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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE NO. DPR-39

AND AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE NO. DPR-48

COMMONWEALTH EDISON COMPANY

ZION NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-295 AND 50-304

1.0 INTRODUCTION

By letter dated July 26, 1996, as supplemented September 3, 1996, September 18, 1996, two submittals dated October 14, 1996, October 22, 1996, two submittals dated November 8, 1996, and December 17, 1996, Commonwealth Edison Company (ComEd, the licensee), proposed changes to the Technical Specifications (TS) for Zion Nuclear Power Station, Units 1 and 2. The requested changes included: (1) establishing new reactor coolant system (RCS) pressure/temperature (P/T) limit curves and low temperature overpressure protection (LTOP) system limits based on new data obtained from Zion's reactor vessel material surveillance program, (2) relocating the P/T limit curves, LTOP system limits, and the reactor vessel specimen withdrawal schedule to the "Zion Nuclear Power Station Pressure and Temperature Limits Report (PTLR)," in accordance with the guidance of Generic Letter (GL) 96-03, (3) changing the withdrawal date for capsule X in the Zion, Unit 2, reactor vessel specimen withdrawal schedule, and (4) referencing the PTLR in the affected limiting conditions for operation and bases. The updated P/T limit curves are effective for up to 32 effective full-power years (EFPY) for Zion Unit 1 and up to 25.63 EFPY for Zion, Unit 2.

The licensee's supplemental submittals of September 3, 1996, September 18, 1996, two submittals dated October 14, 1996, October 22, 1996, two submittals dated November 8, 1996, and December 17, 1996, clarified and provided additional information in support of the initial application for amendment. They did not affect the Commission's initial proposed finding of no significant hazards consideration determination.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the Act) requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative

controls, and states also that the Commission may include such additional TS as it finds to be appropriate. However, the regulation does not specify the particular requirements to be included in a plant's TS.

On July 19, 1995 (60 FR 36959) the staff revised 10 CFR 50.36 and identified four criteria to be used in determining whether a particular matter is required to be included in a limiting condition for operation in the TS, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) 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; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. As a result, existing TS requirements which fall within or satisfy any of the criteria in 10 CFR 50.36 must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other licensee-controlled documents.

3.0 EVALUATION

All components of the reactor coolant system (RCS) are designed to withstand the effects of cyclic loads resulting from system pressure and temperature changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. In accordance with Appendix G to 10 CFR Part 50, TS limit the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation. These limits are defined by P/T limit curves for heatup, cooldown, LTOP, and inservice leak and hydrostatic testing. Each curve defines an acceptable region for normal operation. The curves are used for operational guidance during heatup and cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

In its letter dated March 19, 1996, the licensee requested an exemption from the requirements of 10 CFR Part 50, Appendix G, that would allow it to utilize American Society of Mechanical Engineers (ASME) Code Case N-514 in the determination of LTOP setpoints. In a letter dated May 16, 1996, the staff granted the exemption request.

3.1 Low Temperature Overpressure Protection (LTOP)

3.1.1 LTOP Description

The LTOP system controls RCS pressure at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised or

10 CFR Part 50, Appendix G violated. Zion's LTOP system uses the pressurizer power operated relief valves (PORV) to accomplish this function. The system is manually enabled by operators and uses a single setpoint as the lift pressure for the PORVs. The design basis of Zion's LTOP system considers both mass-addition and heat-addition transients during water solid RCS conditions. The mass-addition analysis accounts for the injection from one charging pump. The heat-addition analysis accounts for heat input from the secondary sides of all steam generators (SG) into the RCS, upon starting a single reactor coolant pump (RCP). The heat-addition transient analysis assumes the secondary side temperatures of the SG are 50 degrees Fahrenheit higher than the RCS temperature.

Zion's proposed LTOP enable temperature and actuation setpoint were established using the methodology presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves," in combination with American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Case N-514, "Low Temperature Overpressure Protection." WCAP-14040-NP-A was approved by the staff in a safety evaluation dated October 16, 1995. Additionally, Zion was granted an exemption from the requirements of 10 CFR Part 50, Appendix G, related to safety margins for P/T limits and allowed to use the methodology in ASME Code Case N-514 in the staff's letter dated July 30, 1996.

