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U-603402
8E-100a
August 25, 2000

Docket No. 50-461

10CFR50.90
10CFR50.12

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

Subject: Clinton Power Station Exemption Request and Proposed Amendment to Facility Operating License No. NPF-62 (LA-00-011)

References:
1. Letter from S. N. Bailey (USNRC) to O. D. Kingsley (ComEd), "Quad Cities – Exemption from the Requirements of 10CFR Part 50, Section 50.60(a) and Appendix G," dated February 4, 2000
2. Letter from S. N. Bailey (USNRC) to O. D. Kingsley (ComEd), "Quad Cities - Issuance of Amendments- Revised Pressure-Temperature Limits," dated February 4, 2000

Dear Madam or Sir:

Pursuant to 10 CFR 50.90, AmerGen Energy Corporation, LLC (AmerGen) hereby applies for amendment of the Clinton Power Station (CPS) Operating License, No. NPF-62. Specifically, AmerGen proposes changes to the Technical Specifications (TS) to revise the reactor vessel pressure/temperature (P/T or P-T) limits specified in TS 3.4.11, "RCS Pressure and Temperature (P/T) Limits" for reactor heatup, cooldown, and critical operation as well as for inservice hydrostatic and leak tests for the reactor coolant system (RCS). Per the proposed changes, the current RCS P/T limits in TS Figure 3.4.11-1, "RCS Pressure Versus Minimum Reactor Vessel Metal Temperature," would be replaced with recalculated RCS P/T limits based, in part, on an alternative methodology.

The alternative methodology used to determine the new P/T limits has been endorsed by the American Society of Mechanical Engineers (ASME) but has not yet received formal approval for generic application by the NRC. Use of the alternative methodology requires an exemption from the current requirements of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," pursuant

AP01

Change: NRC PDR 1 encl 1 w/o prop

to 10 CFR 50.60 (b) and 10 CFR 50.12, "Specific Exemptions." The NRC recently granted such an exemption(s) and approved similar TS changes for the Quad Cities Nuclear Power Station (per References 1 and 2).

In particular, the requested exemption will allow the use of ASME Boiler and Pressure Vessel (B&PV) Code Cases N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," and N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code Section XI, Division 1," in calculating RCS P/T limits. The procedures and methodology that were previously used to calculate the RCS P/T limits for CPS were revised to recalculate the P/T limits, based, in part, on these ASME Code cases.

In addition to application of the above-noted code cases, revision of the RCS P/T limits is necessitated due to re-analysis using a different limiting reactor pressure vessel (RPV) beltline material than what has been previously used, based on a recent, detailed review of RPV fabrication information. Further, re-analysis of the P/T limits was performed using a more conservative, upper-bound neutron fluence value (vice the nominal value used in previous analyses).

The revised P/T limits, as proposed, would yield several benefits. A primary effect of the revised limits is to allow required reactor vessel hydrostatic and leak tests to be performed at a significantly lower temperature. This can significantly reduce critical path time associated with such testing during refueling outages by reducing or eliminating the heatup time required to achieve required test conditions. The safety benefits that may result from this effect include a reduction in the challenges to plant operators associated with maintaining the RCS at higher test temperatures and/or within a narrow temperature band, reduced challenges to personnel safety for inspectors due to lower ambient drywell temperatures, reduced dose to inspectors due to increased inspection effectiveness at the lower ambient drywell temperatures, and increased unavailability of systems connected to the RCS (including the Residual Heat Removal System) because of a reduced heatup and test duration.

The information supporting the proposed TS changes and exemption request is provided in several attachments to this letter:

- Attachment 1 is an affidavit supporting the facts and statements in this letter and its attachments.

- Attachment 2 provides a description and justification for the proposed changes, a finding of no significant hazards consideration and an environmental impact consideration regarding the proposed changes.

- Attachment 3 includes the marked-up and revised TS pages reflecting the requested changes. Attachment 3 also provides, for information only, marked-up pages reflecting changes to be incorporated into the TS Bases pursuant to TS 5.5.11, "Technical Specifications (TS) Bases Control Program."

- Attachment 4 provides the information justifying the Exemption Request.

- Attachment 5 provides General Electric (GE) Nuclear Energy Report GE-NE-B13-02084-00-01, "Pressure-Temperature Curves for AmerGen, Clinton Power Station Using the K_{Ic} Methodology." GE-NE-B13-002084-00-01 contains information that is proprietary to GE. Consistent with the proprietary information notice provided in the preface of the report, AmerGen requests that the information provided by the report be withheld from public disclosure pursuant to 10CFR2.790(a)(4).

Application of the revised P/T limits is desired for the forthcoming refueling outage (RF-7) at CPS, which is scheduled to commence on October 14, 2000. Since, as noted previously, a significant reduction in critical path time can be realized by application of the revised P/T limits (due to the reduced heatup and test time associated with the reactor vessel pressure/leak test), AmerGen respectfully requests NRC review and approval of the requested amendment and exemption by October 26, 2000, which is the currently scheduled date for performance of the reactor vessel pressure/leak test according to the RF-7 outage schedule.

This combined proposed amendment and exemption request has been reviewed by the CPS onsite Facility Review Group and reviewed by the AmerGen Nuclear Review Board (NRB).

Sincerely,

M. T. Coyle
Vice-President

RWC/blf

Attachments

cc: NRC Clinton Project Manager
NRC Resident Inspector, V-690
Regional Administrator – NRC Region III
Illinois Department of Nuclear Safety

AFFIRMATION

Michael T. Coyle, being first duly sworn, deposes and says: That he is Vice President for Clinton Power Station; that this exemption request and application for amendment of Facility Operating License NPF-62 has been prepared under his supervision and direction; that he knows the contents thereof; and that the letter and the statements made and the facts contained therein are true and correct to the best of his knowledge and belief.

Date: This 25TH day of August 2000.

Signed: Michael T Coyle
Michael T. Coyle
Vice President

STATE OF ILLINOIS

} SS.

DEWITT COUNTY

"OFFICIAL SEAL"
Thomas B. Elwood
Notary Public, State of Illinois
My Commission Expires 11/29/2001

Subscribed and sworn to before me this 25th day of August 2000

Thomas B. Elwood
(Notary Public)

PROPOSED CHANGES TO THE CPS TECHNICAL SPECIFICATIONS

INTRODUCTION

In accordance with 10 CFR 50.90, AmerGen proposes to amend the Clinton Power Station (CPS) Operating License. The proposed change is for the Technical Specifications, specifically to Technical Specification (TS) Section 3.4.11, "RCS Pressure and Temperature (P/T) Limits." In particular, AmerGen will replace Figure 3.4.11-1, "RCS Pressure Versus Minimum Reactor Vessel Metal Temperature," for reactor heatup, cooldown, and critical operation as well as for inservice hydrostatic and leak testing of the reactor coolant system (RCS).

Information supporting the proposed TS changes, including a description and discussion of the proposed TS changes, justification for the proposed changes, a safety assessment of the proposed changes, an evaluation for No Significant Hazards Consideration, and the Environmental Impact Consideration, is provided as follows. Information supporting the associated exemption request is in Attachments 4 and 5.

