September 30, 2003

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 1 — ISSUANCE OF AN AMENDMENT REGARDING ANALYTICAL METHODS FOR ROBUST FUEL ASSEMBLY-2 (RFA-2) UPGRADE (TAC NO. MB7746)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 46 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant (WBN), Unit 1. The amendment consists of changes to Technical Specification (TS) 5.9.5, "Core Operating Limits Report," and is in response to your application dated February 14, 2003, as supplemented on June 5 and August 21, 2003.

The requested change would modify TS 5.9.5 to add three additional methodologies in support of the Westinghouse 17x17 RFA-2 fuel design with Intermediate Flow Mixers.

A copy of the safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Margaret H. Chernoff, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures: 1. Amendment No. 46 to NPF-90
2. Safety Evaluation

cc w/enclosures: See next page

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ADAMS Accession No.ML032740199

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DATE	9/26/03	9/30/03	6/30/03	9/29/03

OFFICE	IROB/SC	OGC	PDII-2/SC
NAME	TBoyce	JMoore	AHowe
DATE	9/23/03	9/26/03	9/30/ 03

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
License No. NPF-90

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated February 14, 2003, as supplemented on June 5 and August 21, 2003, 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-90 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 46, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, and shall be implemented during the current refueling outage, which started on September 7, 2003.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Allen G. Howe, Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 30, 2003

ATTACHMENT TO AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

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.

Remove Pages	Insert Pages
5.0-33	5.0-33
B2.0-2	B2.0-2
B2.0-4	B2.0-4
B2.0-6	B2.0-6
B3.2-13	B3.2-13

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

By letter dated February 14, 2003, as supplemented by letters dated June 5 and August 21, 2003 (ADAMS Accession Nos. ML030520332, ML031610659, and ML032380514, respectively), the Tennessee Valley Authority (the licensee) submitted a request for changes to the Technical Specifications (TSs) for Watts Bar Nuclear Plant (WBN), Unit 1.

The requested changes would modify TS 5.9.5, "Core Operating Limits Report (COLR)," to add three additional methodologies in support of the Westinghouse 17x17 RFA-2 fuel design with Intermediate Flow Mixers (IFMs). These three methodologies would be added to the list of U.S. Nuclear Regulatory Commission (NRC) approved methodologies that can be used by the licensee to determine cycle specific core operating limits. These methodologies include the WRB-2M Departure from Nucleate Boiling (DNB) correlation, the Revised Thermal Design Procedure, and the VIPRE-01 methodology. In addition, the licensee is revising the appropriate TS Bases to reflect these changes.

The supplemental letters provided clarifying information that did not expand the scope of the initial amendment as described in the notice and did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The regulatory requirements applicable to the design bases of a fuel system include Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 10, "Reactor Design"; GDC 27, "Reactivity Control System"; and GDC 35, "Emergency Core Cooling Systems"; 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems"; and 10 CFR Part 100, "Reactor Siting Criteria." In addition, the NRC Standard Review Plan (SRP, NUREG 0800), Section 4.2, "Fuel System Design," contains guidelines for the safety review of a fuel design system. The SRP states that the objectives of the fuel design system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always

maintained. NRC-approved Westinghouse methodologies and analyses such as DNB critical heat flux correlations and fuel rod design evaluations may be used to demonstrate that the fuel system design is within the acceptance criteria.

3.0 TECHNICAL EVALUATION

3.1 Fuel Rod Performance

The RFA-2 fuel design changes involve the addition of three IFM grids, mid-grid changes, and thicker guide thimbles and instrument tubes. The transition from the Vantage Plus fuel with Performance Plus features (V+/P+ fuel) to RFA-2 fuel does not result in significant fuel design changes. The design features of the RFA-2 fuel-rod are essentially the same as those features in the V+/P+ fuel rod. The RFA-2 fuel assembly is expected to be compatible with the V+/P+ fuel assembly in fuel performance and structural behavior.