3.1.2 LTOP Enable Temperature

The licensee proposed to change the LTOP enable temperature in the APPLICABILITY section of LCO 3.3.2.G.1 from ≥ 250 degrees Fahrenheit to ≥ 320 degrees Fahrenheit. The LTOP enable temperature is the temperature below which the LTOP system is required to be operable. In accordance with ASME Code Case N-514 and WCAP-14040-NP-A, the minimum allowed LTOP enable temperature is either 200 degrees Fahrenheit or the reactor coolant temperature corresponding to a reactor vessel metal temperature of the limiting (highest) RT_{NDT} + 50 degrees Fahrenheit, whichever is higher. Based on Zion's latest vessel material data, the limiting RT_{NDT} is 233 degrees Fahrenheit. Additionally, licensee calculations show that the temperature difference between the reactor coolant and the most limiting location is 17.66 degrees Fahrenheit and that the uncertainty associated with the wide range RCS cold leg temperature instrumentation is ±7.595 degrees Fahrenheit. Therefore, the minimum allowed LTOP enable temperature for Zion, including instrument uncertainty, is 308.255 degrees Fahrenheit (233 °F + 50 °F + 17.66 °F + 7.595 °F). For an RT_{NDT} value of 233 degrees Fahrenheit, the proposed LTOP enable temperature of 320 degrees Fahrenheit is conservative with respect to the minimum LTOP enable temperature allowed by ASME Code Case N-514 and WCAP-14040-NP-A. Therefore, this change is acceptable.

3.1.3 LTOP Actuation Setpoint

The licensee proposed to change the LTOP actuation setpoint in LCO 3.3.2.G.1.a from ≤ 414 psig to ≤ 407 psig. The LTOP actuation setpoint is the pressure at

which the PORVs will lift, when the LTOP system is enabled, to limit the peak RCS pressure during a pressurization transient.

ASME Code Case N-514 requires that the LTOP system be designed to limit the peak pressure at the controlling location in the reactor to 110 percent of the pressure determined to satisfy the 10 CFR Part 50, Appendix G, P/T criteria. Additionally, since overpressure events most likely occur during isothermal conditions in the RCS, the staff has approved the use of the steady-state 10 CFR Part 50, Appendix G, limits for the design of LTOP. Section 3.2, "COMS Setpoint Determination," of WCAP 14040-NP-A provides discussions of other parameters that need to be considered in determining the LTOP actuation setpoint. Licensee calculations related to the LTOP system were submitted in the October 22, and November 8, 1996, letters. Zion's proposed PORV lift setpoint of 407 psig was based on 110 percent of the steady-state 10 CFR Part 50, Appendix G curve, a minimum head bolt-up temperature of 60 degrees Fahrenheit, a minimum pressurization temperature of 60 degrees Fahrenheit plus instrument uncertainty determined consistent with Instrument Society of America Standard ISA-S67.04-1982, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants," and simultaneous operation of one residual heat removal (RHR) pump and one RCP. In addition, the licensee accounted for the parameters discussed in Section 3.2 of WCAP-14040-NP-A including initial RCS and steam generator parameters, PORV size and lifting characteristics, pressure limits to be protected, instrument uncertainty, and other parameters and conditions.

Zion did not include instrument uncertainty in the bolt-up temperature. However, instrument uncertainty was accounted for in the minimum pressurization temperature to ensure that the RCS is not capable of being pressurized (i.e., the RCS remains vented) until the RCS temperature is greater than or equal to the minimum allowable bolt-up temperature plus instrument uncertainty as determined using a process consistent with ISA S67.04 - 1982. Zion's calculations show that the proposed setpoint of 407 psig protects, as required by the Code Case, the most limiting (lowest) value of pressure on the steady-state Appendix G curve. The most limiting value on Zion's steady-state P/T curves is 470 psig.

The pressure instrument used to actuate LTOP is located downstream of the limiting material in the vessel. Therefore, the dynamic pressure drop results in a pressure at the LTOP location which is less than the actual pressure experienced at the location of the limiting material in the vessel. This pressure drop is a function of flow rate and, therefore, proportional to the number of pumps (RHR pumps and RCPs) running. The dynamic pressure drop accounted for in Zion's calculations bounds plant configurations with one RHR pump and one RCP in operation. To allow operation with different pump configurations and still ensure that the applicable P/T limits are not challenged, Zion procedurally controls the points at which additional pumps may be started. In its letter dated October 22, 1996, in response to questions raised by the staff, the licensee indicated that starting additional pumps is only allowed after the RCS reaches a temperature at which the increase in the allowed 10 CFR Part 50, Appendix G, pressure is sufficient to

overcome the resulting increase in the pressure at the location of the limiting material in the vessel. In this manner, the error introduced in pressure indication as a result of starting additional pumps is accounted for by the difference between the allowed pressure (based on the 10 CFR Part 50, Appendix G curve) and the proposed maximum allowed setpoint pressure of 407 psig.

Based on the above evaluation, the staff finds acceptable the licensee's request to change the LTOP actuation setpoint from ≤ 414 psig to ≤ 407 psig.

