UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated October 2, 1996, as supplemented July 31, 1997, the Consolidated Edison Company of New York, Inc. (the licensee) submitted a request to amend the Pressure and Temperature (P-T) limits in the Technical Specifications (TS) for Indian Point 2 (IP2) (Reference 1). This submittal is related to the exemption request (Reference 2) submitted on October 7, 1997, for using the methodology specified in the Appendix G in the 1996 Addenda to Section XI of the American Society for Mechanical Engineers (ASME) Code (the 1996 methodology) for developing P-T limits for IP2. The exemption request was approved by letter dated January 27, 1998. The July 31, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

The amendment was intended to remove the limiting conditions on the number of effective full-power years (EFPY) for the P-T limits in the current TS. The current P-T limits were generated using the Raju-Newman method, which is different from that of Appendix G and Standard Review Plan (SRP) 5.3.2, for calculating stress intensity factors. The staff rejected the use of the Raj-Newman method in the P-T limits calculation on October 21, 1991, but approved the then proposed P-T limits for reduced EFPYs: 16 EFPYs for the 60 °F/hr curves and 12 EFPYs for the 100 °F/hr curves. The licensee proposed to extend the validity of the P-T limit curves to 21.63 EFPYs by using the Raj-Newman method, and to demonstrate that the Raj-Newman method is equivalent to the 1996 methodology.

The staff evaluates the P-T limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; 10 CFR 50.55a; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and SRP Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2 to review P-T Limit Curves. RG 1.99, Rev. 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1 requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1 requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T Limit submittals, and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limits for the RPV be at least as conservative as those obtained by applying the methodology of

Appendix G to Section XI of the ASME Code. 10 CFR 50.55a specifies the addenda and edition of the ASME Code that is to be utilized by licensees in determining P-T limits. SRP 5.3.2 provides an acceptable method of calculating the P-T limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to the 1989 Edition of 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. 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. This flaw is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. 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 depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

The Appendix G, 10 CFR Part 50 requires that licensees determine the adjusted reference temperature (ART or RT_{NDT}) and the Charpy USE at the maximum postulated flaw depth. 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 (M) term. The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 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 chemistry factor was determined using the tables in RG 1.99, Rev. 2 or surveillance data. The margin term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. RG 1.99, Rev. 2 describes the methodology to be used in calculating the margin term. In addition, Appendix G, 10 CFR Part 50 requires closure flange limitations.

It should be noted that the Combustion Engineering Owners Group (CEOG) provided Report CE NPSD-1039, Rev. 2, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," on July 14, 1997, to the NRC for information only. This report contains weld chemistry data that is applicable to the IP2 RPV. The staff has used information in this report in conducting the review.

2.0 <u>EVALUATION</u>

2.2 <u>The Adjusted Reference Temperature of The Limiting Beltline Material</u>

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the reactor vessel of IP2. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The licensee determined that the material with the highest ART at 21.63 EFPY is the intermediate shell B2002-3, with 0.20% copper (Cu), 0.59% nickel (Ni), and an initial RT_{NDT} of 21 °F. The ART calculated by the licensee using the surveillance data (all three surveillance data are credible) according to Section 2.1 of RG 1.99, Rev. 2 is 193.5 °F for this beltline material. The ART calculated by the staff, using the same methodology, is 183.6 °F. This ART was calculated at 1/4T at 21.63 EFPY with a neutron fluence of 0.488E19 n/cm^2, which was derived linearly from the ID fluence of 1.21E19 n/cm^2 at 32 EFPY from NRC's reactor vessel integrity database (RVID).

The discrepancy in ART values was caused by the licensee's use of the peak 1/4T fluence of 0.585 n/cm^2 in its calculation. If the staff used the same peak fluence, there would be no discrepancy. Therefore, the licensee's ART for Plate B2002-3 is conservative and acceptable.

