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NL-13-003

January 28, 2013

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

SUBJECT: Proposed Technical Specification Changes Regarding RWST Temperature and Containment Pressure in Containment Integrity Analysis
Indian Point Unit Number 3
Docket No. 50-286
License No. DPR-64

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc, (Entergy) hereby requests a change to the Technical Specifications for Indian Point Nuclear Generating Unit No. 3 (IP3). The proposed change will revise Technical Specification surveillance requirement (SR) 3.5.4.1, to limit the maximum Refueling Water Storage Tank (RWST) temperature to $\leq 105^{0}$F; Technical Specification 3.6.4 Limiting Condition for Operation to limit containment pressure to $\leq +1.5$ psig if RWST temperature is > 95^{0}F or containment temperature is >125^{0}F; Technical Specification SR 3.6.3.9 is being deleted and Technical Specification 5.5.15 Containment Leakage Rate Testing Program is being changed to specify the re-analysis value of peak containment pressure. A re-analysis of the Large Break Loss of Coolant Accident with an RWST initial temperature of 105^{0}F and containment initial pressure of 1.5 psig was performed to address mass and energy release errors for containment integrity identified in Nuclear Safety Advisory Letter 11-5. Sensitivity studies establish the bases for the proposed changes. The current analysis of record uses an RWST initial temperature of 110^{0}F and containment initial pressure of 2.5 psig. Since the current Technical Specifications have been determined to be non-conservative, administrative controls will be in place (when RWST temperature exceeds 95^{0}F or containment temperature exceeds 125^{0}F) to operate the plant consistent with the proposed Technical Specifications.

Entergy has evaluated the proposed change in accordance with 10 CFR 50.91(a)(1) using the criteria of 10 CFR 50.92(c) and determined that this proposed change involves no significant hazards, as described in Attachment 1. The marked up Technical Specification pages showing the proposed changes are provided in Attachment 2. The associated Bases changes are provided in Attachment 3 for information. A copy of this application and the associated attachments are being submitted to the designated New York State official in accordance with 10 CFR 50.91.

A001
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Entergy requests approval of the proposed amendment within 12 months and an allowance of 30 days for implementation. There are no new commitments being made in this submittal. If you have any questions or require additional information, please contact Mr. Robert Walpole, Manager, Licensing at (914) 254-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 28, 2013.

Sincerely,

JAV/ai

Attachments:
1. Analysis of Proposed Technical Specification Changes Regarding RWST Temperature and Containment Pressure
2. Marked Up Technical Specification Pages for Proposed Changes Regarding RWST Temperature and Containment Pressure
3. Marked Up Technical Specification Bases Pages for Proposed Changes Regarding RWST Temperature and Containment Pressure

cc: Mr. Douglas Pickett, Senior Project Manager, NRC NRR DORL
Mr. William M. Dean, Regional Administrator, NRC Region 1
NRC Resident Inspectors
Mr. Francis J. Murray, Jr., President and CEO, NYSERDA
Ms. Bridget Frymire, New York State Dept. of Public Service

ATTACHMENT 1 TO NL-13-003

ANALYSIS OF PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING RWST TEMPERATURE AND CONTAINMENT PRESSURE

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

1.0 DESCRIPTION

Entergy Nuclear Operations, Inc (Entergy) is requesting an amendment to Operating License DPR-64, Docket No. 50-286 for Indian Point Nuclear Generating Unit No. 3 (IP3): The proposed amendment will revise Technical Specifications (TS) 3.5.4 surveillance requirement (SR) 3.5.4.1, to limit the maximum Refueling Water Storage Tank (RWST) temperature to ≤105°F; TS 3.6.4 Limiting Condition for Operation (LCO) to limit containment pressure to ≤+1.5 psig if RWST temperature is >95°F or containment temperature is >125°F; TS 3.6.3 SR 3.6.3.9 is being deleted and TS 5.5.15 Containment Leakage Rate Testing Program is being changed to specify the re-analyzed value of peak containment pressure.

