December 3, 2003

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: CHANGES TO PRESSURE-TEMPERATURE CURVES (TAC NO. MB9133)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 220 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated May 28, 2003, as supplemented on June 24, 2003.

The amendment authorizes the applicability of revised pressure-temperature limit curves and the low temperature overpressure protection limits for 20.0 effective full-power years of operation.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 220 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

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Accession Number: ML033370869 *Safety evaluation provided
**See previous concurrence

OFFICE	PDI-1\PM	PDI-1\LA	SC:SRXB	SC:EMCB	SC:IROB**	OGC**	PDI-1\SC
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DATE	11/24/03	11//03	10/10/03	9/29/03	11/19/03	11/24/03	12/03/03

Official Record Copy

DATED: December 3, 2003

AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE NO. DPR-64 INDIAN POINT UNIT 3

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ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 220
License No. DPR-64

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nucleaar Operations, Inc. (the licensee) dated May 28, 2003, as supplemented on June 24, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 220, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by DSkay for/

Richard J. Laufer, Chief, Section I
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 3, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 220

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.4.3-3	3.4.3-3
3.4.3-4	3.4.3-4
3.4.3-5	3.4.3-5
3.4.12-9	3.4.12-9
3.4.12-10	3.4.12-10
3.4.12-11	3.4.12-11
3.4.12-12	3.4.12-12
B 3.4.3-2	B 3.4.3-2
B 3.4.3-9	B 3.4.3-9
B 3.4.12-3	B 3.4.12-3
B 3.4.12-5	B 3.4.12-5
B 3.4.12-11	B 3.4.12-11
B 3.4.12-20	B 3.4.12-20

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE NO. DPR-64

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated May 28, 2003, as supplemented on June 24, 2003, Entergy Nuclear Operations, Inc. (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs). The requested changes would authorize the applicability of revised pressure-temperature (P-T) limit curves and the low temperature overpressure protection (LTOP) limits for 20.00 effective full-power years (EFPYs) of operation. The June 24, 2003, letter provided clarifying information that did not enlarge the scope of the amendment request or change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The NRC staff finds that the licensee in Attachment 1 of the June 12, 2003, supplemental letter identified the applicable regulatory requirements. The regulatory requirements for which the staff based its acceptance are:

a. Appendix G, “Fracture Toughness Requirements,” to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50);

b. 10 CFR 50.61, “Fracture toughness requirements for prevention against pressurized thermal shock events”;

b. Generic Letter (GL) 88-11, “NRC Position on Radiation Embrittlement Of Reactor Vessel Materials And Its Impact On Plant Operations,” dated July 12, 1988;

c. GL 92-01, Revision 1, “Reactor Vessel Structural Integrity,” dated March 6, 1992, and Supplement 1 dated May 19, 1995;

d. Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” May 1988;

e. RG 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, March 2001;

f. Standard Review Plan (SRP) Section 5.3.2, “Pressure-Temperature Limits.”

Appendix G to 10 CFR Part 50 requires that P-T limit curves for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The licensee has currently incorporated the 1989 Edition of the ASME Code into the IP3 licensing basis for defining the ASME Code requirements which apply to the facility’s ASME Code, Section XI program. Hence, with respect to the requirements of Appendix G to 10 CFR Part 50, it is the 1989 Edition of Appendix G to Section XI of the ASME Code which currently applies to the P-T limits in the IP3 TSs.

The Nuclear Regulatory Commission (NRC) staff has approved RPV fluence calculation methodologies which satisfy the requirements of General Design Criteria (GDC) 14, “Reactor coolant pressure boundary,” GDC 30, “Quality of reactor coolant pressure boundary,” and GDC 31, “Fracture prevention of reactor coolant pressure boundary,” in Appendix A to 10 CFR Part 50 and adhere to the guidance in RG 1.190. In this regard, the staff has found that fluence calculations are acceptable if they are done with approved methodologies or with methods which are shown to conform to the guidance in RG 1.190.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. Appendix G to Section XI of the ASME Code requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 on stress intensities resulting from hydrostatic testing. Appendix G to Section XI of the ASME Code also requires a safety factor of 1.0 on stress intensities resulting from thermal loads for normal and transient operating conditions as well as for hydrostatic testing. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress (i.e., of axial orientation). This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to six times its depth. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. The methodology found in Appendix G to Section XI of the ASME Code requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}) at the 1/4T and 3/4T locations. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin term.