The fuel rod mechanical design bases are identical for the RFA-2 and V+/P+ fuel designs, and are based on SRP Section 4.2. The design bases include stress, strain, fatigue, oxidation and hydriding, irradiation growth, rod internal pressure, and hydraulic loads. The licensee performed an evaluation of the fuel-rod design using the approved fuel performance analysis and design (PAD) code and the approved methodology of extended burnup applications in Westinghouse Topical Report WCAP-10125-P-A. The licensee concluded that the design bases and limits were met for the RFA-2 fuel design.

The NRC staff has determined that the licensee used the appropriate methodology for evaluation of fuel rod performance and the appropriate acceptance criteria. Since the acceptance criteria were satisfied, the staff concludes that the amendment is acceptable in this regard.

3.2 Seismic/Loss of Coolant Accident (LOCA) Impact on Fuel Assemblies

Earthquakes and postulated pipe breaks in the reactor coolant system (RCS) would result in external forces on fuel assemblies. Appendix A to SRP Section 4.2 states that fuel system coolability should be maintained and damage should not be so severe as to prevent control rod insertion when required during seismic and LOCA events. The structural integrity of fuel assemblies is analyzed to ensure that external forces do not exceed the maximum allowable grid crushing load such that the resulting damage is minimal, and control rods and thimble tubes remain functional during seismic and LOCA events.

For WBN Unit 1 operation, the most severe loads are induced by scenarios of seismic and LOCA events with a mixed core. The licensee analyzed a mixed core of RFA-2 and V+/P+ fuel assemblies using the approved methodology in WCAP-9401-P-A, selected two limiting mixed core configurations, and used the square-root-of-sum-of-squares (SRSS) method, as described in Appendix A to SRP Section 4.2, to combine the maximum LOCA and seismic impact forces. The results demonstrated that the combined impact forces on grids in different elevations were all below the maximum allowable grid crushing load. Thus, the licensee concluded that grid deformation was within acceptable limits and coolable geometry was maintained under the seismic and LOCA events.

The NRC staff finds that the licensee used an approved methodology for mixed core analysis and the approved SRSS method for combining impact forces. The staff has concluded that the grid impact is acceptable and a coolable geometry will be maintained during seismic and LOCA events for WBN Unit 1. Based on these conclusions, the staff finds that the amendment is acceptable in this regard.

3.3 Nuclear Design

Nuclear design basis, margins and methodology are not affected by the transition to RFA-2 fuel with IFMs. Power distributions, peaking factors, and rod worths are primarily loading pattern dependent. The variations of these core safety parameters are expected to be typical of the normal cycle-to-cycle variations for the standard fuel reloads. The fresh fuel rack criticality analysis is applicable to the RFA-2 fuel. Furthermore, the analysis for fuel misloading is still applicable.

The NRC staff finds that current nuclear design analyses remain applicable in that the design basis, margins and methodology are not affected by the transition to the RFA-2 fuel design, and therefore, the amendment is acceptable in this regard.

3.4 Thermal Hydraulic Design

The existing thermal hydraulic analysis of the V+/P+ fuel is based on the Revised Thermal Design Procedure (RTDP), the WRB-1 DNB correlation, and the VIPRE computer code. Use of RFA-2 fuel will call for the use of the WRB-2M DNB correlation. The WRB-2M correlation database applies to the RFA-2 fuel with IFMs. A 95/95 DNB correlation limit of 1.14 applies to the RFA-2 fuel. The WRB-1 correlation will continue to be used for the V+/P+ fuel, and the W-3 correlation will be used when conditions are outside the range of the WRB-1 and WRB-2M correlations. Uncertainties are combined in the RTDP methodology to obtain the overall DNB uncertainty factor which is used to define the design limit of 1.23. Since parameter uncertainties are statistically combined, the plant safety analyses are performed using nominal values of input parameters.

The hydraulic compatibility of different fuel assembly designs in a mixed core was tested. The results of the testing demonstrated that the RFA-2 fuel design is hydraulically compatible in mixed core applications with the V+/P+ or Vantage 5H fuel previously used in the WBN core. The RFA-2 fuel results in a 10 percent increase in core pressure drop and a 0.6 percent increase in the design core bypass flow. These increases are accounted for in the Nuclear Steam Supply System design parameters.

