

March 31, 1987

Docket No.: 50-416

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Mr. Oliver D. Kingsley, Jr.
Vice President, Nuclear Operations
System Energy Resources, Inc.
Post Office Box 23054
Jackson, Mississippi 39205

Dear Mr. Kingsley:

SUBJECT: CHANGES TO TECHNICAL SPECIFICATIONS REGARDING RPV PRESSURE AND TEMPERATURE LIMITS

RE: GRAND GULF NUCLEAR STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 32 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 11, 1986, as revised January 20, 1987.

This amendment changes the Technical Specifications (TSs) and associated Bases for the reactor pressure vessel (RPV) pressure and temperature limits to be consistent with the limits provided by the vendor for the nuclear steam supply system, General Electric Company.

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

Sincerely,

/s/

Lester L. Kintner, Project Manager
BWR Project Directorate No. 4
Division of BWR Licensing

Enclosures:

1. Amendment No. 32 to License No. NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

MO'Brien
3/24/87

PD#4/PM (C.S.N)
JUnda
3/24/87

LKintner:lb
3/24/87

~~EBXBC
RHermann
3/24/87~~
not needed

OGC
Young
3/25/87

PD#4/D
WButler
3/31/87



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

March 31, 1987

Docket No.: 50-416

Mr. Oliver D. Kingsley, Jr.
Vice President, Nuclear Operations
System Energy Resources, Inc.
Post Office Box 23054
Jackson, Mississippi 39205

Dear Mr. Kingsley:

SUBJECT: CHANGES TO TECHNICAL SPECIFICATIONS REGARDING RPV PRESSURE AND
TEMPERATURE LIMITS

RE: GRAND GULF NUCLEAR STATION, UNIT 1

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

Sincerely,

A handwritten signature in cursive script that reads "L L Kintner".

Lester L. Kintner, Project Manager
BWR Project Directorate No. 4
Division of BWR Licensing

Enclosures:

1. Amendment No. 32 to
License No. NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Oliver D. Kingsley, Jr.
System Energy Resources, Inc.

Grand Gulf Nuclear Station (GGNS)

cc:

Mr. Ted H. Cloninger
Vice President, Nuclear Engineering
and Support
System Energy Resources, Inc.
Post Office Box 23054
Jackson, Mississippi 39205

Mr. C. R. Hutchinson
GGNS General Manager
System Energy Resources, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

Robert B. McGehee, Esquire
Wise, Carter, Child, Steen and Caraway
P.O. Box 651
Jackson, Mississippi 39205

The Honorable William J. Guste, Jr.
Attorney General
Department of Justice
State of Louisiana
Baton Rouge, Louisiana 70804

Nicholas S. Reynolds, Esquire
Bishop, Liberman, Cook, Purcell
and Reynolds
1200 17th Street, N.W.
Washington, D. C. 20036

Office of the Governor
State of Mississippi
Jackson, Mississippi 39201

Mr. Ralph T. Lally
Manager of Quality Assurance
Middle South Utilities System
Services, Inc.
P.O. Box 61000
New Orleans, Louisiana 70161

Attorney General
Gartin Building
Jackson, Mississippi 39205

Mr. John G. Cesare
Director, Nuclear Licensing and Safety
System Energy Resources, Inc.
P.O. Box 23054
Jackson, Mississippi 39205

Mr. Jack McMillan, Director
Division of Solid Waste Management
Mississippi Department of Natural
Resources
Bureau of Pollution Control
Post Office Box 10385
Jackson, Mississippi 39209

Mr. R. W. Jackson, Project Engineer
Bechtel Power Corporation
15740 Shady Grove Road
Gaithersburg, Maryland 20877-1454

Alton B. Cobb, M.D.
State Health Officer
State Board of Health
P.O. Box 1700
Jackson, Mississippi 39205

Mr. Ross C. Butcher
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 399
Port Gibson, Mississippi 39150

President
Claiborne County Board of Supervisors
Port Gibson, Mississippi 39150

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

Mr. James E. Cross
GGNS Site Director
System Energy Resources, Inc.
P.O. Box 756
Port Gibson, Mississippi 39150



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

MISSISSIPPI POWER & LIGHT COMPANY
SYSTEM ENERGY RESOURCES, INC.
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION
DOCKET NO. 50-416
GRAND GULF NUCLEAR STATION, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. NPF-29

1. The Nuclear Regulatory Commission (the Commission) has found that
 - A. The application for amendment by Mississippi Power & Light Company, System Energy Resources, Inc. (formerly Middle South Energy, Inc.) and South Mississippi Electric Power Association, (the licensees) dated November 11, 1986 as revised January 20, 1987, 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:

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 32, are hereby incorporated into this license. System Energy Resources, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

8704060431 870331
PDR ADDCK 05000416
P PDR

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

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 31, 1987

PD#4/D
M O'Brien
3/24/87

PD#4/PM
L Kintner:lb
3/24/87

CGC
3/24/87

PD#4/D
W Butler
3/31/87

WB

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

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 31, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 32

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. Overleaf page(s) provided to maintain document completeness.*

Remove

3/4 4-21
3/4 4-22

B 3/4 4-3
B 3/4 4-4

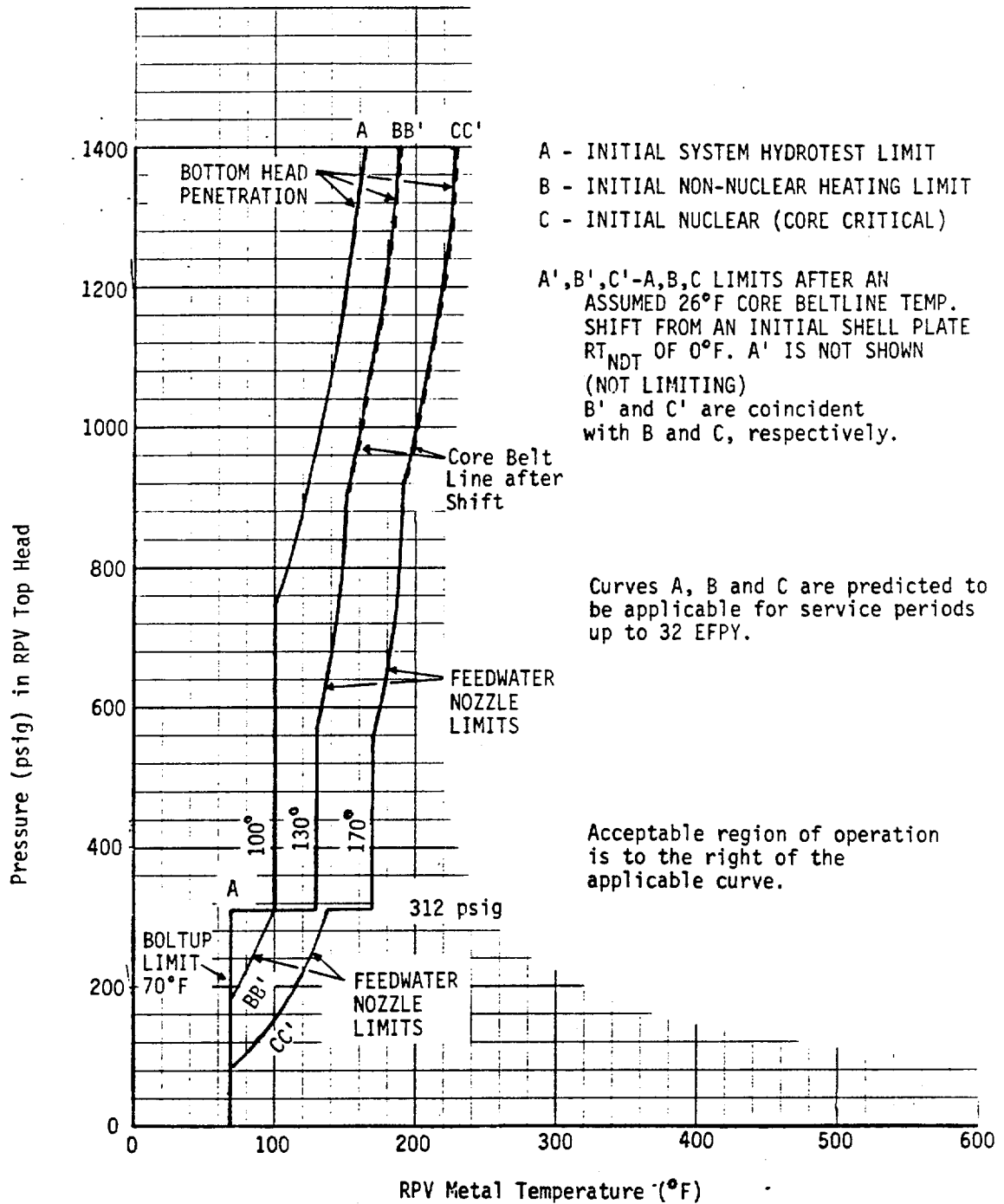
B 3/4 4-5
B 3/4 4-6*

Insert

3/4 4-21
3/4 4-22*

B 3/4 4-3*
B 3/4 4-4

B 3/4 4-5
B 3/4 4-6*



MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

Figure 3.4.6.1-1

TABLE 4.4.6.1.3-1REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

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

REACTOR COOLANT SYSTEM

BASES

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 must be restricted to no more than 800 hours per year, approximately 10 percent of the unit's yearly operating time, since these activity levels increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam line rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

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.