However, after applying the best estimate copper and nickel values in Report CE NPSD-1039, Rev. 2, to IP2 welds, the staff determined that instead of Plate B2002-3, the circumferential weld 9-042 with weld wire heat no. 34B009 is the limiting beltline material. According to the CE report, the best-estimate copper and nickel for this weld is 0.192% Cu and 1.038% Ni. The staff found that the number of EFPYs for weld 9-042 to reach an ART of 193.5 is 18, about 3 years less than the requested 21.63 EFPYs.

Unlike other P-T limits submittals, the licensee did not use the methodology in SRP 5.3.2 to carry out the next step of generating the P-T limits using the ART of the limiting beltline material. Instead, the licensee's proposed P-T limits were generated using the Raju-Newman method which is similar to the methodology in the 1996 Addenda of Section XI of the ASME Code.

2.3 <u>The 1996 Asme Appendix G Methodology and the Raju-Newman Method</u>

The 1996 methodology incorporates the most recent LEFM solutions regarding stress intensity factors due to pressure (K_{Ip}) and radial thermal gradients (K_{It}). These solutions are based on finite element analyses for inside surface flaws performed at Oak Ridge National Laboratories (ORNL), and work published by Electric Power Research Institute (EPRI) for outside surface flaws. The 1996 methodology simply provides better K_{Ip} and K_{It} estimation. It does not reduce safety margins associated with these K_I values. Hence, if applied correctly, it should be adequate for P-T limits application.

The 1996 methodology provides two methods for calculating K_{It} values: one that uses a generic radial thermal gradient (the first method) and the other that uses the worst thermal stress distribution at a specific time during heatup or cooldown from a plant-specific thermal analysis (the second method).

The Raju-Newman method, which uses the finite-element solutions by Raju-Newman, was used to generate the P-T limits in the submittal of 1991. The licensee compared the proposed 21.63 EFPY P-T limits of 1991 (approved by NRC for reduced EFPYs) with those calculated using the 1996 methodology, and reported that they are within 2% of each other and the differences cannot be discerned from the plotted curves.

2.4 <u>Staff's Verification</u>

The licensee is the first to apply a methodology equivalent to the 1996 methodology in its P-T limits submittal, in which the plant-specific thermal stress distribution during heatup and cooldown was used to calculate K_{It}. As a precaution, the staff compared the licensee's results based on the Raju-Newman method with those in the meeting report (Reference 3) dated August 6, 1996, of ASME Section XI Working Group on Operating Plant Criteria. This meeting report contains P-T limits for generic pressurized-water reactor (PWR) vessel by four organizations. One of them is the licensee's contractor, Westinghouse. All four organizations produced nearly identical results. The information in the meeting report played a role in the acceptance of the new Appendix G by the Section XI Subcommittee, the Boiler and Pressure Vessel Main Committee, and the ASME Board of Nuclear Codes and Standards.

The staff presents in the attached Table 1 results of analyses performed by the staff, the ASME Working Group, and the licensee for cooldown (100 °F/hr) using various methodologies. Column 2 is based on SRP 5.3.2 methodology using IP2 plant-specific parameters and was performed by the staff; Column 3 is based on the first method of the 1996 methodology using IP2 plant-specific parameters and was performed by the staff also; Column 4 is based on the second method of the 1996 methodology using generic PWR parameters and was performed by the ASME Working Group; Column 5 is based on the second method of the 1996 methodology using IP2 plant-specific parameters and thermal analysis by the licensee; and Column 6 is based on the Raju-Newman method using IP2 plant-specific parameters and thermal analysis by the licensee. As explained in the footnotes of Table 1 that data from other sources (i.e., Columns 4, 5, and 6) are not direct reproductions. The staff used a new parameter T-RT_{NDT} to replace T so that an adequate comparison can be made. In addition, additional reviews have been conducted to heatup and leak test P-T limits.

