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June 3, 2002
IPN-02-043

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

SUBJECT: Indian Point Nuclear Generating Unit No.3
Docket No. 50-286
Proposed Changes to Technical Specifications:
Relaxation of Pressurizer Water Level Requirement and
Removal of One-Time Allowance for Station Battery Replacement

REFERENCE: (1) NRC letter to Entergy Nuclear Operations, Inc; "Issuance of Amendment 208 to Indian Point 3 Technical Specifications to Allow a One-Time Replacement of Station Batteries While at Power," dated Sept. 19, 2001.

Dear Sir:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc, (ENO) hereby requests the following amendments to the Operating License for Indian Point 3 Nuclear Generating Unit No.3.

Indian Point 3 Technical Specification 3.4.9 requires the pressurizer to be operable with a water level of less than or equal to 58.3% in Modes 1, 2, and 3. The proposed amendment will retain this requirement in Modes 1 and 2, but will establish a level limit of less than or equal to 90% for Mode 3. This new allowance will provide additional operational flexibility and efficiency for performing a plant cooldown.

The proposed change has been evaluated in accordance with 10 CFR 50.91 (a)(1) using the criteria of 10 CFR 50.92 (c) and ENO has determined that this proposed change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

ENO also requests an administrative change to Technical Specification 3.8.4 to remove the one-time note added by Amendment 208 (Reference 1). The station battery replacement project is complete and the note is no longer needed.

A001

There are no new commitments identified in this letter. ENO requests approval of the proposed amendment by January 10, 2003. Once approved, the amendment will be implemented within 30 days. If you have any questions or require additional information, please contact Mr. Kevin Kingsley, NRR Project Manager at 914-734-6034.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 6-3-2002

Very truly yours,

Robert J. Barrett
Vice President, Operations – IP3
Indian Point 3 Nuclear Power Plant

Attachments:
I. Analysis of Proposed Technical Specification Change
II. Proposed Technical Specification and Bases Changes (markup)

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ATTACHMENT I TO IPN-02-043

ANALYSIS OF PROPOSED

TECHNICAL SPECIFICATION CHANGE REGARDING

PRESSURIZER WATER LEVEL REQUIREMENT

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-64, Docket No. 50-286 for Indian Point Nuclear Generating Unit No. 3.

The proposed change to Section 3.4.9 of the Indian Point 3 Technical Specifications will establish a new limit for pressurizer water level (≤ 90%) in MODE 3. The existing level requirement (≤ 58.3%) for MODES 1 and 2 is not being changed. The reason for the proposed amendment is to allow greater operational flexibility and efficiency when performing a plant cooldown.

In addition, an amendment to Technical Specification Section 3.8.4 is requested to remove the one-time note added by Amendment 208. The note was added to support the one-time replacement of station batteries 31 and 32 with the plant on-line. Station battery replacement was successfully performed in February and March 2002, and the note is no longer needed. This is an administrative change because removal of the note will restore the Technical Specification requirements to a condition previously approved by the NRC.

2.0 PROPOSED CHANGE

Indian Point 3 Technical Specification LCO 3.4.9 currently states:

> "The pressurizer shall be OPERABLE with:
>
> a. Pressurizer water level ≤ 58.3%; and ....."
>
> The applicability for this LCO is MODES 1, 2, and 3.

The proposed amendment will revise LCO 3.4.9 to state:

> "The pressurizer shall be OPERABLE with:
>
> a. **Actual** pressurizer water level ≤ 58.3% **in MODES 1 and 2 or ≤ 90% in MODE 3**; and ....."

The applicability statement for this LCO is not being changed.

Indian Point 3 Technical Specification surveillance SR 3.4.9.1 currently states:

> "Verify pressurizer water level is ≤ 58.3%."

The proposed amendment will revise SR 3.4.9.1 to state:

> "Verify **actual** pressurizer water level is ≤ 58.3% **in MODES 1 and 2 <u>OR</u> ≤ 90% in MODE 3**."

The frequency of 12 hours for this surveillance is not being changed.

Proposed changes to the Bases Section 3.4.9 pertaining to the proposed new pressurizer water level limit in Mode 3 are shown in Attachment II.

In summary, the proposed amendment will establish a different limit for pressurizer water level in Mode 3 compared to Modes 1 and 2. The new higher limit in Mode 3 will provide the following benefits:

- accommodate contraction of reactor coolant during cooldown,
- allow greater flexibility for establishing boron concentration required for shutdown margin,
- reduce additional RCS makeup required for establishing the 'pressurizer-solid' condition in Mode 4.

