FPL

JUL 07 2000

L-2000-146
10 CFR §50.90

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, D.C. 20555

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Resubmittal of Proposed License Amendments
"Revised Pressure-Temperature (P/T) Curves, and Cold Overpressure Mitigation System (COMS) Setpoints"

By letter L-2000-028 dated April 27, 2000, in accordance with 10 CFR §50.90, Florida Power and Light Company (FPL) requested that Appendix A of Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4 be amended to extend the heatup, cooldown, and inservice test limitations for the Reactor Coolant System (RCS). Subsequently, previously submitted proposed license amendments have been completely revised and are being resubmitted with this letter for approval. FPL requests that the proposed license amendments be approved by October 31, 2000.

The present pressure-temperature (P/T) limits specified in Technical Specification (TS) 3.4.9.1 and Figures 3.4-2, 3.4-3 and 3.4-4 apply for operation up to 19 effective full power years (EFPY). The proposed amendments will extend the service period for the new P/T limits to a maximum of 32 EFPY.

The proposed amendments also revise Technical Specification 3.4.9.3, Cold Overpressure Mitigation System (COMS) setpoints. COMS is the Westinghouse version of Low Temperature Overpressure Protection (LTOP). The maximum permissible Power Operated Relief Valve (PORV) setpoint for low temperature operation of the RCS is being changed from 415 $\pm$ 15 psig to $\leq$ 468 psig as a result of the P/T limit changes. The enable temperature for the overpressure mitigation system will remain the same at 275°F.

Additionally, the proposed amendments revise TS Surveillance Requirements 4.4.9.3.1a and 4.4.9.3.1d. This revision allows the Analog Channel Operational Test (ACOT) and the backup nitrogen supply to be verified operable up to 12 hours after decreasing the RCS cold leg temperature to less than or equal to the overpressure mitigation system enable temperature.

The proposed changes to the Technical Specifications are also required to meet the requirements of 10 CFR 50.60, and 10 CFR 50, Appendix G. Attached are two exemption requests to 10 CFR 50.60 for the use of the following documents, in lieu of 10 CFR 50, Appendix G:

(i) ASME Section XI Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1;" and

AP01

(ii) ASME Section XI Code Case N-641, "Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection (LTOP) System Requirements, Section XI, Division 1."

FPL has determined that the proposed license amendments do not involve a significant hazards consideration pursuant to 10 CFR §50.92. A description and justification of the amendments request is provided in Attachment 1. The no significant hazards determination in support of the proposed Technical Specifications changes is provided in Attachment 2. Attachments 3 and 4 provide the exemption requests listed above. Attachment 5 provides the proposed revised Technical Specifications pages. Attachment 6 contains the proposed revised Technical Specifications Bases pages for information only.

Additionally, two enclosures are being provided to support the proposed license amendments. Enclosure 1 provides Westinghouse report WCAP-15092, Revision 3, "Turkey Point Units 3 and 4 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation." Enclosure 2 contains Westinghouse report on Low Temperature Overpressure Protection System Systems Setpoints, 32 and 48 Effective Full Power Years for Turkey Point Units 3 and 4.

Enclosure 2 contains information proprietary to Westinghouse Electric Company LLC. It is supported by an affidavit signed by Westinghouse. Accordingly, it is requested that the information in Enclosure 2 be withheld from public disclosure in accordance with 10 CFR 2.790.

The proposed license amendments are similar in nature to other NRC approved industry license amendments related to P/T Curves and LTOP setpoints, such as for Duke Energy Corporation's Oconee Nuclear Station, where requests for exemptions from 10 CFR 50, Appendix G were also submitted along with the license amendment request.

The proposed license amendments have been reviewed by the Turkey Point Plant Nuclear Safety Committee and the FPL Company Nuclear Review Board. In accordance with 10 CFR §50.91(b)(1), a copy of these proposed license amendments is being forwarded to the State Designee for the State of Florida.

Should there be any questions on this request, please contact us.

Very truly yours,

R. J. Hovey
Vice President
Turkey Point Plant

GSS

Attachments

Enclosures

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant
Florida Department of Health

Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendments
"Revised P/T Curves and LTOP Setpoints"

STATE OF FLORIDA)
) ss.
COUNTY OF MIAMI-DADE)

R. J. Hovey being first duly sworn, deposes and says:

That he is Vice President, Turkey Point Plant, of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

RJH

R. J. Hovey

Subscribed and sworn to before me this

7th day of July, 2000.

Cheryl A. Stevenson

Name of Notary Public (Type or Print)

CHERYL A. STEVENSON
NOTARY PUBLIC - STATE OF FLORIDA
COMMISSION # CC929876
EXPIRES 6/19/2004
BONDED THRU ASA 1-888-NOTARY1

R. J. Hovey is personally known to me.

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ATTACHMENT 1

DESCRIPTION OF PROPOSED AMENDMENTS

1.0 Purpose

FPL requests that Appendix A of Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4, respectively, be amended to extend the heatup, cooldown, and inservice test limitations for the Reactor Coolant System (RCS). The present pressure-temperature (P/T) limits specified in Technical Specification 3.4.9.1 and Figures 3.4-2, 3.4-3 and 3.4-4 apply for operation up to 19 effective full power years (EFPY). The service period for the new P/T limits will be extended to a maximum of 32 EFPY.

The proposed amendments also revise Technical Specification 3.4.9.3, Cold Overpressure Mitigation System (COMS) setpoints. COMS is the Westinghouse version of Low Temperature Overpressure Protection (LTOP). The maximum permissible Power Operated Relief Valve (PORV) setpoint for low temperature operation of the RCS is being changed from 415 $\pm$ 15 psig to $\leq$ 468 psig as a result of the pressure-temperature limit changes. Final plant operational setpoints may be conservatively set at values less than these maximum calculated values due to operational and equipment considerations. The enable temperature for the overpressure mitigation system will remain the same at 275^{o}F.

Additionally, the proposed amendments revise TS Surveillance Requirements 4.4.9.3.1a and 4.4.9.3.1d. This revision allows the Analog Channel Operational Test (ACOT) and the backup nitrogen supply to be verified operable up to 12 hours after decreasing the RCS cold leg temperature to less than or equal to the overpressure mitigating system enable temperature.

The P/T limit curves and LTOP setpoint calculations were developed using NRC-approved methodology, with the addition of the following exemption requests:

(a) ASME Code Case N-588 (Reference 1), which considers the circumferential limiting weld materials; and

(b) ASME Code Case N-641 (Reference 2), which provides an alternative to Appendix G-2215 which can be used to determine LTOP effective temperature.

2.0 Background

Appendix G of 10 CFR 50 is the basis for the fracture toughness requirements of the RCS. The pressure-temperature limits identified in Appendix G must be as conservative as the limits obtained by following the methods of analysis and the margin of safety specified in Appendix G of Section XI of the ASME code.

Turkey Point Units 3 and 4 reactor vessels are essentially identical for the purposes of the P/T limit curves and LTOP setpoint. Babcock and Wilcox (B&W) fabricated the Turkey Point Units 3 and 4 reactor vessels using ring forgings joined by submerged arc welds. Therefore, there is only one beltline circumferential weld in the core midplane region. This weld is designated SA 1101 and was fabricated from Page weld wire heat number 71249 for both reactor vessels. Both units have the exact same limiting material circumferential beltline weld (Reference 3). The method used to determine the most limiting material for both units is based on the material properties and projected cumulative fluence. The analysis resulted in developing one set of curves applicable for use on both units. Currently, Turkey Point Units 3 and 4 pressure-temperature limits have been evaluated for operation up to 19 EFPY. Late in the year 2000, both units will reach approximately 19 EFPY (Reference 4) and therefore, the Technical Specification P/T limit curves and LTOP setpoints will require revision.

The methodology for the present P/T limit curves and LTOP setpoints was developed by Westinghouse. Westinghouse was contracted to develop the new curves and LTOP setpoint. The analyses performed comply with the methods of the NRC approved topical report WCAP 14040-NP-A, (Reference 5) with the exception of the Code Cases. The Code Cases contribute to increasing the operating window by reflecting an updated understanding of material properties and operating conditions.

Beltline material properties were supplied to Westinghouse by FPL, and are in agreement with the recently approved and published Reactor Vessel Industry Database (RVID) (Reference 6). There is one additional beltline weld data point from a B&W Owners Group capsule A5 removed from Davis-Besse in 1998 and reported in 1999 (Reference 7). The results from this capsule were analyzed using the credibility criteria of Regulatory Guide 1.99, Revision 2 (Reference 8) and were within the expected uncertainty limits. The Westinghouse topical reports on the Pressure /Temperature Curves, WCAP-15092 Rev. 3 (Reference 9) and the letter report on the P/T calculation (Reference 10) are provided as Enclosures 1 and 2, respectively.

