Mr. Joseph J. Hagan
Vice President, Operations GGNS
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

August 27, 1997

SUBJECT: ISSUANCE OF AMENDMENT NO.132 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1 (TAC NO. M97520)

Dear Mr. Hagan:

The Nuclear Regulatory Commission has issued the enclosed Amendment No.132 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GGNS). This amendment revises the Technical Specifications (TSs) for GGNS in response to your application dated October 22, 1996, as supplemented by letter dated June 26, 1997. (GNRO-96/00120 and GNRO-97/00058, respectively)

The amendment revised Figure 3.4.11-1 and Surveillance Requirements (SR) 3.4.11.1 and 3.4.11.2 of the TSs. The revisions to the pressure temperature (P-T) limits in Figure 3.4.11-1 extend the validity of the P-T limit curves from 10 effective full power years (EFPY) to 32 EFPY in increments. For each EFPY increment, there is a set of P-T limits in Figure 3.4.11-1 valid for the specified range of EFPY (i.e., Figure 3.4.11-1 is 5 figures). Further, paragraphs in the Bases and SRs in the TSs were revised to be consistent with the P-T limits. There are no commitments related to the amendment.

There are no commitments associated with this amendment to the TSs. A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

8/22/97

Jack N. Donohew, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures: 1. Amendment No.132 to NPF-29
2. Safety Evaluation

cc w/encls: See next page

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J. Clifford	T. Harris (TLH3)		

*See previous concurrence
Document Name: GG97520.AMD (EMCB SE dated July 10, 1997)

OFC	PM/PD4-1	LA/PD4-1	OGC*
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 27, 1997

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Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures: 1. Amendment No.132 to NPF-29
2. Safety Evaluation

cc w/encls: See next page

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

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

ENTERGY MISSISSIPPI, INC.

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 132
License No. NPF-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated October 22, 1996, as supplemented by letter dated June 26, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-29 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 132, are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jack N. Donohew

Jack N. Donohew, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 27, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 132

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3.4-27	3.4-27
3.4-28	3.4-28
3.4-31	3.4-31
--	3.4-32
--	3.4-33
--	3.4-34
--	3.4-35
3.4-32	3.4-36

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. <u>AND</u> C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