Overall, these benefits are expected to result in a time savings of 1 – 2 hours for performing a plant cooldown. The LCO and the surveillance are also being revised by the addition of the word 'actual' to emphasize to plant operators that instrument uncertainty is not included in the specified limiting values. The allowance for instrument uncertainty is identified in the Technical Specification Bases.

The administrative change to Technical Specification Section 3.8.4 will remove the one-time notes applied to the completion time for Condition B and the Surveillance Requirements SR 3.8.4.3 and SR 3.8.4.4. The notes to be deleted are shown in Attachment II.

3.0 BACKGROUND

The pressurizer is a component in the reactor coolant system (RCS) that is used to maintain required RCS pressure during steady state operations. The pressurizer also limits pressure changes caused by reactor coolant thermal expansion and contraction during load transients. The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes. The proposed change will not alter these design features of the pressurizer.

The water level in the pressurizer, and the corresponding steam space volume, is maintained by a control system that varies level as a function of reactor coolant average temperature, T_{avg}. At the low power temperature (T_{avg} = 547 ^{0}F) the control system is programmed to maintain level at approximately 23%. At the full power temperature (T_{avg} = 571 ^{0}F) the control system maintains level at approximately 51.3%. The analytical limit of 58.3% specified in LCO 3.4.9 includes an allowance of 7% for instrument uncertainty. This temperature-dependent level program is designed to maintain a constant mass of reactor coolant over the programmed temperature range for power operation. The water level maintained by this program is sufficient so that the pressurizer will not empty on a reactor trip from 100% power. The steam space associated with this water level is sufficient to prevent water relief through the pressurizer safety valves following a loss of load at 100% power. Although the safety function of the pressurizer safety valves (overpressure protection of the RCS pressure boundary) can still be met if water relief occurs, it is preferable to limit operation to steam relief. Water discharge results in higher hydraulic loading on the discharge piping and other components downstream of the safety valves. The proposed change will not alter these design features of the pressurizer level control system.

The design of the pressurizer and the pressurizer level control system are based primarily on thermal-hydraulic conditions that occur when the reactor is operating. The proposed change will not affect the existing requirement in Modes 1 and 2, which are for Power Operation and Plant Startup, respectively. The proposed change only applies to Mode 3, when the reactor is shutdown. In this Mode, a higher initial pressurizer level is acceptable because the potential magnitude of a pressurizer insurge due to thermal expansion of the reactor coolant is much smaller than that which would occur in Mode 1 with the plant at full power.

In addition, Technical Specification 3.4.10, "Pressurizer Safety Valves" and 3.4.12, "LTOP" (Low Temperature Overpressure Protection" remain in effect and are not being modified by the proposed change in pressurizer level. The pressurizer safety valves provide RCS overpressure protection when RCS temperature is above 319 ^{0}F. The LTOP system provides for overpressure protection when the RCS temperature is below 319 ^{0}F. The Bases for this Technical Specification (3.4.12) also explains that administrative controls are in place to limit the potential for RCS pressure exceeding the low temperature pressure limits when RCS temperature is in the range from 319 ^{0}F to 411 ^{0}F. The administrative control consists of a dedicated operator monitoring pressurizer level when level is above a specified limit. These same administrative controls will be referenced in the Bases of Technical Specification 3.4.9.

4.0 TECHNICAL ANALYSIS

When performing a plant cooldown, reactor coolant contracts, resulting in a reduction in pressurizer water level. In order to maintain level in the pressurizer, a net positive addition to the RCS is established by adjusting the charging and letdown flowrates of the Chemical and Volume Control System (CVCS). However, the net positive addition of water may not be sufficient to fully compensate for contraction when performing a cooldown at or near the maximum allowable rate based on metal thermal stress considerations. Operation of the CVCS in this plant condition is also used to establish the higher reactor coolant boron concentrations required to maintain shutdown margin as reactor coolant temperature is reduced. The proposed change to allow a higher pressurizer water level in Mode 3 will provide operators greater flexibility in preparing for and performing a plant cooldown at or near the maximum allowable rate. The higher level will help compensate for reactor coolant contraction and will be less of an operational restriction to the addition of borated water, as needed, to meet shutdown margin requirements. In addition, upon reaching Mode 4 (RCS temperature less than 350 ^{0}F) and placing the residual heat removal system in service, pressurizer level is raised to a 'water-solid' condition. The increase in level is typically performed using the pressurizer spray line to protect the pressurizer surge line from cooling too rapidly. The lower the initial pressurizer level is, the longer this evolution will take. Based on experience during the plant cooldown in May 2000 for Refueling Outage 11, it is estimated that a time savings of approximately 1 – 2 hours is achievable if a higher pressurizer level is allowed in Mode 3.