3.2 Number of Pumps Aligned for Injection into the RCS

The licensee proposed to change LCO 3.3.2.G.2 to limit the number of pumps aligned for injection into the RCS during LTOP operation to one charging pump and no safety injection pumps. This request deletes the allowance to have a safety injection pump aligned for injection.

The mass-addition LTOP analysis accounts for injection from a single charging pump into a water solid RCS. Additionally, during LTOP operation the pressurizer PORVs lift setpoint is lowered from the normal operating setpoint to the LTOP actuation setpoint. At this lower setpoint, the capacity of a safety injection pump is higher than that of a charging pump. Therefore, injection from a safety injection pump would result in a higher mass flow rate into the RCS than a charging pump and, consequently, a higher pressurization rate than assumed in the mass-addition analysis. Limiting the number of pumps aligned for injection into the RCS to one charging pump and deleting the allowance for a safety injection pump is more conservative than the current TS. Moreover, the proposed changes are necessary to ensure consistency between the TS and the mass-addition LTOP analysis. Based on the above evaluation, the staff finds this change acceptable.

3.3 Steam Generator Secondary Side Temperatures When Starting an RCP

The licensee proposed to change LCO 3.3.2.G.3 to require that the temperature of the SG secondary side in any unisolated RCS loop be less than 50 degrees Fahrenheit higher than the RCS temperature when starting an RCP, when no RCPs are running. This is requested to replace the existing TS which places the 50 degrees Fahrenheit limit only on the SG in the loop in which the RCP is to be started.

When an RCP in a single loop is started, flow is achieved through all unisolated loops of the RCS. Flow in the loop in which the RCP is started results directly from the pumping associated with the started pump while flow in the other loops initiates due to backflow conditions. The flow through the RCS loops increases the heat transfer rates between the secondary sides of the SGs and the RCS. Therefore, the heat-addition LTOP analysis accounts for heat input from all SGs upon starting an RCP in a single loop. This analysis further assumes that the secondary sides of all SGs are 50 degrees Fahrenheit higher than the RCS temperature. Existing TS ensure that the temperature between the secondary side of the SG in the loop in which the RCP is to be

started is less than 50 degrees Fahrenheit higher than the RCS temperature. However, the existing TS do not address the secondary side temperatures of the SGs in the other loops. This can lead to starting an RCP with secondary side temperatures in some SGs higher than is analyzed. The proposed change to the LCO 3.3.2.G.3 addresses this situation and brings the LCO into agreement with the heat-addition analysis by requiring that the temperatures of the secondary sides of all SGs be less than 50 degrees Fahrenheit higher than the RCS temperature. Therefore, the staff finds this change acceptable.

3.4 Conclusions Concerning LTOP, Pumps Aligned for Injection into RCS, and Steam Generator Secondary Side Temperature when Starting an RCP

Based on the above discussion, the staff concludes that the following proposed changes are acceptable:

3.4.1 Change the LTOP enable temperature in the APPLICABILITY section of LCO 3.3.2.G.1 from ≥ 250 degrees Fahrenheit to ≥ 320 degrees Fahrenheit.

3.4.2 Change the LTOP actuation setpoint in LCO 3.3.2.G.1.a from ≤ 414 psig to ≤ 407 psig.

3.4.3 Change LCO 3.3.2.G.2 to limit the number of pumps aligned for injection into the RCS during LTOP operation to one charging pump and no safety injection pumps.

3.4.4 Change LCO 3.3.2.G.3 to require that the temperature in the SG secondary in any unisolated RCS loop be less than 50 degrees Fahrenheit higher than the RCS temperature when starting an RCP, when no RCPs are running.

3.5 Review of the Materials Considerations of the Amendment Request

All components of the RCS are designed to withstand the effects of cyclic loads resulting from system pressure and temperature changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. The P/T limit curves for heatup, cooldown, LTOP, and inservice leak and hydrostatic testing are defined by 10 CFR Part 50, Appendix G. Each curve defines an acceptable region for normal operation. The curves are used for operational guidance during heatup and cooldown maneuvering. The pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable P/T region.

3.5.1 Review of PTLR Format and Methodology

3.5.1.1 Definition of the PTLR

The definitions section of the TS was modified to include a definition of the PTLR (TS 1.32a). Figures, values, and parameters for the P/T limits will be relocated to the PTLR in accordance with the methodology of WCAP-14040, Revision 1, approved by the staff for referencing in its letter dated October 16, 1995. This relocation maintains the acceptance limits and the

limits e facility's safety analysis. The figures, values, and parameters for P, its are addressed on a unit-specific basis in that the EFPY for which a given item is valid for each unit is addressed in the PTLR. As noted in the definition, plant operation within these limits is addressed by individual specifications.