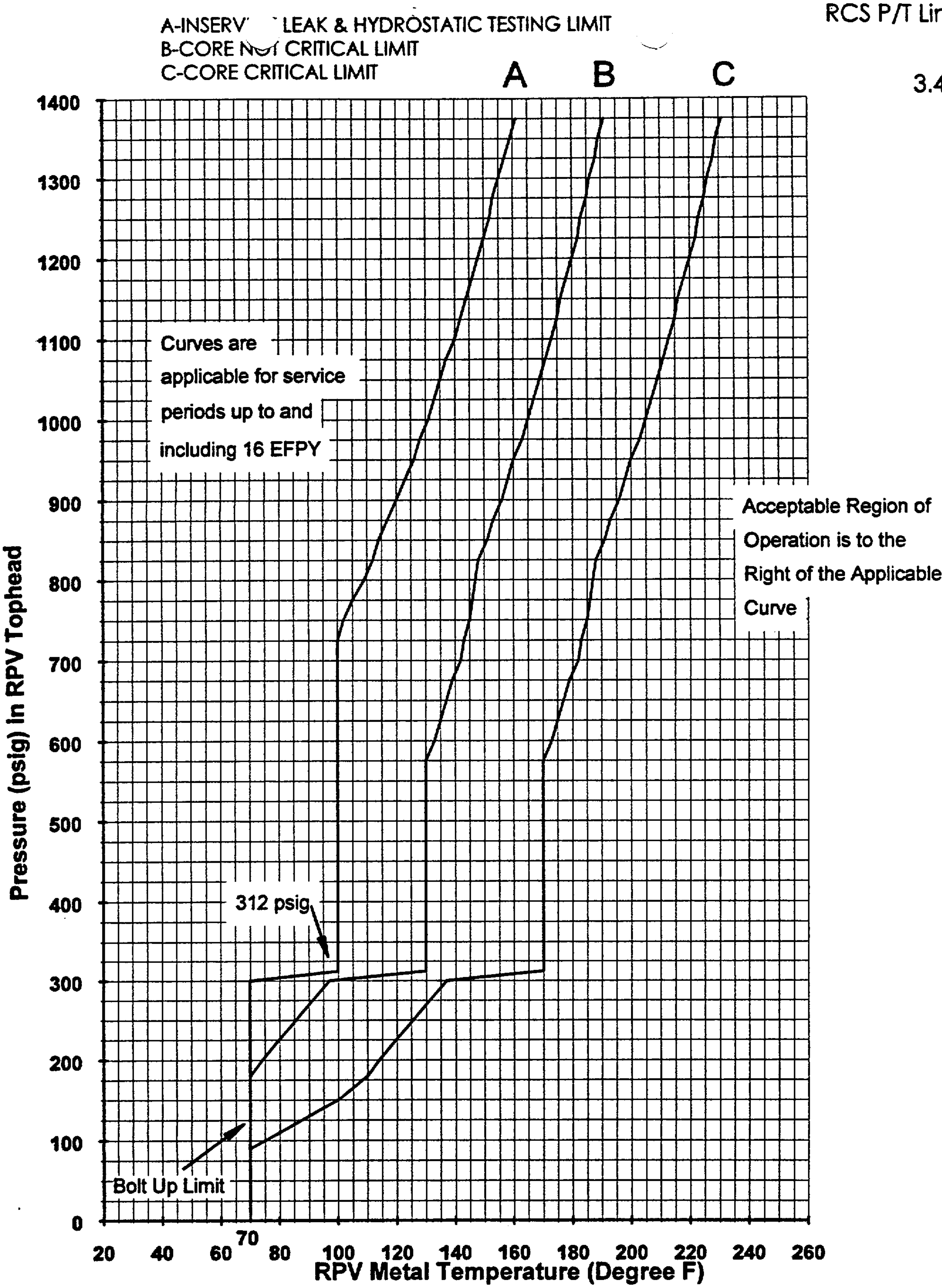
SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --- Verify: a. RCS pressure and RCS temperature are within the limits of the applicable Figure 3.4.11-1 based on the current Effective Full Power Year (EFPY), and b. RCS heatup and cooldown rates are ≤ 100°F in any 1 hour period.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

