FPL

APR 2 7 2000

L-2000-028
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
Proposed License Amendments
"Revised Pressure/Temperature (P/T) Curves, and
Cold Overpressure Mitigation System (COMS) Setpoints"

In accordance with 10 CFR §50.90, Florida Power and Light Company (FPL) requests 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). The present pressure/temperature (P/T) limits specified in Technical Specification (TS) 3/4.4.9, and in TS 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 TS 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 ± 15 psig to ≤ 561 psig, which includes instrument uncertainty of 70 psig, as a result of the P/T limit changes. The enable temperature for the overpressure mitigation system changes from 275°F to 340°F. Accordingly, footnotes of TS 3.4.1.3, and 3.4.1.4.1 for RCS Hot Shutdown and Cold Shutdown, respectively, are being revised to change the RCS average coolant temperature limit from 275 °F to 340 °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 of 340 °F.

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. Therefore, this letter includes three 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;"

(ii) ASME Section XI Code Case N-640, "Alternative Fracture Toughness for Development of P/T Limit Curves for ASME Section XI, Division I;" and

(iii) WCAP-15315 "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants."

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 consideration determination in support of the proposed Technical Specifications changes is provided in Attachment 2. Attachments 3, 4, and 5 provide the exemption requests listed above. Attachment 6 provides the proposed revised Technical Specifications pages. Attachment 7 contains the proposed revised Technical Specifications Bases pages for information only.

Additionally, three enclosures are being provided to support the proposed license amendments. Enclosure 1 provides Westinghouse report WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." Enclosure 2 provides Westinghouse report WCAP-15092, Revision 2, "Turkey Point Units 3 and 4 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation." Enclosure 3 contains the Westinghouse report on Low Temperature Overpressure Protection System Setpoints, 32 and 48 Effective Full Power Years for Turkey Point Units 3 and 4.

Enclosure 3 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 3 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, Bureau of Radiation Control

L-2000-028

Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
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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.

R. J. Hovey

Subscribed and sworn to before me this
27th day of April, 2000.

Cheryl A. Stevenson
Name of Notary Public (Type or Print)

CHERYL A. STEVENSON
MY COMMISSION # CC 564017
EXPIRES: June 19, 2000
Bonded Thru Notary Public Underwriters

R. J. Hovey is personally known to me.

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ENCLOSURES

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 (TS) 3/4.4.9, and in TS 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 TS 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 $\leq$ 561 psig, which includes instrument uncertainty of 70 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 changes from 275^{o}F to 340^{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 COMS 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;

(b) ASME Code Case N-640 (Reference 2), which uses the K_{Ic} curve rather than the K_{Ia} curve to index RT_{ndt} ; and

(c) WCAP-15315 (Reference 3), which is the technical basis for exemption from the 10 CFR 50 Appendix G requirement for the metal temperature of the closure head and flange.

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 curves and COMS 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 mid-plane 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 4). 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 5) and therefore, the Technical Specification P/T limit curves and COMS setpoints will require revision.

The methodology for the present P/T limit curves and COMS setpoints was developed by Westinghouse. Westinghouse was contracted to develop the new curves and COMS setpoints. The analyses performed comply with the methods of the NRC approved topical report WCAP 14040-NP-A, (Reference 6) with the exception of the Code Cases and use of WCAP-15315 methodology. The ASME Code Cases N-588 & N-640, and WCAP-15315 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 7). 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 8). The results from this capsule were analyzed using the credibility criteria of Regulatory Guide 1.99, Revision 2 and were within the expected uncertainty limits. The Westinghouse topical reports on the Pressure/Temperature curves, WCAP-15092 Rev. 2 (Reference 10) and the letter report on the P/T calculation (Reference 11) are provided as Enclosures 2 and 3, respectively.

Three 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-640 and involves the use of the K_{Ic} curve rather than the K_{Ia} curve for indexing RT_{ndt}. The third exemption request uses WCAP-15315 to justify the removal of the 10 CFR 50, Appendix G requirement that the metal temperature of the flange regions must exceed the material unirradiated RT_{ndt} by at least 120 °F for normal operation, when the RCS pressure exceeds 20% of the preservice hydrostatic test pressure. 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.

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 3.4.1.3, Reactor Coolant System Hot Shutdown: Revise the RCS average coolant temperature limit for footnote ** from 275 °F to 340 °F.**

Discussion: The enable temperature has increased to 340 °F. This change is caused by the increase in RT_{ndt} of the circumferential weld. This value is calculated to correspond to the one-quarter thickness (1/4T) Adjusted Reference Temperature at 32 EFPY plus 50 °F. This calculation including the addition of the 50 °F is consistent with the ASME Section XI methodology.

c) **Technical Specification 3.4.1.4.1, Reactor Coolant System Cold Shutdown: Revise the RCS average coolant temperature limit for footnote *** from 275 °F to 340 °F.**

Discussion: The enable temperature has increased to 340 °F. This change is caused by the increase in RT_{ndt} of the circumferential weld. This value is calculated to correspond to the 1/4T Adjusted Reference Temperature at 32 EFPY plus 50 °F. This calculation including the addition of the 50 °F is consistent with the ASME Section XI methodology.