3.5.1.2 References to the PTLR

The following Zion Station TS were revised to replace the P/T limits and the reactor vessel surveillance capsule withdrawal schedule with a reference to the PTLR that provides these limits and the withdrawal schedule:

LCO 3.3.2.A
LCO 3.3.2.B
LCO 3.3.2.G
Figures 3.3.2-1, 3.3.2-2, 3.3.2-3, 3.3.2-4
Tables 3.3.2-1 and 3.3.2-2
LCO 4.3.4.D

3.5.1.3 Reporting Requirements for PTLR

Technical Specification 6.6.1.G, "REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT," was added to the reporting requirements of the administrative controls section of the TS. This specification requires that the PTLR be submitted, upon issuance, to the staff. The report provides the explanations, figures, values, and parameters of the P/T limits until 32 effective full power years (EFPY) for Unit 1 and 25.63 EFPY for Unit 2. Furthermore, this specification requires that the figures, values, and parameters be established using the methodology approved by the staff for this purpose in WCAP-14040-NP-A and be consistent with all the applicable acceptance limits and the limits of the safety evaluation. A change to the PTLR methodology to address the specific case of Zion Unit 1 WF-154 weld is detailed in Section 3.5.2.2 of this SER.

Finally, this specification requires that all changes in values of these limits be documented in the PTLR each effective period and be submitted upon issuance to the staff.

3.5.2 Review of Vessel Material Data

3.5.2.1 General Conclusions

The licensee's method for determining the vessel material data, conforms, in general, to the methodology approved by the staff in WCAP-14040-NP-A which endorses Regulatory Guide 1.99, Revision 2 (RG 1.99, Revision 2), "Radiation Embrittlement of Reactor Vessel Materials." An exception to this was, however, noted by the staff. This exception is addressed in Section 3.5.2.2. A correction based on the staff's evaluation of surveillance data submitted by the licensee is addressed in Section 3.5.2.3. In addition, a significant change (which is in accordance with the accepted methodology) to the initial

reference temperature nil-ductility temperature (IRT_{ndt}) data and margin terms for the limiting material in the Zion station vessels is addressed in section 3.5.2.4.

The staff compared the information supplied by the licensee in this submittal to previously docketed information contained in the staff's Reactor Vessel Integrity Database (RVID). Tables 1 and 2 show any changes (in bold) identified and accepted by the staff in material IRT_{ndt}, chemistry factor (CF), end-of-license (for Zion, Unit 1) or 25.63 EFPY (for Zion, Unit 2) inside diameter (EOL ID) fluence, margin term, and 1/4 T ART. Changes to any additional information in the RVID were determined by applying the calculational methods in RG 1.99, Revision 2.

3.5.2.2 Staff Evaluation of Zion, Unit 1, Weld WF-154

Weld material WF-154, manufactured from weld wire heat number 406L44 and Linde 80 flux, was used in the Zion, Unit 1, vessel to form 82 percent of the inside nozzle beltline to intermediate shell circumferential weld. Since this material was not included in the reactor vessel material surveillance program for either Zion unit, RG 1.99, Revision 2 recommends that the licensee use position 1.1 to evaluate the material's CF. Position 1.1 uses chemical composition (weight percent copper and nickel) to assess a CF for a material. The chemical composition data submitted by the licensee for WF-154 indicated a copper content of 0.31 weight percent and a nickel content of 0.59 weight percent. This gives a CF of 196.7 degrees Fahrenheit for WF-154.

The product of the CF and a fluence factor is the mean value of the adjustment in reference temperature, delta RT_{ndt}. For WF-154 at a neutron fluence of 6.33×10^{18} n/cm^2 at the 1/4 T vessel location, the delta RT_{ndt} is 171.5 degrees Fahrenheit. The adjusted reference temperature (ART) is the sum of the IRT_{ndt} (-5 °F), delta RT_{ndt} (171.5 °F), and a margin term (68.5 °F). This resulted in a calculated ART for this material of 235 degrees Fahrenheit.

The licensee, however, proposed that surveillance data from surrogate weld material manufactured from weld wire heat number 406L44 in the Point Beach, Unit 2 (PB2) surveillance program, be used to determine the CF for the WF-154 weld. The licensee evaluated the CF to be used in the WF-154 ART calculation by using the methodology documented in position 2.1 of RG 1.99, Revision 2. This position of the RG permits the use of surveillance data for determining the CF when two or more credible surveillance data sets become available from the reactor vessel being irradiated. The criteria for determining whether data are credible are identified in the discussion section of the RG. This use of position 2.1 and the PB2 data results in a CF of 203.0 degrees Fahrenheit for the weld heat 406L44 specimens, a CF greater than that found using the generic position 1.1 method (196.7 °F) above.

