Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

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TVA-WBN-TS-02-13

U S. Nuclear Regulatory Commission
ATTN. Document Control Desk
Washington, D. C. 20555

10 CFR 50.90

Gentlemen·

In the Matter of)
Tennessee Valley Authority)

Docket No. 50-390

WATTS BAR NUCLEAR PLANT (WBN) - UNIT 1 - TECHNICAL SPECIFICATION (TS) CHANGE NO. TS-02-13 - REVISE SECTION 5.9.5 TO INCORPORATE ANALYTICAL METHODS FOR ROBUST FUEL ASSEMBLY (RFA)-2 UPGRADE

Pursuant to 10 CFR 50.90, TVA is submitting a request for an amendment to WBN's License NPF-90 to change the Technical Specifications (TS) for Unit 1. The proposed TS change will allow WBN Unit 1 to be refueled and operated using the Westinghouse 17x17 Robust Fuel Assembly-2 (RFA-2) fuel design with Intermediate Flow Mixers (IFMs), commencing with Cycle 6 in September 2003 In conjunction with this fuel change, the WRB-2M Departure from Nucleate Boiling (DNB) correlation is used for the DNB analysis of the RFA-2 fuel. This correlation provides additional DNB margin, has been reviewed and approved by the NRC, and is reflected in the WBN TS and TS Bases changes herein

The significant new mechanical features of the RFA-2 fuel design relative to the current fuel design include (1) structural mid-grid design changes resulting in improved DNB performance and improved resistance to fuel rod fretting, (2) three Intermediate Flow Mixer (IFM) grids to improve DNB performance, and (3) thicker-walled guide thimble and instrumentation tubes to improve fuel assembly stiffness. Although no particular precedent is cited for TVA's application, fuel products with similar fuel assembly design features (including IFMs) have operated successfully in transition cores and full cores at a number of commercial reactors. As discussed herein, the RFA-2 mid-grid design is the culmination of several changes which were evaluated by means of the NRC-approved Fuel Criteria Evaluation Process (FCEP), WCAP-12488-A As required by FCEP, WBN's fuel vendor, Westinghouse Electric Company, notified NRC by letter dated November 13, 2002, of the RFA-2 mid-grid design modifications and the validation of the WRB-2M DNB correlation applicability to the RFA-2 fuel. In addition, the attached report provides a discussion of applicable NRC Safety Evaluation Report (SER) conditional requirements for the DNB analysis of RFA-2 fuel

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Enclosure 1 and the Attachment (Safety Evaluation) to this letter provide the description and evaluation of the proposed TS change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosures 2 and 3 contain copies of the appropriate Unit 1 TS and TS Bases pages, respectively, marked-up to show the proposed changes. Enclosure 4 forwards the revised TS and TS Bases pages for Unit 1 which incorporate the proposed changes.

Section 7 of the enclosed report discusses the evaluation performed for the use of RFA-2 fuel in conjunction with the use of TPBARs (Tritium Producing Burnable Absorber Rods) approved for WBN in Amendment 40, September 23, 2002. Section 7.5 reflects the Refueling Water Storage Tank (RWST) and accumulator minimum boron concentrations for accident mitigation approved for TPBARs. TVA is considering submission of an amended TS to reflect a revised minimum boron concentration range based on a reduced number of TPBARs in the core. No impact to the RFA-2 analyses herein is expected from the change in boron concentration.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the TS change qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The WBN Plant Operations Review Committee and the WBN Nuclear Safety Review Board have reviewed this proposed change and have determined that operation of WBN Unit 1 in accordance with the proposed change will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Tennessee State Department of Public Health.

TVA requests approval of this amendment as soon as practical in support of the RFA-2 fuel manufacturing contract and the Fall 2003 Cycle 5 Refueling Outage, during which it will be installed. Further, TVA requests that implementation of the revised TS be prior to Mode 6 for the refueling outage. TVA is prepared to meet with the Staff if necessary, to facilitate the NRC's review.

There are no regulatory commitments associated with this submittal. If you have any questions about this proposed change, please contact me at (423) 365-1824

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 14th day of February, 2003.

Sincerely,

P. L. Pace
Manager, Site Licensing
and Industry Affairs

Enclosures

1. Description and Evaluation of Proposed Change
   (Attachment - Safety Evaluation)
2. Proposed Technical Specification Changes (mark-up)
3. Proposed Technical Specification Bases Changes (mark-up)
4. Proposed Technical Specification and TS Bases Changes (revised pages)

cc: See page 4

cc (Enclosures):

NRC Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

Mr. K. N. Jabbour, Senior Project Manager
U.S. Nuclear Regulatory Commission
MS 08G9
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2738

U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

Mr. Lawrence E. Nanny, Director (w/o Enclosures 2-6)
Division of Radiological Health
3rd Floor
L & C Annex
401 Church Street
Nashville, Tennessee 37243

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1
DOCKET NO. 390

PROPOSED LICENSE AMENDMENT REQUEST WBN-TS-02-13 - REVISE TS SECTION 5.9.5 TO INCORPORATE ANALYTICAL METHODS FOR ROBUST FUEL ASSEMBLY (RFA)-2 UPGRADE

DESCRIPTION AND EVALUATION OF PROPOSED CHANGE

1.0 DESCRIPTION

The purpose of this letter is to request an amendment to the Operating License NPF-90 and Technical Specifications for Watts Bar Nuclear Plant Unit 1.