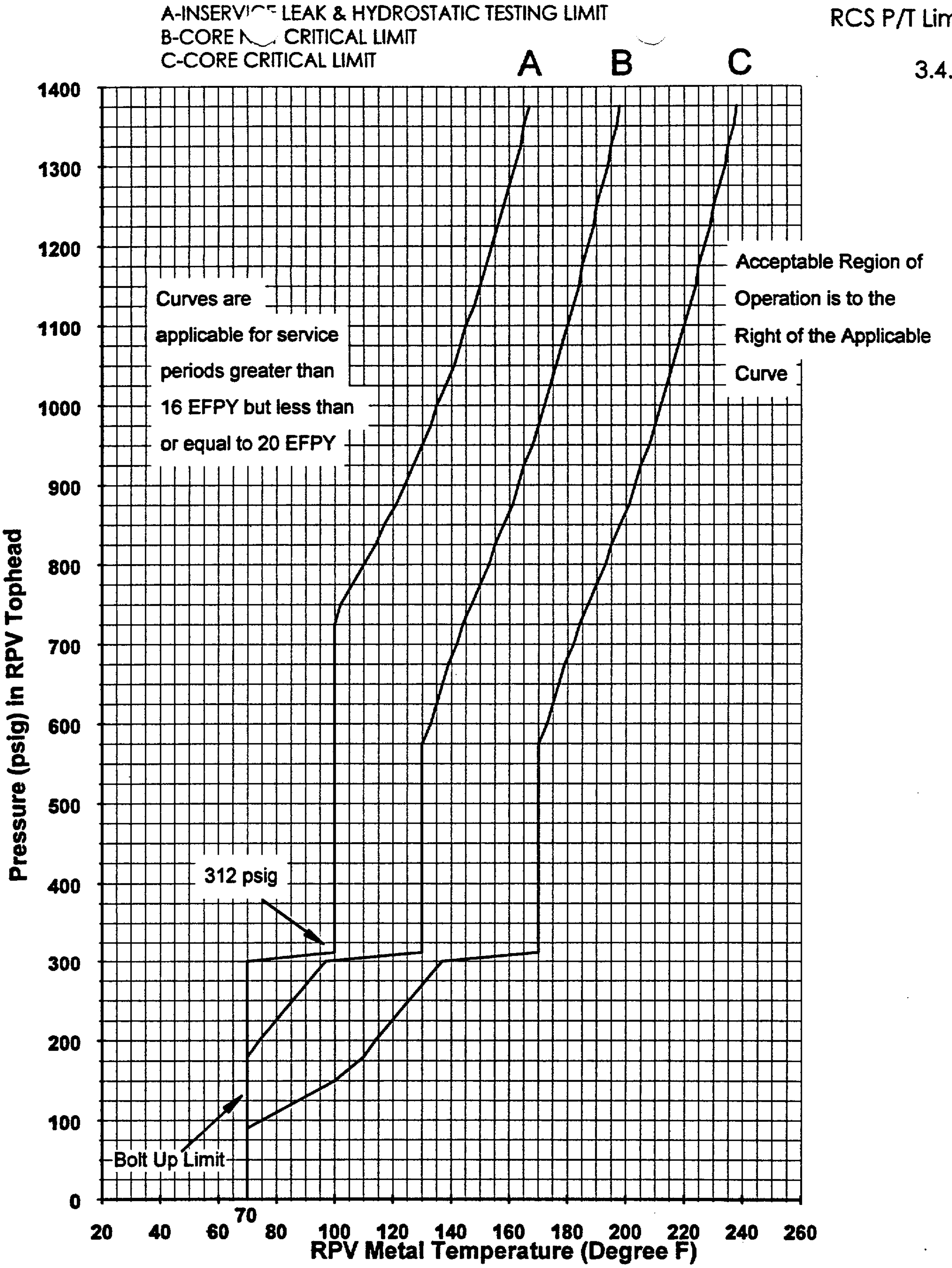
	SURVEILLANCE	FREQUENCY
SR 3.4.11.2	-------------------NOTE------------------- Only required to be met during control rod withdrawal for the purpose of achieving criticality. -- Verify RCS pressure and RCS temperature are within the criticality limits specified in the applicable Figure 3.4.11-1 based on the current Effective Full Power Year (EFPY).	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 with reactor steam dome pressure ≥ 25 psig 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)