Guidance on the determination of ΔRT_{NDT} and the margin term is given in RG 1.99, Revision 2. ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether

the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculational procedures.

10 CFR 50.61 establishes the applicability of RT_{PTS} screening criterion. The relationship between the LTOP enable temperature and the P-T period of validity is a function of the material properties at the end of the proposed period of applicability.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment which are described in the licensee's submittal. The detailed evaluation below will support the conclusion that: (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.

3.1 Licensee Request

In its application dated May 28, 2003, the licensee proposed changes to the IP3 TSs to revise the period of applicability for the P-T and overpressure protection system limits from 16.17 to 20 EFPYs. In a separate letter also dated May 28, 2003, the licensee had requested, pursuant to 10 CFR 50.60(b) and 10 CFR 50.12, an exemption from the requirements of Appendix G to 10 CFR Part 50 to use ASME Code, Section XI, Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME Section XI Division I," in lieu of 10 CFR Part 50, Appendix G, paragraph I. The licensee requested to use ASME Code Case N-640, in conjunction with Appendix G to Section XI of the ASME Code, as the basis for establishing RPV P-T limit curves. ASME Code Case N-640 permits application of the lower bound static initiation fracture toughness (K_{IC}) curve as the basis for establishing the P-T curves in lieu of using the lower bound crack arrest fracture toughness (K_{IA}) curve, which is invoked by Appendix G to Section XI of the ASME Code. All other aspects of the ASME Code, Section XI, Appendix G process for determining P-T limit curves remain unchanged in the licensee's evaluation. On December 2, 2003, the NRC granted the exemption to allow use of ASME Code Case N-640.

The licensing basis for the P-T limit curves at IP3, as given in the TSs, includes Figures 3.4.3-1 through 3.4.3-3. These figures provide P-T limits for normal reactor operation, including heatup and cooldown for normal conditions, with heating rates equal to 20 $^{\circ}$F/hour, 40 $^{\circ}$F/hour, and 60 $^{\circ}$F/hour and cooling rates $\leq$ 100 $^{\circ}$F/hour, as well as for operational leakage test and hydrostatic test conditions. The proposed TS changes replace Figures 3.4.3-1 through 3.4.3-3 with new figures, providing the revised P-T limits for the above heatup and cooldown conditions, and leak/hydro test conditions. The proposed P-T limits would be effective through 20 EFPYs of facility operation.

3.2 Licensee Calculations and Evaluation

The licensee submitted ART calculations and P-T limit curves valid for up to 20 EFPYs of facility operation. Surveillance capsule data from IP3 was reported by the licensee for the lower shell plate B2803-3. The licensee provided two independent sets of ART calculations for plate B2803-3 that were based upon the two different methodologies in RG 1.99, Revision 2 for determining ΔRT_{NDT} and the margin term. The first methodology utilized a value for the CF that was based on the existing surveillance capsule data for the lower shell plate (B2803-3). Using this method, the CF, ΔRT_{NDT} and ART values were obtained by applying the method of Regulatory Position 2.1 from RG 1.99, Revision 2 to the credible data sets for the three pulled capsules. Due to the change in the fluence values for the three withdrawn capsules, the credibility of the data sets for these three capsules was reevaluated according to Criterion 3 in Part B of RG 1.99, Revision 2, and the new capsule surveillance data was deemed credible by the licensee. The second set of ART calculations were based on the use of a CF value that was derived from Table 2 of RG 1.99, Revision 2, as prescribed by Regulatory Position 1.1 of this RG. Using this method, the licensee derived a CF value through linear interpolation between data points in Table 2. This CF value was smaller than the value of the CF determined using the surveillance capsule data. However, due to the significantly larger margin term, the ART values based on this method were larger than those based on the use of the surveillance capsule data, and these larger ART values were used by the licensee for calculating the revised P-T limits for IP3. The licensee used this information to conclude that the most limiting beltline material at the 1/4T and 3/4T locations was the lower shell plate (plate B2803-3). Based on this evaluation, the final ART values for the limiting lower shell plate at the 1/4T and 3/4T locations at 20 EFPY were determined as follows:

	1/4T Location	3/4T Location
Fluence	0.426×10^{19} n/cm^2	0.151×10^{19} n/cm^2
Chemistry Factor	160	160
ΔRT_{NDT}	122.1 °F	80.8 °F
Initial RT_{NDT}	74 °F	74 °F
Margin	34 °F	34 °F
ART	230.1 °F	188.8 °F

For P-T limits during heatup and cooldown transients, the licensee indicated that the through-wall temperature gradients and the stress intensities at the flaw tip, for both the 1/4T and 3/4T postulated flaws, due to thermal loading (i.e., K_{IT}) were unchanged from those used previously. The temperature gradient and thermal stress intensity data were provided in a previous submittal. The licensee proposed the use of a revised methodology for calculating the temperature gradient and thermal stress intensity data. This methodology was approved by the staff for determining through-wall temperature gradients and thermal stress intensities for the development of P-T limits. In this method a heat transfer analysis is performed on the reactor

vessel beltline to generate values for the through-wall temperature gradients. These through-wall temperature gradients are then used to calculate the thermal stress intensities, based on a polynomial fit to the temperature distributions in the RPV wall.

The equation for the stress intensity factor due to pressure loads (K_{IP}) was developed in accordance with the provisions of Appendix G to Section XI of the ASME Code. In accordance with Appendix G to Section XI of the ASME Code, the P-T curves were generated by correlating the stress intensity factors due to thermal and pressure loads (K_{IT} and K_{IP}) with the reference fracture toughness curve, which was derived using the ART values cited above. In calculating the revised P-T limit curves, according to the proposed TS amendment, the licensee invoked the ASME Code Case N-640 modification to the ASME Code, Section XI, Appendix G procedures by using the lower bound K_{IC} fracture toughness curve in lieu of the lower bound K_{IA} fracture toughness curve.

3.2 Staff Evaluation

Vessel Fluence Calculational Methodology

Fluence projections on the RPV were calculated for the uprated power level of 3,068 MWt, which became effective at the onset of Operating Cycle 12. The fluence calculations were done using the two-dimensional discrete ordinates code DORT and the BUGLE-96 cross sections. In these calculations, anisotropic scattering was treated with a P_5 Legendre expansion and the angular discretization with a S_{16} quadrature. Energy and space dependent source distributions and operating temperatures were treated on a cycle-specific basis. In this manner, the preceding 11 fuel cycles (13.7 EFPYs) were calculated to establish the vessel exposure and calculate the projection to the end of the current license.

The methodology summarized above is consistent the guidance of RG 1.190 and, therefore, is acceptable.

In addition, the licensee provided a summary of updated existing calculations regarding IP3, surveillance capsules T, Y, and Z. The measured to calculated values (M/C) are within acceptable limits, increasing confidence in the calculated results.

Fracture Toughness and Evaluation of RPV Materials

Use of the lower bound K_{IC} curve in the development of P-T operating limits is technically correct. The lower bound K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process for an RPV. The NRC staff concluded that P-T curves based on the K_{IC} fracture toughness curve, as referenced by ASME Code-Case N-640, will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operation. In addition, implementation of the proposed P-T curves, as defined by the technical basis supported by ASME Code Case N-640, maintains adequate margins of safety in protecting the RPV from brittle failure.