DESCRIPTION OF THE PROPOSED CHANGES

Attachment 3 indicates the proposed administrative changes to TS 3.4.11 and the replacement of Figure 3.4.11-1 with three figures:

- Figure 3.4.11-1, "Bottom Head and Composite P/T Curves For Pressure Tests [Curve A] Up to 32 EFPY,"
- Figure 3.4.11-2, "Bottom Head and Composite P/T Curves for Core Not Critical Operation [Curve B] Up to 32 EFPY," and
- Figure 3.4.11-3, "Composite P/T Curves for Core Critical Operation [Curve C] Up to 32 EFPY."

The current pressure-temperature (P-T or P/T) limits specified per Figure 3.4.11-1 are indicated via several curves on a single figure, for the various operating and/or test conditions. The current curves are to be replaced with recalculated curves on separate figures, and the associated descriptions contained on the figures are to be revised as well. Revised TS Figure 3.4.11-1 will have a curve for the bottom head region of the vessel and a composite RCS curve (excluding the bottom head) for hydrostatic testing and leak testing conditions for an exposure level up to 32 effective full power years (EFPY). Figure 3.4.11-2 will have a curve for the bottom head region of the vessel and the composite RCS curve (excluding the bottom head) for non-critical operation for up to 32 EFPY. Figure 3.4.11-3 will have a curve for the entire RCS for reactor critical operation for up to 32 EFPY. These curves for specifying the required temperature limits will continue to ensure margin to the brittle fracture temperature, i.e., the nil ductility temperature, for the noted operations or conditions. One of the primary effects of the revised curves is to permit reactor vessel inservice hydrostatic and leak tests to be performed at a lower temperature at applicable vessel pressures.

The revised P/T limits (as proposed) are based, in part, on application of American Society of Mechanical Engineers (ASME) Code Cases N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in the Reactor Vessels" and N-640, "Alternative to Requirement Fracture Toughness for Development of P/T Limit Curves for ASME B&PV Code Section XI, Division 1." These code cases provide alternative methods to those currently approved by the NRC and recognized per 10 CFR 50.60. The use and acceptability of these alternative methods therefore requires an exemption from 10 CFR 50.60 requirements. The request for this exemption is addressed further in Attachment 4.

In addition, a change in the limiting material affects the curves. The limiting material was changed based on a detailed evaluation of the vessel fabrication information as explained below.

BASES FOR THE CURRENT TECHNICAL SPECIFICATION REQUIREMENTS

During all modes of operation, reactor vessel pressure and temperature limits are imposed to ensure that, at the existing pressure, the vessel temperature will not approach the low temperature that could lead to brittle fracture, i.e., the nil ductility temperature. 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," provides the requirement that the pressure and temperature limits as well as the associated vessel surveillance program are consistent with 10 CFR 50 Appendix G, "Fracture Toughness Requirements," and 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10 CFR 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in establishing P/T limits. Appendix G of 10 CFR 50 specifies fracture toughness and testing requirements for reactor vessel material in accordance with the ASME B&PV Code and requires that the beltline material in the surveillance capsules be tested in accordance with Appendix H of 10 CFR 50. Appendix G of 10 CFR 50 also requires the prediction of the effects of neutron irradiation on the vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy. Generic Letter 88-11, "NRC Position on Radiation Embrittlement Of Reactor Vessel Materials And Its Impact On Plant Operations," requests that the methods in Regulatory Guide 1.99, Revision 2, be used to predict the effect of neutron irradiation on the reactor vessel material. Appendix H of 10 CFR 50 requires the establishment of a surveillance program to periodically withdraw surveillance capsules from the reactor vessel.

Pursuant to 10 CFR 50 Appendix G, materials used in the CPS reactor vessel have been tested to determine their initial reference nil ductility transition temperature (RT_{NDT}) and the initial reactor pressure vessel (RPV) P/T limits. Reactor operation and resultant high energy neutron radiation, however, require an adjustment of the reference nil ductility transition temperature as well as the RPV P/T limits based on accumulated reactor operating time. The ART for the beltline material has therefore been predicted using the recommendations of Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Material." Currently, TS Figure 3.4.11-1, curve "A" provides the predicted RCS P/T

limit curves at the end of four, eight, and twelve EFPY for hydrostatic tests and leak tests. TS Figure 3.4.11-1 also has RCS P/T curves "B" and "C" for non-nuclear heating and nuclear (core critical) P/T limits, respectively, as well as an additional curve for the bottom head (BH) region of the reactor vessel.

The current P/T limits for CPS were approved by the NRC in Amendments 51 and 109 of the CPS Operating License. NRC approval of the current P/T limits was based on the conformance of the limits to the requirements of Appendices G and H of 10 CFR 50. The current P/T limits satisfied Generic Letter 88-11 since the method in Regulatory Guide 1.99, Revision 2 was used to calculate ART.

The actual ART of the reactor vessel materials is established periodically by removing and evaluating irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel. Accordingly, the RPV P/T limit curves will be adjusted, as required, based on the specimen data and the recommendations of Regulatory Guide 1.99, Revision 2.

JUSTIFICATION FOR THE PROPOSED TECHNICAL SPECIFICATION CHANGES

AmerGen recently contracted with General Electric Company (GE) to recalculate the P/T limit curves for CPS. The methodology used to generate the new P/T limit curves was similar to the methodology previously used to generate the current P/T limit curves of TS Figures 3.4.11-1. However, several improvements or modifications were made to the P/T limit curve methodology to remove excessive conservatism associated with the current P/T limits.

One improvement was the application of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Cases N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code Section XI, Division 1," and N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1." ASME B&PV Code Case N-640 allows the use of K_{IC} rather than K_{Ia} to determine T-RT_{NDT}. ASME B&PV Code Case N-588 allows the use of an alternative procedure for calculating the applied stress intensity factors for axial and circumferential flaws. A detailed description of the methodology used and the results obtained are contained in Attachment 5 to this letter.

In addition to the application of the code cases, the limiting material for the CPS vessel beltline was changed based on a detailed evaluation of fabrication information. The limiting material was changed to no. 2 shell ring plate 22-3, material heat no. C4380-2 from weld heat no. 76492.

Also, the assumed end-of-life fluence was increased in order to provide conservatism. The end-of-life 1/4T fluence was increased to 6.2×10^{18} n/cm^2 (i.e., upperbound fluence) from 4.6×10^{18} n/cm^2. (An assessment of the code cases, material change, and fluence change is provided later in the safety assessment section of this attachment.)

The potential benefits resulting from the proposed changes include the following.

- Reduction in the challenges to operators during future outages in conducting pressure testing of the reactor coolant system at less than or equal to 212° F, and in maintaining the reactor coolant system within a narrow temperature band,

- Reduction in challenges to personnel safety by conducting inspections at lower coolant temperatures,

- Potential dose savings by increasing the effectiveness of inspectors in the containment at lower ambient temperatures,

- Increased availability of systems that are connected to the RCS (including the Residual Heat Removal system) because of reduced heatup and test duration, and

- Potential outage critical path schedule savings by the reduction of time to achieve reactor coolant system temperature and RPV pressure requirements for testing.

