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September 9, 2008
GO2-08-124

10 CFR 50.90

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

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 LICENSE AMENDMENT REQUEST FOR PROPOSED CHANGES TO COLUMBIA TECHNICAL SPECIFICATION 3.3.6.1; CHANGE GROUP 1 PRIMARY CONTAINMENT ISOLATION VALVES REACTOR WATER LEVEL ISOLATION SIGNAL FROM LEVEL 2 TO LEVEL 1**

References;
1) NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Volume 1, Revision 3.
2) NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6," Volume 1, Revision 3.

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Energy Northwest hereby requests an amendment to the Columbia Generating Station (Columbia) Technical Specifications. The proposed change modifies Technical Specification (TS) 3.3.6.1, Primary Containment Isolation Instrumentation. The proposed change lowers the Group 1 Isolation Valves reactor water level isolation signal from Level 2 (L2) to Level 1 (L1).

Energy Northwest requests approval of this change prior to May 15, 2009. This approval date is requested in order to support implementation during the next refueling outage (R19), scheduled to begin in May 2009. The change will be fully implemented prior to completion of R19. This implementation period will provide adequate time for station documents to be revised using the appropriate change control mechanisms and for the necessary physical modifications to plant systems. If the NRC does not approve the license amendment request (LAR) in time to support R19, Energy Northwest will extend the implementation until the next refueling outage or outage of suitable length to allow completion of the implementation activities.

Energy Northwest is requesting review and approval of this amendment request in accordance with the schedule discussed above due to the nuclear safety benefit the change provides. The Group 1 isolation at L2 closes the main steam isolation valves (MSIVs), removing the availability of both the main condenser and the reactor feedwater (RFW) system. Removing the normal heat sink and normal reactor pressure vessel

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(RPV) pressure and inventory control systems so early in a transient or accident can complicate the scram recovery. Isolating the condenser and the RFW system at L2, while complicating the scram recovery, does not add any commensurate increase in safety. The loss of the main condenser as a heat sink places greater challenges on the RPV safety/relief valves (SRVs) and the primary containment. The loss of RFW means that the operators must depend on high pressure core spray (HPCS) and reactor core isolation cooling (RCIC) systems to restore and maintain RPV level. Following the scram and decrease in RPV level, the operators will have greater opportunity to restore RPV pressure and level control using the normal balance of plant systems if the MSIVs remain open until L1. The proposed change will also decrease the risks associated with inadvertent SRV openings, a stuck open SRV, and anticipated transient without scram (ATWS) scenarios.

Changing the Group 1 isolation signal from L2 to L1 is consistent with the values specified in Reference 1 and 2. As discussed further in the enclosure, several boiling water reactor (BWR) stations have received NRC approval to change the MSIV isolation signal from L2 to L1. All three Browns Ferry units received NRC approval of this change in September 1984. Susquehanna Steam Electric Station, Unit 2, and FitzPatrick Nuclear Power Plant both received approval of the requested change in 1986.

The Enclosure provides a technical and regulatory evaluation of the changes. Proposed TS and Bases page markups and retyped pages are included as attachments to the Enclosure.

This document contains no regulatory commitments.

Should you have any questions or require additional information regarding this matter, please contact Mr. MC Humphreys, Licensing Supervisor, at 509-377-4025.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the date of this letter.

Respectfully,

SK Gambhir
Vice President, Technical Services

Enclosure: Evaluation of the Proposed Change

cc: EE Collins, Jr. - NRC RIV
CF Lyon - NRC NRR
NRC Senior Resident Inspector/988C
RN Sherman - BPA/1399
JO Luce - EFSEC
RR Cowley – WDOH
WA Horin - Winston & Strawn

ENCLOSURE

Evaluation of the Proposed Change

Subject: License Amendment Request For Proposed Changes To Columbia Technical Specification 3.3.6.1; Change Group 1 Primary Containment Isolation Valves Reactor Water Level Isolation Signal From Level 2 to Level 1

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ENCLOSURE

1. SUMMARY DESCRIPTION

This evaluation supports a request to amend the Technical Specifications for Columbia Generating Station (Columbia). The proposed change would revise the Technical Specifications (TS) to lower the Group 1 primary containment isolation valves reactor pressure vessel (RPV) water level isolation signal from Level 2 (L2) to Level 1 (L1). The proposed change would affect TS 3.3.6.1, Primary Containment Isolation Instrumentation.

2. DETAILED DESCRIPTION

2.1 Change to Primary Containment Isolation Instrumentation (TS 3.3.6.1)

This license amendment request (LAR) seeks to revise Table 3.3.6.1-1 "Primary Containment Isolation Instrumentation." Specifically, Item 1.a of Table 3.3.6.1-1 will be revised to reflect a Group 1 isolation signal initiation will occur on "Reactor Vessel Water Level – Low Low Low, Level 1" with the allowable value of "≥-142.3 inches." This value is consistent with the Emergency Core Cooling System (ECCS) Reactor Vessel Water Level - Low Low Low, Level 1 allowable value in TS 3.3.5.1.

