Entergy

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W3F1-2004-0047

May 26, 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Supplement to Amendment Request NPF-38-249,
Extended Power Uprate
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:
1. Entergy Letter dated November 13, 2003, "License Amendment Request NPF-38-249 Extended Power Uprate"
2. NRC Letter dated May 4, 2004, "Waterford Steam Electric Station, Unit 3 (Waterford 3) – Request for Additional Information Related to Revision to Facility Operating License and Technical Specifications - Extended Power Uprate Request (TAC No. MC1355)"

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Operating License and Technical Specifications to increase the unit's rated thermal power level from 3441 megawatts thermal (MWt) to 3716 MWt.

By letter (Reference 2), the Nuclear Regulatory Commission (NRC) staff requested additional information (RAI) related to reactor vessel internals degradation monitoring. Entergy's response to this question is contained in Attachment 1 to this letter. Additionally, Entergy and Westinghouse personnel have discussed the approach taken for vessel internals fluence projections with the NRC staff. Information regarding this subject is provided in Attachment 2.

There are no technical changes proposed. The original no significant hazards consideration included in Reference 1 is not affected by any information contained in this letter. The submittal includes a new commitment as summarized in Attachment 3.

If you have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 26, 2004.

Sincerely,

BLH/DBM/cbh

Attachment:
1. Response to Request for Additional Information
2. Outline of Approach for Vessel Internals Fluence Projection
3. List of Regulatory Commitments

cc: Dr. Bruce S. Mallett
U. S. Nuclear Regulatory Commission
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Arlington, TX 76011

NRC Senior Resident Inspector
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U.S. Nuclear Regulatory Commission
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Attachment 1 To

W3F1-2004-0047

Response to Request for Additional Information

**Response to Request for Additional Information
Related to the Extended Power Uprate**

Reactor Vessel Internals Degradation Monitoring

Question:

The U.S. Nuclear Regulatory Commission (NRC) Review Standard, NRR-RS-001, Revision 0, *Review Standard for Extended Power Uprates* (December 2003), provides the staffs standard review plan for the evaluation of license amendment requests for increasing power above 5 percent of the current core thermal power rated for a U.S. light-water reactor. Note 1 in Matrix 1 of Section 2.1 of NRR-RS-001, Revision 0, indicates that guidance on the neutron irradiation threshold for irradiation assisted stress corrosion cracking (IASCC) of pressurized water reactor (PWR) reactor vessel (RV) internals is given in WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals." WCAP-14577 Revision 1-A states that the threshold fluence level for IASCC in PWR RV internals is $1x10^{21}$ neutrons per square centimeter (n/cm^2) (E > 0.1 MeV). Entergy has indicated that the neutron fluence value for the limiting RV internal component (i.e., the core shroud) will be about $3.5x10^{22}$ n/cm^2 (E > 1 MeV) at the end of the current operating period after the 8 percent extended power uprate has been implemented. Although these values are not expressed on an equivalent basis, it is clear that the projected neutron fluence for the Waterford 3 core shroud is above the NRC's threshold for IASSC of PWR RV internal components made from stainless or nickel-alloy components.

Note 1 in Matrix 1 of Section 2.1 of NRR-RS-001 also indicates that, for management of "... thermal and neutron embrittlement of cast austenitic stainless steel, stress-corrosion cracking, and void swelling [in RV internals under extended power uprated conditions], licensees will need to provide plant-specific degradation management programs or participate in industry programs to investigate degradation effects and determine the appropriate management programs."

Since Entergy has indicated that the post-EPU limiting neutron fluence value will be significantly above the threshold for initiating IASCC in the components, consistent with Note 1 in Matrix 1 above, the NRC staff requests that Entergy either:

1. Submit an inspection program, including the scope, sample size, inspection method, frequency of examination, and acceptance criteria that will be utilized to manage the aging effects for the Waterford 3 RV internals for NRC staff review and approval, or

2. Provide a commitment to participate in the Electric Power Research Institute MRP's research initiatives on age-related degradation of RV internal components.

Response:

As discussed in Attachment 2 to this letter, the estimated post-EPU limiting neutron fluence value actually decreases in comparison to the estimated pre-EPU limiting neutron fluence value. However, Entergy Operations, Inc. (Entergy) is currently an active participant in the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) research initiatives on aging related degradation of reactor vessel internal components (i.e., MRP Reactor Vessel Internals Issues Task Group (ITG)). Entergy commits to continue its active participation in this MRP initiative to determine appropriate reactor vessel internals degradation management programs.

Attachment 2 To

W3F1-2004-0047

Outline of Approach for Vessel Internals Fluence Projection

Outline of Approach for Vessel Internals Fluence Projection

The neutron fluence values applicable to the Waterford Steam Electric Station, Unit 3 (Waterford 3) core shroud were based on the same methodology used in the transport calculations supporting the recent completed reactor vessel surveillance capsule evaluation (Reference 1). These neutron fluence calculations included cycle specific analyses for fuel Cycles 1 through 12.