SAFETY ASSESSMENT OF THE PROPOSED CHANGES

As previously described, recalculation of the P/T limits involved three important changes to the previous approach taken to calculating the limits: (1) application of the noted Code Cases, (2) revision of the limiting beltline material assumed for calculation, and (3) use of a more conservative (upper bound) fluence value. An assessment of each of these changes is provided below.

Application of ASME Code Cases

P/T limits were developed based on the methodology specified in ASME B&PV Code Section XI, Appendix G, as modified by ASME B&PV Nuclear Code Cases N-588 and N-640. Code Case N-588 allows the use of alternate procedures for defining the postulated flaw orientation and for calculating the applied stress intensity factors for the postulated axial and circumferential flaws. Code Case N-640 allows the use of alternate material fracture toughness when determining minimum vessel temperatures, i.e., the use of K_{Ic} rather than K_{Ia} values as defined in ASME B&PV Code Section XI, Appendix A. For the beltline materials, the RT_{NDT} was adjusted based on the analytical methods specified in Regulatory Guide 1.99, Revision 2. Details of the analytical methods and evaluations performed to calculate the P/T limits are provided in "Technical Basis for Revised P-T Limit Curve Methodology," by W. H. Bamford et.al., The 2000 ASME Pressure Vessels and Piping Conference, July 23-27, 2000. (This paper provides the technical basis for revising P/T limits based on Code Case N-640 and presents sample problems for pressurized water reactors (PWRs) and boiling water reactors (BWRs).)

As noted previously, the use of ASME B&PV Code Cases N-640 and N-588 requires prior NRC review and approval of an exemption to 10 CFR 50.60. The justification for the proposed exemption request is contained in Attachment 4 to this letter.

Based on the technical basis provided in "Technical Basis for Revised P-T Limit Curve Methodology," and on the additional justification provided in Attachment 4, AmerGen has determined that these Code Cases maintain an adequate margin of safety for brittle fracture.

Beltline Material Change

The limiting material for the CPS vessel beltline was changed based on a detailed evaluation of vessel fabrication information. The limiting material was changed to no. 2 shell ring plate 22-3, material heat no. C4380-2 from weld heat no. 76492. This is a change from the limiting material previously used for P/T limits determination (per Amendments 51 and 105 of the CPS Operating License). Specifically, material heat no. 76492, Lot L430B27AE is not the limiting material in the beltline region since this material is in a low fluence region of the vessel, and it was not used in the vertical seam welds of the beltline shell courses. In accordance with Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," the predicted ART for heat no. C4380-2, which is the limiting material to be used for the P/T beltline curves, is 52° F.

Consistent with Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," all relevant available vessel material chemistry data was considered for this analysis including recently obtained industry data. Industry vessel material chemistry data from the Boiling Water Reactor Vessel and Internals Project (BWRVIP) that was obtained from Chicago Bridge and Iron (CB&I) was used in addition to the data in the CPS USAR. In particular, data retrieved from CB&I for incorporation into BWRVIP-46, "BWRVIP Updating of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity," dated December 1997, was evaluated.* Vessel material chemistry data for all known lots of material was thus evaluated.

The use of the new limiting material to determine P/T limits is consistent with NRC requirements. Accordingly, an adequate margin of safety is maintained when the ART for the new limiting material is applied.

End-of-Life Fluence Change

In order to increase margin to the brittle fracture temperature, an upper bound fluence value for 1/4T of 6.2×10^{18} n/cm^2 was used to determine the ART for the limiting materials. This ensures that the P/T curves have ample margin to the brittle fracture temperature. This is a conservative increase beyond the nominal fluence value of 4.6×10^{18} n/cm^2 that was used to determine the currently licensed P/T limits. The upper bound fluence conservatively accounts for calculational uncertainty associated with the fluence determinations based on the flux wire measurements that were made in the first refueling outage. The fluence is for end-of-life conditions, i.e., 32 EFPY.

* It should be noted that for the CPS vessel limiting heats, Nos. C4380-2 and 76492, no new vessel material information was retrieved by the BWRVIP project. However, the data in BWRVIP-46 will be affected because the limiting material was changed by this analysis. (Material chemistry data for non-limiting weld heats that are in the CPS vessel were affected by the BWRVIP project, and the impacts are reflected in Attachment 5.)

It should be noted that development of the P/T curves includes consideration of all of the reactor components. Not only are irradiation embrittlement effects in the beltline considered, there are, for example, non-beltline considerations that must be included, including non-beltline discontinuity limits associated with nozzles, penetrations, and flanges that influence the construction of P-T curves. Thus, notwithstanding the above-described changes to the methodology used to calculate the proposed P-T limits, the methodology continues to include consideration of all vessel components for development of the P-T curves, consistent with the requirements of 10CFR50 Appendix G.

The results from the methodology change, material change, and fluence change are provided in Attachment 5. Contingent upon NRC approval of the exemption request, the changes comply fully with NRC requirements.

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

> Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
>
> Create the possibility of a new or different kind of accident from any previously analyzed; or
>
> Involve a significant reduction in a margin of safety.

AmerGen is proposing changes to the Technical Specifications to revise the pressure/temperature (P/T) limits for heatup, cooldown, critical operation and inservice leak and hydrostatic test limitations for the Reactor Pressure Vessel (RPV). The current reactor coolant system (RCS) P/T limits of TS Figure 3.4.11-1, "RCS Pressure Versus Minimum Reactor Vessel Metal Temperature," are to be replaced with recalculated RCS P/T limits that are applicable to the RCS for the life of the plant, i.e., 32 effective full power years (EFPYs).

AmerGen has evaluated the proposed changes to the Technical Specifications (TS) against the above criteria of 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration. The information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for the proposed changes is provided below.

Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed changes to the CPS reactor coolant system (RCS) pressure/temperature (P/T) limits do not modify the boundary, operating pressure, materials or seismic loading of the reactor coolant system. The proposed changes do adjust the P/T limits for radiation effects to ensure that the RPV fracture toughness is consistent with analysis assumptions and NRC

regulations. Thus, the proposed changes do not involve a significant increase in the probability of occurrence of an accident previously evaluated.

The proposed changes do not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the control of radiological consequences is affected. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes to the reactor pressure vessel pressure-temperature limits do not affect the assumed accident performance of any structure, system or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Does the change involve a significant reduction in a margin of safety?

The methodology for determining the RPV/RCS P/T limits ensures that the limits provide a margin of safety to the conditions at which brittle fracture may occur. The methodology is based on requirements set forth in Appendix G and Appendix H of 10CFR50, with reference to the requirements and guidance of ASME Section XI, and on guidance provided in Regulatory Guide 1.99, Revision 2. The P/T limits currently specified in the CPS Technical specification are based on this methodology, as previously approved via Amendments 51 and 109 to the CPS Operating License. The revised P/T limits are also based on this methodology except as modified by application of the noted Code Cases (in addition to the change in the fluence value and beltline material assumed for analysis).

Although the Code Cases constitute relaxation from the current requirements of 10CFR50 Appendix G, the alternatives allowed by the Code are based on industry experience gained since the inception of the 10CFR50 Appendix G requirements for which some of the requirements have now been determined to be excessively conservative. The more appropriate assumptions and provisions allowed by the Code Cases maintain a margin of safety that is consistent with the intent of 10CFR50 Appendix G, i.e., with regard to the margin originally contemplated by 10CFR50 Appendix G for determination of RPV/RCS P/T limits. On this basis, the proposed changes do not involve a significant reduction in the margin of safety.