2.2 Reason for the Amendment

Energy Northwest is submitting the LAR because it will allow more energy to be released to the main condenser, rather than to the suppression pool, prior to isolation of the main steam isolation valves (MSIVs) after a reactor scram. Implementation of this change will allow the plant operators to control the RPV pressure following the initial transient without the use of safety relief valves (SRVs). This reduces the potential to present additional challenges to the plant operations staff and, therefore, reduces the probability of more risk-significant scrams. By removing this energy through the condenser rather than the suppression pool, the nuclear steam supply system (NSSS) safety is improved from the standpoint of reducing SRV challenges (and the potential for stuck open SRVs) and mitigating (along with other systems) the consequences of Anticipated Transient Without Scram (ATWS) events. Additionally, the proposed configuration is typical in other boiling water reactor (BWR) 4 and BWR 5 plants, as well as the BWR 6 plants that General Electric (GE) designed with the Group 1 isolation signal at L1.

3. TECHNICAL EVALUATION

3.1 System Description

The RPV water level initiation signals and alarm functions are defined to provide progressive protection for the reactor for potential variation in RPV coolant inventory. A schematic representation of the various RPV water level initiation signals and alarm functions is shown in Figure 3-1. Table 3-1 shows the RPV water level nominal settings and their primary functions.

One function of RPV L2 is to close the Group 1 isolation valves when the setpoint is reached. The Group 1 isolation valves include the MSIVs and the main steam line drain valves as show in Table 3-2.

3.2 Applicable FSAR Text and Figures

Final Safety Analysis Report (FSAR) Section 7.3.1.1.2, Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) and the referenced sections, tables, and figures describe the purpose, function, and design of the isolation and control systems. Anticipated (moderate frequency) operational occurrences (AOOs), off-design abnormal (unexpected) transients that induce system operating condition disturbances, and postulated accidents of low or extremely low probability (design bases accident [DBA] or limiting faults) are discussed in Chapter 15 of the FSAR. The scope of FSAR Chapter 15 will collectively be referred to as "transient analyses" throughout this document. The design bases for containment systems and ECCS are discussed in FSAR Section 6.2 and 6.3, respectively.

3.3 Analytical Methods, Applicable Standards, Data, and Results

The technical evaluation demonstrates that lowering the RPV water level at which the Group 1 isolation valves initiate from L2 to L1 will not reduce the level of protection offered by the design. The evaluation also supports the conclusion that the proposed amendment warrants a finding of no significant hazards. The technical evaluation considers the effect of the proposed change on the transient analyses for Columbia. The specific events evaluated in the report are the same as those defined in Columbia FSAR Chapter 15 (Reference 6.3).

Energy Northwest examined each transient event with the Group 1 isolation signal at both L2 and L1 to determine if a Group 1 isolation on low RPV water level would occur for a given event. If the Group 1 isolation did not initiate due to L2, the lower initiation setting of L1 would have no impact. This effectively screened out the events that did not need to be reevaluated or considered as a part of this technical evaluation. The results of this screening process are shown in Table 3-3.

3.4 Technical Details in Support of Safety Arguments

As shown in Table 3-3, many transients were not reanalyzed if:

1) L2 was not challenged, or
2) Group 1 isolation was initiated by a signal other than low RPV water level, or
3) the results were bounded by another event.

As such, no further discussion is required for these transients.

Changing the isolation signal for the Group 1 isolation valves from L2 to L1 will impact the following transients:

Event Description	Event Classification	FSAR Section
Loss of Feedwater Flow (LOFF)	AOO	15.2.7,
Loss of Coolant Accident (LOCA)	DBA	15.6.5 and 6.3, and
ATWS - LOFF	Limiting fault	15.8.2.

Energy Northwest evaluated these events against the acceptance criteria defined in Section 15.0.3 of the FSAR (Reference 6.3) for unacceptable results. The criteria for evaluation are summarized as follows:

- Unacceptable Results for Incidents of Moderate Frequency (AOO):
  - Release of radioactive material to the environs that exceeds the limits of 10 CFR 20,
  - Reactor operation induced fuel cladding failure,
  - Nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes, and
  - Containment stresses in excess of that allowed for the transient classification by applicable industry codes.
- Unacceptable Results for DBAs or Limiting Faults:
  - Radioactive material release that results in dose consequences that exceed the requirements of 10 CFR 50.67,
  - Failure of fuel cladding that would cause changes in core geometry such that core cooling would be inhibited,
  - Nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes,
  - Containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required, and
  - Radiation exposure to plant operations personnel in the main control room in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

Energy Northwest also evaluated the results against the acceptance criteria for ATWS events (special events), defined in Section 15.8.0 of the FSAR (Reference 6.3), and summarized as follows:

- The reactor coolant pressure boundary (RCPB) remains below emergency pressure limits.
- The containment pressure remains below design limits. The suppression pool temperature remains below local saturation temperature limits.
- A coolable geometry is maintained.

- Radiological releases are maintained within 10 CFR 50.67 limits.
- Equipment necessary to mitigate the postulated ATWS event provides a high degree of assurance (assurance of function) that it will function in the environment (pressure, temperature, humidity, and radiation) predicated to occur as a result of the ATWS event.

3.4.1 Loss of Feedwater Analysis

As described in the FSAR, during the LOFF event, the RPV water level decreases quickly causing a reactor scram on low water level - Level 3 (L3). After the scram, RPV water level continues to drop until it reaches L2, initiating High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC). The Group 1 isolation is also initiated at L2. At this point in the event, operators will use RCIC to maintain core cooling. With the Group 1 initiation signal lowered to L1, the reactor would not isolate at the same level that initiates the HPCS and RCIC systems. Thus, the lower Group 1 initiation level poses a different challenge to the HPCS and RCIC systems and further evaluation is warranted.