However, position 2.1 of the RG also permits a reduction in the amount of margin if the surveillance data are determined to be credible. The margin term is calculated by taking two times the square root of the sum of the squares of the standard deviation in the material IRT_{ndt} (σ_I) and the standard

deviation in the RT_{ndt} shift (σ_δ). When the surveillance data are credible, the σ_δ term may be reduced to half the value reported in Section C.1.1 of RG 1.99. For weld metal, this means that σ_δ may be reduced from 28 degrees Fahrenheit to 14 degrees Fahrenheit. Thus, although the CF derived from the use of the PB2 surveillance data was greater (203.0 °F vs. 196.7 °F) than the generic position 1.1 value, due to the reduction in the shift margin term for calculating the ART, the ART by this method is 220.3 degrees Fahrenheit.

To permit the licensee to use data from the PB2 surveillance program, the staff requested that the licensee demonstrate that the irradiation environments at the specimen capsule locations were similar between the Zion units and PB2. This comparison was necessary because several variables of the irradiation environment (irradiation temperature, gamma ray flux, neutron spectral balance, damage rate, etc.) may influence the embrittlement response of the surveillance specimens. With the exception of the irradiation temperature parameter, the licensee provided sufficient information for the staff to determine that the irradiation environments of PB2; Zion, Unit 1; and Zion, Unit 2, were similar.

The licensee stated that the irradiation temperature (based on a cycle-by-cycle average of cold leg temperatures) for the Zion vessels was 529 degrees Fahrenheit for both units. The irradiation temperature for the PB2 vessel was given as 542 degrees Fahrenheit. A literature survey on the effect of irradiation temperature on material embrittlement rates for weld materials similar to WF-154 was conducted by the staff. The staff has found that in this temperature regime ($T < 550$ °F) a decrease in irradiation temperature has been associated with an increase in the RT_{ndt} shift at equivalent fluence levels. In EPRI report NP-6114, it was reported that for a large set of surveillance data there is a factor of an additional one degree of RT_{ndt} shift per one degree of irradiation temperature decrease. In addition, ASTM STP-1046, Volume 2 indicated that there is a trend of about 0.7 degrees per degree change in irradiation temperature for Linde 80 welds with chemical compositions similar to those found in the Zion vessels.

The staff has concluded that based on this information, an adjustment in the CF for the Zion welds of +13 degrees Fahrenheit to account for the 13 degrees Fahrenheit difference in irradiation temperature between the PB2 surveillance capsules and the Zion vessel should have been factored into the calculations. Therefore, as shown in Table 1, a CF value of 216.0 degrees Fahrenheit (203.0 °F from surveillance data plus 13 °F from the irradiation temperature difference) is the proper value for weld WF-154. Since this value is dependent upon surveillance data, it is subject to change when additional data become available. The amendments are satisfactory for implementation because this adjustment does not affect the pressure-temperature limit curves. However, depending on future surveillance capsule results, it is possible that it could become limiting. Therefore, in future calculations using surveillance data from Point Beach, Unit 2, an adjustment in the chemistry factor should be made to account for the difference in irradiation temperature. In its letter dated December 17, 1996, the licensee agreed to

this change in methodology. If this causes a change in the limiting weld, ComEd is requested to notify the staff.

Based on the methodologies in RG 1.99, Revision 2, and a 1/4 T EOL fluence of 6.33×10^{18} n/cm^2, the staff calculated the ART of weld WF-154 to be 231.6 degrees Fahrenheit. This value is 11.3 degrees Fahrenheit greater than the ART calculated by the licensee. However, as addressed in Section 2.5.3 of this safety evaluation, the P/T limit curves constructed in the licensee's submittal were based on the selection of the most limiting material ART in either the Zion, Unit 1, vessel at 32 EFPY or in the Zion, Unit 2, vessel at 25.63 EFPY. The limiting material for both units is weld SA-1769 in the Zion, Unit 2, vessel. Its ART was calculated to be 233.0 degrees Fahrenheit. Since the P/T limit curves calculated using the ART of 233.0 degrees Fahrenheit would bound P/T limit curves based on the WF-154 ART of 231.6 degrees Fahrenheit, no adjustment in the P/T limit curves submitted by the licensee is required.