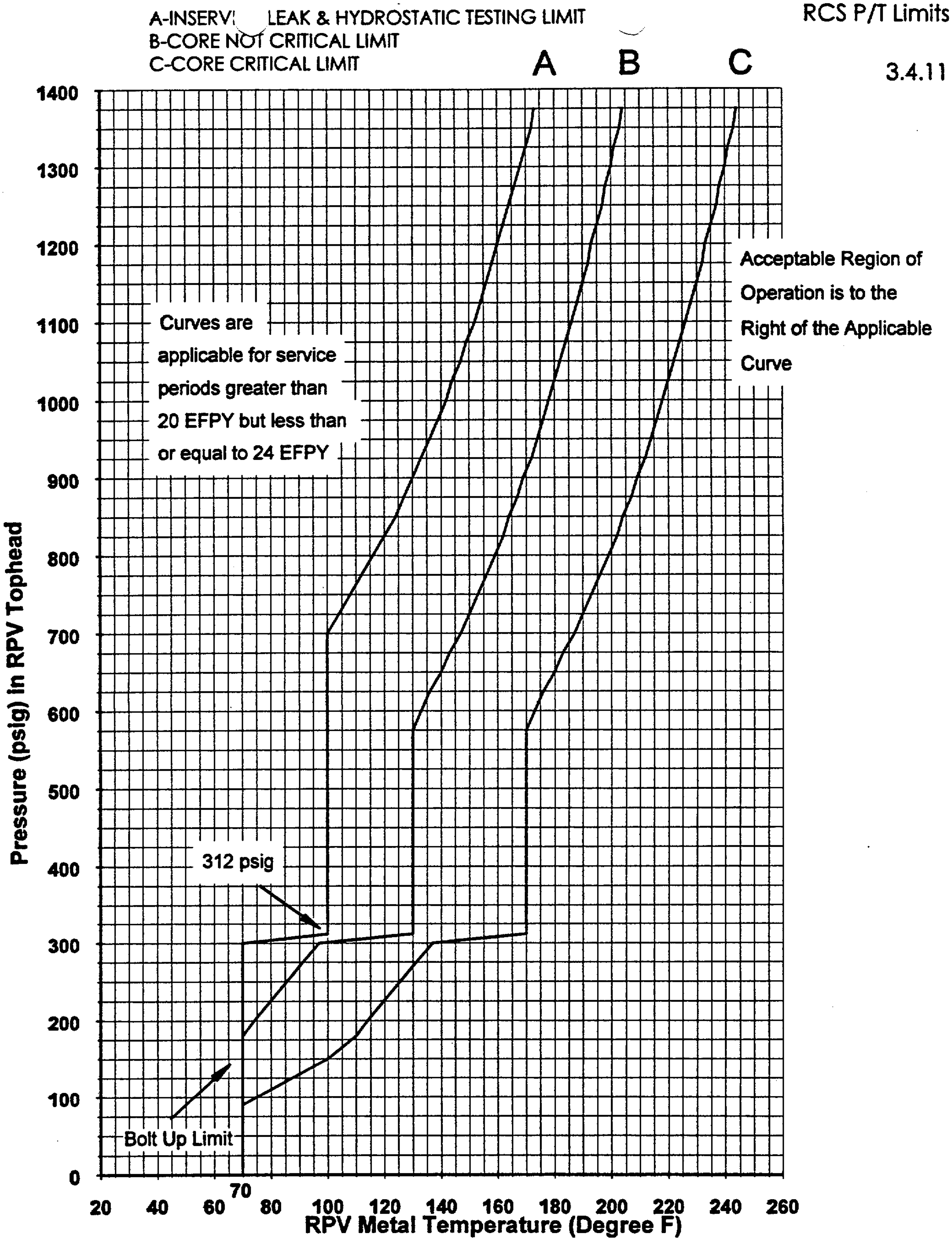


Minimum Reactor Vessel Metal Temperature vs Reactor Vessel Pressure

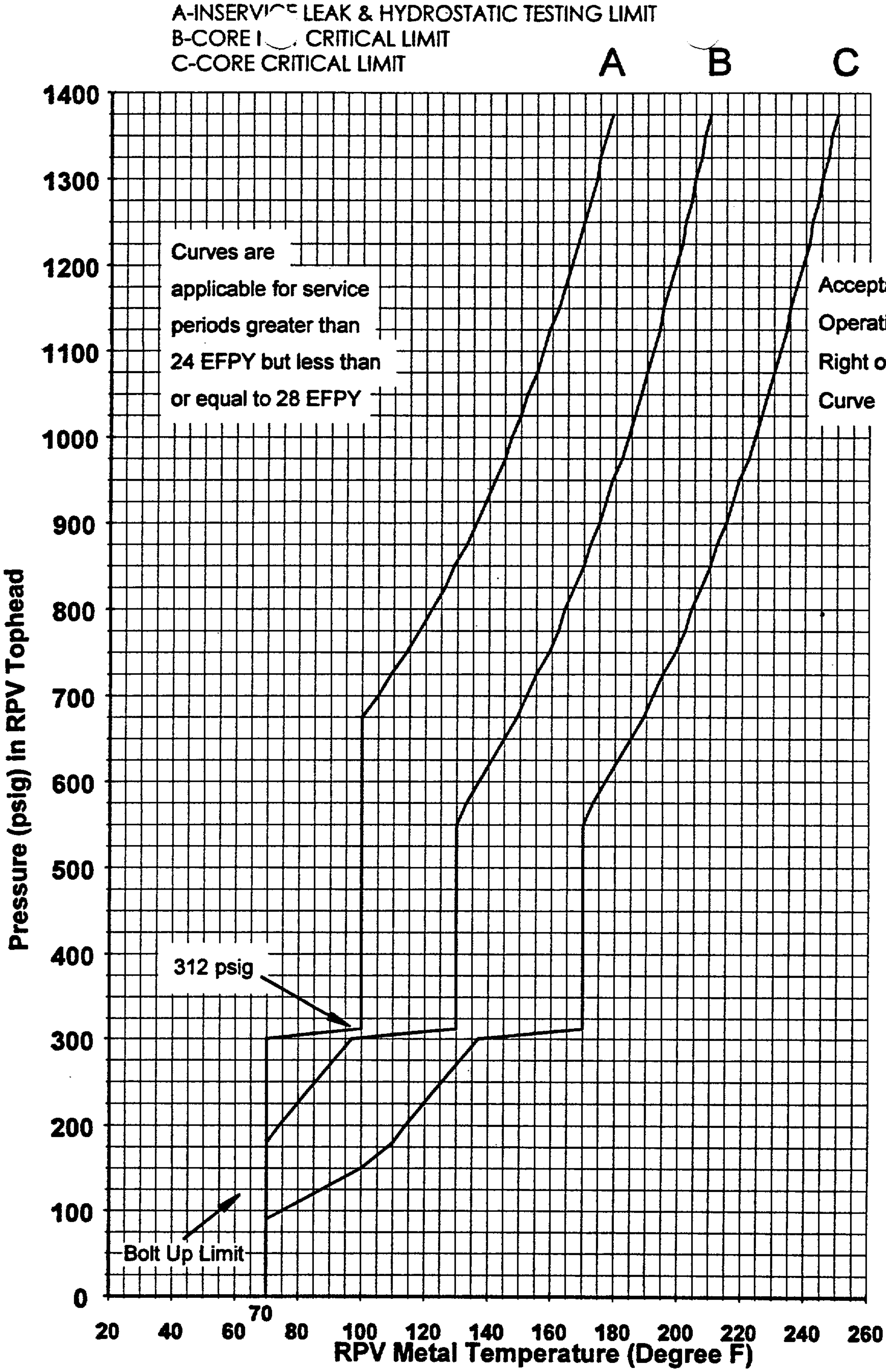
Figure 3.4.11-1 (page 1 of 5)



