March 25, 1991

Docket No. 50-416

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Mr. William T. Cottle
Vice President, Operations GGNS
Entergy Operations, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

Dear Mr. Cottle:

SUBJECT: ISSUANCE OF AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1, REGARDING REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS (TAC NO. 76831)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 75 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated April 26, 1990, as supplemented November 30, 1990.

The amendment changes the Technical Specifications (TS) by revising Figure 3.4.6.1-1 "Minimum Reactor Pressure Vessel Metal Temperature vs. Reactor Vessel Pressure" and associated TS Bases and Surveillance Requirements to reflect the revised methodology of Regulatory Guide 1.99, Revision 2, and revised neutron fluence values for the reactor vessel wall. The revised neutron fluence values were based on an analysis of flux wire dosimeters removed from the reactor during the first refueling outage. The revised pressure/temperature (P/T) limits are applicable for service periods up to 10 effective full power years (EFPY) instead of the presently specified 32 EFPY. You are requested to submit a proposed license amendment providing P/T limits for operation beyond 10 EFPY at least 1 year prior to the time the amendment is required to allow appropriate scheduling of the staff review.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly <u>Federal</u> <u>Register</u> notice.

Sincerely,

Original Signed By:

Lester L. Kintner, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 75 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:PD4-1/LA	:PD4-1/PM	:OGC	:PD4-1/D	:	:	:
NAME	:PNoonan 3/5/91	:LKintner:lh	: RAS w/note slip	:TQuay TPQ	:	:	:
DATE	:2/27/91	:3/5/91	:3/11/91	:3/19/91	:	:	:

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

March 25, 1991

Docket No. 50-416

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Vice President, Operations GGNS
Entergy Operations, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

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Sincerely,

Lester L. Kintner

Lester L. Kintner, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 75 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

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Grand Gulf Nuclear Station

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

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

MISSISSIPPI POWER AND LIGHT COMPANY

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 75
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 April 26, 1990, as supplemented November 30, 1990, 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. 75 , 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

Theodore R. Quay

Theodore R. Quay, Director
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 25, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 75

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. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
3/4 4-19	3/4 4-19
3/4 4-20	3/4 4-20
3/4 4-21	3/4 4-21
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system pressure and reactor vessel metal temperature shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum reactor coolant heatup of 100°F in any one hour period,

b. A maximum reactor coolant cooldown of 100°F in any one hour period,

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 70°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and the reactor coolant system pressure and reactor vessel metal temperature shall be determined to be to the right of the limit lines of Figure 3.4.6.1-1 curves A or B, as applicable, at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system pressure and reactor vessel metal temperature shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. $\leq$ 100°F, at least once per 12 hours.