Energy Northwest contracted with AREVA, the Cycle 19 fuel vendor, to perform the necessary analysis. Based on that analysis, Energy Northwest concluded that the short-term response is benign, since the core inlet subcooling reduction decreases core power. Reaching the L3 RPV water level setpoint initiates the scram. Because of the power reduction, there are no fuel related issues (e.g., minimum critical power ratio [MCPR] or cladding strain) associated with the short-term response and there is no need for cycle-specific analyses. The long-term response is mainly concerned with justifying that the RPV water level does not fall to the top of active fuel prior to level recovery by the RCIC system, assuming no credit for the HPCS system. The HPCS failure was assumed because the system flow is significantly higher than the RCIC system flow.

The general LOFF event sequence is summarized below:

1. Initial core power is 102% of rated thermal power.
2. Feedwater flow is lost.
3. Core power decreases due to the decrease in subcooling.
4. Water level falls as the core continues to generate steam that goes to the turbine. The pressure controller will try to maintain turbine inlet pressure.
5. When the sensed water level reaches Level 4 (L4), in conjunction with a reactor feedwater (RFW) pump trip, the reactor recirculation (RRC) pump speed runs back to the 50% speed setpoint (30 Hz), creating a water level swell that tends to moderate the decrease in actual water level.
6. Core power decreases further due to reduced core flow.
7. When the sensed water level reaches L3, the reactor scrams and the RRC pump speed runs back to the minimum speed setpoint (15 Hz).

8. When the sensed water level reaches L2, the RRC pumps trip and the reduced core flow increases the core void fraction, tending to moderate the decrease in actual water level. At L2 the RCIC system is initiated. At this point, the HPCS would also start and inject at a rate about ten times that of RCIC, but no credit for HPCS is taken in this analysis.
9. The RCIC system takes steam from the RPV and pumps subcooled water into the RPV through the RPV head spray.
10. Steam continues to be created by the decay heat as makeup water is provided by the RCIC system. When the steam mass loss is less than the water gain from the RCIC system, the water level slowly starts to recover.

The analytical bases and assumptions are summarized below.

1. Initial water level is just above the L3.
2. The scram delay after L3 is reached is 5.05 second.
3. The RFW pumps are assumed to ramp down in 1 second.
4. Analytical limit (low) water level setpoints for L3 and L2 are assumed.
5. Initial power is 102%.
6. Decay heat after scram is based on the 10CFR50 Appendix K decay heat used for LOCA analyses.
7. No credit is taken for the HPCS system.
8. RCIC system injects 600 gpm with a 30-sec delay after the L2 initiation signal.
9. The L2 initiation signal delay is 5.05 second.
10. RCIC injection water temperature is 140°F.
11. The RRC pumps runback starts at L3, when the RFW pumps trip.
12. Since the initial water level is near the L3 point (i.e., below the L4 RRC pump runback point), once the RFW pumps trip the RRC pump speed runback begins, at the rate of 5% to 10% per second. Since the runback tends to raise water level (due to voiding in the core as core flow is reduced), the slower runback rate is used. The minimum runback speed is 15 Hz of rated.
13. The L2 RRC pump trip does not occur.

The result of the analysis show that the minimum RPV level falls to -110 inches, well above L1 nominal setpoint of -129 inches and the proposed allowable value of ≥-142.3 inches. The analytical value of -110 inches is 51 inches above the top of active fuel elevation of -161.2 inches. This assures adequate cooling of the core.

The RCIC system flow is sufficient to maintain the reactor water level above L1 by compensating for the steam flow through the turbine bypass valves (BPVs) to the main condenser. Consequently, throughout the event, the RPV level remains above the top of the active fuel. Since the RCIC system is capable of providing adequate core cooling with the Group 1 valve isolation initiation lowered to L1 and since L1 is not reached, the RPV is not isolated from the condenser. The turbine BPVs maintain the RPV pressure at approximately 950 pounds per square inch gauge (psig) and preclude SRV actuation. Suppression pool heat up due to the SRV discharge is therefore avoided.

In sum, with the Group 1 isolation signal lowered to L1, the LOFF event does not result in any temperature, pressure, or water level transient in excess of the criteria for which the fuel, RPV, or containment are designed. Therefore, barrier integrity and functions are maintained. In addition, lowering the Group 1 isolation signal to L1 will reduce the amount of heat discharged to the suppression pool.

3.4.2 LOCA Analysis

Energy Northwest evaluated the impact of lowering the Group 1 isolation signal from L2 to L1 on the LOCA analyses for the ECCS performance evaluation and DBA radiological consequences.

The ECCS LOCA analyses assume the Group 1 isolation occurs at the LOCA initiation. This is conservative relative to closing the MSIVs later in the event, based on either low water level or main steam line pressure. Closing the MSIVs at the LOCA initiation results in a bounding peak cladding temperature (PCT), therefore, the change in Group 1 isolation signal from L2 to L1 has no effect on the ECCS LOCA analysis.

Changing the Group 1 isolation signal from L2 to L1 has no effect on the DBA-LOCA radiological release. In both cases, the L2 and L1 isolations ensure the MSIVs close prior to the start of the radiological release.

Therefore, Energy Northwest concluded that, with respect to LOCA, lowering the Group 1 isolation initiation signal from L2 to L1 is acceptable, because lowering the isolation signal does not challenge the LOCA acceptance criteria for PCT or change the radiological consequences of a DBA-LOCA.