Two exemption requests are being made in support of the proposed license amendments. The first exemption request employs Code Case N-588, which allows the use of circumferential flaws in circumferential welds. The second exemption request employs Code Case N-641, which supplies alternatives to Appendix G-2215 to determine LTOP system effective temperatures. These exemptions are being sought to assure that the pressure differential margin between the reactor coolant pump seals and the ASME Section XI Appendix G limits are maintained, so as to not risk equipment damage or inadvertent PORV actuation.

3.0 PROPOSED TECHNICAL SPECIFICATION CHANGE REQUEST

The following changes to the Technical Specifications are proposed:

a) **Technical Specification INDEX: Revise titles of Figures 3.4-2 and 3.4-3 and delete reference to Figure 3.4-4.**

Discussion: The proposed change is editorial, to ensure consistency with the format of the Technical Specifications.

b) **Technical Specification 3/4.4.9, Pressure/Temperature Limits: Delete reference to Figure 3.4-4.**

Discussion: The new heatup curves have the 60 ^{o}F/hour and 100 ^{o}F/hour rates on one figure. Deletion of Figure 3.4-4 is purely administrative.

c) **Technical Specification Figures 3.4-2, 3.4-3, and 3.4-4: Revise these figures to reflect the Pressure-Temperature Limits applicable for the service period up to 32 EFPY, as opposed to the present 19 EFPY.**

Discussion: The 32 EFPY curves replace the 19 EFPY curves. The new heatup curves have the 60 ^{o}F/hour and 100 ^{o}F/hour rates on one sheet. The 19 EFPY curves were separate for each heatup rate. The curves themselves have no instrument uncertainty built in but the associated LTOP setpoint does have the uncertainty margin included. The new curves are complete to minimum bolt-up temperature whereas the 19-year curves stopped at 80 ^{O}F. The 32 EFPY curves have a compound slope which shows the effect of Code Case N-588. This slope reflects the fact that at low temperatures the forging with its assumed longitudinal reference flaw dominates because hoop stress dominates. The weld is limiting for most of the range.

d) **Technical Specification 3.4.9.3, Overpressure Mitigation System: Revise the power-operated relief valve lift setting from 415 ± 15 psig to ≤ 468 psig.**

Discussion: The proposed PORV lift setpoint for operation of the COMS is ≤ 468 psig. This setpoint represents the analytical limit derived from an evaluation of the revised P/T limit curves and applicable design basis overpressure events, and is adjusted to account for instrument uncertainty. The increased operating window obtained with this setpoint is gained by the use of Code Case N-588.

The proposed changes reflect the addition of the new curves themselves, and the new PORV setpoint limitation. These values are based on the analyses in Enclosures 1 and 2.

e) **Technical Specifications 4.4.9.3.1a and 4.4.9.3.1d, Overpressure Mitigating Systems Surveillance Requirements: Add footnote to these surveillance requirements to allow acceptable surveillance completion.**

Discussion: Technical Specification Surveillance Requirements 4.4.9.3.1a and 4.4.9.3.1d are not needed to be met until 12 hours after decreasing the RCS cold leg temperature to less than or equal to 275 °F. The added footnote is consistent with NUREG-1431, Standard Technical Specifications Westinghouse Plants, Channel Operational Test (COT) Surveillance Requirement 3.4.12.8, which allows performance of the COT within 12 hours subsequent to achieving RCS temperature less than or equal [275 °F]. The need for a twelve hour window to complete the COT and demonstrate operability of the PORV backup nitrogen system is based on the timing sequence of the COMS enable temperature to the Residual Heat Removal (RHR) entry temperature. The RHR entry temperature for Turkey Point Units 3 and 4 is 350 °F, which is 75 °F higher than the COMS enable temperature. Extending the time period for performing these surveillances will allow the Operators to focus on the transition from Mode 3 to Mode 4, and to stabilize the plant on RHR cooling prior to performing any COMS Surveillances.

As an administrative change, the "backup air supply" in TS 4.4.9.3.1d is being revised to read "backup **nitrogen** supply" to reflect the original and current plant configuration where nitrogen has always been used as opposed to the air mentioned in this TS.

Delaying completion of the COT and PORV backup nitrogen operability test for up to 12 hours will not pose a significant safety hazard due to the inherent reliability and redundancy of the Turkey Point Instrument Air System (IAS). The twelve hour time frame considers the unlikelihood of a low temperature overpressure event occurring concurrently with the loss of the highly redundant Instrument Air System.

The Instrument Air System is designed to supply a continuous and reliable air source. The design incorporates sufficient redundancy such that any active component failure will not prevent the system from performing its function. To achieve a continuous reliable source of instrument air, a single air compressor is sized for the expected instrument air demand of both units. Because the Instrument Air System is normally operated with Units 3 and 4 cross-connect valve open, the two systems will function as a common system. Two compressors are provided for each unit - one diesel-driven air compressor and one motor-driven compressor. The diesel-driven air compressors are capable of supplying the required capacity without reliance on external power sources. This arrangement provides up to 300% redundancy in the instrument air supply capacity. Additional reliability is available via a connection to the Units 1 and 2 service air system.

4.0 BASIS FOR PROPOSED CHANGES

The pressure-temperature limit curves are constructed prior to the calculation of the LTOP setpoint. The following paragraphs discuss the data and methods used to develop both the P/T limit curves and LTOP setpoint, which are based upon WCAP-15092, Revision 3.

4.1 Material Properties

The material chemistry data used to construct the P/T limit was taken from the Reactor Vessel Integrity Database (RVID). The calculations of chemistry factor (CF) values for the beltline materials were performed in accordance with Regulatory Guide (RG) 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials. These CF values were generated using either chemistry (copper and nickel values), or using surveillance capsule test data where this data was available and credible.

The credibility criteria of RG 1.99, Rev. 2 were used to determine the credibility of surveillance capsule test data. All forging material was deemed credible with the exception of the

Unit 4 intermediate shell forging for which only one data point was available. The surveillance capsule test data for the girth welds did not meet the credibility requirement so RG chemistry table values were used.

The ratio procedure of RG 1.99 was not used for the beltline weld metal. The surveillance welds demonstrate higher copper values than the best estimate value for the 71249-weld metal and therefore, not using the ratio procedure is conservative. The Adjusted Reference Temperature (ART) of all beltline materials was calculated in accordance with RG 1.99. The controlling values are those with the highest ART in the 1/4 Thickness of the vessel (T) and 3/4 T using either Regulatory Position 1.1 or 2.1 of the RG. These controlling materials were determined to be the circumferential welds and the Unit 4 intermediate shell forging. The CF values for these constituents were determined using the RG 1.99 chemistry tables.

4.2 Fluence Calculations

The predicted fast neutron fluence values at the critical reactor vessel locations for use in the pressure/temperature limit curves are based on methods consistent with Draft Regulatory Guide DG-1053, "Calculations and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The determination of the fluence is based on both calculations and measurements. The fluence prediction is made with calculations, and measurements are used to qualify the calculational methodology.

The projected fluence values include the calculated results as follows:

a) Cycles 1-12 for both units: The projected fluence values were calculated using the Discrete Ordinate Transport (DOT 4.3) computer code for neutron transport analysis and nodal codes for neutron source evaluations. These results were benchmarked against Turkey Point Unit 3 Cycle 10 dosimetry measurements.

b) Cycle 13 to the time of the Thermal Power Uprate for each unit: The projected fluence values are based on the post-uprate neutron flux calculation, corrected by the ratio of rated thermal power between the pre- and post-uprate conditions.

c) Unit 3 Cycle 15 (post-uprate) and Unit 4 Cycle 16 (post-uprate) to current cycle for each unit: The projected fluence values are calculated using Discrete Ordinate Radiation Transport (DORT) computer code for neutron transport analysis, a multi-group cross-section library based on Evaluated Nuclear Data File Version B-VI (ENDF-VI) and Westinghouse Advanced Nodal Code (ANC) computer code. These results were benchmarked against Turkey Point Unit 3

Cycle 15 dosimetry measurements.

The post-uprate flux was then projected to 32 EFPY. The projected fluence values assume the continuance of present low leakage fuel management, and the continued use of hafnium flux suppression assemblies and the associated power restrictions on the core flat locations.

A conservative bounding fluence was used to generate the P/T limit curves.

4.3 Determination of Pressure-Temperature Limits

The basis for the requirements for fracture toughness of ferritic materials in pressure retaining elements of the reactor pressure boundary is the ASME code, Section XI, Appendix G. The proposed P/T curves were developed using Code Case N-588. Code Case N-588 uses circumferential flaw solutions for circumferential welds.