The proposed license amendment will allow WBN Unit 1 to be refueled and operated using the Westinghouse 17x17 Robust Fuel Assembly-2 (RFA-2) fuel design with Intermediate Flow Mixers (IFMs), commencing with Cycle 6 in September 2003. In conjunction with this fuel change, the WRB-2M DNB correlation is used for the DNB analysis of the RFA-2 fuel. This correlation has been reviewed and approved by the NRC (WCAP-15025-P-A, April 1999), provides additional DNB margin, and is reflected in the WBN TS and TS Bases changes herein.

TVA is requesting approval of this amendment as soon as practical in support of the RFA-2 fuel manufacturing contract and the Fall 2003 Cycle 5 Refueling Outage.

2.0 PROPOSED CHANGE

The proposed amendment would revise the Watts Bar Nuclear Plant (WBN) Unit 1 Technical Specifications (TS), Section 5.9.5 (Core Operating Limits Report) to reference analytical methods used to determine core operating limits associated with RFA-2 fuel and the WRB-2M DNB correlation. Corresponding TS Bases changes are also proposed. The TS and TS Bases changes affect the following sections and are illustrated by marked-up and revised pages provided in Enclosures 2 through 4:

<u>Tech Spec:</u>

Section 5.9.5	Adds three references to analytical methods used to determine core operating limits associated with RFA-2 fuel and the WRB-2M DNB correlation.

<u>TS Bases:</u>

Section B 2.1.1 (Page B 2.0-2)	Added the WRB-2M correlation for RFA-2 to reflect the DNB correlation used in the analyses.
Section B 2.1.1 (Page B 2.0-4)	Revised and added design limit DNBR values to reflect the DNB correlation used in the analyses.
Section B 2.1.1	Added a reference for the WRB-2M correlation for RFA-2 to reflect the DNB

(Page B 2.0-6)	correlation used in the analyses
Section B 3.2.2 (Page B 3.2-13)	Added the WRB-2M correlation for RFA-2 to reflect the DNB correlation used in the analyses.

3.0 BACKGROUND AND SUMMARY

Refer to Section 1.0 of the Attachment to this Enclosure.

4.0 TECHNICAL ANALYSIS

Refer to the Attachment to this Enclosure.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

TVA is submitting a request for an amendment to the Watts Bar Nuclear Plant (WBN) Unit 1 Operating License NPF-90 and Technical Specifications (TS). The proposed amendment would allow Unit 1 to be refueled and operated using the Westinghouse 17x17 Robust Fuel Assembly-2 (RFA-2) fuel design with Intermediate Flow Mixers (IFMs). In conjunction with this fuel change, the WRB-2M Departure from Nucleate Boiling (DNB) correlation is used for the DNB analysis of the RFA-2 fuel. This correlation has been reviewed and approved by the NRC and provides additional DNB margin for operation of WBN Unit 1. The amendment would revise the TS to reference analytical methods used to determine core operating limits associated with RFA-2 fuel and the WRB-2M DNB correlation. TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendments(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The Loss of Coolant Accident (LOCA) and non-LOCA transients and accidents which are potentially affected by the parameters and assumptions associated with the use of RFA-2 (including the effects of Tritium Producing Burnable Absorber Rods, TPBARs) have been evaluated/analyzed and all design standards and applicable safety criteria are met. The consideration of these changes does not result in a situation where the design, material, and construction standards that were applicable prior to the change are altered. Therefore, the changes occurring with the use of RFA-2 will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident

   The changes associated with the use of RFA-2 do not affect plant systems such that their function in the control of radiological consequences is adversely affected. TVA's evaluation documents that the design standards and applicable safety criteria limits continue to be met and, therefore, fission barrier integrity is not challenged. The fuel rod design (the first fission product barrier) is not changed. Compared to the current grid design on the resident fuel, the RFA-2 grid design provides improved resistance to fuel rod fretting. The RFA-2 fuel changes have been shown not to adversely affect the response of the plant to postulated accident scenarios. These changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Final Safety Analysis Report (FSAR).

Therefore, since the actual plant configuration, performance of systems, and initiating event mechanisms are not being changed as a result of this evaluation, TVA has concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with the use of RFA-2 do not result in a change to the design basis of any plant component or system. The evaluation of the effects of the use of RFA-2 shows that all design standards and applicable safety criteria limits are met Specifically, the results of the evaluations/analyses lead to the following conclusions:

1. The RFA-2 fuel design for Watts Bar Unit 1 is mechanically compatible with the current fuel assemblies, core components, the control rods and the reactor internals interfaces.
2. The structural integrity of the RFA-2 fuel design has been evaluated for seismic/LOCA loadings for Watts Bar Unit 1. Evaluation of the RFA-2 fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents verified that the fuel assembly design is structurally acceptable
3. The changes to the nuclear characteristics due to the transition to the RFA-2 fuel assembly design will be within the range normally seen from cycle to cycle due to fuel management
4. The RFA-2 fuel assembly design is hydraulically compatible with the current fuel assemblies.
5. The core design and safety analyses documented in this report demonstrate the capability of the core to operate safely at the rated Watts Bar Unit 1 design thermal power with either a mixed core of RFA-2 fuel and the current fuel product or with a full core of RFA-2 fuel.
6. TVA's amendment request establishes a reference upon which to base Westinghouse reload safety evaluations for future reloads with the RFA-2 fuel assembly design.
7. Reload core designs with either a mixed core of RFA-2 fuel and the current fuel product or with a full core of RFA-2 fuel are compatible with the planned introduction of Tritium-Producing Burnable Absorber Rods (TPBARs) into Watts Bar Unit 1.

These changes therefore do not cause the initiation of any accident nor create any new failure mechanisms. All equipment important to safety will operate as designed. Component integrity is not challenged The changes do not result in any event previously deemed incredible being made credible. The use of RFA-2 is not expected to result in more adverse conditions and is not expected to result in any increase in the challenges to safety systems.

Therefore, TVA concludes that this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The margin of safety is maintained by assuring compliance with acceptance limits reviewed and approved by the NRC. All of the appropriate acceptance criteria for the various analyses and evaluations have been met, therefore, there has not been a reduction in any margin of safety.

Therefore, TVA concludes that the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

Regulatory requirements and criteria applicable to the design bases for fuel are provided in the WBN UFSAR Section 4.0. These include the applicable NRC General Design Criteria (GDC) of 10 CFR 50 Appendix A such as GDC-10, "Reactor Design," NRC Standard Review Plan (SRP)-4.2, "Fuel System Design," as well as supplemental criteria such as 10 CFR 50.46, (Acceptance Criteria for Emergency Core Cooling Systems), and applicable NRC-approved Westinghouse methodologies and analyses such as DNB critical heat flux correlations and fuel rod design evaluations.

In particular, as discussed in the attached Safety Evaluation, the RFA-2 mid-grid design is the culmination of several changes which were evaluated by means of the NRC-approved Fuel Criteria Evaluation Process (FCEP), WCAP-12488-A. As required by FCEP, Westinghouse notified NRC by letter dated November 13, 2002, of the RFA-2 mid-grid design modifications and the validation of the WRB-2M DNB correlation applicability to the RFA-2 mid-grid (Refer to Attachment, Reference 8). Section 8.0 of the attached Safety Evaluation provides a discussion of the applicable NRC Safety Evaluation Report conditional requirements for the DNB analysis of RFA-2 Fuel.

In conclusion, based on the considerations discussed above, (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.

6.0 ENVIRONMENTAL IMPACT CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

Refer to Section 9.0 of the Attachment to this Enclosure

ATTACHMENT

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

ROBUST FUEL ASSEMBLY (RFA-2) FUEL UPGRADE - TS-02-13
SAFETY EVALUATION

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1.0 Introduction and Summary

1.1 Introduction

Commencing with Cycle 6, Watts Bar Nuclear Plant Unit 1 will be refueled and operated with the Westinghouse 17x17 Robust Fuel Assembly-2 (RFA-2) fuel design with Intermediate Flow Mixers (IFMs). This report summarizes the evaluations and analyses that were performed to confirm the acceptable use of this fuel product for Watts Bar Unit 1 operation. Sections 2.0 through 6.0 provide the results of the mechanical, nuclear, thermal-hydraulic, accident, and reactor vessel and internals evaluations, respectively. Section 7.0 provides an evaluation of the impact of the RFA-2 fuel upgrade on the analyses supporting the use of Tritium Producing Burnable Absorber Rods (TPBARs), Reference 1. Section 8.0 addresses the NRC SER conditional requirements for the transition to a fuel product with IFMs as well as the NRC SER conditional requirements for the methodologies used to verify the DNB design basis.

This licensing amendment request serves as a reference safety evaluation and analysis report for the region-by-region reload transition from the Watts Bar Unit 1 Cycle 5 core to subsequent cores containing the upgraded features. Thus, the analysis results establish the new Analysis of Record that will be used to support future Watts Bar Unit 1 reload cores, through the transition cycles, with the upgraded fuel features.

The key safety parameters for the analyses performed in support of the fuel upgrade were chosen to maximize the applicability analysis results for future reload cycle evaluations which will be performed utilizing the Westinghouse standard reload methodology (Reference 2). The objective of subsequent cycle-specific reload evaluations will be to verify that applicable safety limits are not exceeded based on the reference analyses currently in the UFSAR or as established in this licensing amendment request.