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

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

e) **Technical Specification Figures 3.4-2, 3.4-3, and 3.4-4: Replace these three current figures with two new Figures 3.4-2, and 3.4-3, 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 °F/hour and 100 °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 COMS 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 °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.

f) **Technical Specification 3.4.9.3, Overpressure Mitigation System: Revise the RCS average coolant temperature limit from 275 °F to 340 °F, and the power-operated relief valve lift setting from 415 ± 15 psig to ≤ 561 psig.**

Discussion: The enable temperature has increased to 340 °F. This change is caused by the increase in RT_{ndt} of the circumferential weld. This value is calculated to correspond to the 1/4T Adjusted Reference Temperature at 32 EFPY plus 50 °F. This calculation including the addition of the 50 °F is consistent with the ASME Section XI methodology.

The proposed PORV lift setpoint for operation of the COMS is ≤ 561 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 Cases N-588 and N-640, and by the elimination of the 10 CFR 50, Appendix G requirement that the metal temperature of the flange regions must exceed the material unirradiated RT_{ndt} by at least 120 °F for normal operation when the RCS pressure exceeds 20% of the preservice hydrostatic test pressure.

The proposed changes reflect the addition of the new curves themselves, the change in enable temperature for COMS, and the new PORV setpoint limitation. These values are based on the analyses in Enclosures 2 and 3.

g) **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 340 °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 air supply is based on the close proximity of the proposed 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 only 10 °F higher than the proposed 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 by the proposed amendments 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. The twelve hour time frame considers the unlikelihood of a low temperature overpressure event occurring concurrently with the loss of the 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 service air system from Turkey Point (Fossil) Units 1 and 2.

4.0 BASIS FOR PROPOSED CHANGES

The pressure/temperature limit curves are constructed prior to the calculation of the COMS setpoint. The following paragraphs discuss the data and methods used to develop both the P/T limit curves and COMS setpoint, which are based upon WCAP-15092, Revision 2.

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 of all beltline materials was calculated in accordance with RG 1.99. The controlling values are those with the highest Adjusted Reference Temperature in the 1/4 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. 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 Cases N-588 and N-640. Code Case N-588 uses circumferential flaw solutions for circumferential welds. Code Case N-640 uses the K_{Ic} from ASME Section XI, Appendix A which is based on the lower bound static K_I values, rather than the K_{Ia} which uses a lower bound of static and dynamic K_I. The flange metal temperature limitation of 10 CFR 50, Appendix G was eliminated by using the methodology of WCAP-15315. 10 CFR 50, Appendix G, Subsection IV.A.2.c states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{ndt} by at least 120°F for normal operation when the operating pressure exceeds 20 percent of the pre-service hydrostatic test pressure.

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:

$$K_{\text{applied}} < K_{\text{material}}$$

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

where C = 2 for service levels 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_{Ic} 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 (RCS) temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T 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 RCS temperature, the change in temperature developed during cooldown results in a higher value of K_{Ic} 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_{Ic} 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_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} 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_{Ic} values do not offset each other. Under these conditions, the pressure/temperature 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 P/T limit curves for both the steady state and finite heatup rate conditions, the final P/T 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 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 the proposed value of 340°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 (Reference 6), which includes instrument uncertainty of 70 psig. The setpoint range was calculated to be ≤ 561 psig.

A footnote is added to Technical Specification Sections 4.4.9.3.1a and 4.4.9.3.1d regarding COMS 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 ≤ 340 °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 close proximity of the proposed COMS enable temperature to the RHR entry temperature. The RHR entry temperature for Turkey Point Units 3 and 4 is 350 °F, which is only 10 °F higher than the proposed 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.

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-640 and WCAP-15315. Code Cases N-588 and N-640 have been approved by ASME Section XI. WCAP-15315, a Westinghouse topical report on removal of the flange metal temperature requirement imposed by 10 CFR 50, Appendix G, has been reviewed by all owners groups. This WCAP is applicable to both BWRs and PWRs. 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 proposed COMS setpoint limitation was determined in accordance with the NRC approved topical report WCAP-14040-NP-A (Reference 6).

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-640, Alternative Reference Fracture Toughness for Development of P/T limit curves Section XI, Division 1.

3. WCAP-15315, Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants, Westinghouse Electric Company LLC, October 1999.

4. FPL letter to the NRC, L-98-155, Response to Request for Additional Information Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," dated July 13, 1998.

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

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

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

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

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

10. WCAP -15092 Revision 2, 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, February 2000.

11. Low Temperature Overpressure Protection System Setpoints, 32 and 48 Effective Full Power Years for Turkey Point Units 3 and 4, Westinghouse Electric Company LLC, January 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 (RCS) pressure boundary (i.e., no change in normal 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 RCS 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-640 have ASME approval and WCAP-15315 is the basis for an industry wide petition for rule making applicable to all PWRs and BWRs.

Delaying performance of two of the COMS surveillances (Power Operated Relief Valve (PORV) Channel Operational Test and the backup nitrogen supply verification) until 12 hours after decreasing the RCS cold leg temperature to ≤ 340 °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 close proximity of the RHR system entry point to the new 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 ≤ 340 °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 ≤ 340 °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 COMS 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 these proposed license amendments and concludes that they meet 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 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 power plant operation 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 the following 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 Exemption Acceptability for:

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-640 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-640, "Alternative Fracture Toughness for Development of Pressure/Temperature (P/T) Limit Curves for ASME Section XI, Division I," in lieu of 10 CFR 50, Appendix G.

Compliance with 10 CFR 50.12 Requirements:

The requested exemption to allow use of ASME Code Case N-640 in conjunction with ASME Section XI, Appendix G to determine the pressure/temperature limits meets the criteria of 10 CFR 50.12 as discussed 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.** The revised P/T limits being proposed for Turkey Point Units 3 and 4, rely in part, on the requested exemption. These revised P/T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME Section XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME Section XI, Appendix G process of determining P/T limit curves remain unchanged with the exception of the flange requirements (limitation imposed by the requirement to consider the reactor vessel flange as the limiting structure). The exemption request for elimination of the flange requirements is provided in Attachment 5. Elimination of the flange requirements complements the use of Code Case N-640.