Table 1 indicates that the P-T limits based on the first method of the 1996 methodology using IP2 plant-specific parameters (Column 3) and the P-T limits based on the second method of the 1996 methodology using generic PWR parameters by the ASME Working Group (Column 4) are very similar to those based on SRP 5.3.2 methodology using IP2 plant-specific parameters (Column 2). The table also indicated that the licensee's P-T limits based on the second method of the 1996 methodology (Column 5) and the P-T limits based on the Raju-Newman method (Column 6), using IP2 plant-specific parameters and thermal analysis by the licensee, are very similar. However, the licensee's P-T limits are less conservative than the P-T limits from other sources in the lower pressure range (< 500 psi). To resolve the discrepancy, the licensee provided an explanation and the plant-specific input parameters to the thermal analysis for IP2 in its response (Reference 4) to staff's request for additional information (RAI). The information in Reference 4 revealed that the major contributor to the discrepancy is the plant-specific temperature-dependent properties used in the submittal instead of the constant property values used in the ASME study (Reference 3). The staff examined these input values and considered them reasonable when compared with those from the generic PWR vessel study in Reference 3. The only exception is the film coefficient. The licensee used 7000 BTU/hr-ft-°F instead of a value of 1000 BTU/hr-ft-°F which were used in the current Appendix G and the ASME study (Reference 3) . The staff determined that the licensee's high film coefficient would reduce significantly the difference between the vessel wall and the water temperatures, but would have little effect on the P-T limits. Since the same computer code was used by Westinghouse for generating the P-T limits in the submittal and in the ASME study (Reference 3), the discrepancy must result from the summing effect of the difference in each input parameter. As stated previously, the IP2 plant-specific input values provided in Reference 4 are reasonable, consequently, the staff determines that the 1996 methodology is an acceptable alternative.

The pressure of the proposed P-T limits using the Raju-Newman method of 1991 is about 10 psi higher than those by the 1996 methodology. This was caused by the slightly different influence coefficients used by these two approaches for K_{Ip} and K_{It} calculations due to evolution of the essentially the same methodology. Since the differences cannot be discerned from the plotted curves, the staff concludes that the licensee's Raju-Newman method is equivalent to the 1996 methodology.

2.5 P-T Limitations Based on Closure Flange Materials And Upper Shelf Energy of Beltline Materials

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P-T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the flange reference temperatures of 60 °F, the minimum allowable temperature of this region is 180 °F. These limits are shown on Figures 3.1.B-1 and 3.1.B-2 of the submittal, and the staff has determined that the proposed P-T limits satisfy the requirements in Section IV.A.2 of Appendix G.

Appendix G also requires that the predicted Charpy USE at end-of-license (EOL) for vessel beltline materials be above 50 ft-lb or that licensees demonstrate that lower values of Charpy USE will provide margins of safety equivalent to those required by Appendix G of Section XI of the ASME Code. This USE requirement is satisfied because all beltline materials have EOL USEs above 50 ft-lb.

3.0 CONCLUSIONS

The staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid for 18 EFPYs. Since the 1996 methodology simply provides better K_{Ip} and K_{It} estimation, and does not reduce safety margins associated with these K_I values, the staff endorsed the use of it as an acceptable alternative to the current Appendix G methodology for P-T limits generation. The staff accepts the Raju-Neuman method because the P-T limits that were generated by it are very close to those produced by the 1996 methodology.

The proposed P-T limits are for 21.63 EFPYs. However, due to the change of the limiting beltline material from Plate B2002-3 to weld 9-042 based on the best estimate copper and nickel values in Report CE NPSD-1039, Rev. 2, the proposed P-T limits is approved for 18 EFPYs for heatup, cooldown, and hydrotest. The P-T limits for 18 EFPYs meet the intent of the beltline material requirements in Appendix G of 10 CFR Part 50. They also satisfy Generic Letter 88-11 because the method in RG 1.99, Rev. 2 was used to calculate the ART. The proposed P-T limits approved for 18 EFPYs may be incorporated into the IP2 Technical Specifications. The proposed changes to the Bases are consistent with Appendix G in the 1996 Addenda and are also acceptable.