1.2 Upgraded Fuel Features

Watts Bar Unit 1 Cycle 6 and subsequent core loadings will have the RFA-2 fuel design that incorporates Intermediate Flow Mixer grids. Throughout this report, the upgraded fuel assembly design is referred to as RFA-2. In the previous five cycles, Watts Bar Unit 1 has been operated with cores of Westinghouse 17x17 VANTAGE 5H fuel without IFMs (Reference 3) and 17x17 VANTAGE+ fuel without IFMs (Reference 4). These existing fuel products are described in the Watts Bar UFSAR. Consistent with the nomenclature in the UFSAR, the 17x17 VANTAGE 5H fuel without IFMs is referred to in this document as V5H. The 17x17 VANTAGE+ fuel without IFMs that has been used in Watts Bar Unit 1 includes PERFORMANCE+ features (Reference 5) and is referred to in the UFSAR and in this document as V+/P+. Throughout this report, the RFA-2 fuel is compared to the current V+/P+ fuel design that was loaded into Cycle 5. All of the V5H and V+/P+ regions that have been used in Watts Bar Unit 1 are mechanically and hydraulically equivalent. Therefore, all conclusions of compatibility between the RFA-2 fuel design and the current V+/P+ fuel design that are made in this report are applicable to any of the prior regions of V5H and V+/P+ that were used in Watts Bar Unit 1.

The significant new mechanical features of the RFA-2 fuel design relative to the current fuel design include:

- structural mid-grid design changes resulting in improved DNB performance and improved resistance to fuel rod fretting,
- three Intermediate Flow Mixer (IFM) grids to improve DNB performance, and
- thicker-walled guide thimble and instrumentation tubes to improve fuel assembly stiffness and to address Incomplete Rod Insertion (IRI) considerations.

In conjunction with this fuel change, the WRB-2M DNB correlation (Reference 6) is used for the DNB analysis of the RFA-2 fuel. This correlation, which has been reviewed and approved by the NRC, provides additional DNB margin.

The structural mid-grid design used in the RFA-2 fuel assembly is a modification of the Low Pressure Drop mid-grid design that was approved by the NRC (Reference 3) for use in the V5H and V+/P+ fuel assembly designs. The RFA-2 mid-grid design is the culmination of several changes which were evaluated by means of the NRC-approved Fuel Criteria Evaluation Process (FCEP), Reference 7. By complying with the requirements of FCEP, it has been demonstrated that the new mid-grid design meets all design criteria of existing tested mid-grids that form the basis of the WRB-2M correlation database and that the WRB-2M correlation with a 95/95 correlation limit of 1.14 applies to the new RFA-2 mid-grid. As required by FCEP, the Westinghouse notification to the NRC of the RFA-2 mid-grid design modifications and the validation of the WRB-2M DNB correlation applicability to the RFA-2 mid-grid was provided in Reference 8.

The IFM grid design used in the RFA-2 fuel assembly is a modification of the initial IFM grid design that was approved by the NRC as part of Reference 3. The applicability of the WRB-2M correlation with a 95/95 correlation limit of 1.14 to the IFM grid design that is used on the RFA-2 fuel assembly was approved by the NRC in Reference 6.

The functional interface of the RFA-2 fuel design with the reactor internals is equivalent to that of the Westinghouse V5H and V+/P+ fuel designs for which Watts Bar Unit 1 is currently licensed. Also, the RFA-2 fuel assembly is designed to be mechanically and hydraulically compatible with the V5H and V+/P+ fuel designs in full or transition cores. The same functional requirements and design criteria previously established for the Westinghouse V5H and V+/P+ fuel assembly designs in References 3 and 4, respectively, remain valid for the RFA-2 fuel assembly design.

1.3 Nuclear Steam Supply System Parameters

The Nuclear Steam Supply System (NSSS) design parameters are the fundamental parameters used as input in all of the NSSS analyses. They provide the primary and secondary side system conditions (temperatures, pressures, and flow) that are used as the basis for all of the NSSS analyses and evaluations. It was necessary to revise these parameters due to the fuel upgrade to RFA-2 with IFMs. The new parameters are identified in Table 1-1. These parameters have been incorporated, as required, into the applicable NSSS systems and components evaluations, as well as safety analyses, performed in support of the RFA-2 fuel upgrade. The basis for the RFA-2 fuel upgrade parameters was the 1.4% Power Uprate Program.

1.3.1 Input Parameters and Assumptions

The NSSS design parameters are determined based on conservative inputs, such as a conservatively low thermal design flow (TDF) and bounding steam generator tube plugging (SGTP) levels, which yield primary- and secondary-side conditions that bound plant operation.

Two cases (corresponding to 0% and 10% steam generator tube plugging) are provided for Watts Bar Unit 1 with the following assumptions common to both cases:

- 17x17 RFA-2 fuel with IFMs (full core),
- Core bypass flow of 9.6% (increased to account for implementation of IFMs),
- Model D3 Steam Generators,
- NSSS power level of 3475 MWt (3459 MWt core power + 16 MWt RCS net heat input),
- Nominal feedwater temperature (T_{feed}) of 441.8°F,
- Thermal Design Flow (TDF) of 93,100 gpm, and
- Vessel Average Temperature (T_{avg}) of 588.2°F.