In determining the lower bound fracture toughness, in the development of P/T operating limits curve, use of the K_{Ic} curve is more technically correct than the use of K_{Ia} curve. The K_{Ic} curve is based on static test results which are much more similar to the slow heat-up and cooldown rates associated with a reactor vessel.

Use of this approach is justified by the initial conservatism of the K_{Ia} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor vessel materials property changes with time and irradiation. Since 1974, knowledge has been gained about the effect of power operations on reactor vessel materials. The additional knowledge demonstrates the lower bound on fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect the public health and safety from potential reactor vessel failure.

The new master curve technology as described in ASTM Standard E-1921, NUREG/CR 5504, and EPRI PWR-MRP-01, Rev. 1 has demonstrated the conservatism in the present K_{Ia} approach.

P/T limit curves based on the K_{Ic} curve will enhance overall plant safety by opening the pressure/temperature operating window with a greater safety benefit in the region of low temperature operations. The two primary safety benefits in increasing the low temperature operating window is a reduction in the challenges to Reactor Coolant System (RCS) Power Operated Relief Valves (PORV) and reduction in the risk of Reactor Coolant Pump impeller cavitation.

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 the following 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)

ASME Section XI, Appendix G, provides procedures for determining allowable loading on the reactor vessel and is approved for that purpose by 10 CFR 50, Appendix G. Application of these procedures in the determination of P/T operating and test curves satisfied the underlying purpose for the following:

1. The RCS pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating

fracture is minimized;

2. P/T operating and test limit curves provide adequate margin in consideration of uncertainties in determining the effects of irradiation on material properties.

The ASME Section XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning reactor vessel materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. This increased knowledge permits a change to the ASME Section XI, Appendix G, requirements via application of ASME Code Case N-640. Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P/T operating limits curve is more technically correct than use of the K_{Ia} curve. The K_{Ic} curve is based on static test results which are much more similar to the slow heat-up and cooldown rates associated with a reactor vessel.

Use of the K_{Ic} maintains the underlying purpose of the ASME Code and the NRC regulations to ensure and acceptable margin of safety. The K_{Ic} is a lower bound curve and it therefore accounts for material uncertainties.

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

The Reactor Coolant System pressure/temperature operating window is defined by the P/T operating and test limit curves developed in accordance with the ASME Section XI, Appendix G procedure. Continued operation of Turkey Point with these P/T limit curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the pressure/temperature operating window. The operating window defines the space between the Cold Overpressure Mitigating System (COMS) setpoint and the minimum pressure for reactor coolant pump operation. The more restricted this space is, the greater the potential for inadvertent PORV actuation or reactor coolant pump damage.

This constitutes an unnecessary burden that can be alleviated by the application of Code Case N-640 in the development of the proposed P/T curves. Implementation of the proposed P/T curves as allowed by ASME Code Case

Code Case N-640 does not reduce the margin of safety. In fact the probability of an inadvertent PORV actuation and probability of reactor coolant pump damage are both reduced. Inadvertent PORV actuation causes a small break LOCA which if the valve fails to reseat can initiate an accident. Reducing this probability is a benefit to public health and safety.

Code Case N-640, Conclusion for Exemption Acceptability:

Compliance with the specified requirements of 10 CFR 50.60 would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-640 allows a reduction in the fracture toughness lower bound used by ASME Section XI Appendix G, in the determination of reactor coolant system pressure/temperature limits. This proposed alternative is acceptable because the Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME Section XI, Appendix G, was approved in 1974. Therefore, application of Code Case N-640 for Turkey Point will ensure an acceptable safety margin. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure 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 Technical Specification 3.4.9.1. Therefore, this exemption does not present an undue risk to the public health and safety.

ATTACHMENT 5

Justification for Reactor Head and Vessel Flange Requirements Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for use of WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," in lieu of 10 CFR 50, Appendix G. The benefit to be gained by use of this exemption is to increase the operating window between the Cold Overpressure Mitigating System (COMS) setpoint (which is based on the Pressure/Temperature (P/T) limit curves), and the minimum pressure requirement for reactor coolant pump operation. Increasing the operating window decreases the possibility of reactor coolant pump damage and inadvertent Power Operated Relief Valve (PORV) actuation.

Compliance with 10 CFR 50.12 Requirements:

The requested exemption to allow use of WCAP-15315 "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," in conjunction with ASME Section XI, Appendix G to determine the pressure/temperature limits meets the criteria of 10 CFR 50.12 as discussed 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.** The revised P/T limits being proposed rely in part on Code Case N-640 which allows the use of the K_{Ic} curve rather than the K_{Ia} curve. The flange requirement is based on the K_{Ia} technology and negates the safety benefit obtained by using K_{Ic} because it restricts the ability to raise the PORV setpoint as the curve rises by pinning the setpoint to the flange rather than the P/T curve. Thus, no matter how good the fracture toughness of the vessel, the P/T limit curve may be superceded by the flange requirement at temperatures below RT_{ndt} + 120^{0} F.