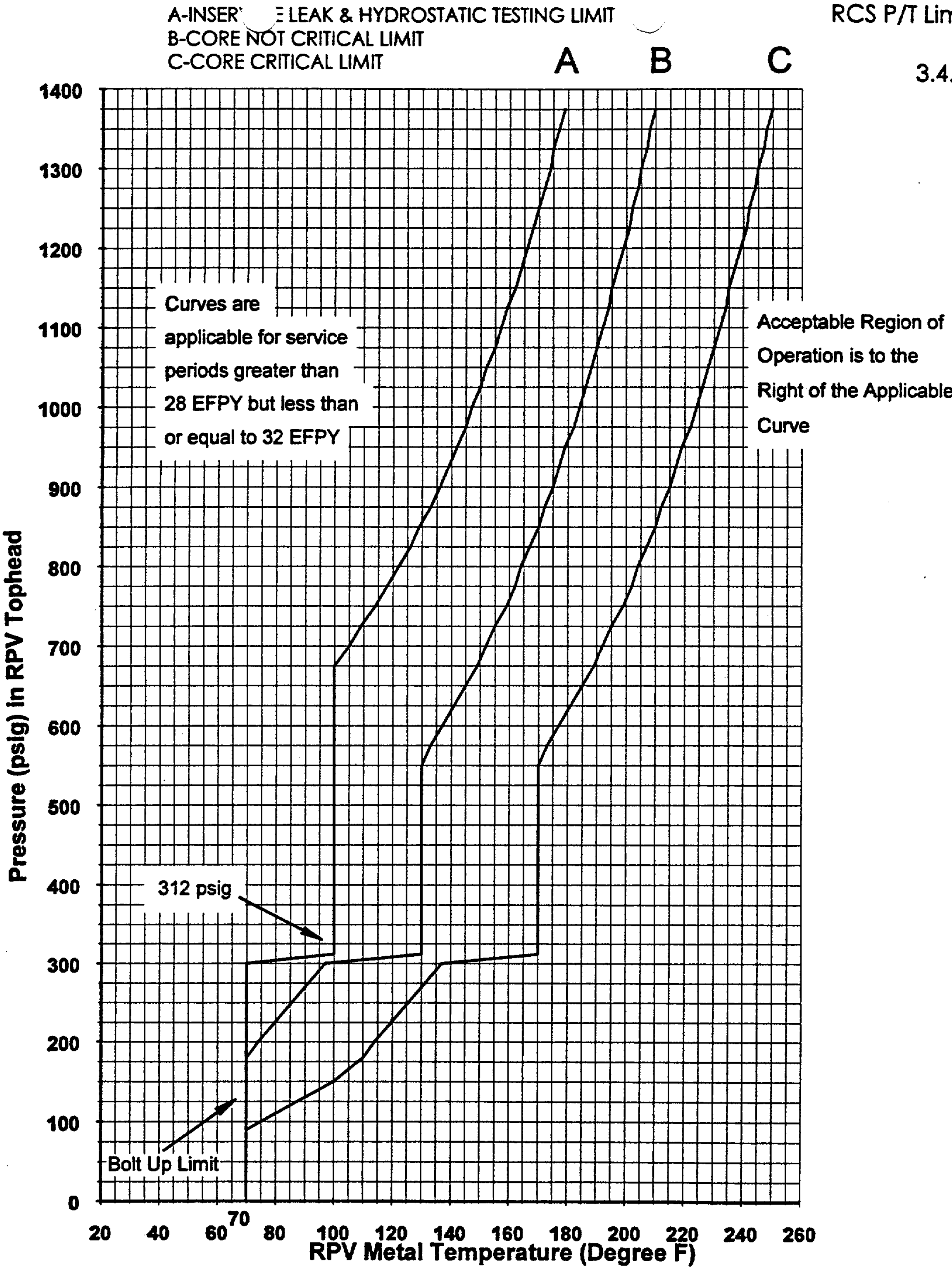
Minimum Reactor Vessel Metal Temperature vs Reactor Vessel Pressure
Figure 3.4.11-1 (page 2 of 5)