ENVIRONMENTAL IMPACT CONSIDERATION

AmerGen has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. AmerGen has determined that these proposed changes meet the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and, as such, that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that these changes are being proposed as an amendment to a license issued pursuant to 10

CFR 50, that the proposed changes are to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR 20, that the proposed changes are to an inspection or surveillance requirement, and that the proposed changes meet the following specific criteria:

(i) The proposed changes involve no significant hazards consideration.

As demonstrated above, these proposed changes do not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed changes are to operational and test limits imposed on the reactor vessel and coolant pressure boundary. The revised limits do not increase the probability or consequences of any accident, as the limits will ensure that adequate margin to brittle fracture is maintained. The revised limits do not affect reactivity or power control of the reactor, do not involve any changes to safety or operational limits with regard to reactor fuel, and do not affect the processing of offgas or any radioactive effluents. Reactor coolant chemistry and radioactive waste treatment systems are also not affected by the proposed changes. On this basis, the proposed changes do not involve any significant increase on the amounts of effluents, nor any change in the types of effluents, that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from these proposed changes.

Marked Up Technical Specification TS 3.4.11 and TS Figure 3.4.11-1;

New TS Figure 3.4.11-1, Figure 3.4.11-2 and Figure 3.4.11-3;

and

"For Information Only" Marked Up TS Bases B 3.4.11

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. ---------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
	C.2 Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ----------	
	Verify:	30 minutes
	a. RCS pressure and RCS temperature are within the limits of Figure 3.4.11-1; and	
	b. RCS heatup and cooldown rates are ≤ 100°F in any one hour period.	

as indicated on the figures

(continued)

Figures 3.4.11-1, 3.4.11-2 and 3.4.11-3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. ----------	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.2 Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ----------	
	Verify: a. RCS pressure and RCS temperature are within the limits of Figures 3.4.11-1, 3.4.11-2 and 3.4.11-3; and b. RCS heatup and cooldown rates are as indicated on the figures.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.2	----------NOTE---------- Only required to be met during control rod withdrawal for the purpose of achieving criticality. ---------- Verify RCS pressure and RCS temperature are within the criticality limits of Figure ~~3.4.11-1~~ 3.4.11-3	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.11.3	----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. ---------- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 100°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.11.4	----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. ---------- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.2	----------NOTE---------- Only required to be met during control rod withdrawal for the purpose of achieving criticality. ---------- Verify RCS pressure and RCS temperature are within the criticality limits of Figure 3.4.11-3.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.11.3	----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. ---------- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 100°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.11.4	----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. ---------- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

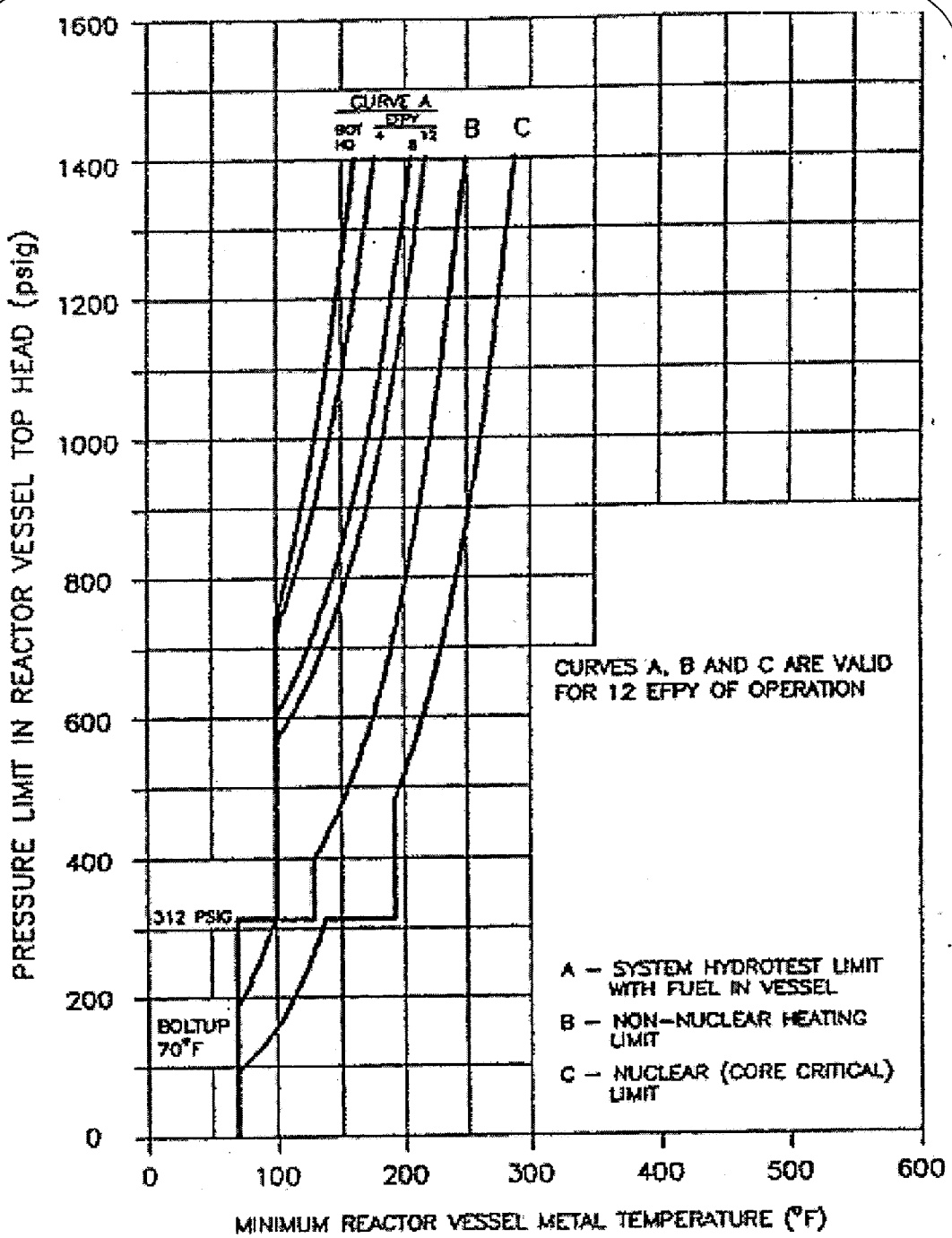

Figure 3.4.11-1 (page 1 of 1)
RCS Pressure Versus Minimum Reactor Vessel Metal Temperature