The actual methodology for pressure-temperature limit curve development is consistent with the overall approach of ASME Section XI, Appendix G and approved Westinghouse topical report WCAP-14040-NP-A, where:

$$K_{applied} < K_{material}$$

$$C \times K_I \text{ (pressure)} + K_I \text{ (thermal)} < K_{Ia}$$

where C = 2 for service level A & B; and C = 1.5 for hydrostatic and leak test conditions.

A one-quarter thickness (1/4T) flaw is assumed.

Cooldown

For the calculation of the allowable pressure versus coolant temperature during cooldown, the 1/4T reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside surface. These stresses increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both isothermal and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest. Furthermore, if conditions exist such that the increase in K_{Ia} exceeds $K_{I\ Thermal}$, the calculated allowable pressure during cooldown will be greater than the steady-state thermal equilibrium value.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant system temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the isothermal situation. It follows that, at any given reactor coolant system temperature, the change in temperature developed during cooldown results in a higher value of K_{Ia} at the 1/4T location for the finite cooldown rates as opposed to the 0 degrees/hr cooldown rate. Furthermore, if conditions exist such that the increase in K_{Ia} exceeds $K_{I\ Thermal}$, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above methods are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative plant operation for the RCS for the entire cooldown period.

Heatup

Three separate calculations are required to determine the limit curves for finite heatup rates. As is performed in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions, assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ia} for the 1/4T crack during heatup is lower than the K_{Ia} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ia} values do not offset each other. Under these conditions, the pressure-temperature limit curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses, which are tensile in nature and therefore, tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the heatup rate and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of the pressure-temperature limit curves for both the steady state and finite heatup rate conditions, the final pressure-temperature limit curves are generated by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is the smallest of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup rate limitations, because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside surface. The associated pressure limit must at all times be based on the analysis of the most critical criterion.

4.4 Determination of Low Temperature Overpressure Protection (LTOP) Setpoint

The LTOP system at Turkey Point is called the Cold Overpressure Mitigation System (COMS). COMS was designed to provide overpressure protection for the reactor vessel from a rapidly propagating brittle fracture. This protection is implemented by choosing a COMS setpoint which prevents the operating unit from exceeding the limits of the pressure/temperature curves. The COMS design basis takes credit for the fact that overpressure events are most likely to occur during isothermal conditions in the RCS. Therefore, it is appropriate to use the steady-state ASME Section XI Appendix G limit.

COMS was designed to mitigate mass input and heat input induced pressure transients during cold shutdown transient and steady state conditions. COMS utilizes the pressurizer Power Operated Relief Valves (PORVs) as the pressure relief path. The following two potential overpressure transients to the reactor coolant system have been identified as the design basis for COMS:

1) The start of an idle reactor coolant pump (RCP) with the secondary water temperature of the steam generators 50° F above the RCS cold leg temperature.

2) The start of a High Head Safety Injection (HHSI) Pump and its injection of water into a water-solid RCS.

The first pressurization transient is characterized as an energy addition event. The second transient is characterized as a mass addition event. Of the two transients, the mass addition event is more limiting for Turkey Point Units 3 and 4.

Technical Specifications 3.4.9.3, 3.4.1.3, and 3.4.1.4.1 provide provisions to isolate the High Head Safety Injection (HHSI) flowpaths to the RCS and to prevent the start of an idle RCP if secondary temperature is more than 50°F above the RCS cold leg temperatures. These requirements are designed to ensure that mass and heat input transients that are more severe than those assumed in the low temperature overpressurization protection analysis cannot occur.

The operability of two PORVs or a reactor coolant system (RCS) vent opening of at least 2.20 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50, when one or more of the RCS cold legs are less than or equal 275°F.

Each PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either:

(1) The start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures including margin for instrument error, or

(2) The start of a HHSI pump and its injection of water into a water-solid RCS.

A range of acceptable setpoints was developed using the methodology outlined in the NRC approved topical report WCAP-14040-NP-A, Revision 2, which includes instrument uncertainty of 70 psig. The setpoint range was calculated to be $\leq$ 468 psig.

A footnote is added to Technical Specification Sections 4.4.9.3.1a and 4.4.9.3.1d regarding Low Temperature Overpressure (LTOP) System surveillances. Per the added footnote, the Channel Operational Test is not required to be performed, and the backup nitrogen supply system is not required to be verified operable until 12 hours after decreasing RCS temperature to ≤ 275 °F during cooldown. The logic of the added footnote is consistent with NUREG-1431, Standard Technical Specifications Westinghouse Plants, Channel Operational Test (COT) Surveillance Requirement 3.4.12.8, which allows performance of the COT within 12 hours subsequent to achieving [275 °F]. The need for a twelve hour window to complete the COT and demonstrate operability of the PORV backup nitrogen system is based on the timing sequence of the COMS enable temperature to the Residual Heat Removal (RHR) entry temperature. The RHR entry temperature for Turkey Point Units 3 and 4 is 350 °F, which is 75 °F higher than the COMS enable temperature. Extending the time period for performing these surveillances will allow the Operators to focus on the transition from Mode 3 to Mode 4, and to stabilize the plant RHR cooling prior to performing any COMS surveillances.

4.5 Enable Temperature

WCAP-15092 (Enclosure 1) calculates an enable temperature of 340 °F for 32 EFPY. However, subsequent to that calculation, approved Code Case N-641 became available. Attachment 4 is an exemption request to use this Code Case.

The enable temperature was calculated using the method described in ASME Code Case N-641, Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection System Requirements, Section XI, Division 1. This Code Case states the following in paragraph 2215.2a, "the LTOP system effective temperature T_{enable} is the temperature at or above which the safety relief valves provide adequate protection against non-ductile failure. LTOP systems shall be effective below the higher temperature determined in accordance with 1 and 2 below.

1. a coolant temperature of 200 °F;

2. a coolant temperature corresponding to a reactor vessel temperature for all vessel beltline materials where T_{enable} is defined for inside axial surface flaws as RT_{ndt} + 40 °F, and T_{enable} is defined for inside circumferential surface flaws as RT_{ndt} - 85 °F."

For Turkey Point, the requirements of paragraph 2 equate to:

T_{enable} (axial flaws) = 124 °F + 40 °F + 24.6 °F = 188.6 °F

T_{enable} (circumferential flaws) = 262 °F - 85 °F + 24.6 °F = 201.6 °F

Where RT_{ndt} is from Tables 8-5 and 8-6 of WCAP 15092 (Enclosure 1). The value of 24.6 °F is a correction factor from the inside wall to the 1/4 T location as described in section 10 of WCAP-15092, rev. 3.

Although an enable temperature of 201.6 °F is justified, Turkey Point will use a more conservative enable temperature of 275 °F, which is the current enable temperature.

5.0 Conclusion

The proposed changes to the Technical Specifications are required to meet 10 CFR 50.60 and 10 CFR 50, Appendix G. The methods applied are in accordance with the ASME Section XI code, with exemptions requested for use of Code Cases N-588 and N-641. Code Cases N-588 and N-641 have been approved by ASME Section XI. Material weld properties are as published in the NRC RVID database as supplemented by a recent data point from the Babcock & Wilcox Owners Group, A-5 capsule from Davis-Besse. The fluence methods used are consistent with Draft Regulatory Guide DG-1053. The COMS setpoint limitation was determined in accordance with the NRC approved topical report WCAP-14040-NP-A.

Therefore, the proposed P/T limit curves and COMS setpoint limitations are acceptable for use at Turkey Point Units 3 and 4.

6.0 REFERENCES

1. Code Case N-588, Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, ASME Section XI, Division 1.

2. Code Case N-641, Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection (LTOP) System Requirements, Section XI, Division 1.

3. L-98-155, Response to RAI to Generic Letter 92-01, Revision 1, Supplement 1.

4. WCAP-14291, Turkey Point Units 3 and 4 Uprate Report

5. WCAP-14040-NP-A, Methodology Used to Develop COMS Setpoints and RCS P/T Curves, 1996.

6. RVID, Reactor Vessel Integrity Database, USNRC, 1999

7. BAW-2360P, Analysis of the A-5 Capsule, MIRVP, June 1999

8. Regulatory Guide 1.99 Revision 2, Radiation Embrittlement of Reactor Vessel Materials.

9. WCAP-15092 Revision 3, Turkey Point Units 3 and 4 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation, Westinghouse Electric Company LLC, May 2000.

10. Low Temperature Overpressure Protection System Systems Setpoints, 32 and 48 Effective Full Power Years for Turkey Point Units 3 and 4, Westinghouse Electric Company LLC, June 2000.

ATTACHMENT 2

No Significant Hazards Consideration Determination

Introduction

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR §50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendments.

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability of occurrence of an accident previously evaluated for Turkey Point is not altered by the proposed amendment to the Technical Specifications. Each accident in the Turkey Point UFSAR was examined with respect to the changes to the proposed Pressure-Temperature (P/T) limit curves and associated Cold Overpressure Mitigation System (COMS) setpoint limitations.