3. **The requested exemption is consistent 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 the following 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 the regulation will continue to be achieved. 10 CFR Part 50, Appendix G contains requirements for pressure/temperature limits for the primary system, and requirements for the metal temperature of the closure head flange and vessel flange regions. The pressure/temperature limits are to be determined using the methodology of ASME Section XI, Appendix G, but the flange temperature requirements are specified in 10 CFR 50, Appendix G. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{ndt} by at least 120°F for normal operation when the operating pressure exceeds 20 percent of the pre-service hydro static test pressure, which is 20% of the pre service hydro static test pressure of 3107 psig, or 621 psig for Turkey Point.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the bolt-up process, outside surface stresses in this region typically reach over 70 percent of the steady state stress, without being at steady state temperature. The margin of 120°F and the pressure limitation of 20 percent of hydro static test pressure were developed in the mid-1970s using the K_{Ia} fracture toughness, to ensure that appropriate margins would be maintained.

Improved knowledge of fracture toughness has led to the recent change to allow the use of K_{Ic} in the development of pressure/temperature curves, as contained in ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1."

The heatup curve using K_{Ic} provides for a higher allowable pressure through the entire range of operating temperatures. For Turkey Point, however, the benefit is negated at temperatures below RT_{ndt} + 120°F because of the flange limitations of 10 CFR Part 50, Appendix G. The flange limitations of 10 CFR 50 were originally developed using the K_{Ia} fracture toughness.

WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements for Operating PWR and BWR Plants," presents an analysis which shows that use of the newly code-accepted K_{Ic} fracture toughness for flange considerations leads to the conclusion that the flange requirement can be eliminated.

Using the K_{Ic} toughness, which has now been adopted by ASME Section XI for P/T limit curves, there is significant margin between the applied stress intensity factor and the fracture toughness at virtually all crack depths. Another objective of the requirements in 10 CFR 50, Appendix G is to assure that fracture margins are maintained to protect against service induced cracking due to environmental effects. Since the governing flaw is on the outside surface (the inside surface is in compression) where there are no environmental effects, there is even greater assurance of fracture margin. Therefore it may be concluded that the integrity of the closure head/flange region is not a concern for any of the operating plants using the K_{Ic} toughness.

Furthermore, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage for this region is less than 0.1, so it may be concluded that flaws are unlikely to initiate in this region.

Therefore, FPL concludes that the underlying purpose of the regulation will continue to be achieved.

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

Currently Turkey Point operates with a COMS setpoint of 415 psig. This was calculated without the use of instrument uncertainty and using the benefit of the 110% of the P/T curve allowable pressure value of Code Case N-514. The minimum pressure required for reactor coolant pump seals is 325 psig. This results in a narrow operating window.

The proposed COMS setpoint limit includes an instrument uncertainty of 70 psig and uses a 100% pressure requirement required by Code Case N-640. Both of these factors, combined with an enable temperature increase from 275 °F to 340 °F, make elimination of the flange limitation desirable.

Application of this methodology will change the pressure difference between the PORV setpoint and the reactor coolant pump seal limit from the present 90 psig to a proposed maximum difference of 236 psig. This change will make a significant improvement in plant safety by reducing the probability of small break LOCA (by challenging the PORVs), and easing the burden on the plant operators.

Conclusion for Exemption Acceptability

Compliance with the 10 CFR 50, Appendix G flange limit would be incongruous with the use of Code Case N-640. Code Case N-640 uses K_{Ic} methodology and the present flange limit uses K_{Ia} methodology. WCAP-15315 provides a valid basis for changing the flange limit. There is positive benefit to the public by reducing inadvertent challenges to the PORVs and increasing the operating margin for reactor coolant pumps.

<u>ATTACHMENT 6</u>

<u>Proposed Technical Specification Pages</u>

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. RHR Loop A, and

e. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

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*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to ~~275°F~~ 340°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The secondary side water level of at least two steam generators shall be greater than 10%.

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable or with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE.

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to ~~275°F~~ 340°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