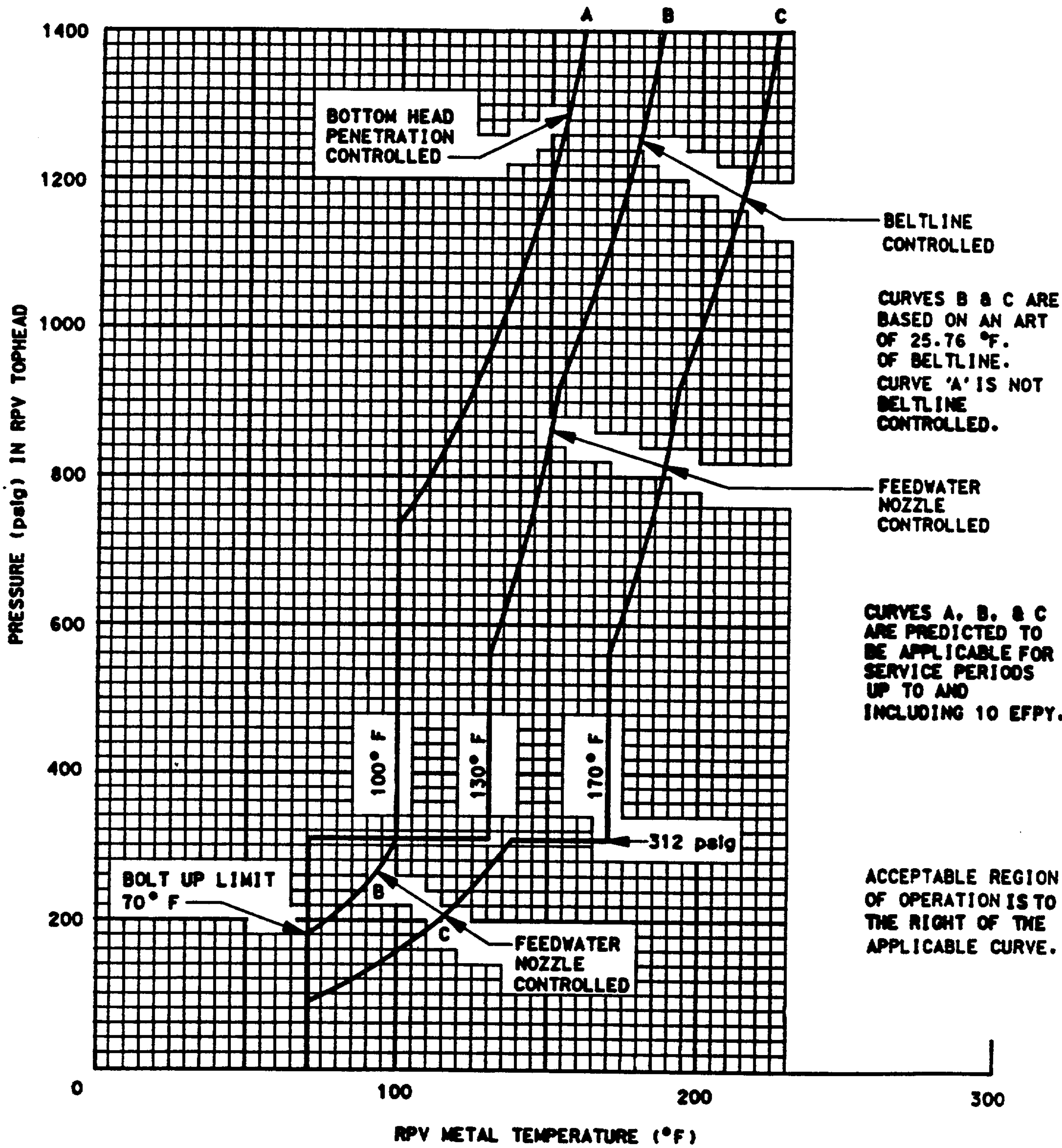
   2. $\leq$ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

4.4.6.1.4 The reactor vessel material specimens shall be removed and examined as a function of time and THERMAL POWER as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1.

4.4.6.1.5 The pressure-temperature limit curves in Figure 3.4.6.1-1 are valid through 10 effective full power years (EFPY) and shall be re-evaluated prior to exceeding 10EFPY.

A - INSERVICE LEAK AND HYDROTEST
B - NON-NUCLEAR HEAT UP & COOLDOWN LIMIT
C - NUCLEAR (CORE CRITICAL) HEAT UP & COOLDOWN LIMIT



MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

FIGURE 3.4.6.1-1

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER		VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1.	131C8981G1-N01	3°	0.46	8
2.	131C8981G1-N02	177°	0.46	24
3.	131C8981G1-N03	183°	0.46	Spare

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 are restricted to no more than 48 consecutive hours.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment.

3/4.4.5 SPECIFIC ACTIVITY (Continued)

The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The RT_{NDT} for welds and base material in the closure flange region is $\leq$ 10°F. The initial hydrostatic test pressure was 1563 psig. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 Mev, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature (ART), based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." Bases Figure B 3/4.4.6-1 has been revised to reflect the analysis of the flux wire dosimeter which was removed during the first refueling outage. The upper bound curve in Figure B 3/4.4.6-1 was used in determining the pressure/temperature limit curves in Figure 3.4.6.1-1. The pressure/temperature limit curves in Figure 3.4.6.1-1 include predicted adjustments for this shift in RT_{NDT} for 10 effective full power years (EFPY) of exposure and shall be adjusted, if required for longer service periods.

PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating in accordance with ASTM E185-73 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and A, for reactor criticality and for inservice leak and hydrostatic testing, have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition, and Addenda through Summer 1978.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication; however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

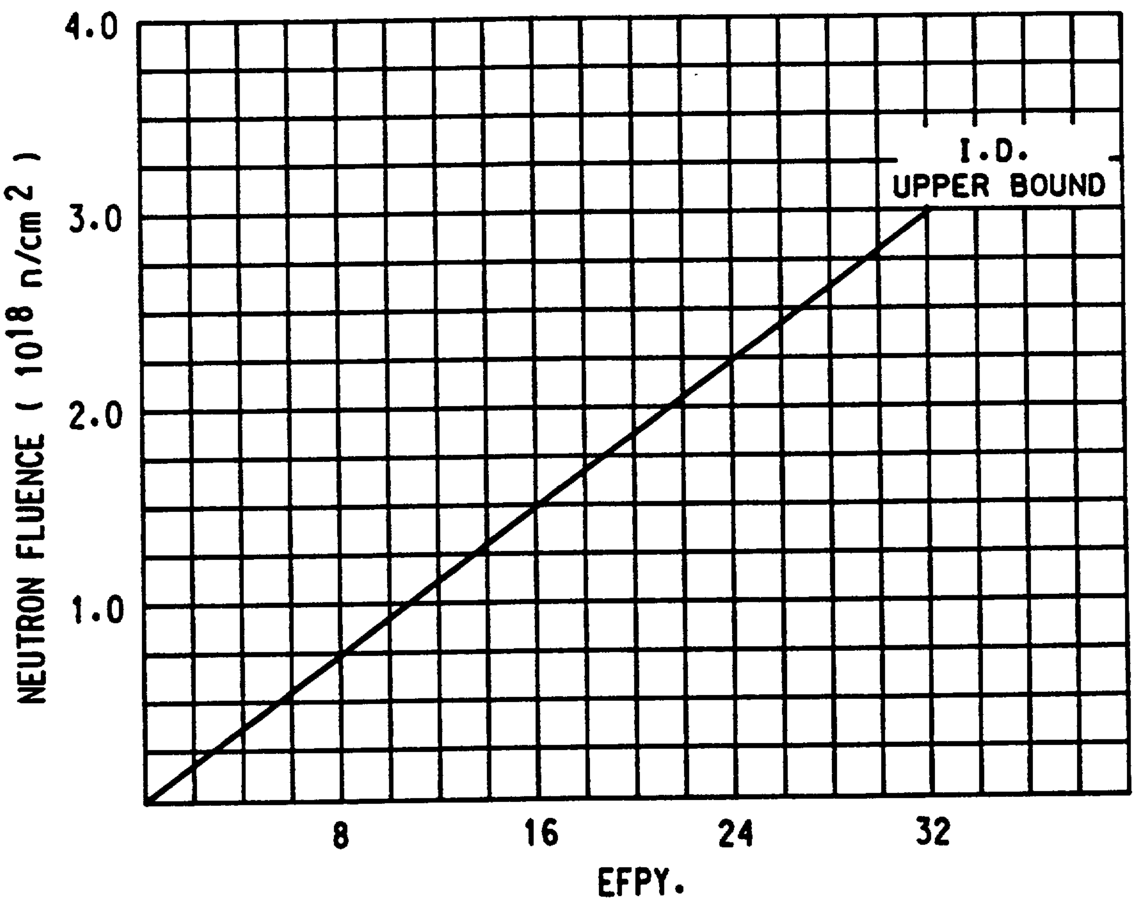
Beltline Component	Weld Seam I.D. or Material Type	Heat No.-Slab No. or Heat No./Lot No.	Cu %	Ni (%)	Starting RT_{NDT} (°F)	Maximum* ΔRT_{NDT} (°F)	Minimum Upper Shelf (ft-lb)	**Maximum ART (°F) at 10EFPY
Plate	SA-533 Gr.B, CL.1 SA-533 Gr.B, CL.1	C2594-2	0.04	0.63	0	8.84	96 (C2594-2)	17.7
Weld	#2 Shell Long. Seams	627260/B322A27AE	0.06	1.08	-30	27.88	88	25.76 (limiting)