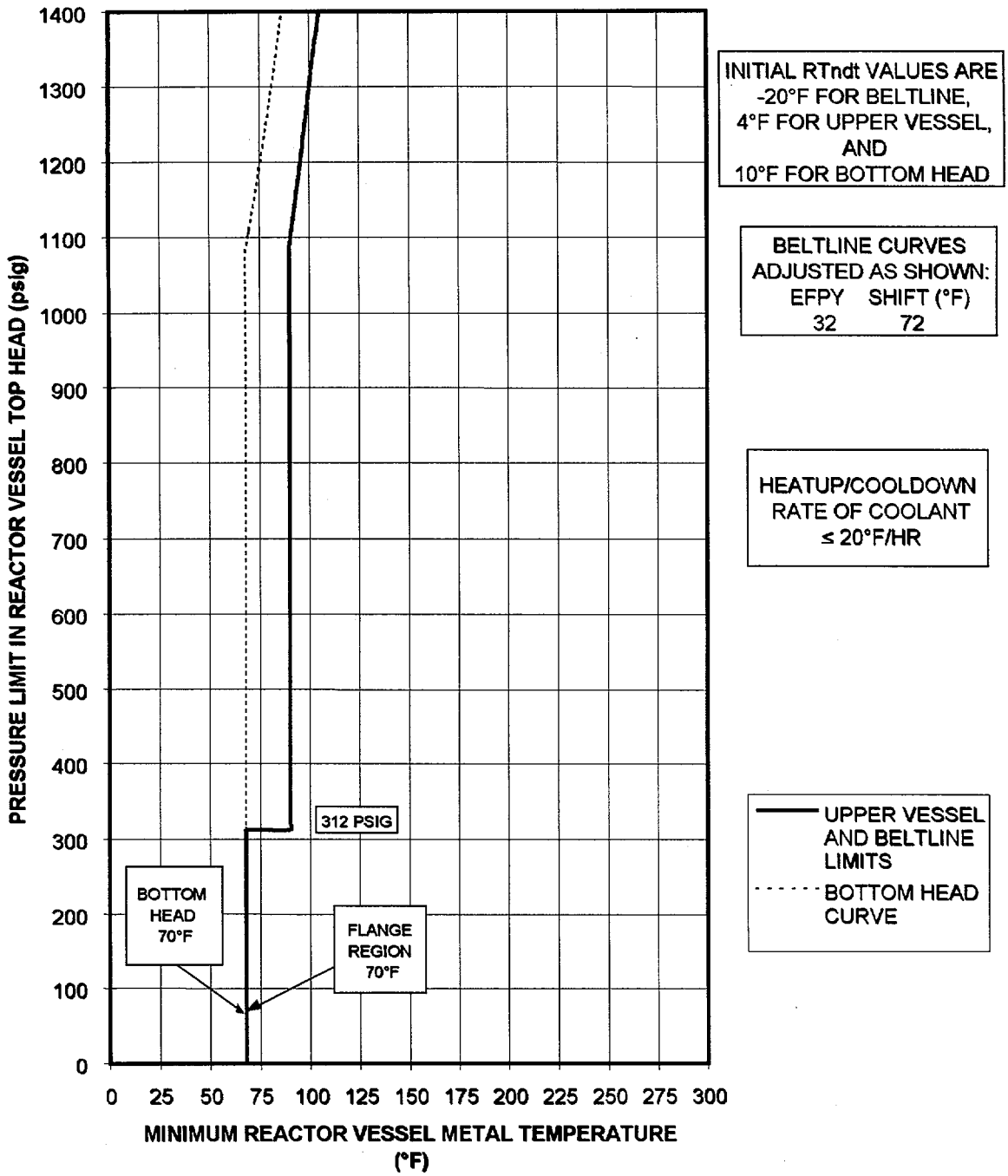

Figure 3.4.11-1
Bottom Head and RCS Composite P/T Curves
for Pressure Tests [Curve A] up to 32 EFPY

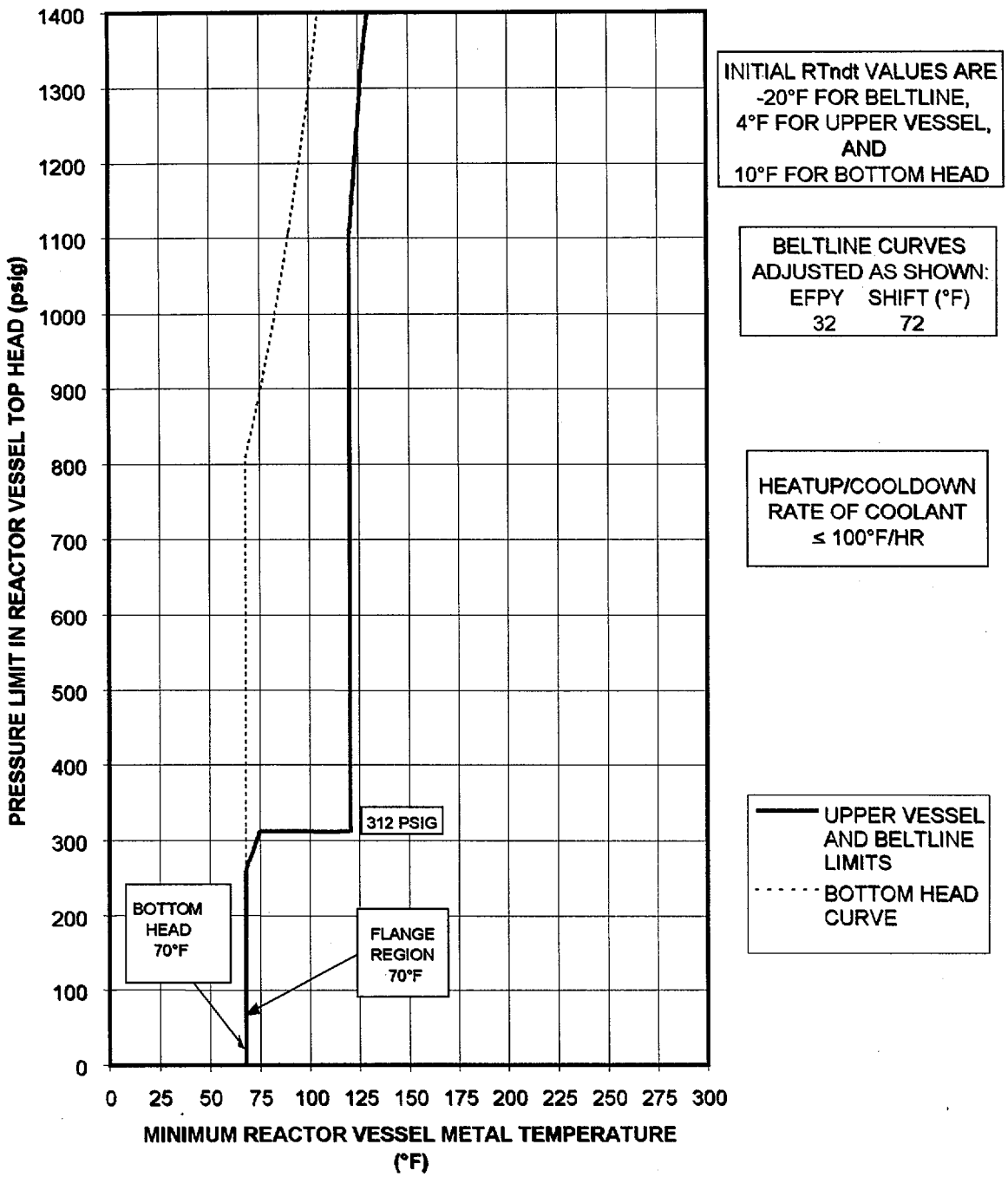

Figure 3.4.11-2
Bottom Head and RCS Composite P/T Curves
for Core Not Critical Operation [Curve B] up to 32 EFPY