The proposed change will allow plant operators to adjust pressurizer level in Mode 3 to $\leq$ 90% in anticipation of performing a plant cooldown. The existing limit of $\leq$ 58.3 % assures that the initial condition assumption regarding pressurizer level remains valid for the limiting accident analyses at full power conditions. Although overpressure protection of the RCS does not depend on operation of the pressurizer level control system, operation at the programmed level does assure that water relief through the pressurizer safety valves is unlikely. In Mode 3, the rate of expansion of reactor coolant in the event of a loss of decay heat removal would be much less significant than with the plant at full power. The proposed change will allow the plant to be operated in Mode 3 (Hot Standby) with a pressurizer level that is higher than that allowed in

Mode 1 (Power Operation) and Mode 2 (Plant Startup). With the higher level, there will still be a steam bubble maintained in the pressurizer.

The intent of LCO 3.4.9 is to ensure that a steam bubble exists in the pressurizer during power operation to provide a cushioning effect during potential overpressure transients. Although the level control system is not credited for the prevention or mitigation of any accidents, pressurizer level is an initial condition assumed in certain accident analyses. The pressurizer level limit of $\leq$ 58.3% specified in LCO 3.4.9 assures that the initial condition assumed for the limiting overpressure events remains valid. The worst-case scenario for these events (loss of load and loss of normal feedwater) occurs at the full power condition because pressurizer insurge is maximized by the thermal expansion of the reactor coolant. The proposed change will not affect the validity of these accident analyses or the assumed initial condition because the change only applies to Mode 3 when the reactor is shutdown.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. 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 change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

Pressurizer water level is an assumed initial condition for certain accident analyses. Plant initial conditions are not accident initiators and do not have an effect on the probability of the accident occurring. The proposed change only revises the specified limit on water level in the pressurizer, so that this change would not affect accident probability.

The specific accidents for which pressurizer water level is an assumed initial condition are a loss of load and a loss of normal feedwater. The limiting accident analysis results occur at full power conditions when the available core thermal power is maximized. The proposed change does not affect the specified pressurizer level limit at any power level from zero to full power. That is, the pressurizer level limit is not being changed in Modes 1 and 2. The proposed change does revise the specified pressurizer water level limit in Mode 3 (Hot Standby) but this does not affect accident analysis results because the limiting analyses will remain those that are postulated to occur in Mode 1 with the plant at full power.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve physical changes to existing plant equipment or the installation of any new equipment. The design of the pressurizer, the pressurizer level control system and the pressurizer safety valves is not being changed and the ability of these systems, structures, and components to perform their design or safety functions is not being affected. The proposed change revises the specified limit on pressurizer water level in Mode 3 (Hot Standby) to allow operators greater flexibility in performing a plant cooldown. The method used in performing the plant cooldown is not being changed. This proposed change does not create new failure modes or malfunctions of plant equipment nor is there a new credible failure mechanism.

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 change involve a significant reduction in a margin of safety?

Response: No.

Pressurizer level is an initial condition assumed in certain accident analyses involving an insurge in the pressurizer and an increasing reactor coolant system (RCS) pressure. These analyses demonstrate that the design pressure for the RCS is not exceeded for the limiting analyses based on the plant at full power. The proposed change does not affect the existing Technical Specification requirement for Mode 1 (Power Operation) or Mode 2 (Plant Startup) and therefore does not affect the assumptions or results of these accident analyses. The margin for RCS design pressure demonstrated by these analysis results is not being reduced. The proposed change only applies to the pressurizer level limit in Mode 3 (Hot Standby) when there is substantially lower thermal energy available to cause rapid expansion of reactor coolant and an insurge to the pressurizer. Protection of the RCS pressure boundary is still maintained by the pressurizer safety valves, which are not being modified by the proposed change in pressurizer water level.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy Nuclear Operations, Inc. concludes that the proposed amendment presents no 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.

5.2 Applicable Regulatory Requirements / Criteria

The proposed changes have been evaluated to determine whether applicable requirements continue to be met. The proposed change is consistent with the Indian Point 3 FSAR and no change to the FSAR is needed to implement the proposed new level limit applicable to Mode 3. Changes to the Technical Specification Bases are proposed to provide information about the limit on pressurizer level in Mode 3. The existing Bases states "The intent of the LCO (e.g. 3.4.9) is to ensure that a steam bubble

exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients." This intent will continue to be satisfied with the proposed change. The new level limit still ensures that a steam bubble exists in the pressurizer in Mode 3. The limit on pressurizer level and the corresponding steam bubble in the pressurizer prior to power operation is not affected, because the existing level limit is not being changed for Modes 1 and 2.