The proposed changes do not impact the integrity of the reactor coolant system pressure boundary (i.e., no change in operating pressure, materials, seismic loading, etc.) and therefore does not increase the potential for the occurrence of a loss of coolant accident (LOCA). The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, or construction standards. The probability of any design basis accident (DBA) is not affected by this change, nor are the consequences of any DBA affected by this change. The proposed P/T limit curves and COMS setpoint limit are not considered to be an initiator or contributor to any accident currently evaluated in the Turkey Point UFSAR.

The curves and setpoint limit were generated in accordance with approved NRC and ASME methodology. Code Cases N-588 and N-641 have ASME Code Committee approval.

Delaying performance of two of the COMS surveillances (PORV Channel Operational Test and the backup nitrogen supply verification) until 12 hours after decreasing the RCS cold leg temperature to ≤ 275 °F during cooldown was also evaluated with respect to the plant accident analyses. The change was determined to not represent a significant increase in the probability or consequences of an accident because a) the likelihood of a low temperature overpressure event occurring concurrently with a loss of the redundant instrument air system is sufficiently small, and b) the existing procedural controls will effectively prevent challenges to the COMS.

Additionally, delaying these surveillances for 12 hours will allow the operators to focus their attention on transitioning the plant to RHR cooling. Given the timing sequence of the RHR system entry point to the COMS enable temperature, the time extension is considered to be a prudent and safety focused change to the method of performing a plant cooldown. The proposed time extension is also consistent with the operational flexibility currently provided in NUREG-1431, Standard Technical Specifications for Westinghouse Plants.

Based on the above, it is concluded that the proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes do not create a new accident scenario. The requirements for the P/T limit curves and low temperature overpressure protection have been in place for some time. The fundamental approach follows approved ASME and Westinghouse topical report methodology. The proposed curves reflect the change in material properties acknowledged and managed by regulation and an upgrade in technology, which has been approved by ASME.

Delaying performance of two of the COMS surveillances (PORV Channel Operational Test and the backup nitrogen supply verification) until 12 hours after decreasing the RCS cold leg temperature to ≤ 275 °F during cooldown was also evaluated with respect to the plant accident analyses. The change was determined to not represent a significant increase in the probability or consequences of an accident because a) the likelihood of a low temperature overpressure event occurring concurrently with a loss of the redundant instrument air system is sufficiently small, and b) the existing procedural controls will effectively prevent challenges to the COMS.

Additionally, delaying these surveillances for 12 hours is consistent with the operational flexibility currently provided in NUREG-1431, Standard Technical Specifications for Westinghouse Plants.

Since no new failure modes are associated with the proposed changes, the activity does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The Technical Specifications for P/T limit curves and COMS setpoints are expiring and must be updated. The COMS setpoint is revised to incorporate additional margin in the instrument uncertainty. Conservative ASME code methods including safety factors have been used. The material properties used are from a much larger database than in past submittals. This results in many more datapoints available for the limiting weld metal than in past submittals. A new master curve of irradiated and unirradiated materials data has been developed for Turkey Point which shows that these curves and associated setpoints are conservative and represent an increase to the margin of safety. The new setpoint limit should reduce the possibility of an inadvertent PORV actuation. They should also reduce the potential for reactor coolant pump impeller cavitation or seal damage when the pumps are operated during low temperature conditions in the RCS. Changing the COMS surveillances to allow completion up to 12 hours after decreasing RCS temperature to $\leq$ 275 °F during cooldown does not result in a reduction in the margin of safety. Acceptability is based on: consistency with NUREG-1431, Standard Technical Specifications Westinghouse Plants, COT Surveillance Requirements; the inherent reliability and redundancy of the Turkey Point Instrument Air System; and the existing procedural controls established to prevent challenges to the LTOP System. The proposed amendments will not involve a significant reduction in the margin of safety.

Summary

Based on the above discussion, FPL has determined that the proposed amendment changes do not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore the proposed changes do not involve a significant safety hazards consideration as defined in 10 CFR 50.92.

Environmental Impact Evaluation

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not:

(i) involve a significant hazards consideration,

(ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(iii) result in a significant increase in individual or cumulative occupational radiation exposure.

FPL has reviewed this proposed license amendment and concludes that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with this request.

ATTACHMENT 3

Justification for ASME Code Case N-588 Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for use of ASME Section XI Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," in lieu of the 10 CFR 50, Appendix G.

Compliance with 10 CFR 50.12 Requirements:

The requested exemption to allow the use of ASME Code Case N-588 to determine stress intensity factors for postulated defects in circumferential welds meets the criteria of 10 CFR 50.12 as addressed below. 10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that:

1. The requested exemption is authorized by law. No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety. 10 CFR 50, Appendix G, requires, in part, that paragraph G-2120 of ASME Section XI, Appendix G, be used to determine the orientation of postulated defects in reactor vessels (RV) when determining pressure-temperature (P/T) limits for the vessel. Paragraph G-2120 defines the maximum postulated defect. The postulated defect is defined as a sharp surface defect, oriented normal (perpendicular to the plane of the material) in the direction of maximum stress, with a length of 1.5 times the section thickness and a depth 0.25 times the section thickness.

Turkey Point Units 3 and 4 reactor vessels were fabricated using ring forgings joined by circumferential girth welds. The section thickness is approximately 8 inches. There are no longitudinal welds in the Turkey Point vessels. Orienting the reference flaw in accordance with the present requirements of paragraph G-2120 causes the defect to terminate in the surrounding relatively ductile forging material. However, the analysis is performed using the most degraded material properties which are those of the girth weld. This has the effect of analyzing a longitudinal weld in a vessel which has no longitudinal welds. It is unlikely to have axial cracks originating from a circumferential weld perpendicular to the weld seam orientation in reactor vessels.

Due to progress made in nondestructive examination (NDE) techniques over the last thirty years, it is very unlikely to have large, undetected defects present in the beltline region of reactor vessels. Both experience and engineering studies indicate that the primary degradation mechanism affecting the beltline region of the reactor vessel is neutron embrittlement. No other service induced degradation mechanism exists in a pressurized water reactor to cause a prior existing defect located in the beltline region of the reactor vessel to grow while in service. Based on these considerations, and the fact that the pressure-temperature limit for reactor operation is the limiting pressure for any of the materials in the vessel, it is not necessary to include additional conservatism in the assumed flaw orientation for circumferential welds. ASME Section XI, Code Case N-588, and the accompanying ASME Appendix G Code change corrected this inconsistency in assumed flaw orientation for circumferential welds in vessels when calculating operating P/T limits.

Code Case N-588 provides benefits in terms of calculating P/T limits by revising the Section XI, Appendix G reference flaw orientation for circumferential welds in reactor vessels. The reference flaw is a postulated flaw that accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. Thus, the intended application of a reference flaw is to account for prior existing defects that could physically exist within the geometry of the weldment. The present ASME Section XI, Appendix G approach mandates the consideration of an axial reference flaw in circumferential welds for purposes of calculating P/T limits. Postulating the ASME Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the vessel thickness, which is much longer than the width of the reactor vessel girth weld. The possibility that an axial flaw may extend from a circumferential weld into a plate/forging or axial weld is already adequately covered by the requirement that axial defects be postulated in plates/forging and axial welds.

The fabrication of reactor vessels for nuclear plants involved precise welding procedures and controls designed to optimize the resulting weld microstructure and to provide the required material properties. These procedural controls were also designed to minimize defects that could be introduced into the weld during the fabrication process. Industry experience with the repair of weld indications found during pre-service inspection, and data taken from destructive examination of actual vessel welds, confirms that any remaining defects are small, laminar in nature, and do not cross transverse to the weld bead orientation. Therefore, any potential defects introduced during the fabrication process, and not detected during subsequent nondestructive examinations, would only be expected to be oriented in the direction of weld fabrication. For circumferential welds this indicates a postulated defect with a circumferential orientation.

ASME Code Case N-588 addresses this issue by allowing consideration of maximum postulated defects oriented circumferentially with circumferential welds. Code Case N-588 also provides appropriate procedures to determine limiting circumferential weld defects and associated stress intensity factors for use in developing reactor vessel P/T limits per ASME Section XI, Appendix G procedures. The procedures allowed by Code Case N-588 are conservative and provide a margin of safety in the development of reactor vessel pressure-temperature operating and pressure test limits which will prevent nonductile fractures.

The proposed P/T limits include restrictions on allowable operating conditions and equipment operability requirements to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, reactor coolant system (RCS) pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits. Therefore, this exemption does not present an undue risk to the public health and safety.

3. The requested exemption is consistent with the common defense and security. The common defense and security are not endangered by this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60. Pursuant to 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of paragraphs:

(a)(2)(ii) - Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

(a)(2)(iv) - The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption.