Non-Beltline Component	Material Type or Weld Seam I.D.	Heat No.-Slab No. or Heat No./Lot No.	Highest Starting RT_{NDT} (°F)
Shell Ring	SA-533 Gr.B, CL.1	C2815-2, C2779-2, C2779-1, C2788-2, C2788-1, C2741-1	+10
Bottom Head Dollar Plate	SA-533 Gr.B, CL.1	A1113-1 C2630-2	0
Bottom Head Radial Plates	SA-533 Gr.B, CL.1	C2539-2, A1145-1	+10
Top Head Dollar Plate	SA-533 Gr.B, CL.1	C2448-3	-30
Top Head Side Plates	SA-533 Gr.B, CL.1	C2944-1	+10
Top Head Flange	SA-508 CL.2	48D1682	-30
Vessel Flange	SA-508 CL.2	48D1141	-30
Feedwater Nozzle	SA-508 CL.2	Forging No. 249A-1, 2, 3, 4, 5, & 6, Q2Q65W	-20
Weld	N/A	N/A	-20***
Closure Stud	SA-540 Gr.B24	84025, 84299	+10

*ΔRT_{NDT} computed based on Regulatory Guide 1.99 Revision 2

**ΔART for 10EFPY calculated based on Regulatory Guide 1.99 Revision 2

***Based on purchase spec. requirements.



BASES FIGURE B 3/4.4.6.1 FAST NEUTRON FLUENCE (E>1MeV) AT VESSEL I.D. AS A FUNCTION OF EFPY.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated April 26, 1990, as supplemented November 30, 1990, the licensee (System Energy Resources, Inc., before June 6, 1990, and Entergy Operations, Inc., on or after June 6, 1990), submitted a request for changes to the Grand Gulf Nuclear Station, Unit 1, Technical Specifications (TS). The staff recommended, and licensee agreed, to editorial changes in the proposed surveillance requirement and Bases for consistency in capitalization, acronyms and phraseology.

The proposed amendment would change the TS by revising Figure 3.4.6.1-1 "Minimum Reactor Pressure Vessel Metal Temperature vs. Reactor Vessel Pressure" and associated TS Bases and Surveillance Requirements to reflect the revised methodology of Regulatory Guide 1.99, Revision 2, and revised neutron fluence values for the reactor vessel wall. The revised neutron fluence values were based on an analysis of flux wire dosimeters removed from the reactor during the first refueling outage. The revised pressure/temperature (P/T) limits are applicable for service periods up to 10 effective full power years (EFPY) instead of the presently specified 32 EFPY.

The licensee previously submitted P/T limit calculations using Regulatory Guide 1.99, Revision 2 to fulfill the requirements of Generic Letter 88-11 on January 9, and February 28, 1989. The staff approved the licensee's calculations and requested the licensee to submit the actual Technical Specification changes in a letter dated February 21, 1990. The proposed P/T limits in this submittal are similar to the limits in the previous submittal except that the licensee changed the P/T limits slightly to include the revised neutron fluence values and to reflect the effect of the revised neutron fluence on the limits.

