August 1, 1990

Docket No. 50-458

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Gulf States Utilities
ATTN: Mr. James C. Deddens
Senior Vice President (RBNG)
Post Office Box 220
St. Francisville, Louisiana 70775

Dear Mr. Deddens:

SUBJECT: RIVER BEND STATION, UNIT 1 - AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. 76851)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 45 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 14, 1990.

The amendment revises the "Minimum Temperature vs. Reactor Pressure" curves in TS 3/4.4.6 and the associated Bases. The changes were made in response to Generic Letter 88-11 which advised licensees to use the methods described in Revision 2 of Regulatory Guide 1.99 to predict the effect of neutron radiation on reactor vessel materials.

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

Sincerely,

Original Signed By

Claudia M. Abbate, Project Manager
Project Directorate IV-2
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 45 to NPF-47
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	PDIV-2/LA	PDIV-2/PM	OGC	PDIV-2/D			
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DATE	07/13/90	07/17/90	07/17/90	07/31/90			

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

GULF STATES UTILITIES COMPANY

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. NPF-47

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

   A. The application for amendment by Gulf States Utilities Company (the licensee) dated May 14, 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, as amended, 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 license 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-47 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 45 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. GSU shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Christopher I. Grimes

Christopher I. Grimes, Director
Project Directorate IV-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 1, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 45

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page. The revised page is identified by Amendment number and contains a vertical line indicating the area of change. The overleaf page is provided to maintain document completeness.

REMOVE	INSERT
3/4 4-23	3/4 4-23
3/4 4-24	3/4 4-24
B3/4 4-5	B3/4 4-5
B3/4 4-6	B3/4 4-6
B3/4 4-8	B3/4 4-8
B3/4 4-9	B3/4 4-9