3.4.3 ATWS – Loss of Feedwater Analysis

Energy Northwest evaluated the impact of lowering the Group 1 isolation signal from L2 to L1 on the ATWS – LOFF event. This evaluation was based on the GE Hitachi Nuclear Energy (GEH) engineering reviews and calculations of the generic evaluation performed in Reference 6.4. Lowering the isolation signal does not challenge the ATWS acceptance criteria, as summarized below:

1. The RRC pump trip at L2 reduces the reactor power because of increased void generation.
2. The initiation of HPCS and RCIC at L2 will make up the RPV inventory lost through the steam lines.
3. The delayed isolation of the Group 1 valves will prevent excessive suppression pool heat up by maintaining the main condenser availability for as long as possible.

Therefore, Energy Northwest concluded that, with respect to ATWS, lowering the Group 1 isolation initiation signal from L2 to L1 is acceptable. For some ATWS events, maintaining the MSIVs open for a longer period of time will reduce the suppression pool

heatup without impacting the dose consequences for the event. The analysis demonstrates that the consequences of these events do not result in any temperature or pressure transient in excess of the design criteria for the fuel, the RPV, or containment. Therefore, barrier integrity and functions are maintained.

3.4.4 Conclusions Based on Transient Analyses

Energy Northwest reviewed the results of the proposed change on the impacted transient analyses, postulated accidents, and special events and concluded that lowering the Group 1 isolation signal to L1 will not cause a reduction in safety margin, a reduction in plant shutdown capability, an increase in radiation release, or a decrease in core cooling capability. Implementation of the proposed license amendment will not result in unacceptable safety consequences as defined in FSAR Sections 15.0.3 and 15.8.0.

The events which initiate the Group 1 isolation due to L2 are the LOFF, the LOCA, and the ATWS-LOFF. The analyses performed in support of this proposed amendment demonstrate that the lower initiation setting of L1 has no impact on the station's ability to meet the acceptance criteria related to radioactive dose, fuel damage, and system and containment stresses. The sequence of events for these analyses are changed such that the Group 1 valve isolation initiates at lower water level later in the event, the SRV opening is reduced or eliminated, and more heat is released to the condenser rather than directed by the SRVs to the suppression pool. The safety margin of the plant as defined in its FSAR Chapter 15 and Sections 6.2 and 6.3 is not reduced and the acceptance criteria as defined in FSAR Sections 15.0.3 and 15.8.0 are met.

3.5 Relationship to Other Relevant Amendments and NRC Issues

- Impact of transition to a new fuel vendor

  The analyses performed in support of this proposed change are based on the current licensing and design bases and do not reflect changes proposed by Energy Northwest for transition to Global Nuclear Fuel (GNF) as the fuel vendor for Cycle 20. The core reload analyses, regardless of fuel vendor, will be performed using conservative inputs and methods approved by the NRC. Therefore, each reload analysis will ensure that any change to the Group 1 isolation initiation does not impact the core limits.

3.6 Summary

To justify the isolation signal change, Energy Northwest evaluated the impact on the safety analyses. To support this change of Group 1 isolation signal, TS 3.3.6.1, Item 1.a of Table 3.3.6.1-1 must be revised as shown on Attachment 1 to change the Group 1 isolation on RPV level from L2 to L1. All other requirements (surveillance intervals, action statements, etc.) remain the same for "Primary Containment Isolation Instrumentation." The analyses of the impacted events demonstrate that when the Group 1 isolation initiation signal is lowered to L1, consequences of LOFF, LOCA, and ATWS-LOFF events do not result in any temperature, pressure, or water level transient in excess of the design criteria for the fuel, RPV, or containment. Therefore, barrier integrity and functions are maintained. The proposed change will allow more energy to be released to the main condenser, rather than to the suppression pool, following a reactor scram. This allows RPV pressure to be controlled following the initial transient using the BPVs, rather than the SRVs. This lessens the potential to present additional challenges to the plant operations staff and, therefore, lessens the chance of more risk-significant scrams. By removing this energy through the condenser rather than the suppression pool, the reactor system safety is improved from the standpoint of reducing SRV challenges (and the potential for stuck open SRVs), and mitigating (along with other systems) the consequences of ATWS events. Therefore, lowering the Group 1 isolation initiation signal to L1 has an insignificant impact to FSAR Chapter 15 events. The events continue to meet all acceptance criteria as defined in FSAR Sections 15.0.3 and 15.8.0.