Minimum Reactor Vessel Metal Temperature vs Reactor Vessel Pressure
Figure 3.4.11-1 (page 3 of 5)



Minimum Reactor Vessel Metal Temperature vs Reactor Vessel Pressure
Figure 3.4.11-1 (page 4 of 5)



Minimum Reactor Vessel Metal Temperature vs Reactor Vessel Pressure
Figure 3.4.11-1 (page 5 of 5)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Reactor Steam Dome Pressure

LCO 3.4.12 The reactor steam dome pressure shall be ≤ 1045 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.12.1 Verify reactor steam dome pressure is ≤ 1045 psig.	12 hours

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 132 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC., ET AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By application dated October 22, 1996, as supplemented by letter dated June 26, 1997 (References 1 and 2), Entergy Operations, Inc. (the licensee) proposed to amend Figure 3.4.11-1, "Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure," and Surveillance Requirements (SR) 3.4.11.1 and 3.4.11.2 of the Technical Specifications (TSs) for Grand Gulf Nuclear Station, Unit 1 (GGNS). The proposed changes to the pressure temperature (P-T) limits in Figure 3.4.11-1 are to extend the validity of the P-T limit curves from 10 effective full power years (EFPY) to 32 EFPY in an initial increment of 6 EFPY, followed by constant increments of 4 EFPY (i.e., Figure 3.4.11-1 will be 5 figures). For each EPFY increment, there is a set of P-T limits in Figure 3.4.11-1 valid for the specified range of EFPY. Further, paragraphs under Bases and Surveillance Requirements in the TSs were revised by the licensee so that they will be consistent with the proposed P-T limits.

Reference 2 was submitted by the licensee in response to the request for additional information from the staff in the letter of April 10, 1997.

The staff evaluated the proposed P-T limits based on the following NRC regulations and criteria:

- 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"
- Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," dated July 12, 1988
- GL 92-01, Revision 1 and Supplement 1, "Reactor Vessel Structural Integrity," dated March 6, 1992, and May 19, 1995
- Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2), "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials"
- Standard Review Plan (SRP) Section 5.3.2, "Pressure-Temperature Limits"

GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1 requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit submittals for TSs. Appendix G to 10 CFR Part 50 requires that P-T limits for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME) Code.

SRP 5.3.2 provides an acceptable method of calculating the P-T limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

The Appendix G, ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or RT_{NDT}) and the Charpy USE at the maximum postulated flaw depth. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term. The ΔRT_{NDT} is a product of a chemistry factor and a neutron fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2 or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2 or surveillance data. The margin term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. RG 1.99, Rev. 2 describes the methodology to be used in calculating the margin term.