1.3.2 Discussion of NSSS Parameter Changes

The RFA-2 fuel upgrade resulted in only minor changes to the NSSS design parameters. The increased core hydraulic resistance associated with the use of IFMs on the RFA-2 fuel assemblies results in an increase in core bypass flow from 9.0% to 9.6%. The core average and outlet temperatures increase slightly due to the increased bypass flow. As shown in Table 1-1, the core average temperature increased by 0.3°F while the core outlet temperature increased by 0.4°F. These changes were evaluated by each of the analytical areas discussed in this report.

The increased core hydraulic resistance does affect the calculated RCS best estimate flows. The minor reduction in the calculated RCS best estimate flow values did not result in a change to the design flow values listed in Table 1-1.

1.4 Conclusions

The results of the evaluations/analyses described herein lead to the following conclusions:

1. The RFA-2 fuel design for Watts Bar Unit 1 Cycle 6 is mechanically compatible with the current fuel assemblies, core components, the control rods and the reactor internals interfaces.
2. The structural integrity of the RFA-2 fuel design has been evaluated for seismic/LOCA loadings for Watts Bar Unit 1. Evaluation of the RFA-2 fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents verified that the fuel assembly design is structurally acceptable.
3. The changes to the nuclear characteristics due to the transition to the RFA-2 fuel assembly design will be within the range normally seen from cycle to cycle due to fuel management.
4. The RFA-2 fuel assembly design is hydraulically compatible with the current fuel assemblies.

5. The core design and safety analyses documented in this report demonstrate the capability of the core to operate safely at the rated Watts Bar Unit 1 design thermal power with either a mixed core of RFA-2 fuel and the current fuel product or with a full core of RFA-2 fuel.
6. This report establishes a reference upon which to base Westinghouse reload safety evaluations for future reloads with the RFA-2 fuel assembly design.
7. Reload core designs with either a mixed core of RFA-2 fuel and the current fuel product or with a full core of RFA-2 fuel are compatible with the planned introduction of Tritium-Producing Burnable Absorber Rods (TPBARs) into Watts Bar Unit 1.

Table 1-1 RFA-2 Fuel Upgrade Program NSSS Design Parameters For Watts Bar Unit 1

	Current Design Basis		RFA-2 Fuel Upgrade	
THERMAL DESIGN PARAMETERS	**Case 1**	**Case 2**	**Case 1**	**Case 2**
NSSS Power, %	100	100	100	100
MWt	3475	3475	3475	3475
10^6 BTU/hr	11,857[1]	11,857[1]	11,857[1]	11,857[1]
Reactor Power, MWt	3459	3459	3459	3459
10^6 BTU/hr	11,803	11,803	11,803	11,803
Thermal Design Flow, Loop gpm	93,100	93,100	93,100	93,100
Reactor 10^6 lb/hr	138.5	138.5	138.5	138.5
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	9.0	9.0	9.6 [2]	9.6 [2]
Reactor Coolant Temperature, °F				
Core Outlet	624.4	624.4	624.8	624.8
Vessel Outlet	619.1	619.1	619.1	619.1
Core Average	592.8	592.8	593.1	593.1
Vessel Average	588.2	588.2	588.2	588.2
Vessel/Core Inlet	557.3	557.3	557.3	557.3
Steam Generator Outlet	557.0	557.0	557.0	557.0
Steam Generator				
Steam Temperature, °F	542.1	538.0	542.1	538.0
Steam Pressure, psia	980	947	980	947
Steam Flow, 10^6 lb/hr total	15.39	15.36	15.39	15.36
Feed Temperature, °F	441.8	441.8	441.8	441.8
Moisture, % max.	0.25	0.25	0.25	0.25
Tube Plugging %	0	10	0	10
Zero Load Temperature, °F	557	557	557	557

HYDRAULIC DESIGN PARAMETERS	Current Design Basis	RFA-2 Fuel Upgrade
Mechanical Design Flow, gpm	105,000	105,000
Minimum Measured Flow, (gpm/total)	379,100 [3]	379,100 [3]

FOOTNOTES:

(1) Includes 16 MWt due to reactor coolant pumps.

(2) Core bypass flow increased by 0.6% to account for IFMs.

(3) Minimum Measured Flow assumes a flow measurement uncertainty of 1.8%.

2.0 Mechanical Design Features

2.1 Introduction and Summary

This section evaluates the mechanical design of the 17x17 RFA-2 fuel product and its compatibility with the V+/P+ fuel design currently in Watts Bar Unit 1. This evaluation addresses the transition from a full core of V+/P+ fuel to a full core of RFA-2 fuel. The RFA-2 fuel assembly has been designed to be compatible with the V+/P+ fuel assembly, the core components, the reactor internals interfaces, the fuel handling equipment and the refueling equipment. From an exterior assembly envelope and reactor internals interface standpoint, the RFA-2 design dimensions are essentially equivalent to the V+/P+ assembly design currently in Watts Bar Unit 1.

The significant new mechanical features of the RFA-2 fuel assembly design relative to the current V+/P+ fuel assembly design include:

- the addition of three Intermediate Flow Mixer (IFM) grids,
- incorporation of thicker-walled guide thimble and instrument tubes, and
- mid-grid design changes to improve mechanical and DNB performance.

The design differences associated with the new mechanical features are listed in Table 2-1.

