressurizer water level is 75% span plus uncertainty. Figure 16-298.1 compares the transient of pressurizer water volume for the DCD case and the sensitivity case which is initiated from 75% span plus uncertainty.

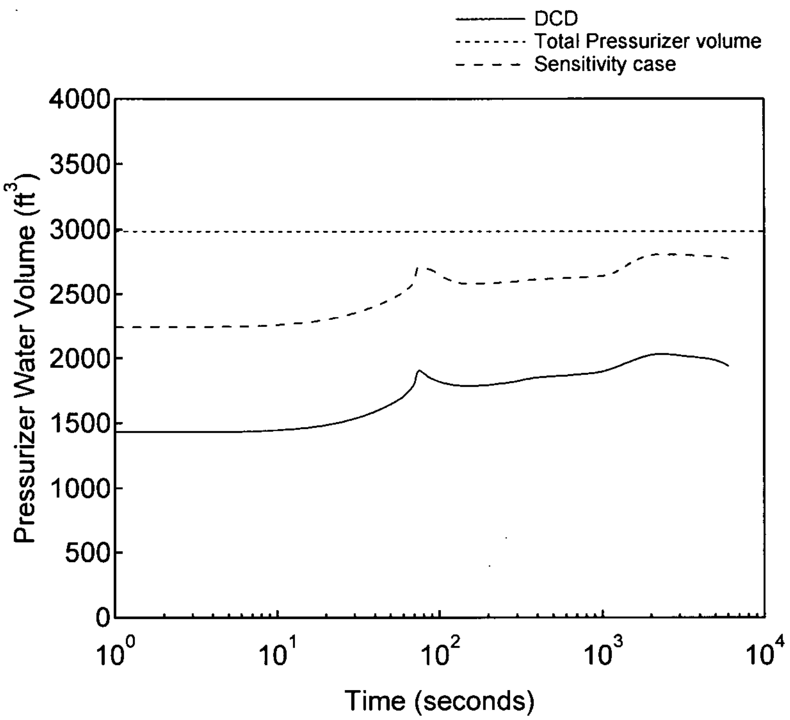

Figure 16-298.1 Pressurizer Water Volume - Loss of Non-Emergency AC Power to the Station Auxiliaries

For the sensitivity case, the increase in pressurizer water volume is 565 ft^3, and the maximum water volume reaches 93.9 %, which is lower than the estimate provided above (95.1%). The reason why the increase in pressurizer water volume is smaller for the sensitivity case is as follows:

> When the reactor trip occurs, pressure and water level decrease due to the decrease in the reactor power, which causes condensation in the saturated steam phase and flashing in the saturated water phase of the pressurizer. Since the sensitivity case has a smaller steam volume and larger water volume due to the higher initial pressurizer water level, the condensation volume in the steam phase is smaller while the flashing volume in the water phase is larger than the DCD case. After reactor trip, the pressurizer water level rises again due to decay heat generation. Since the expanded steam volume is larger due to the previously described reason, the calculated pressurizer water volume increase is lower for the sensitivity case.

Although this effect is small, the result of the sensitivity analysis shows that it is conservative to use the net increase in pressurizer water volume in the DCD case to estimate the maximum water volume when starting from the LCO water level plus uncertainty.

It also should be noted that the Chapter 15 safety analysis will continue to use the nominal initial conditions for pressurizer water level (with uncertainties applied for certain events) which is in accordance with NUREG-0800 Chapter 15 Section 15.0.I.3. This approach is analogous to and consistent with initial conditions for DNB events.

Impact on DCD

DCD Chapter 16, GTS 3.4.9 and B3.4.9 BASES are revised as follows:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 75% (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. AND A.2 Fully insert all rods. AND A.3 Place Rod Control System in a condition incapable of rod withdrawal. AND A.4 Be in MODE 4.	6 hours 6 hours 6 hours 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 ≤ 75% 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).

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

BASES

BACKGROUND (continued)

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 ~~bubble~~ volume 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 pressurizer water level trip setpoint, specified in Table 3.3.1-1 of TS 3.3.1. 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 ($\leq$ 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), result in a continued increase in pressurizer level even after reactor trip, mainly due to the presence of decay heat and reduced secondary

BASES

APPLICABLE SAFETY ANALYSES (continued)

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. Regardless of the initial pressurizer water level in these AOOs, the RCS expansion is essentially the same such that the integrated pressurizer insurge (and outsurge) determines the net increase in pressurizer water level. This net increase in pressurizer level is determined from the limiting Chapter 15 AOO and is the basis for the LCO.

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~~4), 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~~2152 cubic feet, which is equivalent to ~~92~~75%, 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 AOO for this LCO is the loss of non-emergency AC power to the station auxiliaries (Ref. 3).

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%).

The LCO requires three groups of OPERABLE pressurizer heaters,

BASES

LCO (continued)

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 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

BASES

ACTIONS (continued)

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.

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~~ 75% 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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 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. NUREG-0737, November 1980.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.