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.~~

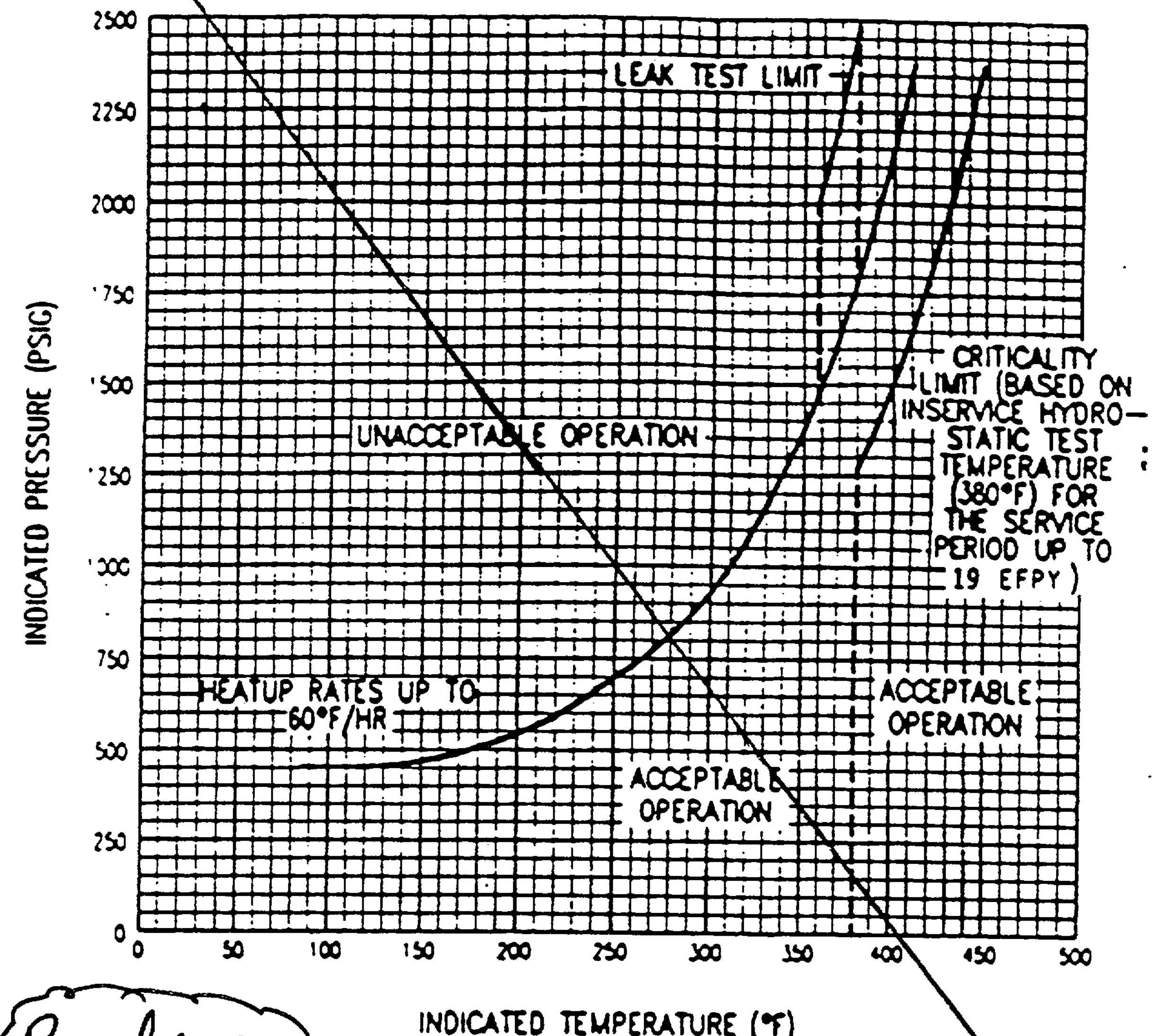
MATERIAL PROPERTY BASIS

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

SERVICE PERIOD: 19 EFPY — RT_{NDT} • 1/4 THICKNESS = 252.5°F

HEATUP RATES: UP TO 60°F/HR — RT_{NDT} • 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.