The staff performed independent calculations of the ART values for all of the IP3 beltline materials using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the licensee’s limiting beltline material for the IP3 RPV is the lower shell plate (plate

B2803-3). Furthermore, the staff verified that the lower shell plate (B2803-3) is represented in the surveillance program for IP3. The staff found that the licensee correctly applied the methodology in Regulatory Position 2.1 of RG 1.99, Revision 2 to the surveillance capsule data by fitting the surveillance capsule data to the equation for ΔRT_{NDT} in RG 1.99, Revision 2 to obtain a value for the CF. The staff verified that the licensee's reevaluation of the surveillance capsule data credibility, due to changes in capsule fluence, was correctly performed according to evaluation criterion 3 in Part B of RG 1.99, Revision 2. The ΔRT_{NDT} data reported by the licensee for the surveillance capsules was verified to be in agreement with the ΔRT_{NDT} data from the original surveillance capsule reports for IP3. Therefore, based on these independent assessments, the staff concluded that the licensee correctly determined values for the CF, ΔRT_{NDT} and ART according to the surveillance capsule methodology of Regulatory Position 2.1 of RG 1.99, Revision 2.

The NRC staff found that the licensee's final use of a CF value from Table 2 of RG 1.99, Revision 2 was justified, due to the significantly higher value for the margin term determined using this method. It was confirmed that the licensee conservatively utilized a final ART value that was based on the use of a CF value derived from Table 2 of RG 1.99, Revision 2, as prescribed by Regulatory Position 1.1 of this RG, and the use of this method resulted in an ART value that was higher than the ART value derived using the surveillance capsule methodology described above. The staff found that the licensee used a margin term that was appropriate based on their use of CF values from Table 2 of RG 1.99, Revision 2 and a plant-specific value of the initial RT_{NDT}. The staff's calculated ART value for the limiting beltline material, based on this methodology, agreed with the licensee's calculated ART value.

Given the acceptability of the licensee's calculated ART value for the limiting beltline material to 20 EFPY, the staff evaluated the licensee's revised P-T limit curves for acceptability by performing a finite set of check calculations based on information submitted by the licensee and by using the methodologies referenced in the ASME Code (as indicated in SRP 5.3.2). The staff verified that the licensee's proposed P-T limit curves satisfied the requirements in Section IV.A.2 of Appendix G to 10 CFR Part 50. Specifically, the staff concluded that the P-T limit curves submitted by the licensee accounted for the limiting conditions defined by the material properties of the limiting beltline materials and were at least as conservative as those that would be generated by the staff's application of the methodology specified in Appendix G to Section XI of the ASME Code, as modified by ASME Code Case N-640.

In addition, Appendix G to 10 CFR Part 50 also imposes a minimum temperature at the closure flange region based on the reference temperature for the flange material. Section IV.A.2 of Appendix G to 10 CFR Part 50 states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature at the closure flange region which is highly stressed by the bolt preload must exceed the reference temperature of the material in that region by at least 160 °F for core critical operation, 120 °F for normal, non-critical core operation, and by 90 °F for hydrostatic pressure tests and leak tests. The staff confirmed the licensee's limiting RT_{NDT} value of 38 °F for the non-beltline flange material, based on information previously reported by the licensee and documented in the staff's Reactor Vessel Integrity Database, as well as the acceptability of this value for the original P-T limits. Based on this limiting flange reference temperature, the staff has determined that the proposed P-T limits have satisfied the above requirements for the closure flange region during all modes of normal operation and for hydrostatic pressure and leak testing.

Based on this independent assessment, the NRC staff concluded that the licensee's proposed P-T limit curves were acceptable for operation of the IP3 RPV through 20 EFPYs of facility operation.

Pressurized Thermal Shock (PTS)

The licensee used NRC staff approved methods for the calculation of the project RPV fluence value to the end-of-life (EOL). The critical material in the beltline region is the lower shell plate B2803-3. In Table 2-3 of the May 28 application, the licensee indicated that this plate will attain an RT_{PTS} value of 269.8 °F at 31.4 EFPYs. The 10 CFR 50.61 screening criterion for plates is 270 °F. Therefore, IP3 cannot operate beyond this vessel exposure without further justification. The licensee has proposed to limit the applicability of the P-T limit curves to 31.4 EFPYs.