2.0 <u>EVALUATION</u>

The staff evaluated the effect of neutron irradiation embrittlement on the beltline materials in the reactor vessels of Grand Gulf. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 32 EFPY is the number (#) 2 shell axial welds, which was fabricated with weld wire heat number 627260, with 0.06% copper (Cu), 1.08% nickel (Ni), and an initial RT_{NDT} of -30°F. The ART calculated by the staff is 57.6°F at 1/4T for the limiting material. The inner-diameter (ID) neutron fluence used in the calculation is 0.25E19 n/cm^2 for 32 EFPY. This ART is the same as that calculated by the licensee. Substituting the ART of 57.6°F into equations in SRP 5.3.2, the staff verified that the proposed P-T limits (for 32 EFPY) for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

The licensee's fluence of 0.25E19 n/cm^2 for 32 EFPY is not consistent with the corresponding value of 0.311E19 n/cm^2 in NRC's reactor vessel integrity database (RVID). The licensee clarified in Reference 2 that the higher fluence value is the upper-bound fluence and does not need to be used in the P-T limits calculation. The licensee's P-T limits consist of 5 sets of curves, with EFPY values varying from 16 to 32 in a constant increment of 4 EFPY. The staff found that a linear relationship was assumed by the licensee to obtain ID fluence values for EFPYs less than 32. For additional assurance, the staff also verified the P-T limits for 16 EFPY in a manner similar to what is described above.

The major portion of the proposed P-T limit curves is defined by the beltline limiting material. However, certain part of the P-T limit curves is either bottom head or feedwater nozzle controlled. The methodology for these non-beltline regions was documented in Reference 2, and was found to be consistent with that used to support the P-T limits of Clinton (1996), Oyster Creek (1997), and Hatch (1997) nuclear power plants. This methodology was based on a finite element analysis of a generic BWR (Boiling Water Reactor) Type 6 vessel for an applied pressure of 1593 psig for the bottom head, and for the most severe transient with cold 40°F feedwater injection at normal operating condition (551.4°F and 1050 psig) for the feedwater nozzle. GGNS is a BWR Type 6 plant. The results from the generic analysis were then adjusted according to the plant-specific initial RT_{NDT} value of the bottom head or feedwater nozzle material. The staff determines that this methodology is consistent with WRC Bulletin 175 and the Appendix G of the ASME Code, and is acceptable to the staff.

Appendix G of 10 CFR Part 50 also imposes P-T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states, "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure." The staff examined the current and the proposed P-T limits, and found that this portion of the curves (straight line segments) remains the same. Therefore, the staff determined that the proposed P-T limits satisfy the requirements in Sections IV.A.2 and IV.A.3 of Appendix G.

Appendix G also requires that the predicted USE at end-of-license (EOL) for vessel beltline materials be above 50 ft-lb or that licensees demonstrate that lower values of Charpy USE will provide margins of safety equivalent to those required by Appendix G of Section XI of the ASME Code. The EOL USE for the limiting material (#2 shell axial welds with heat number 5P6214B) is 79 ft-lb, and satisfies the USE requirement of Appendix G.

3.0 CONCLUSIONS

The staff has reviewed the information provided by the licensee in its letters of October 22, 1996, and June 26, 1997 (References 1 and 2). The staff has determined that the licensee used methodologies consistent with Appendix G to 10 CFR Part 50, Appendix G to Section XI of the ASME Code, and SRP Section 5.3.2. Based on this determination, the licensee's proposed sets of P-T limits for heatup, cooldown, leak test, and criticality, which are valid for their specified EFPY ranges, may be incorporated into the TSs for GGNS. Further, the proposed bases and surveillance requirements are consistent with the proposed P-T limits. Therefore, the proposed changes to the TSs are acceptable.

4.0 REFERENCES

1. October 22, 1996, letter from J. J. Hagan, (Entergy) to USNRC Document Control Desk, subject: "Grand Gulf Nuclear Station - Pressure-Temperature Limit Curves; Proposed Amendment to the Operating License." (GNRO-96/00120)

2. June 26, 1997, letter from W. K. Hughey, (Entergy) to USNRC Document Control Desk, subject: "Grand Gulf Nuclear Station - Responses to NRC Questions Requested in a NRC Letter Dated April 10, 1997, Related to Pressure-Temperature Limit Curves." (GNRO-97/00058)

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding

(62 FR 8797). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: S. Sheng

Date: August 27, 1997