3.5.2.3 Staff Evaluation of Zion, Unit 2, Plate C4007-1

The licensee determined that the CF and margin values for plate C4007-1 should be 81.6 degrees Fahrenheit and 34 degrees Fahrenheit, respectively. These values are based on the chemical composition (0.12 weight percent copper and 0.53 weight percent nickel) of the plate and are in accordance with section 1.1 of RG 1.99, Revision 2. The licensee determined that the surveillance data from plate C4007-1 were not credible and did not utilize the surveillance data to determine the CF for plate C4007-1.

The staff evaluated the CF for Zion, Unit 2, plate material C4007-1 using surveillance data from the Zion, Unit 2, surveillance program. The staff examined whether the data are credible with respect to the credibility criteria in RG 1.99, Revision 2. Of particular interest was criterion 3 which stated that the scatter of delta RT_{ndt} values about a best-fit line should be less than 17 degrees Fahrenheit, the 1σ standard deviation value for the base metal population used in the development of RG 1.99, Revision 2. The licensee had determined that the data were not credible because 2 data points out of 6 for fitting the best-fit line had fallen between the 1σ value of 17 degrees Fahrenheit and the 2σ value of 34 degrees Fahrenheit.

Since a normal distribution of surveillance data would result in 4 of the 6 data points under 1σ and 2 of the 6 between 1σ and 2σ, the staff concluded that the surveillance data from the Zion, Unit 2, program for plate C4007-1 are credible and provided a valid determination of the material's CF. Hence, in accordance with the methodology in Section 2.1 of RG 1.99, Revision 2, the CF for plate C4007-1 should be 88.1 degrees Fahrenheit instead of the CF of 81.6 degrees Fahrenheit proposed by the licensee.

Section 2.1 of RG 1.99, Revision 2 would permit the amount of margin for plate C4007-1 to be reduced from 34 degrees Fahrenheit to 17 degrees Fahrenheit when surveillance data are used to calculate the CF. However, the standard deviation for the licensee's data was calculated by the staff to be 15.2

degrees Fahrenheit. Due to this large amount of scatter, the staff concludes that the margin value for plate C4007-1, which is included to account for uncertainties in the fluence values and calculational procedures, should be 34 degrees Fahrenheit instead of 17 degrees Fahrenheit. These changes modify the ART for this material. However, plate C4007-1 is not the limiting material and its ART value will not affect the P/T limit curves submitted by the licensee. Therefore, the licensee is not required to change its CF and margin values for Plate C4007-1.

3.5.2.4 IRT_{ndt} and Margin Changes for Zion, Unit 2, Weld SA-1769

Zion, Unit 2, weld SA-1769, manufactured using weld wire heat number 71249 and Linde 80 flux, is the limiting material for the P/T limit curves analysis for both Zion station units. Therefore, changes to its material property data will affect the station's P/T limit curves. The licensee has proposed to change the IRT_{ndt} for this material from a previously reported generic value of -5 degrees Fahrenheit to a material specific value of +10 degrees Fahrenheit. This change was based on data from EPRI report NP-373 for welds manufactured from weld wire heat number 71249. The value of +10 degrees Fahrenheit has been previously accepted by the NRC for welds made from the same weld wire heat number and flux type in the Ginna; Point Beach, Unit 1; Turkey Point 3; and Turkey Point 4 vessels.

The use of material specific data also permits a change to the margin term for the SA-1769 weld. When using the generic IRT_{ndt} value of -5 degrees Fahrenheit, a margin of 68.5 degrees Fahrenheit (σ_I of 19.7 °F and a σ_Δ of 28 °F) was calculated for SA-1769. The material specific IRT_{ndt} value of +10 degrees Fahrenheit permits the use of σ_I of 0 degrees Fahrenheit and a resulting margin of 56 degrees Fahrenheit. This reduction in σ_I is consistent with the methodology in RG 1.99, Revision 2. The value of σ_Δ is unaffected by the change in IRT_{ndt}.

Since these changes to the IRT_{ndt} and margin term for SA-1769 are consistent with data previously accepted by the staff for sister plant welds and with RG 1.99, Revision 2, the staff has concluded that they are acceptable.

3.5.3 Review of the Construction of the P/T Limit Curves

The staff performed independent calculations to assess the P/T limit curves proposed by the licensee for inclusion in the Zion PTLR. The staff's assessment was performed for a limiting ART of 233.0 degrees Fahrenheit. This ART bounds the material properties for all vessel materials in the Zion, Unit 1, vessel to 32 EFPY and all vessel materials in the Zion, Unit 2, vessel to 25.63 EFPY. The staff's calculations were based on the methodologies of Appendix G to Section XI of the ASME Code, and Standard Review Plan (NUREG-0800) Section 5.3.2. The staff calculated P/T limit curves for the following transient conditions: 60 °F/HR reactor cooldown, 100 °F/HR reactor cooldown, 60 °F/HR reactor heatup, and 100 °F/HR reactor heatup.