Since there is an increase in flow resistance because of the IFMs in the RFA-2 fuel, there will be a mixed core penalty of 12 percent with the first transition core. The flow will be directed to the lower resistance fuel. As the fuel population of RFA-2 fuel increases, the effects will decrease resulting in a decrease of the penalty until a full core of RFA-2 fuel is in the core and there is no longer a penalty.

Based on its review, the NRC staff finds that the V+/P+ and RFA-2 fuel assemblies are hydraulically compatible, and sufficient DNB margin is available to cover the applicable penalties. The staff has determined that the licensee is applying an approved methodology within its approved range of applicability. Since the results calculated with this methodology

met the applicable acceptance criteria, the staff concludes that the amendment is acceptable in this regard.

3.5 Accident Analyses

The licensee evaluated non-LOCA accidents, and stated that the transient evaluations and analyses demonstrate that all applicable safety analysis acceptance criteria continue to be met for the intended RFA-2 fuel upgrade implementation at WBN Unit 1. According to the licensee, the results and conclusions presented in the Updated Final Safety Analysis Report (UFSAR) remain valid, and are acceptable.

The licensee evaluated the LOCA accidents for RFA-2 fuel with IFMs. Three large-break LOCA (LBLOCA) cases were analyzed using the Analysis of Record (AOR) reference calculation as the base case. Case 1 analyzed a full core of RFA-2 fuel. Case 2 analyzed a potential limiting transition core case with RFA-2 fuel only in the hot assembly position with the remainder of the core being V+/P+ fuel with 8000 MWD/MTU minimum burnup. Case 3 analyzed a transition core case where RFA-2 fuel is in the hot assembly position and in the average fuel assemblies under the guide tubes, with the remaining core being V+/P+ fuel with 8000 MWD/MTU burnup.

The licensee stated that the best estimate LBLOCA evaluation for the transition cores and the full core of RFA-2 fuel remains in compliance with the requirements of 10 CFR 50.46 and is acceptable.

For the small-break LOCA (SBLOCA), the licensee stated that changes resulting from the introduction of RFA-2 fuel are expected to have a negligible effect on the analysis results. Therefore, no changes are necessary to the SBLOCA AOR. The primary factors affecting the calculations for the post-LOCA long-term core cooling subcriticality requirement are the volumes and boron concentrations of the RCS, the refueling water storage tank (RWST), and the accumulators. The licensee stated that the implementation of IFMs will have no effect on these volumes and concentrations. Accordingly, long-term core cooling subcriticality will not be affected, and is acceptable.

The hot-leg switchover time is dependent on the power level and volumes and boron concentrations of the RWST and accumulators. The licensee stated that the RCS volume will decrease by an insignificant amount due to the additional grids and increased outside diameter of the thimble tubes. Accordingly, there would be no adverse effect on the post-LOCA hot leg switchover time for implementation of RFA-2 for the Westinghouse emergency core cooling system evaluation model, and it is acceptable.

The NRC staff finds that the transient and LOCA evaluations and analyses demonstrate that all applicable safety analysis acceptance criteria continue to be met for operation with the RFA-2 fuel design and, therefore, this amendment is acceptable in this regard.

3.6 Topical Report Conditions

The WBN Unit 1 fuel upgrade to the RFA-2 design includes the use of IFMs, which was reviewed and approved in WCAP-10444-P-A. The licensee responded to the conditions for application of this topical report set forth in the staff's Safety Evaluation for the Report. The licensee also responded to the conditions in three, methodology-related, approved topical

reports. The WRB-2M correlation was reviewed and approved in WCAP-15025-P-A; the subchannel analysis code VIPRE-01 was reviewed and approved in WCAP-14565-P-A; and the RTDP methodology was approved in WCAP-11397-P-A. The three methodology-related topical reports would be added to WBN TS 5.9.5.

3.6.1 WCAP-10444-P-A Conditions

1. The statistical convolution method described in WCAP-10125 for the evaluation of initial fuel rod to nozzle growth has not been approved. This method may not be used in VANTAGE 5.

   The licensee stated that the statistical convolution method was not used. Instead, the licensee used the worst-case fabrication tolerances to assess fuel rod performance on a cycle-specific basis.