Figure 3-1

Columbia Reactor Vessel Water Level Functions

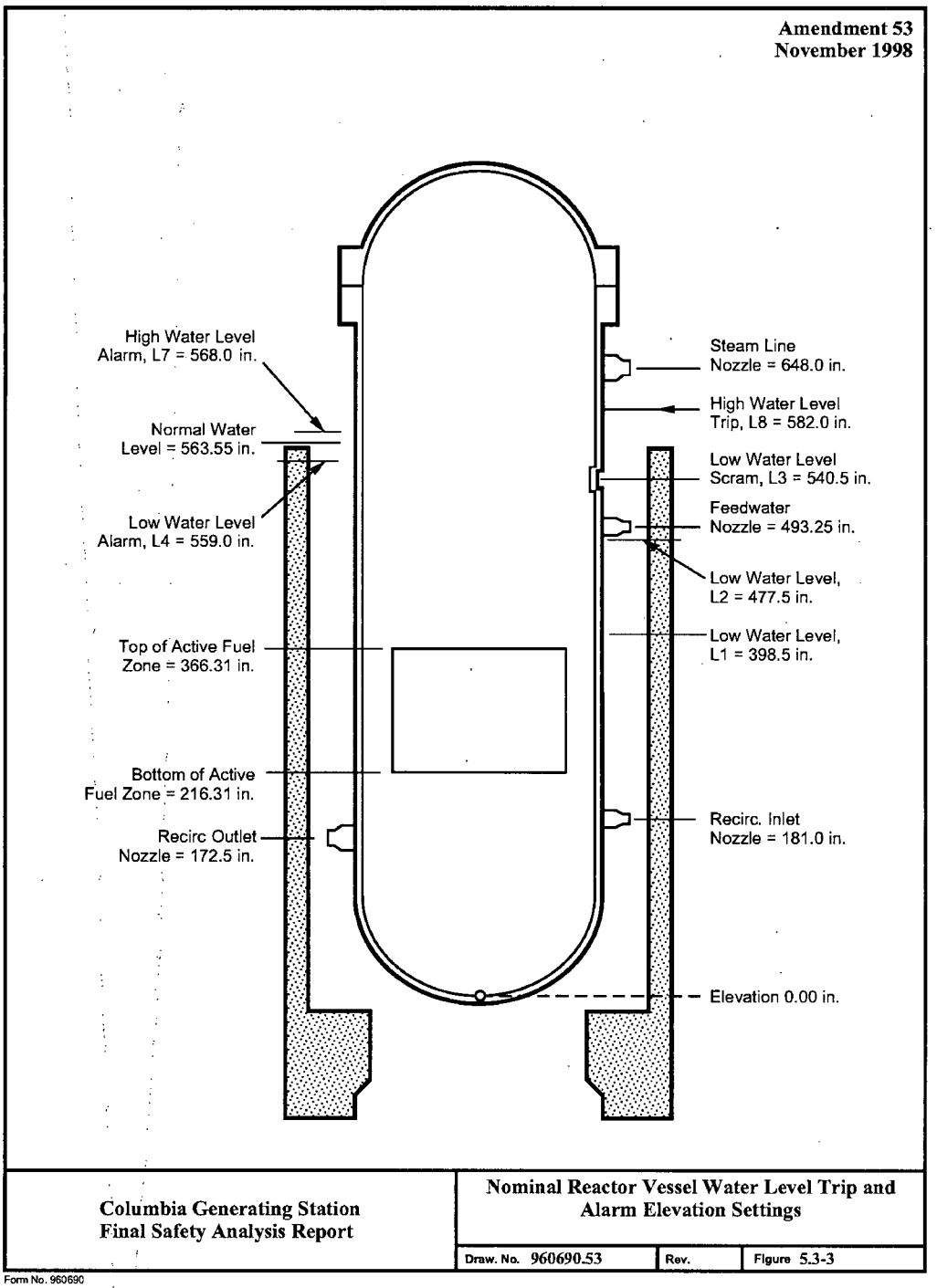

Table 3-1

Columbia Reactor Vessel Water Level Functions

Reactor Vessel Water Level	Elevation from Instrument Zero[a] (in)	Description of Major Functions
8	54.5	Reactor Feed Water Pump Trip Close Main Steam Turbine Valves Trip RCIC[b] and Close HPCS[b] Injection Valve
7	40.5	High Water Level Alarm
6,5	---	Normal Water Level
4	31.5	Low Water Level Alarm
3	13.0	Scram Reactor
2	-50	Initiate RCIC and HPCS Current isolation signal to close Group 1 Isolation Valves Trip Recirculation Pumps
1	-129	Initiate RHR and Core Spray Proposed new isolation signal for Group 1 Isolation Valves

a Instrument zero is 527.5 inches from the vessel bottom

b RCIC = Reactor Core Isolation Cooling

HPCS = High Pressure Core Spray

Table 3-2

Columbia Group 1 Isolation Valves

Valve Function	Valve Number
MS lines drain inboard	MS-V-16
MS lines drain outboard	MS-V-19
MS line A inboard MSIV	MS-V-22A
MS line B inboard MSIV	MS-V-22B
MS line C inboard MSIV	MS-V-22C
MS line D inboard MSIV	MS-V-22D
MS line A outboard MSIV	MS-V-28A
MS line B outboard MSIV	MS-V-28B
MS line C outboard MSIV	MS-V-28C
MS line D outboard MSIV	MS-V-28D
MS line A drain isolation	MS-V-67A
MS line B drain isolation	MS-V-67B
MS line C drain isolation	MS-V-67C
MS line D drain isolation	MS-V-67D