2.2 Compatibility of Fuel Assemblies

The RFA-2 design for Watts Bar Unit 1 Cycle 6 and subsequent cycles incorporates three ZIRLO™ Intermediate Flow Mixing (IFM) grids. The RFA-2 fuel assembly skeleton is the same length as the V+/P+ fuel assembly. A comparison of the two fuel assemblies is shown in Figure 2-1. The changes from the V+/P+ design to RFA-2 design are listed in Table 2-2. The RFA-2 fuel assembly is designed to be mechanically and hydraulically compatible with the V+/P+ fuel assembly in full or transition cores. The same functional requirements and design criteria as previously established for the Westinghouse V+/P+ fuel assembly remain valid for the RFA-2 fuel assembly.

2.2.1 Fuel Rods

The fuel rod in the RFA-2 fuel design for Cycle 6 is identical to the V+/P+ fuel rod that was used in Region 7 for Cycle 5. Therefore, the mechanical design bases for the RFA-2 fuel rod remain unchanged from those used for the V+/P+ fuel rods in the Unit 1 Cycle 5 core.

2.2.2 Grid Assemblies

<u>RFA-2 Mid-grid Design</u>

The principle changes made to the modified mid-grid for RFA-2 versus the current Low Pressure Drop (LPD) mid-grid are changes to the vane pattern, vane length, spring window cut-outs, spring and dimple contact areas, and the incorporation of the anti-snag outer grid strap design. Compared to the current LPD grid, the RFA-2 changes reduce the potential for high frequency vibration and improved resistance to fuel rod fretting. The RFA-2 mid-grids are made from ZIRLO™ material. The mid-grids in the RFA-2 design are not rotated because of the revised vane pattern.

For the RFA-2 grid design, which has to accommodate a thicker guide thimble and instrumentation tube, the grid cells at the guide thimble and instrumentation tube locations are embossed (radiused) to accept the larger diameter tubes. In addition, the sleeve and the sleeve notch on the strap will also be bigger than the current sleeve to provide clearance for the larger diameter thimble tube. This improvement addresses high thimble tube loading forces during skeleton fabrication.

The RFA-2 mid-grid design was evaluated by means of the NRC-approved Fuel Criteria Evaluation Process (FCEP), Reference 7. As required by FCEP, the Westinghouse notification to the NRC of the RFA-2 mid-grid design modifications and the validation of the WRB-2M DNB correlation applicability to the RFA-2 mid-grid was provided in Reference 8.

RFA-2 IFM Grid Design

The IFM grids are located in the three uppermost spans between the ZIRLO™ mixing vane structural mid-grids (i.e., grids 4, 5, 6 and 7 in Figure 2-1). The primary function of the IFM grids is to provide enhanced mid-span flow mixing in the hottest fuel assembly spans. The mixing vanes on the RFA-2 IFM design are identical to the mixing vanes on the RFA-2 mid-grid design. Each IFM grid cell contains four dimples which are designed to prevent fuel rod contact with the mixing vanes. This simplified cell support arrangement allows for a shortened grid height (compared to the mid-grid design) so that the IFM can accomplish its flow mixing objective with minimal pressure drop.

The IFM grids are fabricated from ZIRLO™. This material was selected to take advantage of the inherent low neutron capture cross-section. Although these grids are not intended to function as structural components, the presence of the three IFM grids provides a small stiffening effect in the upper half of the RFA-2 fuel assembly compared to the V+/P+ design.

Top / Bottom Grid Design

The top Inconel non-mixing vane grid of the RFA-2 fuel assembly design is identical in design to the top Inconel non-mixing vane grid of the V+/P+ fuel assembly. Due to the new thimble dashpot OD, a new bottom grid insert tubing design is required for the bottom Inconel non-mixing vane grids. The insert tube ID was increased to interface with the larger thimble tube and the guide thimble end plug was modified to a slip fit interface with the thimble tube. This results in a minimal diameter increase locally at the weld and additional margin for fit up in the insert assembly. Both Inconel end grids continue to use the same grid cell support configuration as the ZIRLO™ mid-grids (i.e., six support locations per cell - four dimples and two springs).

Protective Grid Design

The Inconel protective grid is not changed in the RFA-2 fuel assembly design except for a new insert tubing design (similar to the bottom grid) which is due to the new larger thimble dashpot OD.

2.2.3 Guide Thimble and Instrumentation Tubes

The guide thimble tube diameter and the instrumentation tube diameter of the RFA-2 skeleton are different from those in the V+/P+ design. The guide thimble and instrumentation tube wall thickness in the RFA-2 design is increased by 25% to improve fuel assembly stiffness and to address incomplete rod insertion (IRI) considerations. The guide thimble tube and the instrument tube in the RFA-2 design have the same length as of those for the V+/P+ design. The general design bases for the RFA-2 guide thimble and instrumentation tubes remain the same as those given in Reference 9.