Replace

FIGURE 3.4-2

TURKEY POINT UNITS 3 & 4

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

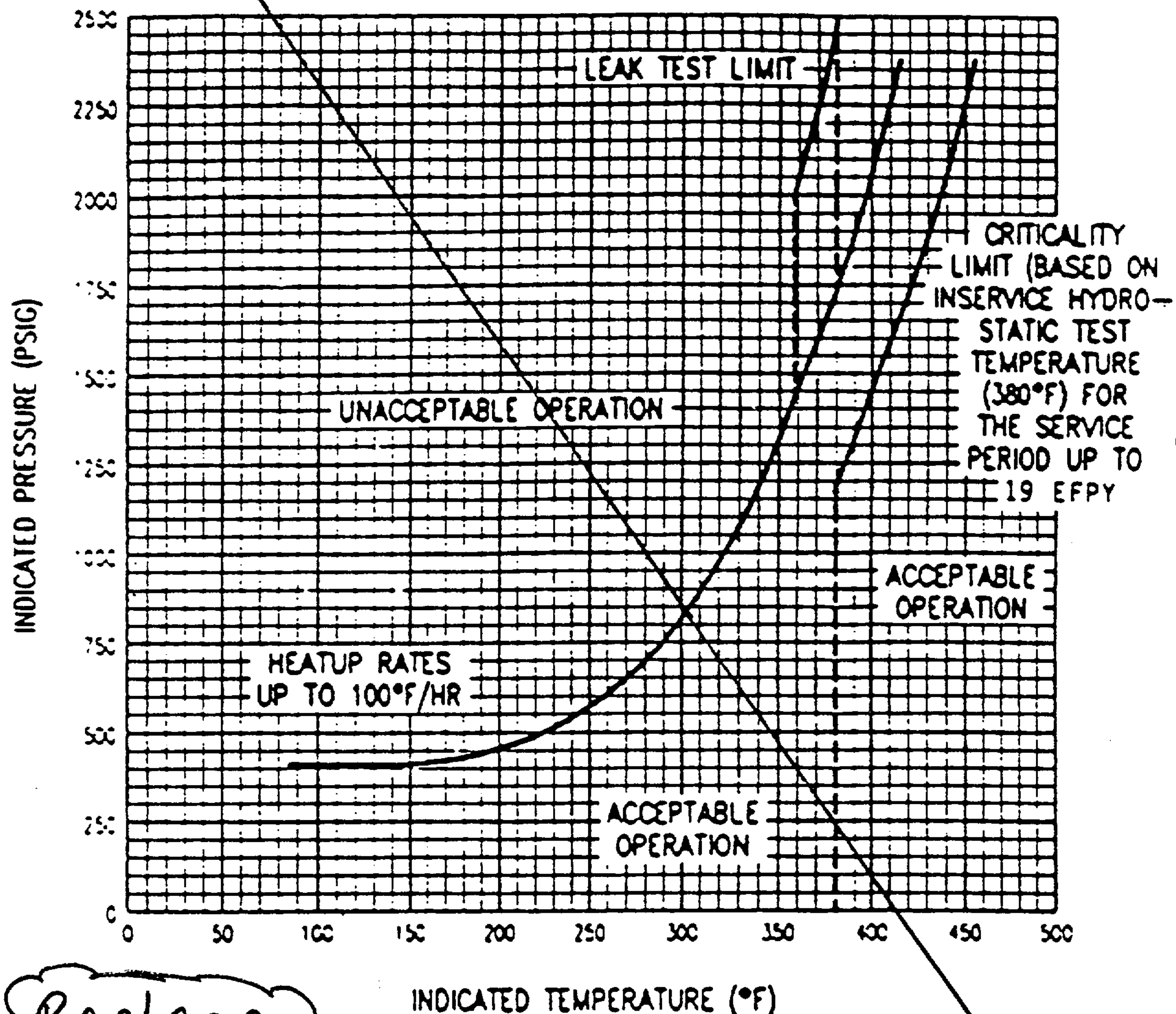
MATERIAL PROPERTY BASIS

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

SERVICE PERIOD: 19 EFPY — RT_{NDT} @ 1/4 THICKNESS = 252.5°F

HEATUP RATES: UP TO 100°F/HR — RT_{NDT} @ 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.



Replace

FIGURE 3.4-3

TURKEY POINT UNITS 3 & 4

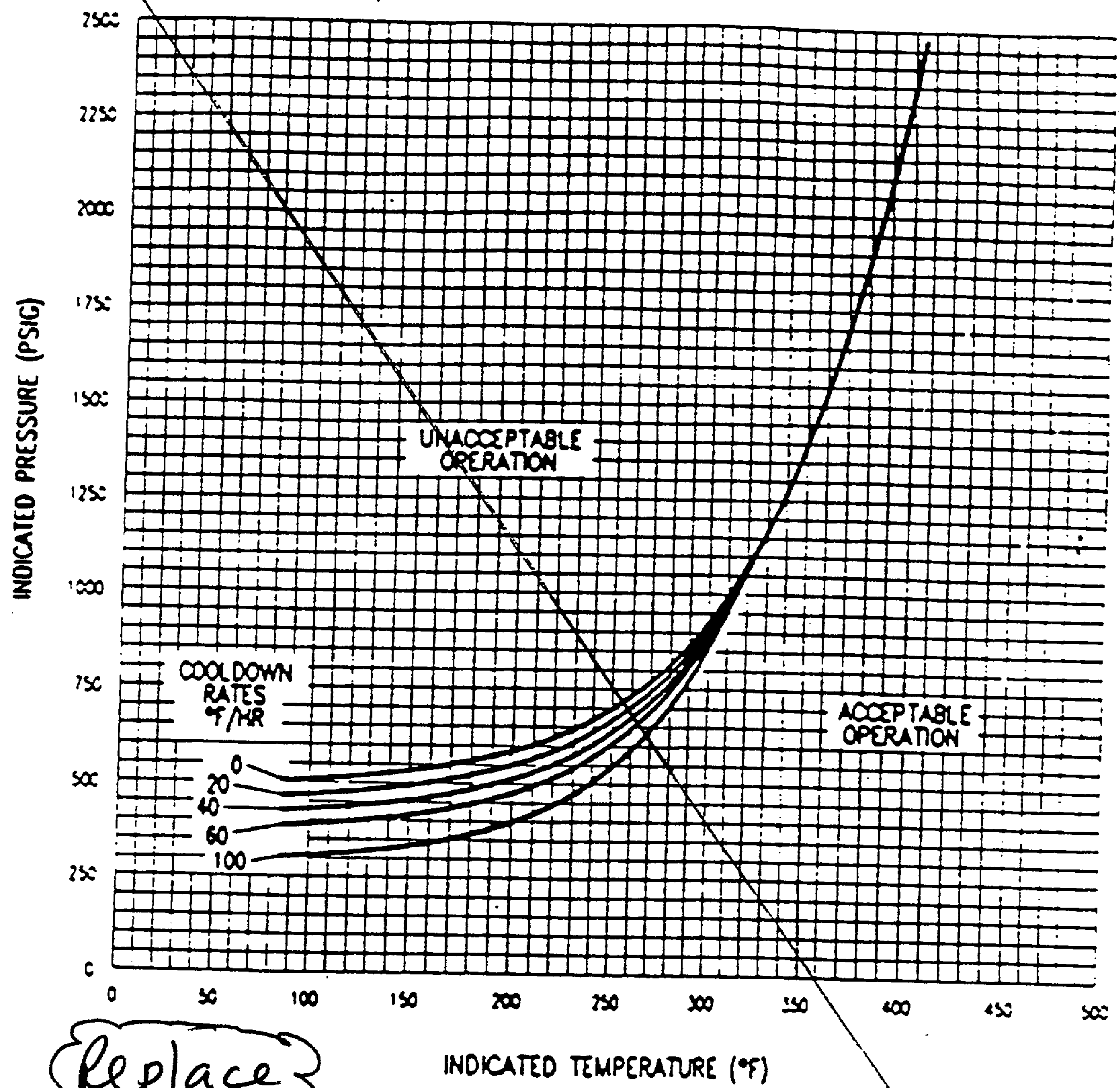
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (100°F/hr) - APPLICABLE UP TO 19 EFPY

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

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

SERVICE PERIOD: 19 EFPY
COOLDOWN RATES: UP TO 100°F/HR

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.