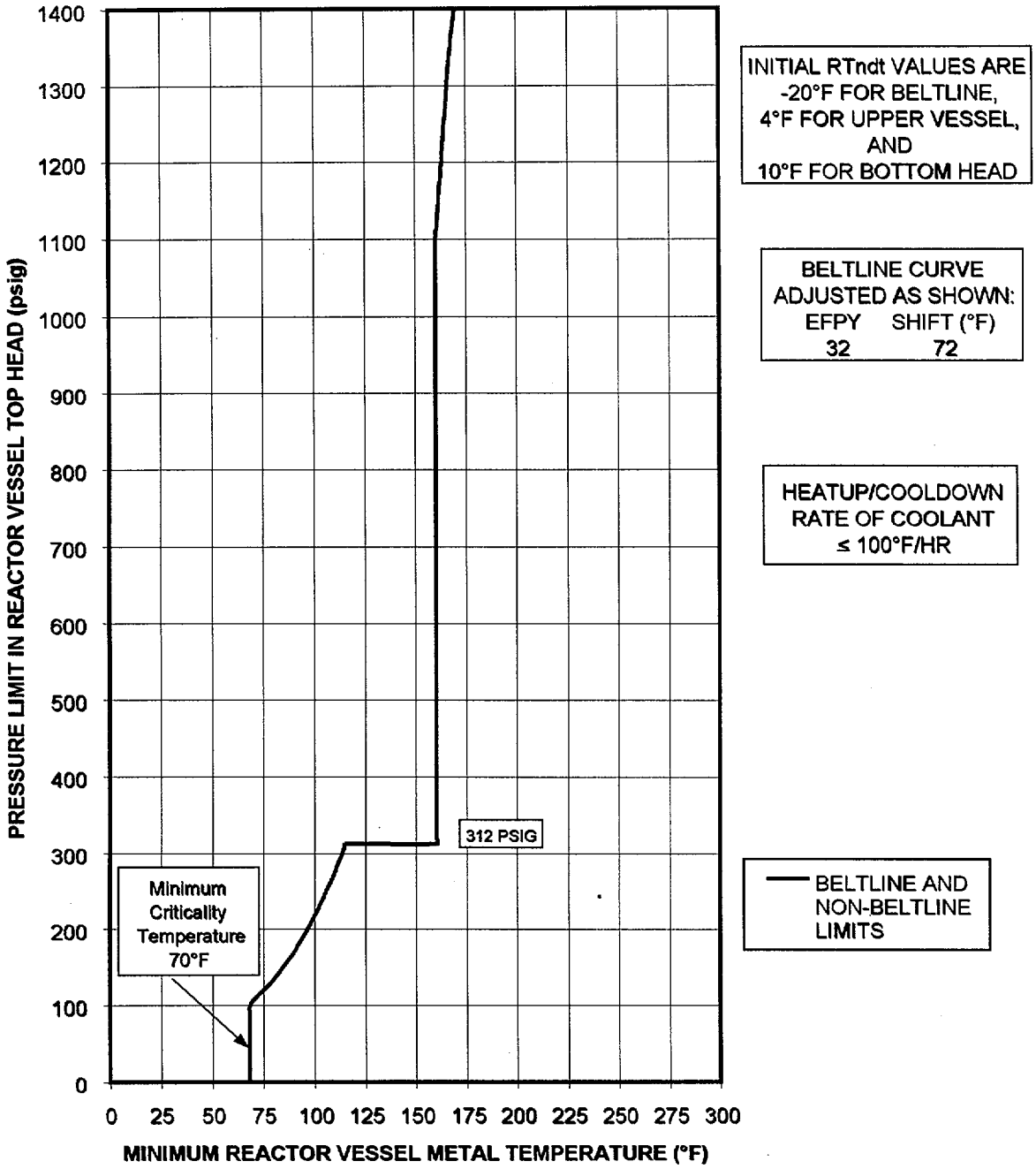

Figure 3.4.11.3
RCS Composite P/T Curves for Core Critical Operation [Curve C] up to 32 EFPY

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.11-1, 3.4.11-2, and 3.4.11-3 contain composite

Figure 3.4.11-1 contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing. The P/T limit curves are valid for 12 Effective Full Power Years (EFPY) of operation. Curves B and C are based on core beltline conditions with an assumed 130°F shift from an initial weld RT_{NDT} of -30°F. Curve A includes beltline adjusted reference temperatures (ARTs) of 58°F for 4 EFPY, 88°F for 8 EFPY, and 100°F for 12 EFPY. In addition, Curve A includes a separate P/T limit curve for the reactor pressure vessel bottom head to account for the fact that during leak and hydrostatic pressure testing, the bottom head temperature may be cooler than the higher elevations of the vessel if the recirculation pumps are either stopped or operating at low speed, and injection through the control rod drives is used to pressurize the vessel.

limits are based on a calculated 72

32 The and an -20 52°F Figure 3.4.11-1

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

(continued)

BASES (continued)

BACKGROUND (continued)

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and 10 CFR 50, Appendix H (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

With regard to the reactor vessel material specimen capsule withdrawal schedule, NRC staff review and approval of any change to this schedule is required prior to implementation. Furthermore, changes to the capsule removal schedule that do not conform with ASTM E-185 (Ref. 3) require NRC approval in the form of a license amendment as described in NRC Administrative Letter 97-04 (Ref. 10).

(continued)

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BASES

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limits include the Reference 1 requirement that they be at least 40°F above the heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leak and hydrostatic testing.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 7 establishes the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

methodology for determining the P/T limits. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The elements of this LCO are:

a. RCS pressure, temperature, and heatup or cooldown rate are within the limits during RCS heatup, cooldown, and inservice leak and hydrostatic testing.

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is within the limit during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow.

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel meets the limit during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow.

d. RCS pressure and temperature are within the criticality limits prior to achieving criticality.

e. The reactor vessel flange and the head flange temperatures are within the limits when tensioning the reactor vessel head bolting studs.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leak and

(continued)

BASES

LCO (continued)

hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves. In addition, administrative limits have been imposed to restrict the rate of temperature changes to ≤ 20°F in any one hour period when operating between Curve A and Curve B or C, as applicable, of Figure 3.4.11-1. This additional limitation on temperature changes is imposed due to the reduced margin to the limits and the desire to maintain RCS temperature essentially constant during pressurization for hydrostatic testing.

Figure 3.4.11-2 or Figure 3.4.11-3

the curves in Figure 3.4.11-1

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

A.1 and A.2

Operation outside the P/T limits while in MODE 1, 2, or 3 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

Condition A is modified by a Note requiring Required Action A.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With the reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by bringing the plant to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 200°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

RCS temperature conditions are determined by measuring the metal temperature of the reactor vessel flange surfaces, bottom head outside surface, bottom head inside surface (as measured by the bottom head drain temperature), and reactor recirculation loop temperature. Verification that operation is within limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction of minor deviations.

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

This SR has been modified by a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.1 (continued)

With regard to RCS pressure, temperature, and heatup and cooldown rates values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 11).

