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U. S. Nuclear Regulatory Commission
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Subject: River Bend Station
Docket No. 50-458
License No. NPF-47
Additional Information Related to License Amendment
Request (LAR) 99-22, "Request for a Revision to the Reactor
Vessel Material Surveillance Program Capsule Withdrawal Schedule"

File No.: G9.5, G9.4.2

Reference: Entergy Operations, Inc. (EOI) Letter to NRC, RBG-45151, dated October 25, 1999

RBF1-00-0006
RBG-45225

Ladies and Gentlemen:

In the referenced letter, EOI requested a license amendment to defer withdrawal of the first River Bend Station (RBS) reactor vessel surveillance capsule. An extension of the capsule withdrawal would allow time for the NRC to review an industry-proposed program for monitoring the material of boiling water reactor (BWR) vessels. Specifically, the proposed deferral for RBS capsule withdrawal was at 13.4 effective full power years (EFPY). After review and further discussion with the industry group and the NRC, EOI now proposes that the first RBS capsule be withdrawn during the tenth refueling outage (RF-10), scheduled for Fall 2001, at approximately 11.5 EFPY (see proposed change to Technical Requirements Manual enclosed).

The Boiling Water Reactor Vessel and Internals Program (BWRVIP) was organized to address reactor vessel and internals issues affecting U.S. BWRs. The BWRVIP Assessment Committee recently developed a plan to address the requirements in 10 CFR 50, Appendix H, for surveillance of reactor vessel material related to monitoring radiation embrittlement. The BWRVIP surveillance plan (BWRVIP-78) was submitted to the NRC on December 22, 1999. Based on criteria delineated in the program plan (e.g., chemistry match, excellent baseline

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data, and same fabricator), the RBS capsules were initially selected as representative of several reactor vessels and were included in the proposed schedule for withdrawal, test, and analysis under the integrated surveillance program (ISP). As discussed in the referenced request, EOI is participating in the BWRVIP and intends to participate in the ISP described in BWRVIP-78 as finally approved by the NRC and implemented by the industry participants.

The current RBS withdrawal schedule of 10.4 EFPY requires removal of the first capsule in RF-9, March 2000. A deferral to RF-10 would allow time for NRC review and approval of BWRVIP-78, and thus, allow full RBS participation in the final NRC-approved ISP. If RBS capsules are not selected for testing and analysis in the final ISP, appropriate licensing actions will be taken to revise the RBS Technical Requirements Manual withdrawal schedule (TRM Table 3.4.11-1) to reflect the final NRC-approved ISP plan.

While the request for deferral to RF-10 is related to the ISP plan, technical justification for the deferral was provided in the referenced request. The justification was based on the following points:

- *General Electric (GE) report GE-NE-B1301807-02, April 1996, "Surveillance Specimen Program Evaluation for River Bend Station,"* provided results of an evaluation of the RBS shift in RT_{NDT} using actual measured surveillance data from other BWRs. Based on the measured value, the predicted shift for RBS, calculated in accordance with NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988, is expected to conservatively bound measured values. In addition, by comparing the surveillance results for plants with fluence levels near that of RBS at 10.4 EFPY (Tables 3-1 and 3-2 of the report, Plants AJ and AF), the measured shifts are less than the predicted RG 1.99, Revision 2, shift plus margin values by an average of 54°F (base metal) and 38°F (weld material).

- *GE-NE-B1301807-02*, Figure 4-1, shows that the reference fracture toughness (i.e., the crack arrest fracture toughness) used in calculations for the RBS pressure/temperature curves (ASME Code, Section III/XI, Appendix G) is conservatively bounded by the static crack initiation fracture toughness. At 12 EFPY, the static crack initiation fracture toughness for RBS is 2.4 times the crack arrest fracture toughness for a pressure test temperature of 169°F and a vessel adjusted reference temperature of 65°F. The report concludes that the combination of lower bound fracture toughness, the RBS operating characteristics, and the conservative fracture toughness values indicate that the RBS vessel fracture toughness is not a significant concern over the life of the plant.

- The current BWR Owners' Group Supplemental Surveillance Program (SSP) includes a surveillance specimen that will be analyzed in 2000, with results factored into the ISP. The limiting RBS beltline material is included in four of the seven SSP capsules, including the one being analyzed in 2000. Though the RBS plate specimens are not included in the SSP capsules, materials having similar chemistry to the RBS vessel will provide data useful to RBS.