Reduction of the P-T applicability from 31.4 to 20 EFPYs

The applicability of the P-T curves is limited by the value of the LTOP enable temperature. As irradiation embrittles the material of the vessel, the LTOP enable temperature becomes higher. The current limit of 319 °F was retained as an interim enable temperature for operational reasons.

The licensee determined the maximum operating time for which 319 °F is a valid enable temperature for P-T limits calculated using ASME Code Case N-640. This is accomplished by interpolating of the 1/4T RT_{NDT} versus operating time. The staff notes that the function of limiting-flaw (1/4T) stress versus operating time does not include temperature instrument error. The instrument error is 14.4 °F, and therefore, at 319 °F the enable temperature is 304.6 °F. At 34.7 EFPYs the enable temperature is 339.2 °F or 324.8 °F, without instrument error. (Note that 324.8 - 304.6 = 20.2.) The 34.7 EFPY 1/4T RT_{NDT} is 250.4 °F. Therefore, the RT_{NDT} of the new operating time is 250.4 - 20.2 = 230.2 °F. Interpolation of the 1/4T RT_{NDT} versus operating time yields 20 EFPYs.

The staff finds the above process to be reasonable and to have used information derived with NRC staff approved methods, and therefore, is acceptable.

Secondary Side Temperature

IP3 TSs impose restrictions on the pressurizer level and the primary to secondary ΔT as a function of T_{cold} before a reactor coolant pump (RCP) can be restarted. The appropriate pressurizer level allows the anticipated primary water expansion to be accommodated and the T_{cold} prevents the RCS heatup to increase to the point that the LTOP is automatically disabled. To this end, the licensee established the secondary side temperature limitations (SSTLs) curve. The SSTL is defined with three points, each derived assuming an actual maximum ΔT of 100 °F and the extremes of instrument errors. The licensee does not anticipate primary-to-secondary temperature differences greater than 100 °F. The derivation of the SSTLs is reasonable and is designed to prevent the pressurizer power operated relief valves from being challenged, and therefore is acceptable.

Pressurizer Limitations for LTOP Inoperable

The IP3 TSs provide for operator actions to protect the vessel from overpressure in case the LTOP becomes inoperable. This is also achieved by creating a pressurizer bubble sufficient to keep the RCS pressure below the LTOP setpoint for 10 minutes. The staff considers 10 minutes as adequate time for operator action to interrupt the injection transient. The assumption is that either a single charging pump is capable of injecting, or all three charging pumps are capable of injecting, or one safety injection pump is capable of injecting. All other combination of safety or charging pump injection are considered non credible because of TS limitations.

Pressurizer volumes are calculated taking into account charging or safety injection pump performance, initial RCS pressure and temperature. Pump performance and instrument uncertainties are derived from original plant data. The calculation is reasonable, the data credible, and therefore, the proposed pressurizer limitations for LTOP inoperable are acceptable.

3.9 Summary

The staff reviewed the information submitted by the licensee to support TS changes associated with extension of the P-T limits. The revised P-T curves are calculated to the end of the current license at approximately 34.7 EFPYs. The licensee requested to maintain the LTOP enable setpoint of 319 °F and to limit the applicability of the revised P-T limits to 20 EFPYs. At 20 EFPYs, the licensee can adjust the LTOP enable temperature and continue to use the current P-T curves. The staff reviewed the fluence methodologies, P-T limit curve applicability, EOL projected RT_{PTS} values and recalculation of the EFPYs corresponding to the LTOP enabling temperature of 319 °F. The staff concluded that: (1) the methodology used for the fluence evaluation conforms with the guidance in RG 1.190 and is acceptable, (2) the P-T and LTOP limit applicability was conservatively estimated and is acceptable, and (3) the TS changes correctly reflect the proposed changes, and thus, are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (68 FR 43389). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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.

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