SR 3.4.11.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

This SR has been modified by a Note that requires this Surveillance to be met only during control rod withdrawal for the purpose of achieving criticality.

With regard to RCS pressure and temperature values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 12).

SR 3.4.11.3 and SR 3.4.11.4

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 8) are satisfied.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.3 and SR 3.4.11.4 (continued)

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.11.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.11.3 and SR 3.4.11.4 have been modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4 during recirculation pump start. In MODE 5, the overall stress on limiting components is lower; therefore, ΔT limits are not required.

With regard to temperature difference values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Refs. 13, 14).

SR 3.4.11.5, SR 3.4.11.6, and SR 3.4.11.7

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. SR 3.4.11.5 allows up to 10% of the reactor vessel head bolting studs to be fully tensioned with flange temperatures < 70 °F. This allows the closure flange O-rings to be sealed to support raising reactor water level to assist in warming the flanges. When in MODE 4 with RCS temperature ≤ 80°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 90°F, monitoring of the flange temperature is required every 12 hours to ensure the temperatures are within limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.5, SR 3.4.11.6, and SR 3.4.11.7 (continued)

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

With regard to reactor vessel flange and head flange temperature values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 15).

SR 3.4.11.8 and SR 3.4.11.9

Differential temperatures within the applicable limits ensure that thermal stresses resulting from increases in THERMAL POWER or recirculation loop flow during single recirculation loop operation will not exceed design allowances. Performing the Surveillance within 15 minutes before beginning such an increase in power or flow rate provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the change in operation.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.11.9 is to compare the temperatures of the operating recirculation loop and the idle loop.

Plant specific test data has determined that the bottom head is not subject to temperature stratification with natural circulation at power levels as low as 30% of RTP and with any single loop flow rate greater than or equal to 30% of rated loop flow. Therefore, SR 3.4.11.8 and SR 3.4.11.9 have been modified by a Note that requires the Surveillance to be met only when THERMAL POWER or loop flow is being increased when the above conditions are not met. The Note for SR 3.4.11.9 further limits the requirement for this Surveillance to exclude comparison of the idle loop temperature if the idle loop is isolated from the RPV since the water in the loop cannot be introduced into the remainder of the Reactor Coolant System.

(continued)

The P/T limits for this Technical Specification are discussed in more detail in Reference 18.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.8 and SR 3.4.11.9 (continued)

With regard to temperature difference values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Refs. 16, 17).

REFERENCES

1. 10 CFR 50, Appendix G.

2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

3. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests For Light-Water Cooled Nuclear Power Reactor Vessels."

4. 10 CFR 50, Appendix H.

5. Regulatory Guide 1.99, Revision 2, May 1988.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. NEDO-21778-A, "Transient Pressure Rises Affecting Fracture Toughness Requirements for BWRs," December 1978.

8. USAR, Section 15.4.4.

9. USAR, Section 5.3.

10. NRC Administrative Letter 97-04, "NRC Staff Approval for Changes to 10 CFR Part 50, Appendix H, Reactor Vessel Surveillance Specimen Withdrawal Schedules."

11. Calculation IP-0-0036.

12. Calculation IP-0-0037.

13. Calculation IP-0-0038.

14. Calculation IP-0-0039.

(continued)

BASES

REFERENCES (continued)

15. Calculation IP-0-0040.

16. Calculation IP-0-0041.

17. Calculation IP-0-0042.

18. GE-NE-B13-02084-00-01, "Pressure-Temperature Curves for AmerGen, Clinton Power Station Using the K_{Ic} Methodology."

REQUESTED EXEMPTION

Code Cases N-640 and N-588 are addressed separately below.

Justification for Use of Code Case N-640

10 CFR 50.12(a) Requirements

The requested exemption to allow use of ASME B&PV Code Case N-640 in conjunction with ASME B&PV Code XI, Appendix G to determine the pressure/temperature limits for the reactor pressure vessel 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 the following are met.

1. The requested exemption is authorized by law:

   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 pressure/temperature (P/T) limits being proposed for Clinton Power Station rely in part on the requested exemption. In accordance with Code Case N-640 the revised P/T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME B&PV Code, Section XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia} fracture toughness curve of ASME B&PV Code, Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. Except for the changes in Code Case N-588, the other margins involved with the ASME B&PV Code, Section XI, Appendix G process of determining P/T limit curves remain unchanged.

   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 the K_{Ia} curve. The K_{Ic} curve models the slow heat-up and cooldown process of a reactor pressure 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 pressure vessel material fracture toughness. Since 1974, additional knowledge has been gained about the fracture toughness of reactor pressure vessel materials and their fracture response to applied loads. As described in "Technical Basis for Revised P-T Limit Curve Methodology," the additional knowledge demonstrates that the lower bound fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect against potential reactor pressure vessel failure, and that the lower bound K_{Ic} fracture toughness provides an adequate margin of safety to protect against potential reactor pressure vessel failure.

The use of P/T curves based on the K_{Ic} fracture toughness limits will enhance overall plant safety by opening the P/T operating window especially in the region of low temperature operations. Safety benefits that would be realized during the pressure test include a reduction in the challenges to operators in maintaining a high temperature in a limited operating window, personnel safety while conducting inspections in primary containment at elevated temperatures, and increased availability of plant systems, including the residual heat removal system, due to reduction of the heatup and test time.

Based on the above, this exemption does not present an undue risk to the public health and safety.

3. The requested exemption will not endanger the common defense and security:

This exemption request concerns the revision of operating and test limits for the Clinton Power Station commercial power reactor in accordance with industry-proposed guidance and has no impact on common defense and security. Therefore, the common defense and security are not endangered by approval of this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60:

In accordance with 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. The requested exemption meets the special circumstances of the following paragraphs of 10 CFR 50.12.

(a) (2) (ii) – demonstrates the underlying purpose of the regulation will continue to be achieved;

(a) (2) (iii) – would result in undue hardship or other cost that are significant if the regulation is enforced and;

(a) (2) (v) – will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

Each of the above paragraphs is addressed below.

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

ASME B&PV Code, Section XI, Appendix G, provides procedures for determining allowable loading on the reactor pressure 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 limit curves satisfy the underlying requirement that the reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the reactor pressure vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized, and that the 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 B&PV Code, Section XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning reactor pressure 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 relaxation of the ASME B&PV Code, Section XI, Appendix G, requirements via application of ASME B&PV Code Case N-640, while maintaining the underlying purpose of the ASME B&PV Code and the NRC regulations to ensure an acceptable margin of safety.

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

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 B&PV Code, Section XI, Appendix G procedure. Continued operation of Clinton Power Station, with these P/T curves without the relief provided by ASME B&PV Code Case N-640 would unnecessarily restrict the pressure-temperature operating window. This restriction challenges the operations staff during pressure tests to maintain a high temperature within a limited operating window. It also subjects inspection personnel to increased safety hazards while conducting inspections of systems at elevated temperatures.

This constitutes an unnecessary burden that can be alleviated by the application of ASME B&PV Code Case N-640 in the development of the proposed P/T curves. Implementation of the proposed P/T curves as allowed by ASME B&PV Code Case N-640 does not significantly reduce the margin of safety below that established by the original requirement.