2. For each plant application, it must be demonstrated that the LOCA/seismic loads considered in WCAP-9401 bound the plant in question; otherwise, additional analysis will be required to demonstrate the fuel assembly structural integrity.

   The licensee assessed the structural integrity of the RFA-2 fuel assembly with IFMs considering the lateral effects of a LOCA and seismic accident. The licensee's results showed the structural acceptability for an all RFA-2 core and a transition core consisting of both RFA-2 fuel assemblies and VANTAGE+ assemblies.

3. An irradiation demonstration program should be performed to provide early conformation performance data for the VANTAGE 5 design.

   The licensee stated that an irradiation demonstration program for both VANTAGE 5 and RFA-2 fuel assemblies has been successfully completed.

4. For those plants using the ITDP [Improved Thermal Design Procedure], the restrictions numerated in Section 4.1 of this report [WCAP-10444 SER] must be addressed and information regarding measurement uncertainties must be provided.

   The licensee stated that ITDP is not being used. The RTDP described in WCAP-11397 is being used.

5. The WRB-2 correlation with a DNBR [Departure from Nucleate Boiling Ratio] limit of 1.17 is acceptable for application to 17X17 VANTAGE 5 fuel. Additional data and analyses are required when applied to 14X14 or 15X15 fuel with an appropriate DNBR limit. The applicability range of WRB-2 is specified in Section 4.2.

   The licensee stated that WRB-2M DNB correlation described in WCAP-15025-P-A is being used in this application. In its submittal, the licensee addressed the specific conditions for use of WCAP-15025-P-A.

6. For 14X14 and 15X15 VANTAGE 5 fuel designs, separate analyses will be required to determine a transitional mixed core penalty. The mixed core penalty

and plant-specific safety margin to compensate for the penalty should be addressed in the plant TS Bases.

This condition is not applicable since all fuel in WBN Unit 1 is 17X17.

7. Plant specific analysis should be performed to show that the DNBR limit will not be violated with the higher value of F-DeltaH.

   This condition is not applicable since the F-DeltaH limit is not being increased.

8. The plant-specific safety analysis for the steam supply system piping failure event should be performed with the assumption of loss of offsite power if that is the most conservative case.

   The licensee stated that an assessment of the steamline break transient was performed. The assessment concluded that the transient response would be insignificantly impacted by the changes introduced by the RFA-2 fuel design.

9. With regard to the RCS pump shaft seizure accident, the fuel failure criterion should be the 95/95 DNBR limit. The mechanistic method mentioned in WCAP-10444 is not acceptable.

   The licensee stated that the mechanistic approach was not used in the locked rotor accident analysis, but rather any fuel rod violating the 95/95 DNBR limit was assumed to fail.

10. If a positive MTC [Moderator Temperature Coefficient] is intended for VANTAGE 5, the same positive MTC consistent with the plant Technical Specification should be used in the plant specific safety analysis.

    The licensee stated that a positive MTC is not used at WBN Unit 1. All of the analyses are performed consistent with the MTC limits given in the COLR.

11. The LOCA analysis performed for the reference plant with higher F(Q) of 2.55 has shown that the PCT [Peak Clad Temperature] limit of 2200 degrees Fahrenheit is violated during transitional mixed core. Plant specific LOCA analysis must be done to show that at the appropriate value of F(Q), the 2200 degrees Fahrenheit criteria can be met during the use of a transitional mixed core.

    The licensee stated that the WBN Unit 1 LOCA analyses were performed with consideration of transition core effects. The licensee performed a transition core study which concluded that WBN Unit 1 remains in compliance with the requirements of 10 CFR 50.46 based on the AOR for both transition cores and full RFA-2 cores based on a LOCA F(Q) of 2.50 and F-DeltaH of 1.65.

12. The NRC staff's SER on Westinghouse's extended burnup topical report WCAP-10125 is not yet complete; the approval of the VANTAGE 5 design for operation to extended burnup levels is contingent on NRC approval of WCAP-10125. However, VANTAGE 5 fuel may be used to those burnups to which Westinghouse fuel is

presently operating. The staff's review of the Westinghouse extended burnup topical report has not identified any safety issues with operation to the burnup value given in the extended burnup report.