A re-analysis of the large break loss-of-coolant accident (LOCA) was performed to correct methodology errors in the long-term mass and energy (M&E) releases for containment integrity analysis. The RWST temperature and containment pressure changes are necessary as a result of the re-analysis to maintain the peak containment pressure at about the same value as the current analysis of record.

The specific proposed changes are listed in the following section.

2.0 PROPOSED CHANGES

The proposed TS changes are as follows:

Change SR 3.5.4.1 from

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	-----------------NOTE------------------------------ Not required to be performed when ambient air temperature is ≥ 35°F and ≤ 110°F if heating steam supply isolation valves are locked closed. --- Verify RWST borated water temperature is ≥ 35°F and ≤ 110°F.	24 hours

To

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	------------------NOTE-------------------------------- Not required to be performed when ambient air temperature is ≥ 35°F and ≤ 95°F if heating steam supply isolation valves are locked closed. -- Verify RWST borated water temperature is ≥ 35°F and ≤ 105°F.	24 hours

Change TS LCO 3.6.4 from

Containment pressure shall be ≥ -2.0 psig and ≤ +2.5 psig.

To

Containment pressure shall be maintained as follows:

a. If RWST temperature is > 95^{0}F or containment temperature is > 125^{0}F, Containment pressure shall be ≥ -2.0 psig and ≤ +1.5 psig.

b. If RWST temperature is ≤ 95^{0}F and containment temperature is ≤ 125^{0}F, Containment pressure shall be ≥ -2.0 psig and ≤ +2.5 psig.

Delete SR 3.6.3.9

~~SR 3.6.3.9~~	~~Verify the combined leakage rate for all containment bypass leakage paths is ≤ 0.6 L_a when pressurized to ≥ 42.42 psig.~~	~~In accordance with the Containment Leakage Rate Testing Program~~

Change 5.5.15, Containment Leakage Rate Testing Program, from

The calculated peak containment internal pressure for the design basis loss of coolant accident, Pa, is 42.0 psig. The containment design pressure is 47 psig.

To

The calculated peak containment internal pressure for the design basis loss of coolant accident, Pa, is 42.38 psig. The containment design pressure is 47 psig.

The marked up Technical Specification pages showing these changes are in Attachment 2. The associated changes to the Technical Specification Bases, to be made after approval using the 10 CFR 50.59 process, are in Attachment 3 for information.

3.0 BACKGROUND

Nuclear Safety Advisory Letter 11-5 (NSAL-11-5, Reference 1) identified Westinghouse methodology errors in the long-term mass and energy (M&E) releases during a large break loss-of-coolant accident (LOCA). These impacted containment integrity analysis for Indian Point Unit 3 (IP3).

The four issues listed below impact the IP3 long-term LOCA M&E release calculation utilizing the Westinghouse containment analysis methodology;

- The reactor vessel modeling did not include all the appropriate vessel metal mass available from the component drawings. This discrepancy results in an inaccurate vessel metal mass that affects the amount of reactor vessel stored energy initially available in the M&E model.
- The reactor vessel model did not include the appropriate amount of vessel metal mass in the reactor vessel barrel/baffle downcomer region. Differences were identified in the calculated metal mass and surface area input values. Increases in the barrel/baffle metal mass impact the initial energy stored within the reactor vessel.
- The long-term LOCA M&E release analysis was initialized at a non-conservative (low) steam generator (SG) secondary pressure condition. This input value determines the initial SG secondary side temperature and pressure used in the long-term LOCA M&E release calculations. The pressure at the exit of the SG outlet nozzle was incorrectly used as the SG secondary side pressure, as opposed to the correct, higher tube bundle pressure.
- An error was found in the EPITOME computer code that is used to determine the M&E release rate during the long-term (i.e., post-reflood) SG depressurization phase of the LOCA transient. The error results in an underestimated energy release in the long-term, post-reflood phase of the transient.