In a teleconference with NRC Staff on January 5, 2000, questions on the method for calculating fluence for RBS were posed. Flux and fluence calculations for RBS are performed in accordance with accepted industry standards. The methods are described in RBS Updated Safety Analysis Report (USAR), Sections 4.1.4.5, 4.3.2.8, and 5.3.1.6.2 (copy enclosed). The peak fluence of 6.6×10^{18} n/cm^2 at vessel inside surface at 32 EFPY, used in calculations of RBS adjusted reference temperatures, is based on RBS actual first cycle flux wire measurements *(see GSU response to NRC Generic Letter 88-11, dated May 14, 1990)*. Actual RBS fluence is predicted to be bounded by the calculated value, and the RBS adjusted reference temperatures calculations, performed in accordance with NRC RG 1.99, Revision 2, are considered to be accurate and conservative. This peak fluence has been used in previous revisions to the RBS pressure/temperature limits *(reference Amendment 45, dated August 1, 1990, Amendment 92, dated February 13, 1997, and Amendment 93, dated April 14, 1997, to the RBS Operating License)*.

To assist the Staff in its review of License Amendment Request 99-22, the following listing provides a complete description of the reference information highlighted in italics above.

1. RBS letter from J. C. Deddens to NRC dated May 14, 1990, requesting changes to the RBS reactor vessel pressure-temperature limits in response to Generic Letter 88-11 (including General Electric (GE) Report, SASR 89-20, Rev. 1). NRC Staff review and approval of the requested changes is documented in the Safety Evaluation enclosed with Amendment 45 and NRC letter from Claudia M. Abbate to James C. Deddens dated August 1, 1990.
2. RBS letter from John R. McGaha, Jr. to NRC dated August 29, 1996, LAR 96-35, "Request for a Revision to the Reactor Vessel Material Surveillance Program Capsule Withdrawal Schedule" (including GE Report GE-NE-B1301807-02). NRC Staff review and approval of the requested changes is documented in the Safety Evaluation enclosed with Amendment No. 92 and NRC letter from David L. Wigginton to John R. McGaha, Jr. dated February 13, 1997.
3. RBS letter from John R. McGaha, Jr. to NRC dated January 10, 1997, LAR 96-09, submitting revised Pressure-Temperature Limits valid through 12 EFPY. NRC Staff review and approval of the requested changes is documented in the Safety Evaluation enclosed with Amendment No. 93 and NRC letter from David L. Wigginton to John R. McGaha, Jr. dated April 14, 1997.

No new commitments are included herein. If you have further questions, contact Mr. Bill Fountain of my staff at 225-381-4625.

Sincerely,

Rick J. King

RJK/wjf
Enclosures

cc:

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ENCLOSURE 1

LAR 99-22

TRM TABLE 3.4.11-1 MARK-UP

Table 3.4.11-1
REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM
WITHDRAWAL SCHEDULE

CAPSULE WITHDRAWAL	WITHDRAWAL TIME - EFPY
First	~~10.4~~11.5
Second	15
Third	Standby

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ENCLOSURE 2

RBS USAR

Sections 5.3.1.6.2, 4.1.4.5, & 4.3.2.8

First capsule - 6 EFPY, or at the time when the accumulated neutron fluence of the capsule exceeds $5x10^{22}$ n/m^2 ($5x10^{18}$ n/cm^2), or at the time when the highest predicted ΔRT_{NDT} of all encapsulated materials is 28°C (50°F), whichever comes first.

Second capsule - 15 EFPY, or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first.

Third capsule - EOL (not less than once or greater than twice the peak EOL vessel fluence. This may be modified on the basis of previous tests. This capsule may be held without testing following withdrawal).

Fracture toughness testing of irradiated capsule specimens is to be in accordance with requirements of 10CFR50, Appendix H.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Sections 4.1.4.5 and 4.3.2.8.
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The peak fluence at the inside surface of the vessel beltline shell is $6.6(10)^{18}$ n/cm^2 after 40 yr of service. All predictions of radiation damage to the reactor vessel beltline material were made using peak fluence values.