In addition, administrative controls already in place as explained in the Bases to Technical Specification 3.4.12 will continue to be in effect. These administrative controls provide additional assurance that the 10 CFR 50 Appendix G criteria for RCS pressure - temperature limits are met.

ENO has determined that the proposed changes do not require any exemptions or relief from regulatory requirements other then the change requested to Technical Specification Section 3.4.9. The proposed change to the Technical Specification does not affect conformance to any design criteria described in the FSAR and the revised Technical Specification will continue to satisfy Criterion 2 of 10 CFR 50.36.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) 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.

6.0 PRECEDENCE

The existing Technical Specification 3.4.9 for Indian Point 3 is modeled after the Standard Technical Specification (Reference 1), which requires a plant specific value for the pressurizer level limit. The value to be used depends on the plant specific value assumed as an initial condition for accident analyses involving a pressurizer insurge resulting from thermal expansion of the reactor coolant (Reference 2). The limiting scenario occurs with the plant at full power when the amount of thermal energy imparted to the reactor coolant is maximized. The proposed change only applies to Mode 3 when the plant is not at power and minimal thermal energy is available to contribute to rapid expansion of the reactor coolant inventory. A similar approach for specifying this requirement was approved by the NRC for Point Beach, Docket Numbers 50-266 / 301.

The Technical Specification Task Force traveler, TSTF-162 (Reference 3) documents a change to the Bases Section 3.4.9 to clarify that the maximum pressurizer level limit is based on ensuring that a steam bubble exists in the pressurizer. The TSTF also states that "the maximum pressurizer water level is not credited in any safety analyses." Although pressurizer level may be assumed in the safety analyses, acceptable analysis results may not depend on this value because the pressurizer safety valves will continue to perform their safety function of preventing the system pressure from exceeding the system safety limit which is 110% of the

RCS design pressure. The proposed change in level for Mode 3 still ensures that a steam bubble exists in the pressurizer for this plant condition.

In some cases, plant technical specifications may specify an upper limit of approximately 90% pressurizer level for all three modes of applicability (Modes 1, 2, and 3). Examples of this include Diablo Canyon, Wolf Creek, and Callaway, Docket Numbers 50-275 / 323, 50-482, and 50-483, respectively. The proposed change for Indian Point 3 will retain the lower limit on pressurizer level for Modes 1 and 2, corresponding to the initial condition assumed in the accident analyses with the plant at full power. The proposed change for Indian Point 3 only provides for an increase in the allowable level to 90% for Mode 3.

7.0 REFERENCES

1. NUREG 1431, "Standard Technical Specifications for Westinghouse Plant," Revision 1, dated April 1995.

2. NRC letter to Power Authority of the State of New York (licensee of Indian Point 3 at the time of the letter); "Request for Additional Information Regarding Standard Technical Specification Conversion," dated July 9, 1999.

3. Industry / TSTF Standard Technical Specification Change Traveler, TSTF-162; "Maximum Pressurizer Water Level Limit Bases," Revision 0, approved by NRC October 3, 1997.

ATTACHMENT II TO IPN-02-043

MARKUP OF TECHNICAL SPECIFICATION AND BASES

FOR PROPOSED CHANGE REGARDING

PRESSURIZER WATER LEVEL REQUIREMENT

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

in MODES 1 and 2 or ≤ 90% in Mode 3

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 58.3%, and

Actual

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 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 with reactor trip breakers open.	6 hours
	AND	
	A.2 Be in MODE 4.	12 hours
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 ≤ 58.3% .	12 hours
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is ≥ 150 kW.	24 months

actual

in MODES 1 and 2 OR ≤ 90% in MODE 3

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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 emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble 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.

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 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

(continued)

BASES

BACKGROUND (continued)

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.

Pressurizer heaters are powered from either the offsite source or the diesel generators (DGs) through the four 480V vital buses as follows: bus 2A (DG 31) supports 485 kW of pressurizer heaters; bus 3A (DG 31) supports 555 kW of pressurizer heaters; bus 5A (DG 33) supports 485 kW of pressurizer heaters; and, bus 6A (DG 32) supports 277 kW of pressurizer heaters.

APPLICABLE SAFETY ANALYSES

DELETE

In MODES 1, 2, and 3, the LCO requirement for a steam 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 and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present. The required pressurizer level of $\leq$ 58.3% is the analytical limit used as an initial condition in the accident analysis. An additional margin should be allowed for instrument error.

INSERT A

Safety analyses presented in the FSAR (Ref. 1) do 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.

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36. 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), is the reason for providing an LCO.