The analysis of record (AOR) peak containment pressure is 40.38 psig for the double-ended hot leg (DEHL) break and 42.00 psig for the double-ended pump suction (DEPS) break, respectively (Reference 4).

4.0 TECHNICAL ANALYSIS

Westinghouse re-analyzed the containment integrity analysis with the errors identified in Section 3.0 corrected in the long-term LOCA M&E model. Further, the re-analysis assumed a RWST initial temperature of 105^{0}F and a containment initial pressure and temperature of 1.5 psig and 130^{o}F, respectively.

The analysis of record for containment integrity is based on the limiting single failure of 32 emergency diesel generator (EDG) coincident with loss of offsite power. As noted in FSAR Section 14.3.6.2.2, "The minimum ECCS case was based upon a diesel train failure (which leaves available as active heat removal systems one containment spray pump and four RCFCs)".

With the error corrections of NSAL-11-5, and assuming the same initial conditions as the analysis of record (RWST temperature = 110^{0}F, containment pressure = 2.5 psig, containment temperature = 130^{0}F), a peak containment pressure of 44.26 psig was calculated for the DEPS in Reference 2. While this pressure is well below the containment design pressure of 47 psig, it was desired to maintain peak containment pressure at or about the current analysis of record value for purposes of other programs. Consequently, sensitivity studies were performed in Reference 3 to evaluate the impacts of initial RWST temperature, initial containment pressure, and initial containment temperature on peak containment pressure. These studies showed that with an initial RWST temperature of 105^{0}F, initial containment pressure of 1.5 psig, and initial containment temperature of 130^{0}F, the peak containment pressure would be 39.71 psig for the DEHL break and 42.38 psig for the DEPS break. As shown in Table 1 below, the most limiting peak containment pressure is slightly higher than the analysis of record, and will have an insignificant impact on other programs. There are no changes to design, and the revised analysis is consistent with the plant configuration for equipment availability and the peak containment pressure remains well below the design pressure of 47 psig.

Table 1 - Comparison of Peak Containment Pressure

Peak Containment Pressure	Analysis of Record [psig]	Error Correction and RWST=105^{0}F, containment pressure = 1.5psig [psig]
Double-Ended Hot Leg (DEHL) Break	40.38	39.71
Double-Ended Pump Suction (DEPS) Break	42.00	42.38

The sensitivity studies also demonstrated that a 5^{0}F decrease in initial RWST temperature would result in 0.46 psi reduction in peak containment pressure, a 5^{0}F decrease in initial containment/accumulator temperature would result in 0.57 psi reduction in peak containment pressure and a 0.25 psi decrease in initial containment pressure would result in 0.30 psi decrease in peak containment pressure. As mentioned above, the case in Reference 2 which had initial conditions of 110^{0}F for RWST temperature, 2.5 psig for initial containment pressure and 130^{0}F for initial containment/accumulator temperature resulted in a peak containment pressure of 44.26 psig. Reducing the RWST temperature for this case to 95^{0}F would result in a peak containment pressure of 42.88 psig [44.26-(3x0.46)] and reducing the containment/accumulator temperature from 130^{0}F to 125^{0}F would result in a peak containment pressure of 42.31 psig (42.88-0.57). Thus, with a RWST temperature $\leq$95^{0}F and containment /accumulator temperature of $\leq$125^{0}F, an initial containment pressure of 2.5 psig would result in an acceptable peak containment pressure. It should be noted that while the accumulator is located in the containment, due to its lower elevation, it is expected to be at a lower temperature than the containment average temperature.

Section 5.5.15 for the Containment Leakage Rate Testing Program has been revised to reflect the change in the peak calculated pressure from 42.0 psig to 42.38 psig. The peak pressure is not an accident initiator. The increase in peak pressure does not result in an increase in doses since the pressures used for the Type A, Type B and Type C tests are above 42.38 psig. The containment Appendix J test also exceeded this value. The increase in calculated pressure does not affect systems, components or tests.