10 CFR 50.12(a)(2)(ii):

The underlying purpose of 10 CFR 50, Appendix G and ASME Section XI, Appendix G, is to satisfy the requirement that: (1) the reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure that when stressed the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and (2) P/T operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

Application of Code Case N-588 to determine the P/T operating and test limit curves per ASME Section XI, Appendix G, provides appropriate procedures to determine the limiting maximum postulated defects and to consider these defects in the P/T limits. The Turkey Point Units 3 and 4 reactor vessels only contain circumferential welds. Therefore, this application of the code case more adequately describes the actual conditions present while maintaining the safety factors originally contemplated for plates/forgings and axial welds.

The vessel will behave in a non-brittle manner under operating conditions and all irradiation effects are considered. The state of stress is better described by the use of Code Case N-588. Therefore use of Code Case N-588, as described above, satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety.

10 CFR 50.12(a)(2)(iv):

The RCS pressure-temperature operating window is defined by the P/T operating and test curves developed in accordance with the ASME Section XI, Appendix G procedure. Continued operation with these P/T curves without the relief provided by ASME Code Case N-588 would unnecessarily restrict the pressure-temperature operating window for Turkey Point Units 3 and 4. Existing Cold Overpressure Mitigating System (COMS) guidelines will reduce the potential for an undesired challenge to the reactor coolant system power operated relief valve (PORV) and supply additional margin between the reactor coolant pump seal limit and the COMS setpoint.

The present methodology provides a restrictive setpoint which constitutes an unnecessary burden that can be alleviated by the application of Code Case N-588 in the development of the proposed P/T curves. Implementation of the proposed P/T curves as allowed by ASME Code Case N-588 does not reduce the margin of safety originally contemplated by either the NRC or ASME. Code Case N-588 decreases the possibility of inadvertent PORV actuation or reactor coolant pump damage and thereby, is of benefit to the public safety.

Code Case N-588, Conclusion for Exemption Acceptability:

Compliance with the specified requirements of 10 CFR 50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-588 allows postulation of a circumferential defect in circumferential welds to be considered in lieu of requiring the defect to be oriented across the weld from one plate or forging to the adjoining plate or forging. This circumstance was not considered at the time ASME

Section XI, Appendix G was developed and imposes restrictions on P/T operating limits beyond those originally contemplated.

This proposed alternative is acceptable because the code case maintains the relative safety margin commensurate with that which existed at the time.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits specified in the Technical Specifications. Therefore, this exemption does not present an undue risk to the public health and safety.

Attachment 4

Justification for ASME Code Case N-641 Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for use of ASME Section XI Code Case N-641, "Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection (LTOP) System Requirements, Section XI, Division 1", as an alternative to method endorsed in 10 CFR 50, Appendix G.

Compliance with 10 CFR 50.12 Requirements:

The requested exemption to allow the use of ASME Code Case N-641 to determine the LTOP system enable temperature meets the criteria of 10 CFR 50.12 as addressed below. 10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that:

1. **The requested exemption is authorized by law.** No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the Commission under 10 CFR 50.12.

2. **The requested exemption does not present an undue risk to the public health and safety.** 10 CFR 50, Appendix G, requires, in part, that paragraph G-2215 of ASME Section XI, Appendix G, be used to determine the effective coolant temperature range of the LTOP system. G-2215 states that for plants that have LTOP systems, the system shall be effective at coolant temperatures less than 200 °F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{ndt} +50 F, whichever is greater. This temperature is based on an axially located flaw.

   Turkey Point Units 3 and 4 were fabricated using ring forgings joined by circumferential girth welds. The section thickness is approximately 8 inches. There are no longitudinal welds in the Turkey Point vessels. It is unlikely to have axial cracks originating from a circumferential weld perpendicular to the weld seam orientation in reactor vessels. Code Case N-588 allows the reference flaw in circumferential welds to be located circumferentially rather than axially.

   Because the currently defined value of T_{enable} is based on the stress intensity factor for an axially oriented flaw, the current definition of T_{enable} is inadequate in plants where Code Case N-588 is applied. Using the current code method (paragraph G-2215) would cause T_{enable} to approximate 350 °F which would correspond to initiation of residual heat removal. This is not desirable because operation of the units should provide the flexibility of preventing the Reactor Control Operators from performing two competing tasks in parallel.

Reducing the enable temperature would not present undue risk to the public, it would in fact reduce risk, as described below.

3. **The requested exemption will not endanger the common defense and security.** The common defense and security are not endangered by this exemption request.

4. **Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60.** Pursuant to 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of paragraphs:

   (a)(2)(ii) - Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule;

   (a)(2)(iv) - The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption.

10 CFR 50.12(a)(2)(ii):

The underlying purpose of 10 CFR 50, Appendix G and ASME Section XI, Appendix G, is to satisfy the requirement that: (1) the reactor coolant system pressure boundary be operated in a regime having sufficient margin to ensure that when stressed the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and (2) P-T operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

Application of Code Case N-641 to determine the LTOP T enable provides appropriate procedures to determine the limiting temperature below which, protection is required against overpressure conditions. Sufficient margin remains to assure the Turkey Point reactor vessels behave in a non-brittle manner.

The Turkey Point Units 3 and 4 reactor vessels only contain circumferential welds. Therefore, this application of the code case more adequately describes the actual conditions present while maintaining the safety factors originally contemplated for plates / forgings and axial welds.

Therefore, use of Code Case N-641, as described above, satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety.

10 CFR 50.12(a)(2)(iv):

The reactor coolant system pressure-temperature operating window is defined by the P-T operating and test curves developed in accordance with the ASME Section XI, Appendix G procedures. Operation with these P-T curves without the relief provided by ASME Code Case N-641 would unnecessarily restrict the pressure-temperature operating window for Turkey Point Units 3 and 4. The proposed (LTOP) guidelines will increase the operating window by lowering the temperature regime in which LTOP is operable. It will reduce the potential for an undesired challenge to the reactor coolant system by supplying additional margin between the reactor coolant seal limit and the COMS setpoint.

The current methodology provides a restrictive enable temperature which constitutes an unnecessary burden that can be alleviated by the application of Code Case N-641. Implementation of the proposed enable temperature as allowed by ASME Code Case N-641 does not reduce the margin of safety originally contemplated by either the NRC or ASME. Code Case N-641 would allow Turkey Point to leave the current enable temperature unchanged reducing the need for operator retraining in this area.

Code Case N-641, Conclusion for Exemption Acceptability:

Compliance with the specified requirements of 10 CFR 50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-641 presents a benefit in safety to the public in that initiation of RHR and enabling of LTOP do not coincide. Code Case N-641 allows determination of an enable temperature for LTOP which considers postulation of a circumferential defect in circumferential welds to be considered in lieu of requiring the defect to be oriented across the weld from one plate or forging to the adjoining plate or forging. This is consistent with Code Case N-588.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits specified in the Technical Specifications. Therefore, this exemption does not present an undue risk to the public health and safety.

ATTACHMENT 5

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 ~~and 3.4-4~~ during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 5°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3 ~~and 3.4-4~~.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
INITIAL RT_{NDT}: 10°F

SERVICE PERIOD: 19 EFPY

HEATUP RATES: UP TO 60°F/HR

RT_{NDT} ● 1/4 THICKNESS = 252.5°F

RT_{NDT} ● 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.