Table 3-3

Evaluation of Group 1 Isolation for FSAR Transient Analysis

FSAR Section	Event Description	Event Classification	Disposition/Comment
15.1.1	Loss of Feedwater Heating	AOO	No impact/ Level 2 not challenged.
15.1.2	Feedwater Controller Failure	AOO	Potential impact on long term level control is bounded by impact on loss of feedwater flow.
15.1.3	Pressure Regulator Failure – Open	AOO	No impact/ MSIVs close on low steam line pressure.
15.1.4	Inadvertent Safety/Relief Valve Opening	AOO	No impact/ Level 2 not challenged.
15.1.6	Inadvertent RHR Shutdown Cooling Operation	AOO	No impact/ Level 2 not challenged.
15.2.1	Pressure Regulator Failure - Closed	AOO	No impact/ Level 2 not challenged.
15.2.2	Generator Load Rejection	AOO	No impact/ Level 2 not challenged.
15.2.3	Turbine Trip	AOO	Potential impact on long term level control is bounded by impact on loss of feedwater flow.
15.2.4	Main Steam Isolation Valve Closures	AOO	No impact/ MSIVs close at event initiation.
15.2.5	Loss of Condenser Vacuum	AOO	No impact/ MSIVs close on low condenser vacuum.
15.2.6	Loss of Alternating Current Power	AOO	No impact/ MSIVs close on loss of power.
15.2.7	Loss of Feedwater Flow	AOO	No impact on core uncovery/level control. Additional evaluation provided in Section 3.4.1.
15.2.9	Failure of RHR Shutdown Cooling	AOO	No impact/ MSIVs close on loss of power.

Table 3-3 (Continued)

Evaluation of Group 1 Isolation for FSAR Transient Analysis

FSAR Section	Event Description	Event Classification	Disposition/Comment
15.3.1	Recirculation pump trip *a)* Single-pump trip *b)* Two-pump trip	 *a)* AOO *b)* AOO	 *a)* No impact/ Level 2 not challenged. *b)* Potential impact on long term level control is bounded by impact on loss of feedwater flow.
15.3.2	Recirculation Flow Control Failure – Decreasing Flow	AOO	No impact/ water level control is maintained.
15.3.3	Recirculation Pump Seizure *a)* Two-loop operation *b)* Single-loop operation	 *a)* Infrequent incident *b)* Limiting fault	 *a)* Potential impact on long term level control is bounded by impact on loss of feedwater flow. *b)* No impact.
15.3.4	Recirculation Pump Shaft Break	Infrequent incident	Potential impact on long term level control is bounded by impact on loss of feedwater flow.
15.4.1	Rod Withdrawal Error – Low Power	Infrequent incident	No impact/ Level 2 not challenged.
15.4.2	Rod Withdrawal Error – at Power	AOO	No impact/ Level 2 not challenged.
15.4.4	Startup of Idle Recirculation Pump	AOO	No impact/ Level 2 not challenged.
15.4.5	Recirculation Flow Control Failure with Increasing Flow	AOO	No impact/ Level 2 not challenged.
15.4.7	Misplaced Bundle Accident	Infrequent incident	No impact/ Level 2 not challenged.
15.4.9	Control Rod Drop Accident	Limiting fault	No impact/ Level 2 not challenged.
15.5.1	Inadvertent High-Pressure Core Spray Startup	AOO	No impact/ Level 2 not challenged.
15.6.2	Instrument Line Pipe Break	Limiting fault	No impact/ Level 2 not challenged.

Table 3-3 (Continued)

Evaluation of Group 1 Isolation for FSAR Transient Analysis

FSAR Section	Event Description	Event Classification	Disposition/Comment
15.6.4	Steam System Piping Break Outside Containment	Limiting fault	No impact/ MSIVs close on high steam flow.
15.6.5	LOCA	Design basis accident	No impact on radiological evaluation. See Section 3.4.2.
15.6.6	Feedwater Line Break	Limiting fault	Potential impact remains bounded by the limiting recirculation line break LOCA.
15.7.3	Postulated Radioactive Release Due to Liquid Radwaste Tank Failure	Limiting fault	No impact/ MSIV operation does not impact this event.
15.7.4	Fuel Handling Accident	Limiting fault	No impact/ MSIV operation does not impact this event.
15.7.5	Spent Fuel Cask Drop Accident	Not required	No impact/ Event not analyzed.
15.8.1	ATWS – Inadvertent Control Rod Withdrawal	Limiting fault	No impact/ Level 2 not challenged.
15.8.2	ATWS - Loss of Feedwater	Limiting fault	No impact on core uncovery/level control. Additional evaluation provided in Section 3.4.3.
15.8.3	ATWS - Loss of Alternating Current Power	Limiting fault	No impact/ MSIVs close on loss of power.
15.8.4	ATWS – Loss of Electrical Load	Limiting fault	No impact/ MSIVs close on loss of power.
15.8.5	ATWS – Loss of Condenser Vacuum	Limiting fault	No impact/ MSIVs close on low condenser vacuum.
15.8.6	ATWS – Turbine Trip	Limiting fault	Potential impact on core uncovery is bounded by ATWS loss of feedwater flow event.
15.8.7	ATWS – Closure of Main Steam Isolation Valves	Limiting fault	No impact/ MSIVs close at event initiation.

Table 3-3 (Continued)

Evaluation of Group 1 Isolation for FSAR Transient Analysis

FSAR Section	Event Description	Event Classification	Disposition/Comment
15.8.8	ATWS – Inadvertent Opening of Relief Valve	Limiting fault	No impact/ MSIV closure on low level disabled in analysis. MSIVs close on low steam line pressure.
15.8.9	ATWS – Pressure Regulator Failure – Open (PREGO)	Limiting fault	No impact/ MSIVs close on low steam line pressure.
5.2.2	ASME Over pressurization *a)* MSIV closure *b)* Turbine governor or throttle valve closure	Special event	 *a)* No impact/ MSIVs close at event initiation. *b)* Peak pressure during event occurs prior to water level approaching Level 2.
6.2	Containment Analysis	Design basis accident	No impact/ The small delay in the MSIV isolation allows removal of more energy from the containment prior to containment isolation and reactor depressurization.
6.3	ECCS Analysis	Design basis accident	No impact/ Cycle 19 LOCA analysis not impacted. See Section 3.4.2.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements and Criteria

The NRC acceptance criteria for primary containment related to the proposed license amendment are based on the General Design Criteria (GDC), 10 CFR 50.36, and 10 CFR 50.46 and are summarized as follows.