Based on the above, the surveillance requirement for RWST found in TS 3.5.4 has been lowered to an acceptance criteria of 105°F as the maximum allowable temperature for which an analysis has been done. The note concerning the temperature at which this monitoring must be performed has also been decreased to 95°F since this is the RWST temperature at which the containment pressure must be reduced. The RWST will not exceed this temperature until the outside temperature reaches it. This change does not affect the probability of an accident and is consistent with accident analyses so no doses would increase. There are no changes to the operation of any systems or components or any tests.

Based on the above, the surveillance requirement for Containment Air Temperature found in TS 3.6.5 LCO and Condition B need not be lowered to an acceptance criteria of 125°F since the pressure limit of 42.38 psig is met with a Containment air temperature of 130°F, a RWST temperature of 105°F and the initial Containment air pressure of 1.5 psig (as required with the RWST above 95°F or the Containment air temperature above 125°F). There are surveillances every 24 hours for RWST and Containment temperature.

Based on the above, the Containment pressure limits will be maintained using two sets of values. When the RWST temperature is below 95°F and the Containment temperature is below 125°F, the peak internal pressure can be ≤ 2.5 psig. When either temperature exceeds that value, the containment pressure must be ≤ 1.5 psig. This change does not affect the probability of an accident and is consistent with accident analyses so no doses would increase. There are no changes to the operation of any systems or components or any tests.

Indication of containment pressure is available in the control room with an uncertainty of +/- 1.5 psi with all indicators operable. With this indication, containment purging from the control room at ≤ 1 psi currently satisfies the accident analysis initial condition of 2.5 psig when RWST temperature is ≤ 95°F and containment/accumulator temperature is ≤125°F. When RWST temperature is > 95°F or containment/accumulator temperature is >125°F, containment pressure indication with a higher accuracy instrument will assure the accident initial conditions are maintained. This will be controlled by a local mounted high accuracy containment pressure indicator. With this indication, which will have an uncertainty of at least +/- 0.5 psi, containment purging at ≤ 1.0 psi would satisfy the accident analysis initial conditions. The TS change allows for continued monitoring of the containment pressure from the control room except maybe for a few hot days during the summer, when operators would be required to monitor the containment pressure from the locally mounted instrument.

SR 3.6.3.9 is being deleted as it is redundant to TS 5.5.15. When IP3 converted from custom TS to Standard Technical Specifications (STS) (Reference 9), SR 3.6.3.9 was modified. STS SR 3.6.3.11 was intended to measure bypass leakage from a shield building. Specifically, STS SR 3.6.3.11 which states "Verify the combined leakage rate for all shield building bypass leakage paths is ≤ [La] when pressurized to ≥ [psig]" was modified to state "Verify the combined leakage rate for all containment bypass leakage paths is ≤ 0.6La when pressurized to ≥ 42.42 psig." IP3 has no shield building, and the STS SR for bypass leakage from a shield building does not apply.

Since leakage rate testing of the containment is specified in TS 5.5.15, Containment Leakage Rate Testing Program, SR 3.6.3.9 is redundant and may be deleted.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy has evaluated the safety significance of the proposed changes to the Indian Point 3 Technical Specifications. The proposed changes have been evaluated according to the criteria of 10 CFR 50.92, "Issuance of Amendment". The changes to SR 3.6.3.9 and TS 5.5.15 are considered editorial changes, in that SR 3.6.3.9 is being deleted due to not being applicable to IP3 and redundancy to TS 5.5.15 and TS 5.5.15 is being changed to specify the re-analyzed value of peak containment pressure. Entergy has determined that the subject changes do not involve a Significant Hazards Consideration as discussed below:

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

   No. The proposed change would not change the current limiting EDG failure but limits the RWST temperature to $\leq 105^0$F and containment pressure to ≤ 1.5 psig (when RWST temperature is >95^0F or containment/accumulator temperature is >125^0F). The proposed change also removes a redundant TS for Containment testing and corrects the peak pressure in the containment testing program. The initial conditions assumed in accident analysis are not accident initiators so the probability of an accident does not increase. The change in initial conditions compensates for the error corrections and maintains the post accident containment pressure within 0.38 psig of the current value and within Containment testing limits and therefore does not increase the probability or consequences of a previously evaluated accident. Therefore the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   No. The change to the initial conditions assumed in the analysis for peak containment pressure, the removal of a redundant Technical Specification and the correction to the peak pressure limit in the Containment testing program do not create the possibility of a new or different accident. There are no changes to design or operating procedures that could create a new or different kind of accident since the changes only affect the initiating conditions. The revised analysis is consistent with the available equipment following the postulated worst case single failure.

   Therefore the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   No. The change in peak containment pressure is from 42 psig to 42.38 psig as a result of the error corrections of NSAL-11-5 and change to the initial conditions for the RWST temperature and containment pressure. There is an insignificant impact on other programs due to change in peak containment pressure, which remains well below the containment design pressure of 47 psig. Therefore there is no significant reduction in a margin.

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

Based on the above, Entergy concludes that the proposed amendment to the Indian Point 3 Technical Specifications 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 plant will continue to meet Criterion 2 of 10 CFR 50.36 which says "A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The current version of TS SR 3.5.4.1 and TS LCO 3.6.4 are being changed to be consistent with the input assumptions used in the re-analysis (Reference 3) to address errors identified in NSAL-11-5. The change to SR 3.6.3.9 and TS 5.5.15 are editorial in that SR 3.6.3.9 is being deleted due to redundancy to TS 5.5.15 and TS 5.5.15 is being changed to specify the re-analyzed value of peak containment pressure. The same codes and methods were used in the re-analysis as was done for Stretch Power Uprate License Amendment Request (LAR) of Reference 4 which contained a proprietary report, Reference 5, and which was approved as Amendment 225 by the NRC in Reference 6. Section 6.5.3.5.1 of Reference 4 stated: "The associated single failure assumption is the failure of a diesel to start, resulting in one train of ECCS and containment safeguards equipment being available. This combination results in a minimum set of safeguards equipment being available."

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.

5.3 Environmental Considerations

The proposed changes to the IP3 Licensing Basis do 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

A Technical Specification revision to change the RWST temperature was found for Indian Point

Unit 2, which revised the RWST temperature from 100^{0}F to 110^{0}F at the time of the Stretch Power Uprate submittal (Reference 7) and approved as Amendment 241 by the NRC in Reference 8.

7.0 REFERENCES

1. Nuclear Safety Advisory Letter, "Westinghouse LOCA Mass and Energy Release Calculation Issues," NSAL-11-5, dated July 25, 2011.

2. Letter from Edward P. Shields (Westinghouse) to Nasser Nik (Entergy), "LOCA Mass and Energy Analysis," INT-12-3, dated February 24, 2012.

3. Letter from Edward P. Shields (Westinghouse) to Nasser Nik (Entergy), "LOCA Mass and Energy Analysis," INT-12-8, dated April 23, 2012.

4. Entergy Letter NL-04-069 to NRC, "Proposed Changes to Technical Specifications: Stretch Power Uprate (4.85%) and Adoption of TSTF-339," dated June 3, 2004.

5. Indian Point Nuclear Generating Unit No. 3 Stretch Power Uprate NSSS and BOP Licensing Report, WCAP-16212-P, dated June 2004.

6. NRC Letter to Entergy, Indian Point Nuclear Generating Unit No 3 - Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters (TAC No. MC3552), March 24, 2005.

7. Entergy Letter NL-04-005 to NRC, "Proposed Changes to Technical Specifications: Stretch Power Uprate Increase of Licensed Thermal Power (3.26%)," dated January 29, 2004.