(continued)

INSERT A FOR PAGE B 3.4.9 - 2

In Modes 1, 2, and 3, the LCO requirement on pressurizer water level ensures that a steam bubble exists in the pressurizer. In addition, the safety analyses for loss of load and for loss of normal feedwater include an analytical limit of 58.3% as an initial condition assumption. The analyses assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present. The limiting scenario for these accident analyses is with the plant at full power. Therefore, the LCO requirement specified for MODE 1 ensures that the pressurizer initial condition assumption remains valid. An additional margin on the analytical limit must be allowed for instrument error.

INSERT B FOR PAGE B 3.4.9 – 3

When RCS temperature is below 411 ^{0}F, administrative controls in the Technical Requirements Manual (Ref. 3) are used to limit the potential for exceeding 10 CFR 50, Appendix G limits.

BASES

LCO

The LCO requirement for the pressurizer to be OPERABLE with water level less than or equal to 58.3% ensures that a steam bubble exists. The required pressurizer level of ≤ 58.3% is the analytical limit used as an initial condition in the accident analysis. An additional margin of approximately 7% ~~should~~ **must** be allowed for instrument error (i.e., the indicated level should not exceed 51.3%).

(for MODES 1 and 2) or less than or equal to 90% (for MODE 3)

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 is also consistent with analytical assumptions.

,for MODES 1 and 2 or 83%, for MODE 3

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW, capable of being powered from either the offsite power source or the emergency power supply. Each of the 2 groups of pressurizer heaters should be powered from a different DG to ensure that the minimum required capacity of 150 kW can be energized during a loss of offsite power condition assuming the failure of a single DG. 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 value of 150 kW is sufficient to maintain pressure and 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 MODES 1 and 2. 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.

INSERT B

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

(continued)

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BASES

APPLICABILITY (continued)

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 and A.2

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.

If the pressurizer water level is not within the limit, action must be taken to place the plant in a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and 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 that the redundant heater group is still available and the low probability of an event during this period. Pressure control may be maintained during this time using remaining 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

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

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 separately 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 has been shown by operating experience to be acceptable.

REFERENCES

1. FSAR, Section 14.

2. NUREG-0737, November 1980.

3. IP3 Technical Requirements Manual.

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The following four DC electrical power subsystems shall be OPERABLE:

Battery 31 and associated Battery Charger;
Battery 32 and associated Battery Charger;
Battery 33 and associated Battery Charger; and
Battery 34.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DC electrical power subsystem 34 inoperable.	A.1 Declare Inverter 34 inoperable and take Required Actions specified in LCO 3.8.7, Inverters-Operating.	2 hours
B. One DC electrical power subsystem (31 or 32 or 33) inoperable.	B.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours *
C. Required Action and Associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

DELETE ASTERISK

*See next page for one-time allowed completion time

DELETE

LCO 3.8.4.Action B Completion Time (continued)

*On a one-time (per battery)only basis for Station batteries 31 and 32, the batteries may be inoperable for up to 10 days each, as necessary, to allow on-line replacement of the batteries. The time period during which this allowance may be exercised will end on May 31, 2002. The following additional requirements shall also be met to invoke this extended one-time allowed outage time: No risk significant planned maintenance or testing activities, which may impact AC or DC normal or emergency distribution sources or ESF systems, shall be performed during this replacement period.

DELETE

DELETE PAGE

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage on float charge is within the following limits: a. ≥125.7 V for battery 31; b. ≥123.5 V for battery 32; and c. ≥ 127.8 V for batteries 33 and 34.	31 days
SR 3.8.4.2	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify each battery charger supplies its associated battery at the voltage and current adequate to demonstrate battery charger capability requirements are met.	24 months
SR 3.8.4.3	----------NOTES---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4.* **DELETE ASTERISK** ---------- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test or a modified performance discharge test.	24 months

(continued)

* This battery surveillance may be performed on a one-time only basis during replacement of Station batteries 31 and 32 when the unit is in Mode 1, 2, 3, or 4 in order to support the one-time allowed outage time change of 10 days, as indicated in section 3.8.4.B. This testing shall be done when the battery is disconnected from the DC bus.

DELETE

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.4	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4.* ---------- Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test. **DELETE ASTERISK**	60 months <u>AND</u> 12 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating
SR 3.8.4.5	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration.	24 months

* This battery surveillance may be performed on a one-time only basis during replacement of Station batteries 31 and 32 when the unit is in Mode 1, 2, 3, or 4 in order to support the one-time allowed outage time change of 10 days, as indicated in section 3.8.4.B. This testing shall be done when the battery is disconnected from the DC bus.

DELETE