5.3.1.6.3 Predicted Irradiation Effects on Vessel Beltline Materials

Estimated maximum changes in vessel beltline RT_{NDT} (initial reference temperature) values as a function of the 32 effective full power year (EFPY) fluence are listed in Table 5.3-1. The predicted peak 32 EFPY fluence at the inside surface of the vessel beltline is $6.6(10)^{18}$ n/cm^2. Transition temperature changes and changes in upper shelf energy were calculated in accordance with the rules of Regulatory Guide 1.99, Revision 2. Reference temperatures were established in accordance with 10CFR50, Appendix G, and NB-2330 of the ASME Code.

The lead factors for each surveillance capsule are 0.67 vessel i.d. and 0.89 1/4 T. Due to the geometry, the lead factors for all three capsule specimens will be the same.
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4.1.4.3 Reactor Systems Dynamics

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The analysis techniques and computer codes used in reactor system dynamics are described in Section 5 of Reference 10. Section 4.4.4.6 also provides a complete stability analysis for the reactor coolant system.

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4.1.4.4 Nuclear Engineering Analysis

The analysis techniques are described and referenced in Section 4.3.3. The codes used in the analysis are:

Computer Code	Function
Lattice physics model	Calculates average few-group cross sections, bundle reactivities, and relative fuel rod powers within the fuel bundle.
BWR reactor simulator	Calculates three-dimensional nodal power distributions, exposures, thermal-hydraulic characteristics as burnup progresses.

4.1.4.5 Neutron Fluence Calculations

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Neutron vessel fluence calculations were carried out using a two-dimensional, discrete ordinates, S_n transport code with general anisotropic scattering.

This code is a widely used discrete ordinates code which will solve a wide variety of radiation transport problems. The program will solve both fixed source and multiplication problems. Slab, cylinder, and spherical geometries are allowed with various boundary conditions. The fluence calculations incorporate, as an initial starting point, neutron fission distributions prepared from core physics data as a distributed source. Anisotropic scattering was considered for all regions. The cross sections were prepared with 1/E flux weighted, P_1 matrices for anisotropic scattering but did not include resonance self-shielding factors.

Fast neutron fluxes at locations other than the core midplane were calculated using a second two-dimensional, discrete ordinate code. This second two-dimensional code is used to solve smaller sized problems, and is similar to the two-dimensional code used for the vessel neutron fluence calculations.

The fast neutron flux calculations are used to establish the ratio of flux between the surveillance capsule locations and the location of peak vessel inside surface flux, known as the lead factor. Use of the lead factor is discussed in Section 4.3.2.8.

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4.3.2.7.2 Thermal Hydraulic Stability

This subject is covered in Subsection 4.4.4.6.

4.3.2.8 Vessel Irradiations

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The lead factor was calculated using the two-dimensional discrete ordinates transport code described in Section 4.1.4.5. The discrete ordinates code was used in a distributed source mode with cylindrical geometry. The geometry described seven regions with the core modeled as two homogenized regions. The coolant water region between the core and the shroud was described containing saturated water at 550°F and 1,050 psi. Subcooled water at 530°F and 1040 psia was used for the coolant between the shroud and the vessel. The material compositions for the stainless steel in the shroud and the carbon steel in the vessel contain the mixtures by weight as specified in the ASME material specifications for ASME SA240, 304L, and ASME SA533, Grade B. In the region between the shroud and the vessel, the presence of the jet pumps was ignored. A diagram showing the regions and dimensions modeled is shown in Fig. 4.3-23.

The distributed source, which can be separated in space and energy, was obtained from the core power shape and the neutron spectra. The integral over position and energy is normalized to the total number of neutrons in the core region. The core region is defined as a 1-cm thick disc with no transverse leakage. The power in this core region is set equal to the maximum power in the axial direction. The optimum axial power distribution is shown in Figure 4.3-13.

Dosimetry located on the inside surface of the vessel was removed after the first fuel cycle and tested to determine the flux at that location. The lead factor relating the dosimeter location to the peak location was used to calculate the peak vessel inside surface flux. Assuming an 80% capacity factor, or 32 effective full power years (EFPY) in 40 years of operation, the fluence for this operating period was estimated. The measured dosimeter flux, calculated peak flux and fluence are shown in Table 4.3-5. The calculated neutron flux leaving the cylindrical core is listed in Table 4.3-6.

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