WCAP-10125 has subsequently been approved by NRC staff.

13. Recently, a vibration problem has been reported in a French reactor having 14-foot assemblies; vibration below the fuel assemblies in the lower portion of the reactor vessel is damaging the movable incore instrumentation probe thimbles. The staff is currently evaluating the implications of this problem to other cores having 14-foot long fuel bundle assemblies. Any limitations to the 14-foot core design resulting from the staff evaluation must be addressed in plant specific evaluations.

    This condition is not applicable to WBN Unit 1, which has 12-foot long fuel assemblies.

3.6.2 WCAP-15025 Conditions

1. Since WRB-2M was developed from test assemblies designed to simulate Modified Vantage 5H fuel the correlation may only be used to perform evaluations for fuel of this type without further justification. Modified Vantage 5H fuel with or without modified intermediate flow mixer grids may be evaluated with WRB-2M.

    The licensee stated that the structural mid-grid design used in the RFA-2 fuel design is a minor modification of the Modified Low Pressure drop mid-grid design addressed in WCAP 15025. The RFA-2 mid-grid design was reviewed in accordance with NRC-approved Fuel Criteria Evaluation Process. Westinghouse notified the NRC of the RFA-2 mid-grid design modifications and validation of the WRB-2M correlation via a letter dated November 13, 2002, in accordance with the Fuel Criteria Evaluation Process.

2. Since WRB-2M is dependent on calculated local fluid properties these should be calculated by a computer code that has been reviewed and approved by the NRC staff for that purpose. Currently WRB-2M with a DNBR limit of 1.14 may be used with the THINC-IV computer code. The use of VIPRE-01 by Westinghouse with WRB-2M is currently under separate review.

    The licensee stated that the analysis of RFA-2 fuel was based on the VIPRE code and the WRB-2M DNB correlation with a 95/95 correlation limit of 1.14. This has been approved by the NRC staff.

3. WRB-2M may be used for PWR [pressurized water reactor] plant analyses of steady state and reactor transients other than loss of coolant accidents. Use of WRB-2M for loss of coolant accident analysis will require additional justification that the applicable NRC regulations are met and the computer code used to calculate local fuel element thermal/hydraulic properties has been approved for that purpose.

    The licensee stated that the WRB-2M correlation is not used for the loss of coolant accident analysis for the RFA-2 fuel in WBN Unit 1.

4. The correlation should not be used outside its range of applicability defined by the range of the test data from which it was developed.

   The license stated that the application of the WRB-2M correlation to the RFA-2 fuel upgrade in WBN Unit 1 is consistent with the range of parameters specified in Table 4-1 of WCAP-15025-P-A.

3.6.3 WCAP-14565 Conditions

1. Selection of the appropriate CHF [Critical Heat Flux] correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal.

   The licensee stated that the WRB-2M correlation with a 95/95 correlation limit of 1.14 was used in the DNB analysis for the RFA-2 fuel upgrade in WBN Unit 1. Westinghouse notification to the NRC of the validation of the WRB-2M DNB correlation applicability to the RFA-2 mid-grid was provided to NRC staff in a letter dated November 13, 2002. The use of plant specific hot channel factors and other fuel dependent parameters in the DNB analysis was previously reviewed and approved in License Amendment No. 31.

2. Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE.

   The licensee stated that the core boundary conditions for the VIPRE calculations for the RFA-2 fuel upgrade are all generated from NRC-approved codes and analysis methodologies. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology.

3. The NRC staff's generic SER for VIPRE set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification.

   The licensee stated that the WRB-2M correlation limit of 1.14 is used for the DNB analyses of the RFA-2 fuel in WBN, Unit 1. The WRB-1 correlation with a limit of 1.17 continues to be used for the VANTAGE-5H and VANTAGE+ designs that have previously been loaded into WBN Unit 1.

4. Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC staff's generic review of VIPRE did not extend to post-CHF calculations. VIPRE does not model the

time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained.

The licensee stated that application of VIPRE to the RFA-2 fuel upgrade in Watts Bar Unit 1 did not include usage in the post-CHF range.

3.6.4 WCAP-11397 Conditions

1. Sensitivity factors for a particular plant and their ranges of applicability should be included in the Safety Analysis Report or reload submittal.