The curves calculated by the licensee were conservative compared to those determined by the staff. The licensee provided amended copies of these curves in its October 14, 1996, submittal to address typographical and editorial concerns raised by the staff. These changes did not affect the technical validity of the P/T limit curves themselves. As such, the staff has concluded that the P/T limit curves, as amended in the licensee's October 14, 1996 submittal, are acceptable.

3.5.4 Significance of Relocation of Requirements

Relocation of the P/T curves and LTOP setpoints does not eliminate the requirement to operate in accordance with the limits specified in Appendix G to 10 CFR Part 50. The requirement to operate within the limits in the named report or pressure temperature limits report (PTLR) is specified in and controlled by the TS. Only the figures, values, and parameters associated with the P/T limits and LTOP setpoints are to be relocated to the PTLR. The PTLR review process requires that changes to the methodology be approved by the NRC. Further, when changes are made to the figures, values, and parameters contained in the PTLR, the PTLR is to be updated and submitted to the NRC upon issuance.

On this basis, the NRC staff concludes that the licensee provided an acceptable means of establishing and maintaining the detailed values of the P/T limit curves and LTOP system limits. Further, because plant operation continues to be limited in accordance with the requirements of Appendix G to 10 CFR Part 50 and the P/T and LTOP system limits in the TS were established using a methodology approved by the NRC, these changes will not impact plant safety.

The staff also concludes that the relocated requirements discussed above relating to the P/T limits and LTOP system limits are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Accordingly, the staff concludes that the proposed changes are acceptable and that these requirements may be relocated from the TS to the PTLR.

3.5.5 Review of the Modifications to the licensee's 10 CFR Part 50, Appendix H, Program

The licensee also proposed to modify the Zion Station Reactor Vessel Material Surveillance program, a requirement of 10 CFR Part 50, Appendix H. Currently, the licensee's program for Zion, Unit 2, calls for the removal of surveillance capsule X after 14 EFPY. The licensee has proposed, based on changes in the projected fluence at the capsule location due to state-of-the-art fast neutron transport methodology and the use of low-leakage core designs, to defer the removal of capsule X until 19 EFPY.

The staff examined this change with respect to the capsule withdrawal schedule identified in ASTM standard E185-82 which is referenced by 10 CFR Part 50,

Appendix H. Based upon the requirements of E185-82, capsule X should be withdrawn from the Zion, Unit 2, vessel after being irradiated to a neutron fluence (E > 1 MeV) not less than once or greater than twice the peak EOL vessel inside diameter fluence. The licensee's analysis showed that the peak vessel inside diameter fluence at 32 EFPY would be 1.52×10^{19} n/cm^2. The licensee also showed that the capsule X fluence at 19 EFPY would be 2.97×10^{19} n/cm^2. Since 2.97×10^{19} n/cm^2 is between 1.52×10^{19} n/cm^2 and 3.04×10^{19} n/cm^2, the staff has concluded that this change meets the capsule withdrawal schedule of E185-82 and is, therefore, acceptable.

3.6 Conclusions Concerning the Vessel Material Data

3.6.1 The format and methodology proposed for the licensee's PTLR is acceptable. However, the PTLR methodology must be changed to permit the use of surveillance data from the PB2 vessel for Zion, Unit 1, weld WF-154. This change must account for differences in the irradiation temperature between the two vessels and is detailed in Section 3.5.2.2 of this safety evaluation. This change to the PTLR methodology to account for the differences in the irradiation temperature between the two vessels does not need to be submitted to the staff for approval since it is already evaluated and found to be acceptable in this safety evaluation. The staff expects that the licensee will document this adjustment in its PTLR methodology, acquire new data from the PB2 surveillance program in a timely manner, and assess its impact on the Zion, Unit 1, P/T limit curves and pressurized thermal shock (PTS) analysis. Changes in the licensee's assessment of the material condition of weld WF-154 should be submitted to the staff as required by the facility license and 10 CFR Part 50.61.

3.6.2 With the exception of the effects of changes to the CFs for the Zion, Unit 2, C-4007-1 plate material (88.1 °F) and the Zion, Unit 1, WF-154 weld (209.7 °F), the data for the Zion vessel material properties submitted by the licensee were acceptable. However, these changes did not affect the bounding P/T limit curve analysis submitted by the licensee because these are not the limiting materials in the P/T limit curve analysis.

3.6.3 The P/T limit curves constructed by the licensee for an ART of 233.0 degrees Fahrenheit satisfy 10 CFR Part 50, Appendix G; Appendix G to Section XI of the ASME Code, and SRP Section 5.3.2. Therefore, they provide acceptable regions of operation for the Zion, Unit 1, vessel to 32 EFPY and the Zion, Unit 2, vessel to 25.63 EFPY.