Replace

FIGURE 3.4-4

TURKEY POINT UNITS 3 & 4

REACTOR COOLANT SYSTEM COOLDOWN 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

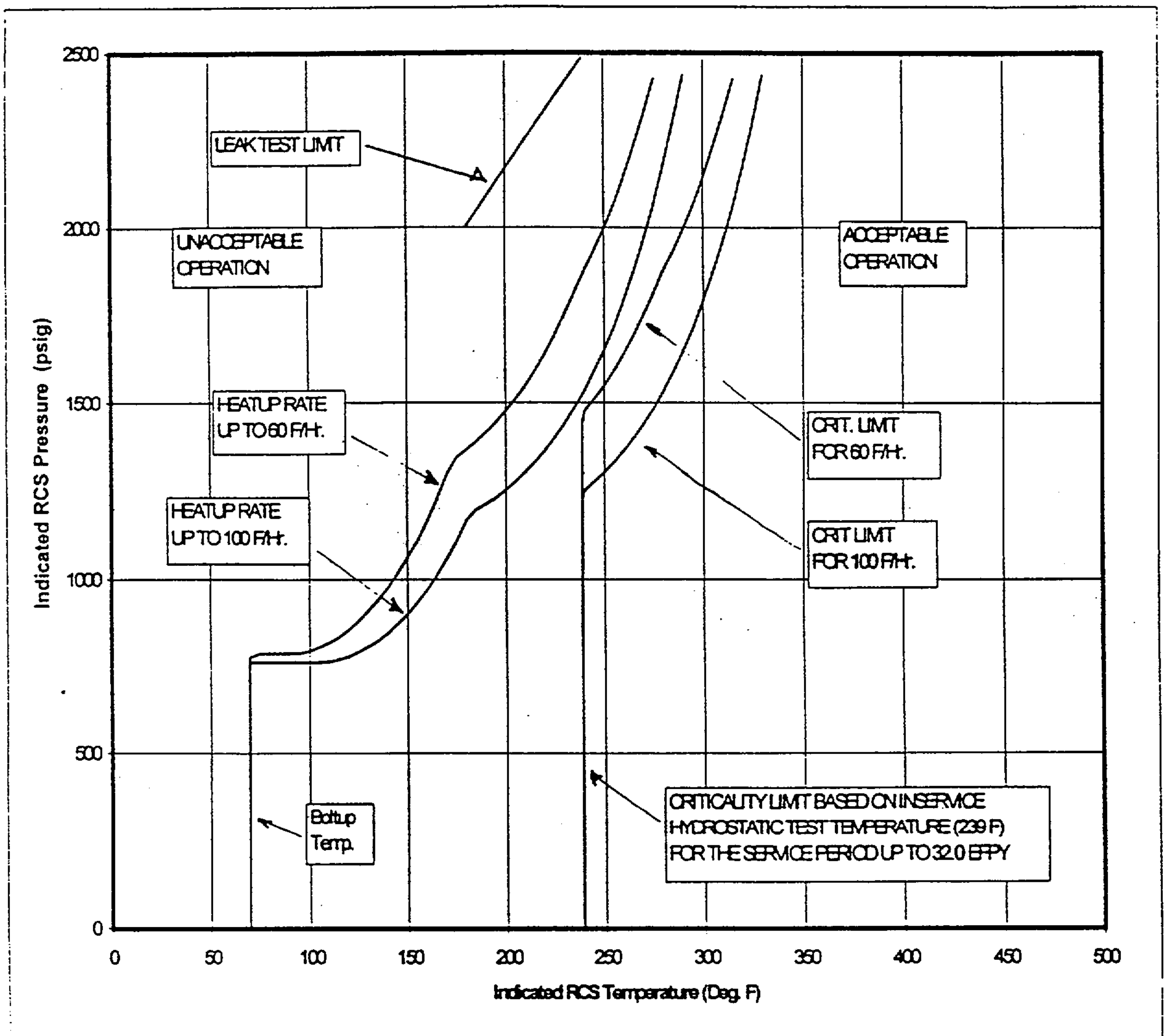


FIGURE 3.4-2 Turkey Point Units 3 and 4 Reactor Coolant System Heatup Limitations (Heatup Rate of 60 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate/Lower Shell Circumferential Weld Seams (Ht. # 71249)
LIMITING ART VALUES AT 32 EFPY: 1/4T, 262°F
3/4T, 218°F