REACTOR COOLANT SYSTEM

BASES

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 $< 10^{\circ}F$. 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, based upon the fluence, phosphorus 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 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/temperature limit curve, Figure 3.4.6.1-1, curves A', B' and C', includes predicted adjustments for this shift in RT_{NDT} for the end of life fluence. Curves B' and C' are coincident with curves B and C, respectively.

REACTOR COOLANT SYSTEM

BASES

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 1.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, 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 %	P (%)	Starting	Maximum**	Minimum	Maximum EOL
					RT _{NDT} (°F)	ΔRT _{NDT} (°F)	Upper Shelf (ft-lb)	RT _{NDT} (°F)
Plate	SA-533 Gr.B, CL.1 SA-533 Gr.B, CL.1	C2594-2	0.04	0.012	0	+26	96 (C2594-2)	+26*
Weld	#2 Shell Long. Seams	627260/B322A27AE	0.06	0.020	-30	44	N/A	+14
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					

*Combination of the highest starting RT_{NDT} plate and the highest ΔRT_{NDT} plate.

**These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}.

***Based on purchase spec. requirements.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 32 TO FACILITY OPERATING LICENSE NO. NPF-29

MISSISSIPPI POWER & LIGHT COMPANY

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated November 11, 1986 as revised January 20, 1987, Mississippi Power & Light Company, (the licensee)* requested an amendment to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GGNS-1). The proposed amendment would change the Technical Specifications (TSs) and associated Bases for the reactor pressure vessel (RPV) pressure and temperature limits to be consistent with the limits provided by the vendor for the nuclear steam supply system, (NSSS), General Electric Company.

2.0 EVALUATION

The present RPV pressure and temperature limits in the Technical Specifications (Figure 3.4.6.1-1 "Minimum Reactor Pressure Vessel Metal Temperature vs. Reactor Vessel Pressure") were included in the full power operating license for GGNS-1 issued November 1, 1984. During a recent review of changes to the Final Safety Analysis Report (FSAR), the licensee found that the RPV limit curves in the updated FSAR (which are identical to TS Figure 3.4.6.1-1) did not correspond to the correct RPV limit curves provided by the NSSS vendor, the General Electric Company (GE). The limit curves affected are the curves below 312 psig for non-nuclear heating (Curve B) and nuclear heating with the core critical (Curve C). The system hydrotest limit (Curve A) is not affected by the error.

The present erroneous curves are non-conservative with respect to the correct GE supplied curves. The licensee stated that a GE analysis has concluded that while operation with the present curves does not significantly impact plant safety, the limit curves in the TSs should be corrected.

*On December 20, 1986, the Commission issued License Amendment No. 27 which authorized the transfer of control and performance of licensed activities from Mississippi Power & Light Company to System Energy Resources, Inc. (SERI). "The licensee" refers to Mississippi Power & Light Company before December 20, 1986 and to SERI on or after December 20, 1986.

The basis for this conclusion is that for a normal heatup and pressurization, the GGNS-1 operating procedures result in plant conditions which avoid the restricted area on the proposed correct curves. A review of plant records by the licensee confirmed that the plant has not operated in the proposed new restricted areas of the limit curves. Furthermore, the licensee has stated that operating procedures for GGNS-1 were changed to use the GE supplied curves until TSs are changed to incorporate them. The staff has reviewed the submittals for this amendment and concludes that the GE supplied limit curves are the correct curves and therefore, the proposed changes to the TSs are acceptable.

In addition to the changes to Curves B and C on TS Figure 3.4.6.1-1, the reference to NEDO-21778-A in Figure 3.4.6.1-1 and in Bases 3/4.4.6 would be deleted because Appendix G to 10 CFR 50 supersedes NEDO-21778-A as a basis for Curve C. Another change to the Bases is to delete the statement "as well as adjustments for possible errors in the pressure and temperature sensing instruments" because errors in RPV pressure and temperature measurements were not included in the calculation of limit curves A', B', and C'. The staff agrees with the deletion of NEDO-21778-A and reference to measurement errors for the reasons stated above.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. 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.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (52 FR 5863) on February 26, 1987, and consulted with the state of Mississippi. No public comments were received, and the state of Mississippi did not have any comments.

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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and the security nor to the health and safety of the public.

Principal Contributor: Felix Litton, Engineering Branch, DBL

Dated: March 31, 1987