2.2.4 Top / Bottom Nozzle

RFA-2 Top Nozzle Design

The Reconstitutable Top Nozzle (RTN) for the RFA-2 fuel assembly design is similar to the RTN in the V+/P+ design. The same insert is used since the insert ID was large enough to accommodate the larger RFA-2 guide thimble. However, the RTN bulge joint height was slightly modified to accommodate the bulging process of a thicker guide thimble. As a result, all joints including the RTN joint were reviewed and/or tested to assure design load requirements continue to be met. The top nozzle assembly flow holes, the thermal and hydraulic characteristics, and the method of attachment to the guide thimble tubes are all unchanged from the RTN design that is currently in Watts Bar.

RFA-2 Bottom Nozzle Design

The debris filter bottom nozzle (DFBN) of the RFA-2 fuel assembly design is identical to the DFBN in the V+/P+ design except for the enlarged counter bore to accommodate the larger instrumentation tube diameter for the RFA-2 design.

2.3 Mechanical Performance

The RFA-2 design changes associated with the addition of three IFM grids, the mid-grid changes, and the incorporation of thicker-walled guide thimble and instrument tubes do not significantly influence the V+/P+ fuel assembly structural characteristics that were determined by prior mechanical testing. The same functional requirements and design criteria as previously established for the Westinghouse V+/P+ fuel assembly remain valid for the RFA-2 fuel assembly.

Hydraulic testing was performed to demonstrate compatibility of different fuel assembly designs in mixed core applications. From the test results, it was concluded that the RFA-2 fuel assembly design is hydraulically compatible in mixed core applications with the V5H or V+/P+ fuel assemblies previously used in Watts Bar.

For a mixed core of RFA-2 and V+/P+ fuel assemblies, the higher resistance RFA-2 fuel assemblies impose an increased lift force on the remaining V+/P+ fuel assemblies. There is sufficient margin in the V+/P+ hold-down springs design to meet the hold-down spring design criteria for the higher lift forces that can occur during a transition core due to the higher pressure drop associated with the RFA-2 fuel with IFMs.

Therefore, the RFA-2 fuel assembly structural behavior and projected performance is compatible with the current V+/P+ fuel assembly design.

2.4 Fuel Rod Performance

The transition to RFA-2 fuel will result in no fuel rod design change. The only impact is a negligible variation in the core mass flow rate due to the introduction of Intermediate Flow Mixers (IFM). The design features in the RFA-2 fuel rod for Cycle 6 are identical to the V+/P+ fuel rod that was used in Region 7 for Cycle 5. Therefore, fuel rod performance for all Watts Bar fuel can satisfy the NRC Standard Review Plan (SRP) fuel rod design bases on a region-by-region basis. These same bases are applicable to all fuel rod designs, including the Westinghouse V+/P+ and RFA-2 fuel designs. The design bases for Westinghouse fuel are discussed in Reference 7.

There is no impact from a fuel rod design standpoint due to having fuel with more than one type of geometry simultaneously residing in the core during the transition cycles except for the negligible variation in the core mass flow rate. The mechanical fuel rod design evaluation for each region incorporates all appropriate design features of the region, including any changes to the fuel rod or pellet geometry from that of previous fuel regions (such as the presence of axial blankets or changes in the fuel rod and plenum length, for example). Analysis of Integral Fuel Burnable Absorber (IFBA) rods includes any geometry changes necessary to model the presence of the burnable absorber, and conservatively models the gas release from the IFBA coating. Fuel performance evaluations are completed for each fuel region to demonstrate that the design criteria will be satisfied for all fuel rod types in the core under the planned operating conditions. Any changes from the plant operating conditions originally evaluated for the mechanical design of a fuel region (for example, a power uprating or a change to the core flow rate) are addressed for all affected fuel regions as part of the reload safety evaluation process when the plant change is to be implemented.

Fuel rod design evaluations for Watts Bar are performed using the NRC-approved models (References 4 and 10) and NRC-approved design criteria and methods (References 11 and 12) to demonstrate that the SRP fuel rod design criteria are satisfied. No new fuel rod design methods are required to support the RFA-2 fuel upgrade in Watts Bar Unit 1. Independent of the fuel upgrade, the improved fuel performance analysis and design model (PAD 4.0) will be implemented for the verification of fuel rod design criterion for Cycle 6. PAD 4.0 has been approved by the NRC (Reference 10) and will be used in place of PAD 3.4 (Reference 13).

2.5 Seismic/LOCA Impact on Fuel Assemblies

The RFA-2 fuel assembly responses under seismic and LOCA load conditions were analyzed to support the Watts Bar Unit 1 fuel upgrade. The grid loads resulting from a combined safe shutdown earthquake and a LOCA loading condition are well below the allowable grid strengths for both homogeneous core and transitional mixed cores with the V+/P+ fuel design that is currently used in Watts Bar Unit 1. The results show adequate grid load margin and that the core coolable geometry and control rod insertion requirements are met.

An evaluation of the structural integrity of the RFA-2 fuel with IFMs was performed considering the lateral effects of a LOCA and a seismic accident. Although the IFM grids are not intended to function as structural components, they provide a small stiffening effect near the top of the fuel assembly. The grid load results show that RFA-2 fuel assembly has ample margin in withstanding the faulted condition transient loads.