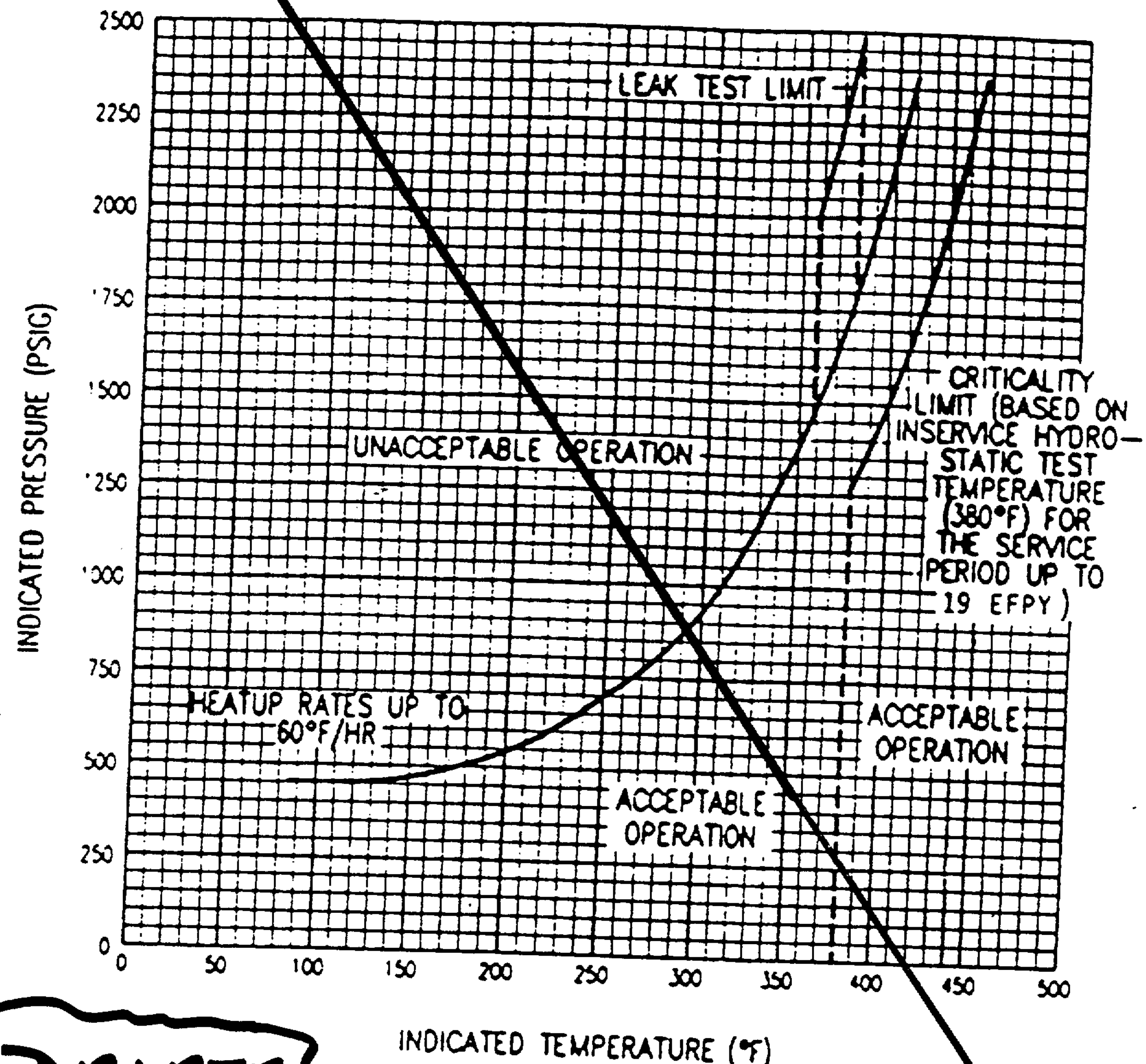

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FIGURE 3.4-2

TURKEY POINT UNITS 3 & 4

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (60°F/hr) - APPLICABLE UP TO 19 EFPY

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate/Lower Shell Circumferential Weld Seams (Ht. # 71249)

LIMITING ART VALUES AT 32 EFPY: 1/4T, 262°F

3/4T, 218°F

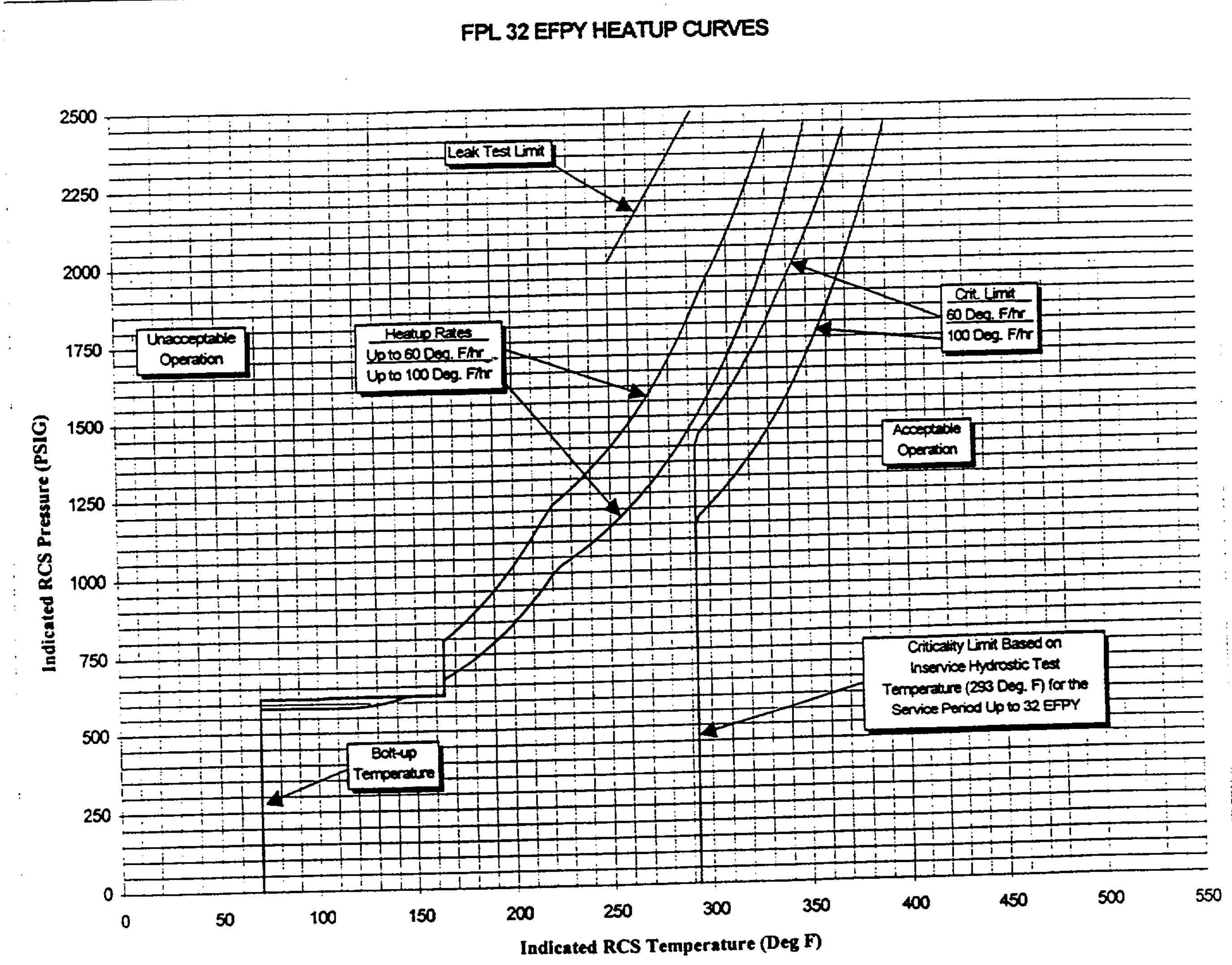

FIGURE 3.4-2 Turkey Point Units 3 and 4 Reactor Coolant System Heatup Limitations (Heatup Rate of 60 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
INITIAL RT_{NDT}: 10°F

SERVICE PERIOD: 19 EFPY

HEATUP RATES: UP TO 100°F/HR

RT_{NDT} @ 1/4 THICKNESS = 252.5°F

RT_{NDT} @ 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.