2.0 EVALUATION

2.1 Revised Neutron Fluence in Reactor Vessel

The estimate of the 10.0 EFPY peak azimuthal fluence to the pressure vessel is based on an end of Cycle 1 flux wire measurement from a flux wire dosimeter attached to the surveillance capsule at 3° on the azimuthal. The first cycle lasted for 0.93 EFPYs. The fluence measurement was based on the measured decay rate of the reaction Fe-54(n,p)Mn-54. This is the most common reaction for this type of measurement. The Mn-54 half life is 312.2 days and is therefore suitable for a one cycle irradiation. However, the actual irradiation time in this instance was about 3 years. The analysis accounted for the measurement uncertainties which were estimated at ±25% (2 sigma). The modeling of the flux attenuation was based on the two-dimensional DOT neutron transport code in (R,O) geometry. One-eighth core geometry was used with 26 energy groups, P_3 angular scattering expansion and S_8 quadrature approximation. This calculation was primarily used for the estimation of the azimuthal flux distribution required to translate the 3° wire flux measurement to the peak flux at 45°. The 0.93 EFPYs of the first cycle fluence was then scaled to 10 and 32 EFPYs at the clad-vessel interface. Finally, the method suggested in Regulatory Guide (RG) 1.99, Rev. 2, was used to estimate the corresponding values at ¼T, i.e. one quarter thickness of the pressure vessel.

We have reviewed the information submitted by letter dated November 30, 1990, regarding the determination of the pressure vessel fluence at the peak azimuthal location ¼T thickness at 10 EFPYs, required for the determination of the P/T limit curves. We find the measurement of the flux to be acceptable, because it used the standard reaction for similar determinations and has a reasonable uncertainty associated with it. The translation of that measurement was made with the DOT code using acceptable approximations. The estimation of the ¼T thickness fluence from the inner diameter value was performed using the RG 1.99, Rev. 2 method. Finally, the value proposed was the upper limit rather than the nominal estimate which is very conservative. Therefore, we find the proposed fluence of 2.1x10+18 n/cm^2 end-of-life at ¼T, peak azimuthal location with E greater than 1.0 MeV, acceptable.

2.2 Revised Pressure/Temperature Limit Curves

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Grand Gulf reactor vessel. 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 adjusted reference temperature (ART) at 10 EFPY at 1/4T (T = reactor vessel beltline thickness) was the longitudinal weld 627260 with 0.06% copper (Cu), 1.08% nickel (Ni), and an initial reference temperature (RT_{NDT}) of -30°F. The material with the highest ART at 10 EFPY at 3/4T was plate C2594-2 with 0.04% Cu, 0.63% Ni, and an initial RT_{NDT} of 0°F.

The licensee has not removed any surveillance capsules from the Grand Gulf reactor vessel; however, a flux wire dosimeter was removed during the first refueling outage and used to recalculate neutron fluence (Section 2.1). For the limiting beltline material at 1/4T, longitudinal weld 627260, the staff calculated the ART to be 25.2°F at 10 EFPY. For the limiting beltline material at 3/4T, longitudinal plate C2594-2, the staff calculated the ART to be 11.2°F at 10 EFPY. Based on the licensee revised neutron fluence of 2.1E18 n/cm^2 at end of life, the staff calculated a fluence of 6.5E17 n/cm^2 at 1/4T and 2.9E17 n/cm^2 at 3/4T for the 10 EFPY P/T limits.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 25.76°F at 10 EFPY at 1/4T for the same limiting weld metal. Substituting the staff calculated ART of 25.2°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50 except that the cooldown temperature at 1000 psi is about 14°F lower (less conservative) than the staff's calculated temperature. The staff judges that a difference of 14°F in the high pressure region of the P/T limits is acceptable in this particular case because at the lower pressure region of the cooldown limit curve, the licensee's temperatures are conservative.

In addition to beltline materials, 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.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 pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of -30°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The material with the lowest initial USE is the girth weld with an initial USE of 79 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of the girth weld metal at the end of life will be approximately 68 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 10 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The limits also satisfy Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Grand Gulf 1 Technical Specifications.

The staff recommends that in the future the licensee should use the procedures in NUREG-0800, Standard Review Plan (SRP) Section 5.3.2 to verify its P/T limits because, for the cooldown limits, the SRP 5.3.2 calculation shows a higher (more conservative) temperature at 1000 psi than the licensee's calculated temperature.

3.0 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 comment.

4.0 ENVIRONMENTAL CONSIDERATION

This 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 the 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 off site, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 6873). Accordingly, this 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 this amendment.

5.0 CONCLUSION

The staff 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 this amendment will not be inimical to the common defense and the security, or to the health and safety of the public.

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