8. NRC Letter to Entergy, Indian Point Nuclear Generating Unit No 2 - Issuance of Amendment Re: 3.26 Percent Power Uprate (TAC No. MC1865), October 27, 2004.

9. NRC Letter to Entergy Issuing Amendment For Conversion to Improved Standard Technical Specifications (TAC No. MA4359), dated February 27, 2001.

ATTACHMENT 2 TO NL-13-003

MARKED UP TECHNICAL SPECIFICATION PAGES

FOR PROPOSED CHANGES REGARDING RWST TEMPERATURE AND

CONTAINMENT PRESSURE

Changes indicated by lineout for deletion and Bold/Italics for additions

Unit 3 Affected Pages:
3.5.4-2
3.6.4-1
3.6.3 -6
5.0-31

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	----------NOTE---------- Not required to be performed when ambient air temperature is ≥ 35°F and ≤ ~~110~~ ***95***°F if heating steam supply isolation valves are locked closed. ---------- Verify RWST borated water temperature is ≥ 35°F and ≤ ~~110~~ ***105***°F.	24 hours
SR 3.5.4.2	Verify RWST borated water level is ≥ 35.4 feet.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2400 ppm and ≤ 2600 ppm.	31 days
SR 3.5.4.4	Perform CHANNEL CHECK of RWST level	7 days
SR 3.5.4.5	Perform CHANNEL CALIBRATION of RWST level switch and ensure the low level alarm setpoint is ≥10.5 feet and ≤12.5 feet.	184 days
SR 3.5.4.6	Perform CHANNEL CALIBRATION of RWST level transmitter and ensure the low level alarm setpoint is ≥10.5 feet and ≤12.5 feet.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ***maintained as follows:*** ~~≥ -2.0 psig and ≤ +2.5 psig.~~

a. If RWST temperature is > 95^0F or containment temperature is > 125^0F, Containment pressure shall be ≥ -2.0 psig and ≤ +1.5 psig.

b. If RWST temperature is ≤ 95^0F and containment temperature is ≤ 125^0F, Containment pressure shall be ≥ -2.0 psig and ≤ +2.5 psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	24 months
SR 3.6.3.7	Verify each 10 inch containment pressure relief line isolation valve is blocked to restrict valve opening to ≤ 60 degrees.	24 months
SR 3.6.3.8	Perform one complete cycle of each manually operated containment isolation valve on essential lines.	24 months
~~SR 3.6.3.9~~	~~Verify the combined leakage rate for all containment bypass leakage paths is ≤ 0.6 L_a when pressurized to ≥ 42.42 psig~~	~~In accordance with the Containment Leakage Rate Testing Program~~
SR 3.6.3.~~10~~*9*	Verify leakage rate into containment from isolation valves sealed with service water system is within limits.	In accordance with the Containment Leakage Rate Testing Program

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

cooler unit when pressurized at $\geq$ 1.1 P_a. This limit protects the internal recirculation pumps from flooding during the 12-month period of post accident recirculation.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10CFR50, Appendix J.

The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42.~~0~~***38*** psig. The containment design pressure is 47 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

5.5.16 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation System (CRVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

(continued)

ATTACHMENT 3 TO NL-13-003

MARKED UP TECHNICAL SPECIFICATION BASES PAGES

FOR PROPOSED CHANGES REGARDING RWST TEMPERATURE AND

CONTAINMENT PRESSURE

Changes indicated by lineout for deletion and Bold/Italics for additions

Unit 3 Affected Pages:

B 3.5.4-3
B 3.5.4-6
B 3.6.2-2
B3.6.3-10
B3.6.3-11
B3.6.3-16
B3.6.3-17
B 3.6.4-1

ENTERGY NUCLEAR OPERATIONS, INC.
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APPLICABLE SAFETY ANALYSIS (continued)

to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The RWST level required by Technical Specifications includes allowances for instrument accuracy, the unusable volume in the RWST, and the maximum volume expected to remain in the RWST when the plant is switched from the injection to recirculation modes of operation.