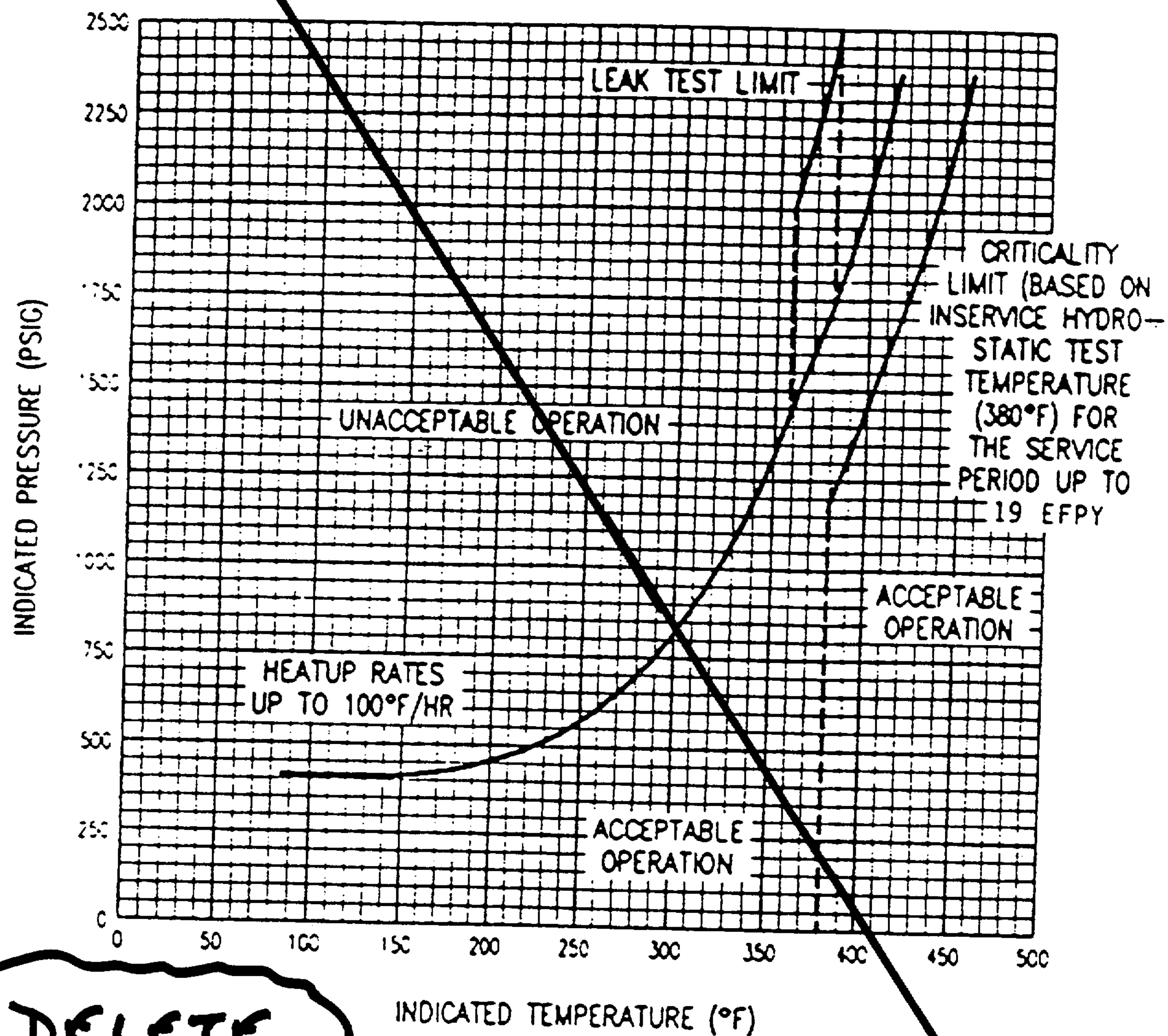

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FIGURE 3.4-3

TURKEY POINT UNITS 3 & 4

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (100°F/hr) - APPLICABLE UP TO 19 EFPY

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate/Lower Shell Circumferential Weld Seams (Ht. # 71249)

LIMITING ART VALUES AT 32 EFPY: 1/4T, 262°F

3/4T, 218°F

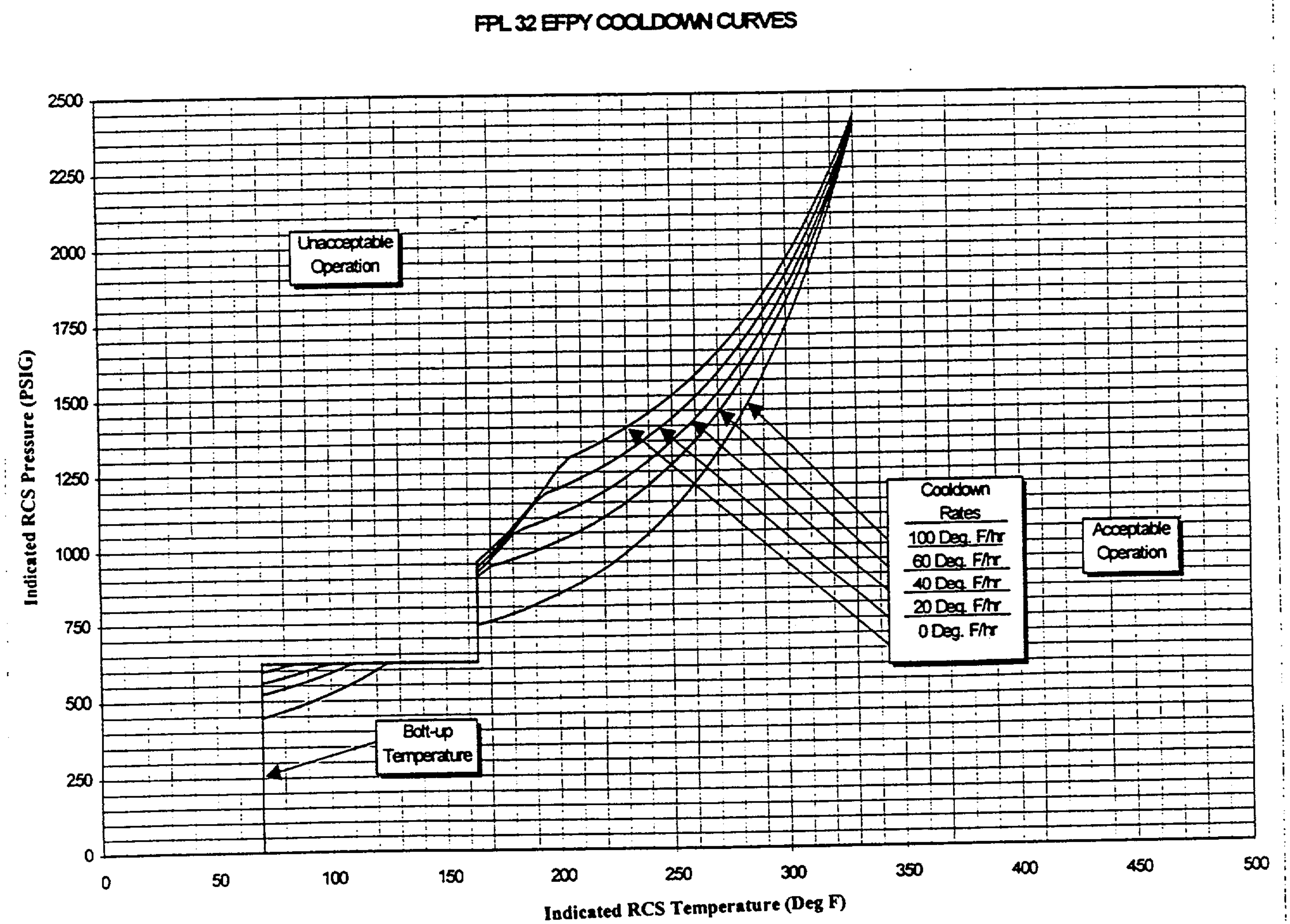


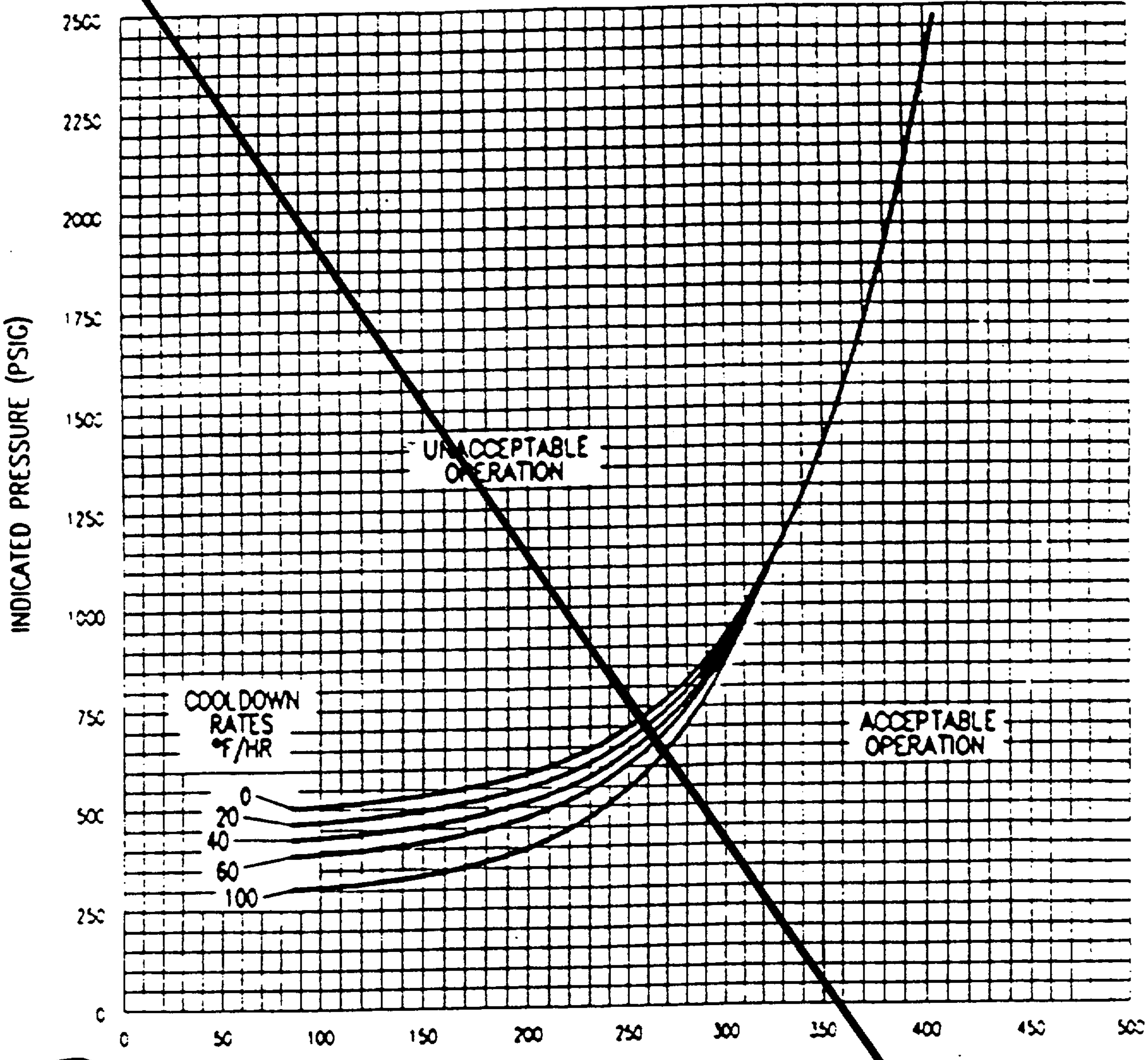
FIGURE 3.4-3 Turkey Point Units 3 and 4 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
INITIAL RT_{NDT}: 10°F
SERVICE PERIOD: 19 EFPY
COOLDOWN RATES: UP TO 100°F/HR