Projections for operation beyond Cycle 12 were based on the assumptions documented in Reference 1. Specifically, the Cycle 12 spatial power distribution, water temperatures, and reactor power level were used directly for the Cycle 13 analysis and the length of Cycle 13 was assumed to be 490 EFPD. Both reflect the same (identical) power distribution data based on the 1.5% uprate (3441 MWt).

Beginning with the onset of Cycle 14, an increase in reactor power level to 3716 MWt was assumed to represent operation at the additional approximate 8% uprate. The spatial power distributions (including pin-wise power distributions for the peripheral assemblies) and water temperatures for all future operating cycles were the same as used in Reference 1. Additionally, a 5% positive bias was applied to the power generated in the peripheral fuel assemblies for cycles fourteen and beyond. (The intent of the peripheral bias was to include some flexibility in the design of future fuel cycles.)

It should be noted from Table 6-1 of WCAP-16002 the overall trend of decreasing neutron flux to the vessel with time even after implementation of the 8% uprate planned for Cycle 14. This trend is entirely due to the transition from out-in type core loading to in-in-out type core loading. This can readily be seen by comparing the power distributions for the first cycle given in Figures 4.3-3 and 4.3-4 of the Waterford 3 Final Safety Analysis Report (FSAR) with the power distributions from a more recent cycle (e.g. Cycle 12 appearing in Figures 4.3A-8 thru 4.3A-10 of the Waterford 3 FSAR). For example, these figures show that the ratio of the relative power for Box 2 Cycle 12 to Cycle 1 is 0.40. This ratio is consistent with the ratio of 0.43 for flux at the 14 degree point given in Table 6-1 of WCAP-16002.

Methodology

As was the case for the Capsule W263 evaluations, the methodology used to complete the fluence calculations for the shroud conform to the requirements specified in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 4). In particular, all of the transport calculations were carried out using the DORT discrete ordinates code Version 3.1 (Reference 2) and the BUGLE-96 cross-section library (Reference 3). The BUGLE-96 library provides a 67 group coupled neutron - gamma ray cross-section data set produced specifically for light water reactor (LWR) applications. In the analysis for Waterford 3, anisotropic scattering was treated with a P_5 legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. Energy and space dependent core power distributions as well as system operating temperatures were treated on a fuel cycle specific basis. The core power distributions and system operating temperatures were the same as used in Reference 1.

Neutron Fluence Calculation for Core Internals

The output of the neutron fluence calculations is a synthesized three-dimensional fluence distribution within the shroud geometry. This synthesized distribution was scanned to determine the location of the maximum neutron exposure at the inner surface of the shroud. The result of this procedure yielded a maximum fast neutron (E > 1.0 MeV) fluence projection of 3.82e+22 n/cm^2 for the shroud material following 32 effective full power years (40 calendar years) of operation.

Comparison to Design Fluence

The original design fluence projection for the shroud was based on the assumption that the neutron flux calculated for Cycle 1 was characteristic of the entire 40 year (32 EFPY) irradiation period. The current projections are based on the actual operating history to date (including the incorporation of low leakage fuel management) plus the uprated conditions through the same 40 year period as discussed above. A comparison of the original projections with the results of the current analysis are provided as follows for 32 EFPY of operation:

Original design projections -	4.80 E+22 n/cm^2
Current projections including uprate-	3.82 E+22 n/cm^2

This comparison supports the previous statement that the actual fluence to the internals after uprate is less than the original design projection. The decrease was expected and is attributed to the use of low leakage fuel management for the majority of the operating period. Thus, when the actual plant specific operating environment, including both the power uprate and the use of low leakage fuel management, is considered, it is evident that the reactor internals materials will experience less exposure to fast neutron irradiation than was indicated by the original design projections.

References:

1. WCAP-16002-NP, "Analysis of Capsule 263□ from the Entergy Operations Waterford Unit 3 Reactor Vessel Radiation Surveillance Program," March 2003.
2. RSICC Computer Code Collection CCC-650, "DOORS 3.1 One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," "Radiation Shielding Information Center, Oak Ridge National Laboratory, August 1996.
3. RSICC Data Library Collection DLC-185, "BUGLE-96 Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," Radiation Shielding Information Center, Oak Ridge National laboratory, March 1996.
4. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

Attachment 3 To

W3F1-2004-0047

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Entergy Operations, Inc. (Entergy) is currently an active participant in the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) research initiatives on aging related degradation of reactor vessel internal components (i.e., MRP Reactor Vessel Internals Issues Task Group (ITG)). Entergy commits to continue its active participation in this MRP initiative to determine appropriate reactor vessel internals degradation management programs.		X	NA