The upper limit on boron concentration of 2600 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 35°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of ~~110~~***105***°F is used in the LOCA containment integrity analysis. Exceeding this temperature will result in higher containment pressures due to reduced containment spray cooling capacity. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

Following a LOCA, switchover from the injection phase to the recircu***l***ation phase must occur before the RWST empties to prevent damage to the pumps and a loss of cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment to support recirculation pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST.

(continued)

BASES

ACTIONS (continued)

C.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume) or B (e.g., two level alarms inoperable), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

D.1 and D.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, 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 at least MODE 3 within 6 hours and to MODE 5 within 36 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 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST ***(a reduced temperature is used to assure restrictions on allowable Containment operating pressures are met)*** and the heating steam isolation valves are locked closed. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.

(continued)

BACKGROUND (continued)

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident. In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as La = 0.1% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure Pa = 42.~~0~~***38*** psig following a DBA (LBLOCA or MSLB). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36.

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not

(continued)

ACTIONS

C.1 and C.2 (continued)

flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system. The closed system must meet the requirements of Reference 3.

Required Action C.2 is modified by a Note that applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1

With the containment bypass leakage rate not within limit of ~~SR 3.6.3.9~~, ***TS 5.5.15,*** the assumptions of the safety analyses are not met. Therefore, the leakage must be restored to within limit within 4 hours. Restoration can be accomplished by isolating the penetration(s) that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed

(continued)

BASES

ACTIONS

D.1 (continued)

to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration(s) and the relative importance of containment bypass leakage to the overall containment function.

With the hydrostatically tested valve leakage not within limit of SR 3.6.3.~~10~~***9***, the potential exists for flooding the Containment Recirculation Pumps during long term post-accident cooling. The 72 hour Completion Time is reasonable because of the low probability of an event occurring during this period.

E.1 and E.2

If the Required Actions and associated Completion Times are not met, 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 at least MODE 3 within 6 hours and to MODE 5 within 36 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 3.6.3.1

Each 36 inch containment purge supply and exhaust isolation valve (FCV-1170, FCV-1171, FCV-1172, and FCV-1173) is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A containment purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.6.3.9~~

~~This SR ensures that the combined leakage rate of all containment leakage paths is less than or equal to the specified leakage rate for those paths that are not sealed by the Isolation Valve Seal Water System or sealed by the RHR system or sealed by the service water system. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves.~~

~~This testing is performed in accordance with the requirements, Frequency and acceptance criteria required by Specification 5.5.15, Containment Leakage Rate Testing Program. This program was established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by IP3 specific approved exemptions. This program conforms to guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995." In the event containment isolation valve leakage results in exceeding the overall containment leakage rate, entry into the applicable Conditions and Required Actions of LCO 3.6.1 is required.~~

SR 3.6.3.~~10~~***9***

The Containment Leakage Rate Testing Program includes verification that inleakage rate from the containment isolation valves sealed with service water is maintained at a level that will prevent flooding the internal recirculation pumps for the full 12-month period of post accident recirculation. This inleakage test has specific acceptance criteria ($\leq$ 0.36 gpm per fan cooler unit when pressurized at $\geq$ 1.1 P_a) specified in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.~~10~~**9** (continued)

Containment Leakage Rate Testing Program and the results for this inleakage test are not counted against the acceptance criteria for the Type B and C tests that are also performed as part of the SR.

REFERENCES

1. FSAR, Section 14.
2. FSAR, Section 6.
3. Standard Review Plan Section 6.2.4.
4. FSAR, Section 5.2.
5. Generic Issue B-24.
6. Safety Evaluation Report for IP3 Amendment 195.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). The containment can withstand an internal vacuum of 3 psig. The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. Cycle specific analysis results indicate that the worst case peak containment pressure could result from either a loss of coolant accident or a steam line break inside containment (Ref. 1).