   The licensee stated that the sensitivity factors were evaluated using the WRB-2M DNB correlation and the VIPRE code for parameter values applicable to the RFA-2 fuel in WBN Unit 1. These factors were used to determine the maximum design limit DNBR, which is included in the WBN Unit 1 UFSAR.

2. Any changes in DNB correlation, THINC-IV correlations, or parameter values listed in Table 3-1 of WCAP-11397 outside of previously demonstrated acceptable ranges require re-evaluation of the sensitivity factors and of the use of Equation (2-3) of the topical report.

   The licensee's response states that, in addition to the response to Condition 1 above, justification for use of the WRB-2M correlation and VIPRE-01 code is provided in the body of their amendment request.

3. If the sensitivity factors are changed as a result of correlation changes or changes in the application or use of the THINC code, then the use of an uncertainty allowance for application of Equation (2-3) must be re-evaluated and the linearity assumption made to obtain Equation (2-17) of the topical report must be validated.

   The licensee stated that Equation (2-3) of WCAP-11397-P-A and the linearity approximation made to obtain Equation (2-17) have been shown to be valid for the combination of WRB-2M and the VIPRE code. The VIPRE model used in this application does not differ significantly from that used in the WCAP.

4. Variances and distributions for input parameters must be justified on a plant-by-plant basis until generic approval is obtained.

   The licensee stated that the plant specific variances and distributions for this application were justified in the report supporting License Amendment No. 31.

5. Nominal initial condition assumptions apply only to DNBR analyses using RTDP. Other analyses, such as overpressure calculations, require the appropriate conservative initial condition assumptions.

The licensee stated that nominal initial conditions were only applied to DNBR analyses which used RTDP.

6. Nominal conditions chosen for use in analyses should bound all permitted methods of plant operation.

   The licensee stated that bounding nominal conditions were used in the DNBR analyses that were based on RTDP.

7. The code uncertainties specified in Table 3-1 (+/- 4 percent for THINC-IV and +/- 1 percent for transients) must be included in the DNBR analyses using RTDP.

   The licensee stated that the code uncertainties were included in the DNBR analyses.

Based on its review of the information supplied by the licensee, the NRC staff has concluded that the licensee has demonstrated that it will meet the restrictions given in the staff safety evaluations. Accordingly, the staff finds that the topical reports are acceptable for referencing in the WBN TS and the three methodology-related topical reports can be added to TS 5.9.5.

3.7 Tritium Producing Burnable Absorber Rods (TPBARs) Evaluation

On September 23, 2002, the NRC staff issued Amendment No. 40, allowing WBN Unit 1 to load and irradiate up to 2304 TPBARs in the reactor core for the purpose of producing tritium for the U.S. Department of Energy. The licensee provided descriptions of the methods and analyses used to demonstrate acceptability of 2304 TPBARs in Westinghouse Topical Report NDP-00-0344, Revision 1. Westinghouse 17 x 17 VANTAGE+ fuel assemblies were considered in this Topical Report and associated analyses.

To support the use of 17 x 17 RFA-2 fuel design with IFMs for WBN Unit 1, the licensee assessed the use of TPBARs in conjunction with the RFA-2 fuel. The licensee’s assessment of TPBARs and RFA-2 fuel encompassed mechanical compatibility, nuclear design aspects, thermal-hydraulic design aspects, and impacts on the UFSAR Chapter 15 safety analyses. In general, the use of RFA-2 fuel will result in only minor effects on tritium production core (TPC) designs when compared to VANTAGE+ TPC designs.

The mechanical features of the RFA-2 fuel assembly have been designed to be compatible with existing core components, including the TPBAR design. As such, the licensee verified that necessary clearances will be maintained throughout cycle life, that TPBAR holddown spring assembly force remains acceptable throughout cycle life, and that the use of TPBAR assemblies in conjunction with RFA-2 fuel has no impact on the fuel assembly structural integrity. Based on its assessment of the mechanical design features, the licensee concluded that the design basis analyses and evaluations performed for WBN Unit 1 in NDP-00-0344, Revision 1, remain applicable for RFA-2 fuel. Because these evaluations remain applicable for RFA-2 fuel designs, the staff finds that implementation of RFA-2 fuel at WBN is compatible with the use of TPBARs in WBN Unit 1.