3.6.4 The change to the Zion, Unit 2, reactor vessel surveillance program (the removal of capsule X at 19 EFPY) meets the capsule withdrawal schedule of ASTM standard E185-82 and is, therefore, acceptable.

Table 1: Changes in RVID data for Zion, Unit 1						
Beltline Material ID	Heat No.	IRT_{ndt} (°F)	CF (°F)	EOL ID Fluence (x 10^{19} n/cm²) (1)	Margin (°F)	1/4 T ART (°F)
Int. Shell Axial Welds WF-4	8T1762	-5	152.25	**0.599**	68.47	**172.9**
Int. Shell Axial Welds WF-8	8T1762	-5	152.25	**0.599**	68.47	**172.9**
Int. Shell Plate	B7835-1	-20	**80.8 (2)**	**1.39**	**34 (2)**	**90.8**
Int. Shell Plate	C3795-2	-10	80.8	**1.39**	34	**100.8**
Lower Shell Axial Welds WF-8	8T1762	-5	152.25	**0.599**	68.47	**172.9**
Lower Shell Plate	B7823-1	-20	87.4	**1.39**	34	**97.1**
Lower Shell Plate	C3799-2	-20	104.5	**1.39**	34	**113.3**
Middle Circ. Weld WF-70	72105	-26	**225.5 (3)**	**1.39**	**28 (3)**	**216.3**
Upper Circ. Weld WF-154	406L44	-5	**216.0 (4)**	**1.05**	**48.33 (4)**	**231.6**
Upper Shell Forging	ANA 102	20	37.0	**1.05**	34	**86.3**

(1) New fluence values were reported in Attachment F, Attachment 2 to the licensee's submittal dated July 26, 1996, "WCAP-14666, Revision 0, Zion Units 1 and 2 Radiation Analysis and Neutron Dosimetry Evaluation."

(2) A conservative CF value from the tables of Position 1.1 of RG 1.99 Revision 2 was assumed because the licensee determined that the value based on surveillance data (76.7 °F) was not credible.

(3) CF and margin based on credible surveillance data.

(4) CF and margin based on credible surveillance data from the Point Beach Unit 2 reactor, including a +13 degrees Fahrenheit shift to account for differences in cold leg (irradiation) temperature (Point Beach, Unit 2 = 542 degrees Fahrenheit, Zion Units = 529 degrees Fahrenheit).

Table 2: Changes in RVID data for Zion, Unit 2

Beltline Material ID	Heat No.	IRT_{ndt} (°F)	CF (°F)	25.63 EFPY ID Fluence ($x 10^{19}$ n/cm^2) (1)	Margin (°F)	1/4 T ART (°F)
Int. Shell Plate	B8006-1	10	81.8	1.25	34	119.2
Int. Shell Plate	B8040-1	-10	96.4	1.25	34	112.7
Int./Lower Shell Circ. Weld SA-1769	71249	+10 (2)	182	1.25	56 (2)	233.0
Lower Int. Shell Axial Weld WF-29	72102	-5	174.1	0.431	68.47	173.8
Lower Nozzle Belt Forging	ZV-3855	10	58	0.950	34	92.9
Lower Shell Axial Welds WF-70	72105	-26	225.5 (3)	0.431	28 (3)	144.9
Lower Shell Plate	B8029-1	-10	81.2	1.25	34	98.7
Lower Shell Plate	C4007-1	10	88.1 (4)	1.25	34 (4)	125.1
Nozzle/Int. Shell Circ. Weld WF-200	821T44	-5	177.95	0.950	68.47	213.7

(1) New fluence values were reported in Attachment F, Attachment 2 to the licensee's submittal dated July 26, 1996, "WCAP-14666, Revision 0, Zion Units 1 and 2 Radiation Analysis and Neutron Dosimetry Evaluation." The 25.63 EFPY ID fluence is given to be consistent with the ARTs in the last column. EOL ID fluences are given in the RVID.

(2) Based on additional investigation by the licensee into IRT_{ndt} data from other welds manufactured for sister plants with the same weld wire heat. Documented in EPRI report NP-373.

(3) CF and margin based on credible surveillance data.

(4) NRC determined that the surveillance data reported by the licensee was credible and that the CF proposed by the licensee (81.4 °F) based on the tabulated values of RG 1.99, Revision 2, Position 1.1, would not be conservative. The margin term was not reduced because of a large standard deviation (15.2 °F) in the licensee's data.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (61 FR 50341). Accordingly, the amendments meet 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 amendments.

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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