- GDC-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment;
- GDC-33, insofar as it requires that the system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary;
- GDC-35, insofar as it requires that the system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that:
  - Fuel and clad damage that could interfere with continued effective core cooling is prevented and
  - Clad metal-water reaction is limited to negligible amounts;
- GDC-50, insofar as it requires that the containment and its internal components be able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA;
- GDC-54 insofar as it requires that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems;
- 10 CFR 50.36, insofar as it specifies those requirements that should be included in TS; and
- 10 CFR 50.46, insofar as it requires that the ECCS be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of 10 CFR 50.46.

The objective of Item II.K.3.16 of NUREG-0737, "TMI Action Plan Requirements," (Reference 6.1) is to reduce the potential challenges and failures of SRVs. In response to this requirement, the BWR Owners Group (BWROG) submitted the results of a feasibility study and evaluation of various actions and modifications, which might reduce the challenges and failures of SRVs. One recommendation was to lower the MSIV closure signal from L2 to L1. The Columbia response to NUREG-0737, Item II.K.3.16 did not include implementation of the isolation signal change from L2 to L1 because Energy Northwest had installed SRVs with improved reliability. However, the change was implemented at a number of other BWR 4 and BWR 5 plants, as described in Section 4.2 and Reference 6.2.

4.2 Precedent

The proposed change is similar to license amendments 50 and 33 (ML021130102), issued to Commonwealth Edison Company on May 6, 1987, for LaSalle County Station Units 1 & 2, respectively. The 1987 amendment "revised the LaSalle County Station, Units 1 and 2 Technical Specifications to change the Group I Main Steam Isolation Valves' closure signal from Reactor Pressure Vessel Level 2 to Level 1." Like Columbia, LaSalle is a BWR 5 plant with a Mark II containment structure. The NRC issued similar amendments to the following BWR 4 plants with Mark I containments:

- Cooper Nuclear Station, Amendment 83, dated May 4, 1983 (ML021350409)
- Browns Ferry, Units 1, 2, and 3, Amendments 112, 106, and 80, dated September 19, 1984 (ML013650008)
- Peach Bottom Atomic Power Station, Units 2 and 3, Amendments 111 and 115, dated October 2, 1985 (ML021570504)
- Susquehanna Steam Electric Station, Unit 2, Amendment 25, dated April 1, 1986 (ML010180043) (Susquehanna has a Mark II containment.)
- FitzPatrick Nuclear Power Plant, Amendment 103, dated December 19, 1986 (ML010610096)

4.3 Significant Hazards Consideration

Pursuant to 10 CFR 50.90, Energy Northwest hereby requests an amendment to the Columbia TS. The proposed changes modify TS 3.3.6.1, "Primary Containment Isolation Instrumentation." The proposed change lowers the primary containment Group 1 isolation valves low RPV water level isolation signal from L2 to L1.

Energy Northwest has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Lowering the Group 1 isolation signal does not increase the probability of an accident, it changes only the level at which the isolation valves close. Isolation of the Group 1 valves occurs in response to lowering RPV water level during some transient events. As such, the isolation of Group 1 valves on lowering water level, which occurs in response to transients, is not an initiator of any transient or accident previously evaluated. Because the isolation of Group 1 valves on low water level occurs in response to some transients and is not an initiator of a transient event, lowering the level at which this isolation occurs does not impact the probability of an accident previously evaluated.

During some transients, delayed closure of the Group 1 isolation valves will reduce the chances of SRV actuation following an event by allowing the main condenser to remain available longer, without increasing the dose consequences of an event. Analyses performed show that lowering of the Group 1 isolation signal to L1 has no impact on the FSAR Chapter 15 events in terms of RPV limits, ability to maintain necessary coolant inventory, or fission product release. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   While the proposed change is a change to the Group 1 isolation initiation signal, the other requirements (surveillance intervals, action statements, etc.) remain the same for "Primary Containment Isolation Instrumentation." The methods used to test and determine operability of the instrumentation providing the low water level initiation for Group 1 isolation valves are unaffected by this change. This change does not change any equipment function, change the potential failure modes of any equipment, or alter any existing logic. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change to the Group 1 isolation signal from L2 to L1 allows more energy to be released to the main condenser (and reduces the amount potentially added to the suppression pool) after a reactor scram. This allows the operations staff and the turbine BPVs to control RPV pressure following the initial transient without the use of SRVs. This reduces the potential of additional challenges to the operations staff and plant equipment and therefore, reduces the probability of more risk-significant scrams. By removing this energy through the condenser rather than the suppression pool, the change requested improves reactor system safety from the standpoint of reducing SRV challenges (and the potential for stuck open SRVs). The analyses for transients and accidents that involve the Group 1 isolation demonstrate that the isolation occurs on signals other than low water level, or that adequate core cooling capability is maintained so that RPV water level does not decrease below acceptable levels. The analyses of the impacted events demonstrate that when the Group 1 isolation signal is lowered to L1, consequences of LOFF, LOCA, and ATWS-LOFF events do not result in any temperature, pressure, or

water level transient in excess of the design criteria for the fuel, RPV, or containment. Therefore barrier integrity and functions are maintained. For these reasons, the margin of safety is not reduced for any impacted event. Implementation of the proposed amendment would improve the margin of safety, in terms of reducing the probability of risk-significant scrams and reducing the amount of energy required to be absorbed by the suppression pool for some events. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Energy Northwest concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

The proposed amendment does not involve (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (3) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

6.1 "Clarification of TMI Action Plan Requirements," Office of Nuclear Reactor Regulation, US Nuclear Regulatory Commission, NUREG-0737, November 1980 - TASK II.K.3.16 - Reduction of Challenges and Failures of Relief Valves-Feasibility Study and System Modification.