The initial pressure condition used in the containment analysis was +~~2~~***1***.5 psig. ***Sensitivity studies have demonstrated that if RWST temperature is ≤ 95°F and containment/accumulator temperatures are ≤125°F, an initial containment pressure of +2.5 psig would also be acceptable.*** This analysis concluded that the containment design pressure of 47 psig would not be exceeded for either a major loss-of-coolant accident or for a main steam line break accident. The containment analysis results are presented in Reference 1 and the current value for peak containment pressure is listed in Specification 5.5.15, "Containment Leakage Rate Testing Program."

(continued)

Letter #: NL-12-xxx **(IP3 TS Change)**

Response Due: 10-17-2012

Subject: Proposed Technical Specification Changes Regarding RWST Temperature and Containment Pressure / Temperature in Containment Integrity Analysis

Date Issued for Review: 10-26-2012

Correspondence Preparer / Phone #: A Irani x6618

Section I **Letter Concurrence and Agreement to Perform Actions**

POSITION / NAME	**Action** (concurrence, certification, etc.)	**Signature** (sign, interoffice memo, e-mail, or telecom)
S Prussman - Licensing	Concurrence	e-mail: comments included
J Chang – Fuels & Analysis	Concurrence/Certification	[signature] 11/8/2012
J Hill – I&C	Concurrence/Certification	e-mail: comments included
R Walpole - Man Licensing	Concurrence	[signature]
M Lewis - Ops	Concurrence/~~Certification~~	e-mail
P Conroy – NSA	Concurrence	[signature] 1-28-13
T McCaffrey - Design Man	Concurrence	e-mail: comments included
~~D Williams - Maint~~	~~Concurrence~~	
OSRC	Recommendation	1/24/13 Approved ~~1/24/13~~ mtg 13-03 comment incorporated
J Ventosa – Site VP	Signature	
M Woodby – Dir Eng	Information	
V ANDREOZZI ~~R Burroni~~ – Systems Man	Information	
M Tesoriero – P& C Man	Information	

COMMENTS

Section II **Correspondence Screening**

Does this letter contain commitments? **If "yes," identify the commitments with due dates in the submittal and in Section III. When fleet letters contain commitments, a PCRS LO (e.g., LO-LAR, LO-WT) should be initiated with a CA assigned to each applicable site to enter the commitments into the site's commitment management system.**	Yes No	☐ ☒
Does this letter contain any information or analyses of new safety issues performed at NRC request or to satisfy a regulatory requirement? **If "yes," reflect requirement to update the UFSAR in Section III.**	Yes No	☐ ☒
Does this letter require any document changes (e.g., procedures, DBDs, FSAR, TS Bases, etc.), if approved? **If "yes," indicate in Section III an action for the responsible department to determine the affected documents. (The Correspondence Preparer may indicate the specific documents requiring revision, if known or may initiate an action for review.)**	Yes No	☒ ☐
Does this letter contain information certified accurate? **If "yes," identify the information and document certification in an attachment. (Attachment 9.5 must be used.)**	Yes No	☒ ☐

Section III **Actions and Commitments**

Required Actions *Note: Actions needed upon approval should be captured in the appropriate action tracking system*	**Due Date**	**Responsible Dept.**
LO-LAR-2012-157 CA#4	10/30/2013	Licensing

Commitments *Note: When fleet letters contain commitments, a PCRS LO should be initiated with a CA assigned to each applicable site to enter the commitments into the site's commitment management system.*	**Due Date**	**Responsible Dept.**

Section IV **Final Document Signoff for Submittal**

Correspondence Preparer **A Irani**	Ardesar Irani 1/14/2013
Final Submittal Review *(optional)*	
Responsible Department Head	[signature] 1/22/13