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

The requested exemption provides only temporary relief, since AmerGen anticipates that the provisions of Code Case N-640 will be incorporated into (or reconciled with) the requirements of 10CFR50 Appendix G, based on ongoing industry efforts to do so. NRC approval of the code Case is pending, but additional action may be required to allow use of the Code Case without requiring an exemption to 10CFR50 Appendix G. The estimated time for such actions to be completed is unknown, and therefore, the effective period of time that the exemption would be effective is indefinite.

ASME B&PV Code Case N-640, Conclusion for Exemption Acceptability:

Compliance with the specified requirement of 10 CFR 50.60(a) would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME B&PV Code Case N-640 allows a reduction in the lower bound fracture toughness used in ASME B&PV Code, Section XI, Appendix G, in the determination of reactor coolant system P/T limits. This proposed alternative is acceptable because the ASME B&PV Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME B&PV Code, Section XI, Appendix G, was approved in 1974. Therefore, application of ASME B&PV Code Case N-640 for Clinton Power Station will ensure an acceptable margin of safety and does not present an undue risk to the public health and safety.

Justification for the Use of Code Case N-588

10 CFR 50.12(a) Requirements:

The requested exemption to allow use of ASME B&PV Code Case N-588 to determine stress intensity factors for postulated flaws and postulated flaw orientation for circumferential welds 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 the following is satisfied:

1. The requested exemption is authorized by law:

   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 that Article G-2120 of ASME B&PV Code, Section XI, Appendix G, be used to determine the maximum postulated defects in reactor pressure vessels (RPV) for determination of the vessel pressure-temperature limits. These limits are determined for normal operation and pressure/leak test conditions. Article G-2120 specifies, in part, that the postulated defect be in the surface of the RPV material and normal (i.e., perpendicular in the plane of the material) to the direction of maximum stress. ASME B&PV Code, Section XI, Appendix G, also provides methodology for determining the stress intensity factors for a maximum postulated defect normal to the maximum stress. The purpose of this article is, in part, to ensure the prevention of non-ductile fractures by providing procedures to identify the most limiting postulated fractures to be considered in the development of P/T limits.

   Code Case N-588 provides benefits, in terms of calculating P/T limits, by revising the Article G-2120 reference flaw orientation for circumferential welds in reactor pressure 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 defects that could physically exist within the geometry of the weldment. The current ASME B&PV Code Section XI, Appendix G approach mandates the consideration of an axial reference flaw in circumferential welds for purposes of calculating the P/T limits. Postulating such a reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the postulated length of the flaw is 1.5 times the reactor pressure vessel wall thickness, which is much longer than the width of circumferential welds. 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 defects be postulated in plates/forgings and axial welds. The fabrication of reactor pressure 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 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, in-service non-destructive examinations and data taken from destructive examination of actual reactor pressure vessel welds, confirms that any remaining defects are small, laminar in nature, and do not cross transverse to the weld bead. Therefore, any postulated defects introduced during the fabrication process, and not detected during subsequent non-destructive 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 B&PV Code Case N-588 addresses this issue by allowing consideration of maximum postulated defects oriented circumferentially in circumferential welds. ASME B&PV Code Case N-588 also provides appropriate procedures for determining the stress intensity factors for use in developing reactor pressure vessel P/T limits per ASME B&PV Code, Section XI, Appendix G procedures. The procedures allowed by ASME B&PV Code Case N-588 are conservative and provide a margin of safety in the development of reactor pressure vessel P/T operating and pressure test limits, for prevention of non-ductile fracture of the reactor pressure vessel.

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, RCS pressure and temperature must be maintained within the heatup and cooldown rate-dependent P/T limits specified in TS 3.4.11, "Reactor Coolant System." Based on the above, this requested exemption does not present an undue risk to the public health and safety.

3. <u>The requested exemption will not endanger the common defense and security</u>:

This exemption request only concerns the revision of operating and test limits for the Clinton Power Station commercial power reactor in accordance with industry-proposed guidance and has no impact on common defense and security. Therefore, the common defense and security are not endangered by this exemption request.

4. <u>Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60</u>:

In accordance with 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. The requested exemption meets the special circumstances of the following paragraphs of 10CFR50.12:

(a)(2)(ii) - demonstrates that the underlying purpose of the regulation will continue to be achieved;

(a)(2)(iii) - would result in undue hardship or other cost that are significant if the regulation is enforced and;

(a)(2)(v) - will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

Each of the above paragraphs is addressed below.

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

The underlying purpose of 10 CFR 50, Appendix G and ASME B&PV Code, Section XI, Appendix G, is to satisfy the underlying requirement that:

1) The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure that when stressed the reactor pressure 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 limit curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

Application of ASME B&PV Code Case N-588 when determining P/T operating and test limit curves per ASME B&PV Code, Section XI, Appendix G, provides appropriate procedures for determining limiting maximum postulated defects and considering those defects in the P/T limits. This application of the code case maintains the margin of safety originally contemplated when ASME B&PV Code, Section XI, Appendix G was developed.

Therefore, use of ASME B&PV Code Case N-588, as described above, satisfies the underlying purpose of the ASME B&PV Code and the NRC regulations to ensure an acceptable level of safety.

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

The Reactor Pressure Vessel P/T operating window is defined by the P/T operating and test limit curves developed in accordance with the ASME B&PV Code, Section XI, Appendix G procedure. Continued operation of with these P/T limit curves without the relief provided by ASME B&PV Code Case N-588 would unnecessarily restrict the P/T operating window for Clinton Power Station. This restriction challenges the operations staff during pressure tests to maintain a high temperature within a limited operating window. It also subjects inspection personnel to increased safety hazards while conducting inspections of systems at elevated temperatures.

This constitutes an unnecessary burden that can be alleviated by the application of ASME B&PV Code Case N-588 in the development the proposed P/T curves. Implementation of the proposed P/T limit curves as allowed by ASME B&PV Code Case N-588 does not reduce the margin of safety originally contemplated by either the NRC or ASME.

10CFR50.12(a)(2)(v):

The requested exemption provides only temporary relief, since AmerGen anticipates that the provisions of Code Case N-640 will be incorporated into (or reconciled with) the requirements of 10CFR50 Appendix G, based on ongoing industry efforts to do so. NRC approval of the code Case is pending, but additional action may be required to allow use of the Code Case without requiring an exemption to 10CFR50 Appendix G. The estimated time for such actions to be completed is unknown, and therefore, the effective period of time that the exemption would be effective is indefinite.

ASME B&PV Code Case N-588, Conclusion for Exemption Acceptability:

Compliance with the specified requirements of 10 CFR 50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME B&PV 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 B&PV Code, 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 margin of safety commensurate with that which existed at the time ASME B&PV Code, Section XI, Appendix G, was approved in 1974. Therefore, application of ASME B&PV Code Case N-588 for Clinton Power Station will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor pressure vessel failure.

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 P/T limits specified in TS Section 3.4.11. Therefore, this exemption does not present an undue risk to the public health and safety.

GE Nuclear Energy Report

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"Pressure-Temperature Curves for AmerGen, Clinton Power Station Using the K_{Ic} Methodology"

(GE Proprietary Information)