RT_{NDT} @ 1/4 THICKNESS = 252.5°F
RT_{NDT} @ 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.



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INDICATED TEMPERATURE (°F)

FIGURE 3.4-4

TURKEY POINT UNITS 3 & 4

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS (100°F/hr) - APPLICABLE UP TO 19 EFPY

OVERPRESSURE MITIGATING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The high pressure safety injection flow paths to the Reactor Coolant System (RCS) shall be isolated, and below an RCS average coolant temperature of 275°F at least one of the following Overpressure Mitigating Systems shall be OPERABLE:

a) Two power-operated relief valves (PORVs) with a lift setting of ≤ 468 ~~415 ± 15~~ psig, or

b) The RCS depressurized with a RCS vent of greater than or equal to 2.20 square inches.

APPLICABILITY: MODES 4 (below an RCS average coolant temperature of 275°F), 5, and 6 with the reactor vessel head on.

ACTION:

a. With the high pressure safety injection flow paths to the RCS unisolated, restore isolation of these flow paths within 4 hours.

b. With one PORV inoperable in MODE 4 (below an RCS average coolant temperature of 275°F), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.

c. With one PORV inoperable in MODES 5 or 6 with the reactor vessel head on, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.20 square inch vent within a total of 32 hours, or (3) complete depressurization and venting of the RCS through at least one open PORV and associated block valve within a total of 32 hours.

d. With both PORVs inoperable, either restore one PORV to OPERABLE status or complete depressurization and venting of the RCS through at least a 2.20 square inch vent within 24 hours.

e. In the event either the PORVs or a 2.20 square inch vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence. A Special Report is not required when such a transient is the result of water injection into the RCS for test purposes with an open vent path.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST * on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV block valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

d. While the PORVs are required to be OPERABLE, the backup ~~air~~ nitrogen supply shall be verified OPERABLE at least once per 24 hours.*

4.4.9.3.2 The 2.20 square inch vent shall be verified to be open at least once per 12 hours** when the vent(s) is being used for overpressure protection.

4.4.9.3.3 Verify the high pressure injection flow path to the RCS is isolated at least once per 24 hours by closed valves with power removed or by locked closed manual valves.

* Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ 275 °F.

** Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

ATTACHMENT 6

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SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the RCS are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are induced by normal load transients, reactor trips and startup and shutdown operations. During RCS heatup and cooldown, the temperature and pressure changes must be limited to be consistent with design assumptions and to satisfy stress limits for brittle fracture.

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron irradiation damage is also greatest at the inside surface location, the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

XI

PRESSURE/TEMPERATURE LIMITS (Continued)

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 ~~to 3.4-4~~ for the service period specified thereon: and 3.4-3

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   and 3.4-3

   b. Figures 3.4-2 ~~to 3.4-4~~ define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, the version of the ASTM E185 standard required by 10 CFR 50, Appendix H, and in accordance with additional reactor vessel requirements.

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The properties are then evaluated in accordance with Appendix G of the ~~1983~~ Edition of Section ~~III~~ of the ASME Boiler and Pressure Vessel Code and the additional requirements of 10 CFR 50, Appendix G and the calculation methods described in Westinghouse Report ~~GTSD-A-1.12, "Procedure for Developing Heatup and Cooldown Curves~~." INSERT (A)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of ~~19~~ effective full power years (EFPY) of service life. The ~~19~~ EFPY service life period is chosen such that the limiting RT_{NDT}, at the 1/4T location in

32

<u>INSERT (A)</u>

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown limit Curves."

PRESSURE/TEMPERATURE LIMITS (Continued)

the core region is greater than the RT_{NDT}, of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3 ~~and 3.4-4~~ are composite curves prepared by determining the most conservative case with either the inside or outside wall controlling, for any heatup rate up to 100 degrees F per hour and cooldown rates of up to 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of predicted adjusted reference temperature at the end of the applicable service period (~~19~~ 32 EFPY).

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Tables B 3/4.4-1 and B 3/4.4-2. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and chemistry factors of the material has been predicted using Regulatory Guide 1.99, Revision 2, dated May 1988, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3, ~~and 3.4-4~~ include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period.

The actual shifts in RT_{NDT}, of the vessel materials will be established periodically during operation by removing and evaluating, in accordance with the version of the ASTM E185 standard required by 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel.

Since the limiting beltline materials (Intermediate to Lower Shell Circumferential Weld) in Units 3 and 4 are identical, the RV surveillance program was integrated and the results from capsule testing is applied to both Units. ~~The surveillance capsule "T" results from Unit 3 (WCAP-8631) and Unit 4 (SWRI 02-4221) and the capsule "V" results from Unit 3 (SWRI 06-8576~~ were used with the methodology in Regulatory Guide 1.99, Revision 2, to provide

(INSERT)
All available surveillance capsule results for the Unit 3 and 4 reactor vessel

PRESSURE/TEMPERATURE LIMITS (Continued)

limiting material properties information for generating the heatup and cooldown curves in Figures 3.4-2, and 3.4-3 ~~and 3.4-4~~. The integrated surveillance program along with similar identical reactor vessel design and operating characteristics allows the same heatup and cooldown limit curves to be applicable at both Unit 3 and Unit 4.

XI Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 ~~and Westinghouse Report GTSB-A-1.12, "Procedure for Developing Heatup and Cooldown Curves~~."

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. XI Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{IT} \leq K_{IR} \qquad (2)$$

PRESSURE/TEMPERATURE LIMITS (Continued)

Finally, the 10 CFR 50 Appendix G rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. The rule states that the minimum metal temperature for the flange regions should be at least 120 F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig). Since the limiting RT_{NDT} for the flange regions for Turkey Point Units 3 and 4 is 44 F, the minimum temperature required for pressure of 621 psig and greater based on the Appendix G rule is 164 F. The heatup and cooldown curves as shown in Figures 3.4-2 ~~to 3.4-4~~ and 3.4-3 clearly satisfy the above requirement by ample margins.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

OVERPRESSURE MITIGATING SYSTEM

The Technical Specifications provide requirements to isolate High Pressure Safety Injection from the RCS and to prevent the start of an idle RCP if secondary temperature is more than 50°F above the RCS cold leg temperatures. These requirements are designed to ensure that mass and heat input transients more severe than those assumed in the low temperature overpressurization protection analysis cannot occur.

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.20 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 275°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures including margin for instrument error, or (2) the start of a HPSI pump and its injection into a water-solid RCS. When the PORVs or 2.2 square inch area vent is used to mitigate a plant transient, a Special Report is submitted. However, minor increases in pressure resulting from planned plant actions, which are relieved by designated openings in the system, need not be reported. INSERT (B) - here

REACTOR MATERIAL SURVEILLANCE PROGRAM

Each Type I capsule contains 28 V-notch specimens, ten Charpy specimens machined from each of the two shell forgings. The remaining eight Charpy specimens are machined from correlated monitor material. In addition, each

<u>**INSERT (B)**</u>

The Overpressure Mitigation System setpoint includes an allowance for instrument uncertainty.

L-2000-146

ENCLOSURE 1

WCAP-15092 Revision 3,

Turkey Point Units 3 and 4

WOG Reactor Vessel 60-Year Evaluation

Minigroup Heatup and Cooldown

Limit Curves for Normal Operation,

Westinghouse Electric Company LLC,

May 2000.