4.0 REFERENCES

1. October 2, 1996, letter from Stephen E. Quinn, (ConEd) to USNRC Document Control Desk, subject: "Indian Point 2 Plant - Proposed Changes to Technical Specifications Regarding Pressure-Temperature (P-T) Limits (Heatup and Cooldown Curves)."

2. October 7, 1997, letter from P. H. Kinkel, (ConEd) to USNRC Document Control Desk, subject: "Request for Exemption from the Requirements of 10 CFR 50.60: Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation, to Use the 1996 Addenda of ASME Section XI, Appendix G, Article G-2000: Vessels."

3. August 6, 1996, Meeting Report of ASME Section XI Working Group on Operating Plant Criteria, Portland, Oregon.

4. July 31, 1997, letter containing licensee's response to staff RAI from Stephen E. Quinn, (ConEd) to USNRC Document Control Desk, subject: "Supplement to the Indian Point Unit No. 2 License Amendment Request Proposing Changes to the Technical Specifications Regarding Pressure-Temperature (P-T) Limits."

Table 1
A Comparison of P/(T-RTndt) Limits from Different Sources
(100 DEG F/hr Cooldown)

P	T-RTndt[1] (SRP 5.3.2 - Staff)	T-RTndt[1] (First Method of the 1996 Methodology - Staff)	T-RTndt[2] (Second Method of the 1996 Methodology - ASME Working Group)	T-RTndt[3] (Second Method of the 1996 Methodology - Licensee)	T-RTndt[4] (Raju-Newman Methodology-Licensee)
300	-14.8	-12.0	-10.1	-52.67	-60.17
400	15.5	17.0	11.4	- 0.64	- 3.88
500	36.5	37.4	34.3	27.15	24.83
600	52.5	53.1	51.5	46.50	44.60
700	65.6	65.9	65.8	61.23	59.83
800	76.8	76.7	74.1	73.35	72.15
900	86.2	86.0	85.0	83.58	82.52

1. Performed by the staff.

2. Extrapolating from the figure on the page after Page 7 of the ASME meeting report (not very accurate).

	P
T(1/4t) - RTndt = (135.0 + 33.9) - 179 = -10.1	300
T(1/4t) - RTndt = (156.5 + 33.9) - 179 = 11.4	400
T(1/4t) - RTndt = (179.6 + 33.7) - 179 = 34.3	500
T(1/4t) - RTndt = (196.8 + 33.7) - 179 = 51.5	600
T(1/4t) - RTndt = (211.1 + 33.7) - 179 = 65.8	700
T(1/4t) - RTndt = (219.4 + 33.7) - 179 = 74.1	800
T(1/4t) - RTndt = (230.3 + 33.7) - 179 = 85.0	900

3. Results from using new Appendix G methodology - to obtain the metal temperature at 1/4t, a del(T) of 30 deg F was added to the coolant temperature from Table 6 of Attachment III of the submittal.

   Tw = T(1/4t) -(Cooling Rate)(Thickness^2)[1-(1-1/4)^2]/[2x(Thermal Diffu.)]
   = T(1/4t) -(100/60)(8.7^2)[1-(1-1/4)^2]/[2x0.92]
   = T(1/4t) - 30

   An example of the calculation:
   At 600 P, T=210 °F. $T_{1/4t}$ - RTndt = (210 + 30) - 193.5 = 46.5

4. Results from using Raju-Newman methodology - to obtain the metal temperature at 1/4t, a del(T) of 30 deg F was added to the coolant temperature from Table 7 of Attachment III of the submittal.

5.0 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.

6.0 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 (61 FR 58901). 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.

7.0 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.

Principal Contributor: S. Sheng

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AMENDMENT NO.195 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
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