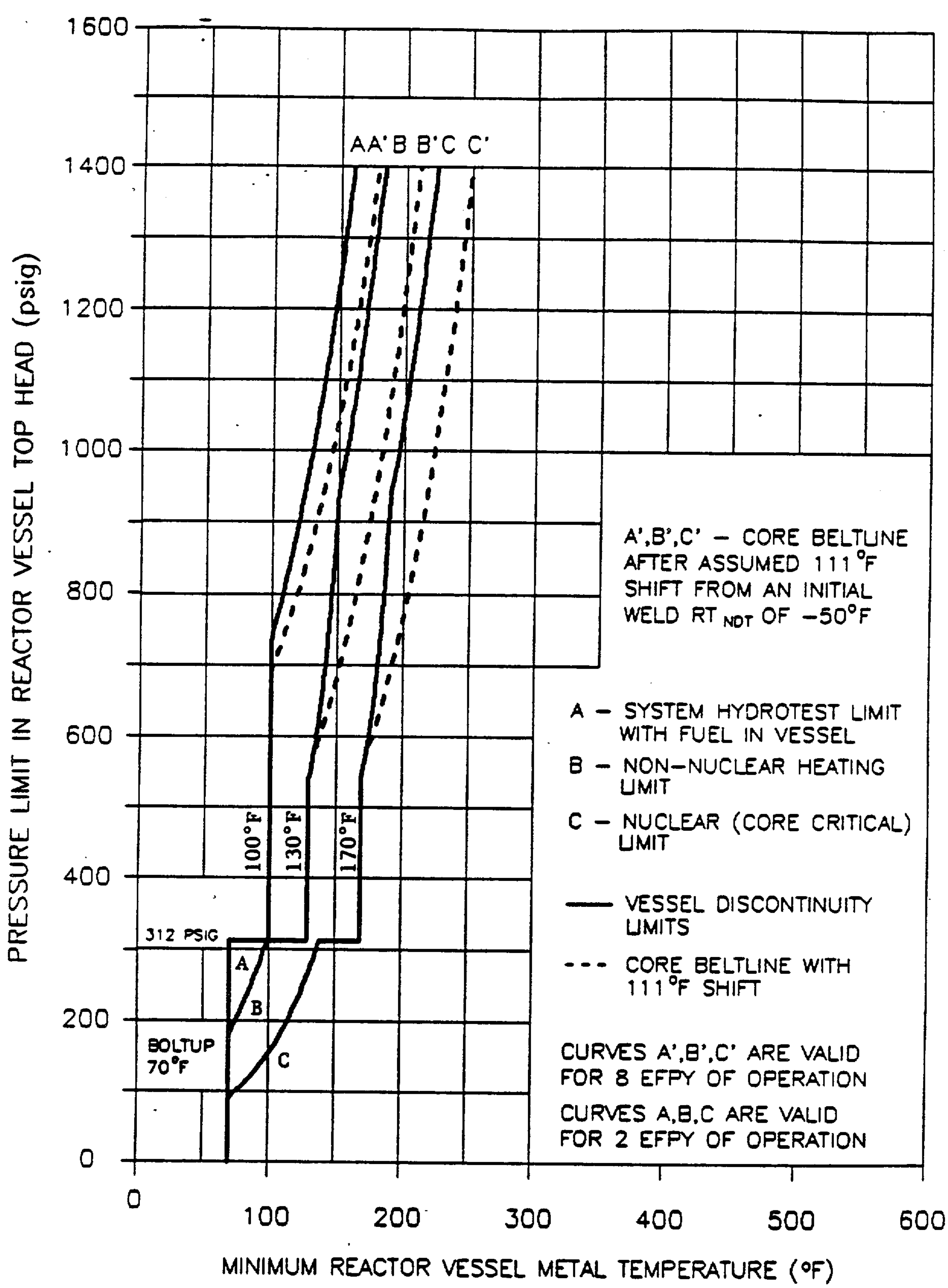


FIGURE 3.4.6.1–1
MINIMUM TEMPERATURE REQUIRED VS REACTOR PRESSURE

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR AT I.D./¼T	WITHDRAWAL TIME (EFPY)
1	3°	0.67/0.89	6
2	177°	0.67/0.89	15
3	183°	0.67/0.89	Standby

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 thermally 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 thermally 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. Consequently, 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 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, 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." 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 conditions at 8 EFPY.

PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in RT_{NDT} of the vessel material will be determined periodically during operation by removing and evaluating, in accordance with ASTM E185 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 Figure 3.4.6.1-1, curves C, and C', and A 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. However, single failure considerations require that two valves be OPERABLE. 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. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

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 adequate cooling 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.

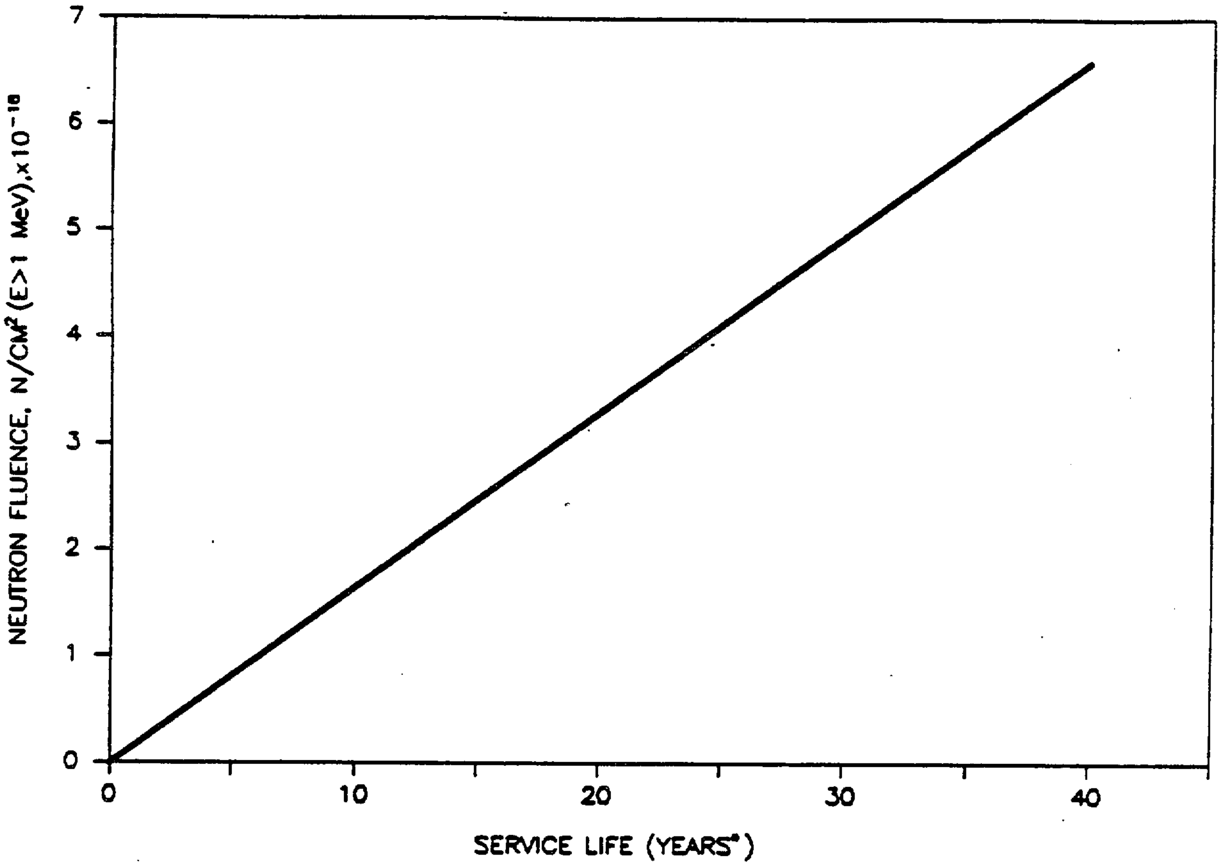
BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