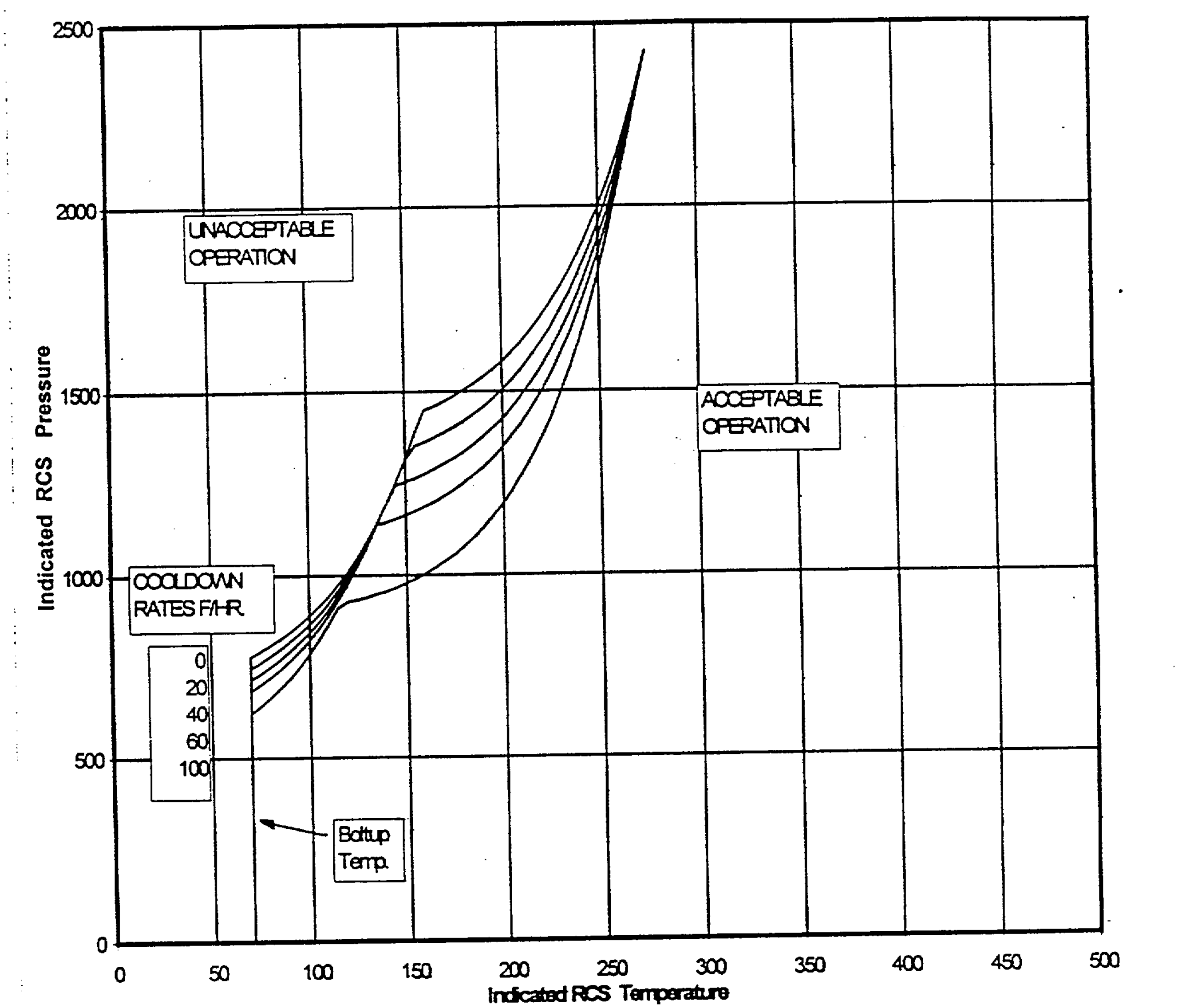


FIGURE 3.4-3 Turkey Point Units 3 and 4 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

REACTOR COOLANT SYSTEM

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~~ 340°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 ~~415 ± 15~~ ≤ 561 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~~ 340°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~~ 340°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.

REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

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 $\leq 340^\circ$ 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 7

For Information Only
Markup of Technical Specification Bases Pages

B 3/4 4-1
B 3/4 4-7
B 3/4 4-8
B 3/4 4-9
B 3/4 4-12
B 3/4 4-13
B 3/4 4-14
B 3/4 4-15

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the applicable design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, three reactor coolant loops provide sufficient heat removal capability for removing core decay heat in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single active failure considerations require that at least two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat, but all combinations of two loops, except two RHR loops, provide single active failure protection.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but the unavailability of the steam generators as a heat removing component, requires that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

340°F

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 275°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. The 50°F limit includes instrument error.

The Technical Specifications for Cold Shutdown allow an inoperable RHR pump to be the operating RHR pump for up to 2 hours for surveillance testing to establish operability. This is required because of the piping arrangement when the RHR system is being used for Decay Heat Removal.

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, Appendex H, and in accordance with additional reactor vessel requirements.

XI

The properties are then evaluated in accordance with Appendix G of the ~~1983~~ 1996 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~~ 32 effective full power years (EFPY) of service life. The ~~19~~ 32 EFPY service life period is chosen such that the limiting RT_{NDT}, at the 1/4T location in

Insert (A)

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

PRESSURE/TEMPERATURE LIMIT~~S~~ (Continued)

limiting material properties and information for generating the heatup and cooldown curves in Figures 3.4-2, 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.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section ~~III~~ XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 ~~and Westinghouse Report GTSD-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~~ XI as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. 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}~~ K_{Ic} for the metal temperature at that time. ~~K_{IR}~~ K_{Ic} is obtained from the reference fracture toughness curve, defined in ~~Appendix G to the~~ ASME Code Case N-640. The ~~K_{IR}~~ K_{Ic} curve is given by the equation:

$$K_{Ic}\ \cancel{K_{IR}} = \cancel{26.78}\ 33.2 + \cancel{1.223}\ 20.734 \exp\left[\cancel{0.0145}\ 0.02\left(T - RT_{NDT} \cancel{+ 160}\right)\right] \qquad (1)$$

Where: K_{Ic} ~~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:

$$C\,K_{IM} + K_{IT} \le \cancel{K_{IR}}\ K_{Ic} \qquad (2)$$

PRESSURE/TEMPERATURE LIMITS (Continued)

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{IT} = the stress intensity factor caused by the thermal gradients,

K_{IC} K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IC} $\cancel{K_{IR}}$ is determined by the metal temperature at the tip of the postulated flaw, ~~the~~ appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw 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, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location K_{IC} for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a

PRESSURE/TEMPERATURE LIMITS (Continued)

cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done 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 vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} 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 such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature 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. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

K_{IC}

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw 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 thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

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 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~~ 340°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

Insert (B)

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