The licensee assessed the nuclear design impacts of RFA-2 fuel on TPC designs by reanalyzing the first transition, second transition, and equilibrium cycle TPC core designs. The

licensee incorporated the same methodology used to support the original TPBAR amendment. The licensee's evaluation demonstrated that the impacts of RFA-2 fuel on nuclear design of a TPC for parameters such as power distributions, peaking factors, and TPBAR performance (tritium production) will be very similar to core designs utilizing VANTAGE+ fuel. The primary impact of RFA-2 fuel is a small decrease in excess core reactivity that results in a slightly lower critical boron concentration (28 parts per million) in the equilibrium cycle. Based on this analysis, the licensee concluded that key safety parameters for RFA-2 TPC designs will be comparable to those for Vantage+ TPC designs previously evaluated. Additionally, key safety analysis parameter values are reviewed for each core design as part of the standard reload safety evaluation process to determine the impact on safety analyses. Because these minor changes are accommodated and evaluated by the reload safety evaluation process using approved methods and cycle-specific values of input parameters, the staff finds that WBN Unit 1 TPC designs comprised of RFA-2 fuel and TPBARs will satisfy appropriate nuclear design criteria and safety analysis limits.

The licensee assessed the impacts of RFA-2 TPC designs on the core thermal-hydraulic design criteria. These assessments demonstrated that the core bypass flow limits will be satisfied for RFA-2 TPC designs, that TPBAR guide thimble boiling criteria will be satisfied, and that DNB criteria will be satisfied. Similar to the nuclear design assessment, cycle-specific analyses are performed for all core designs as part of the reload safety evaluation process, using NRC-approved methodology with appropriate cycle-specific value input parameters. Therefore, the staff finds that WBN Unit 1 TPC designs comprised of RFA-2 fuel and TPBARs will satisfy appropriate thermal-hydraulic criteria and safety analysis limits.

The licensee assessed the impacts of RFA-2 TPC designs on the transient and accident analyses. WBN Unit 1 incorporates a Reload Safety Analysis Checklist (RSAC) and key safety analysis parameter methodology as part of its reload design process. In this methodology, key safety analysis parameter values for a core design are compared to the parameter values assumed in the AOR. If the parameters are not bounded by those assumed in the analyses, then the core design is modified or the impacted safety analyses are reanalyzed such that acceptance criteria are satisfied. The licensee previously demonstrated that, relative to non-LOCA analyses, as long as the RSAC parameters are satisfied on a cycle-to-cycle basis, the results of the safety analyses are unaffected by the proposed reload core design. Because the RFA-2 TPC designs will be assessed for each cycle as part of this reload safety evaluation process, the non-LOCA safety analysis acceptance criteria will continue to be satisfied.

The TPC LBLOCA and SBLOCA analyses are performed using the LOCTA_JR computer code, with thermal-hydraulic boundary conditions derived from the non-TPC large and small break LOCA AORs. The staff previously approved this methodology and the use of LOCTA_JR for licensing applications as discussed in the Safety Evaluation dated September 23, 2002. For RFA-2 TPC designs, the licensee determined that the WBN Unit 1 LBLOCA AOR remains bounding for RFA-2 fuel, and that the RFA-2 fuel will have a negligible effect on the SBLOCA AOR. Because the thermal-hydraulic boundary conditions used in LOCTA_JR remain bounding, no changes to the current TPC LBLOCA or SBLOCA analyses are necessary, and the results of these analyses remain acceptable. Additionally, because the use of RFA-2 fuel has a negligible impact on core reactivity, there is no effect on the post-LOCA subcriticality and hot leg switchover time evaluations.

The NRC staff has determined that the licensee is applying approved methodologies and inputs in its assessment of the use of TPBARs in conjunction with the use of RFA-2 fuel. Since the results of the assessment using the approved methodologies and inputs met the acceptance criteria, the staff concludes that the amendment is acceptable in this regard.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes administrative requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). 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.

6.0 CONCLUSION

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

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