LIMITING BELTLINE COMPONENT	WELD SEAM OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	Ni(%)	STARTING RT_{NDT}(°F)	ΔRT^{*}_{NDT}(°F)	AVG. UPPER SHELF (FT-LBS)	MAXIMUM RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	C3138-2	0.08	0.63	+9	75	79	84
Weld	SHELL COURSE No.2 Vertical Seam 3	5P6756/ Lot 0342	0.09	0.92	-50	153	97	103

NOTE:* These values are given only for the benefit of calculating the 32 EFPY RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT}(°F)
Shell Ring	SA 533 GrB Cl.1	ALL HEATS	+10
Bottom Head Dome	SA 533 GrB Cl.1	ALL HEATS	+10
Bottom Head Torus	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Dome	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Torus	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Flange	SA 508 Cl.2	ALL HEATS	+10
Vessel Flange	SA 508 Cl.2	ALL HEATS	+10
Feedwater Nozzle	SA 508 Cl.2	ALL HEATS	-20
Weld	LOW ALLOY STEEL	ALL HEATS	-20
Closure Studs	SA 540 GRADE B23 or B24	ALL HEATS	Meets requirement of 45 ft-lbs and 25 mils lateral expansion at + 10°F



BASES FIGURE B 3/4.4.6-1
FAST NEUTRON FLUENCE (E>1 MEV) AT VESSEL I.D.
AS A FUNCTION OF SERVICE LIFE*

* AT 90% OF RATED THERMAL POWER AND 90% AVAILABILITY

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. NPF-47

GULF STATES UTILITIES COMPANY

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

INTRODUCTION

By letter dated May 14, 1990, Gulf States Utilities Company (GSU) (the licensee) requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The proposed amendment was submitted in response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement on Reactor Vessel Materials and Its Effect on Plant Operations," and requested revisions to the pressure/temperature (P/T) limits in the River Bend Station, Unit 1 Technical Specifications, Section 3.4. The revisions also change the effectiveness of the current P/T limits from 8.8 to 8 effective full power years (EFPY). The proposed P/T limits were developed based on Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest. The proposed P/T limit curves also include a set of curves for 2 EFPY in addition to a set for 8 EFPY, but since the plant has already achieved 2.7 EFPY, the staff reviewed only the set for 8 EFPY.

To evaluate the P/T limits, the staff used the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the United States. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the River Bend reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff determined that the material with the highest ART at 8 EFPY at 1/4T (T = reactor vessel beltline thickness) was weld 5P6756/0342 with 0.09 percent Cu (copper), 0.92 percent Ni (nickel), and an initial unirradiated RT_{ndt} (nil-ductility transition reference temperature) of -50°F. At the 3/4T the limiting material was plate C3138-2 with 0.08 percent Cu, 0.63 percent Ni, and an initial unirradiated RT_{ndt} of 9°F.

The licensee is scheduled to remove the first surveillance capsule from the River Bend reactor vessel at 6 EFPY. The staff determined that all surveillance capsules contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material at 1/4T, weld 5P6756/0342, the staff calculated the ART to be 60.8°F at 8 EFPY. For the limiting beltline material at 3/4T, plate C3138-2, the staff calculated the ART to be 42.6°F. The staff used a neutron fluence of 1.2E18 n/cm^2 at 1/4T and 6.2E17 n/cm^2 at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because no surveillance capsule has been removed from the River Bend reactor pressure vessel.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 60.5°F at 8 EFPY at 1/4T for the same limiting weld metal. The staff judges that a difference of 0.3°F between the licensee's ART of 60.5°F and the staff's ART of 60.8°F is acceptable. Substituting the ART of 60.8°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.

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 percent 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. 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 10°F, the staff 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 plate C3138-2 with 79.3 ft-lb. Using Figure 2 of RG 1.99, Rev. 2, the staff predicted that the USE for plate C3138-2 after 32 EFPY would be 65.8 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 8 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Therefore, the proposed P/T limits are acceptable and may be incorporated into the River Bend Technical Specifications.

ENVIRONMENTAL CONSIDERATION

The amendment involves a change in 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 and changes in surveillance requirements. 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 exposures. 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. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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.

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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. The staff therefore concludes that the proposed changes are acceptable.

Dated: August 1, 1990

Principal Contributors: M. Huang
J. Tsao