6.2 "RPV Water Level Setpoint for MSIV Closure," General Electric Company, SIL No. 367, December 1981.

6.3 "Columbia Generating Station, Final Safety Analysis Report," Amendment 59, December 2007.

6.4 Low-Low Set Logic and Lower MSIV Water Level Trip for BWRs With Mark I Containment, NEDE-22223, September 1982.

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Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low, Level ~~2~~ 1	1,2,3	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ ~~-58~~ -142.3 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 804 psig
c. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 124.4 psid
d. Condenser Vacuum – Low	1,2[a], 3[a]	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 7.2 inches Hg vacuum
e. Main Steam Tunnel Temperature – High	1,2,3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
f. Main Steam Tunnel Differential Temperature – High	1,2,3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 90°F
g. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA
2. Primary Containment Isolation					
a. Reactor Vessel Water Level – Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 9.5 inches
b. Reactor Vessel Water Level – Low Low, Level 2	1,2,3	2[e]	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -58 inches
c. Drywell Pressure – High	1,2,3	2[e]	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 1.88 psig

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(a) With any turbine throttle valve not closed.

(e) Also required to initiate the associated LOCA Time Delay Relay Function pursuant to LCO 3.3.5.1.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for process and all instrument uncertainties, except drift and calibration. The trip setpoints are derived from the analytic limits, corrected for process and all instrument uncertainties, including drift and calibration. The trip setpoints derived in this manner provide adequate protection because all instrumentation uncertainties and process effects are taken into account.

Certain Emergency Core Cooling Systems (ECCS) and RCIC valves (e.g., minimum flow) also serve the dual function of automatic PCIVs. The signals that isolate these valves are also associated with the automatic initiation of the ECCS and RCIC. Some instrumentation and ACTIONS associated with these signals are addressed in LCO 3.3.5.1, "ECCS Instrumentation," and LCO 3.3.5.2, "RCIC System Instrumentation," and are not included in this LCO.

In general, the individual Functions are required to be OPERABLE in MODES 1, 2, and 3 consistent with the Applicability for LCO 3.6.1.1, "Primary Containment." Functions that have different Applicabilities are discussed below in the individual Functions discussion.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Main Steam Line Isolation

1.a. Reactor Vessel Water Level – Low Low, Level ~~2~~ Low 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level – Low Low, Level ~~2~~ Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level – Low Low, Level ~~2~~ Function associated with isolation is assumed in the analysis of the

Low 1 1 Low

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level – Low Low, Level ~~2~~ (continued)

recirculation line break (Ref. 1). The isolation of the MSL on Level ~~2~~ supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four differential pressure switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level – Low Low, Level ~~2~~ Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level – Low Low, Level ~~2~~ Allowable Value is chosen to be the same as the ECCS Level ~~2~~ Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure – Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hour if the pressure loss is allowed to continue. The Main Steam Line Pressure – Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 4). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hour) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four sensors that are connected to the MSL header. The sensors are arranged such that, even though physically separated from each other, each sensor is able to detect low MSL pressure.

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Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -142.3 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 804 psig
c. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 124.4 psid
d. Condenser Vacuum – Low	1,2[a], 3[a]	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 7.2 inches Hg vacuum
e. Main Steam Tunnel Temperature – High	1,2,3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
f. Main Steam Tunnel Differential Temperature – High	1,2,3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 90°F
g. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA
2. Primary Containment Isolation					
a. Reactor Vessel Water Level – Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 9.5 inches
b. Reactor Vessel Water Level – Low Low, Level 2	1,2,3	2[e]	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -58 inches
c. Drywell Pressure – High	1,2,3	2[e]	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 1.88 psig

(continued)

(a) With any turbine throttle valve not closed.

(e) Also required to initiate the associated LOCA Time Delay Relay Function pursuant to LCO 3.3.5.1.

Table 3.3.6.1-1 (page 2 of 4)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment Isolation (continued)					
d. Reactor Building Vent Exhaust Plenum Radiation – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 16.0 mR/hr
e. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA
3. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 250 inches wg
b. RCIC Steam Line Flow – Time Delay	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 3.00 seconds
c. RCIC Steam Supply Pressure – Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 61 psig
d. RCIC Turbine Exhaust Diaphragm Pressure – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 20 psig
e. RCIC Equipment Room Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 180°F
f. RCIC Equipment Room Area Differential Temperature – High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 60°F
g. RWCU/RCIC Steam Line Routing Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 180°F
h. Manual Initiation	1,2,3	1[(b)]	G	SR 3.3.6.1.6	NA
4. RWCU System Isolation					
a. Differential Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 67.4 gpm

(continued)

(b) RCIC Manual Initiation only inputs into one of the two trip systems.