The evaluation of the RFA-2 fuel assembly in accordance with NRC requirements as given in SRP 4.2 Appendix A, shows that the RFA-2 fuel assembly is structurally acceptable for Watts Bar Unit 1. The grid loads evaluated for the LOCA and seismic events and combined by the square root sum of the squares (SRSS) method identified in SRP 4.2 are less than the allowable limit. Thus, the core coolable geometry is maintained. The stresses in the fuel assembly components resulting from the seismic and LOCA induced deflections and the vertical impact loads are within acceptable limits. The reactor can be safely shutdown under the combined faulted condition loads.

2.6 Core Components

The RFA-2 design is compatible with the existing core component designs that are used in Watts Bar. The RFA-2 thimble tubes retain the same ID as the V+/P+ thimble tubes and, therefore, provide sufficient clearance for insertion of B_4C control rods, Wet Annular Burnable Absorber (WABA) rods, source rods, Tritium Producing Burnable Absorber Rods (TPBARs), and dually compatible thimble plugs to assure the proper operation of these core components.

Figure 2-1 Comparison of the 17x17 RFA-2 and V+/P+ Fuel Assemblies

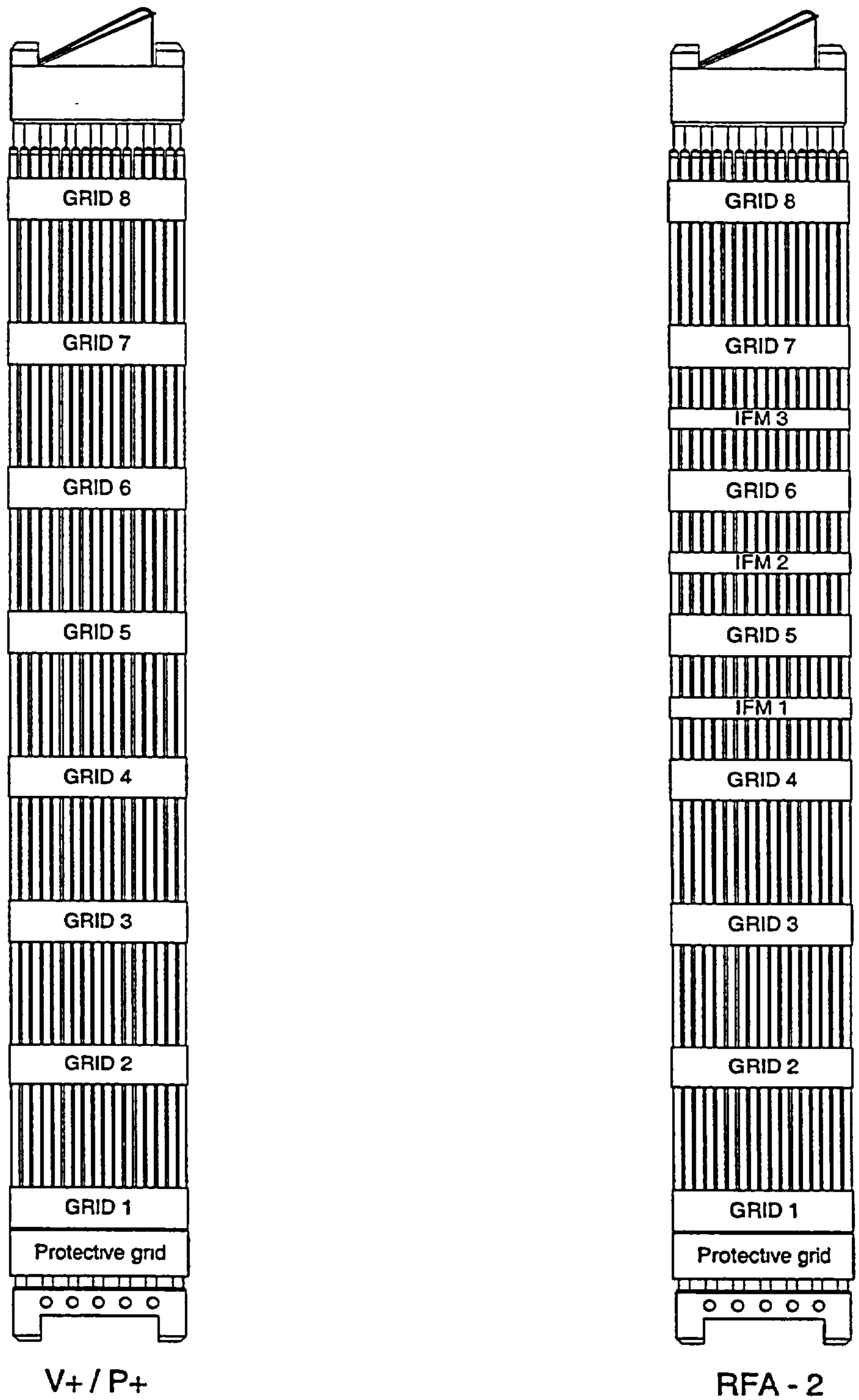

Table 2-1 Comparison of RFA-2 and V+/P+ Fuel Assembly Mechanical Design Parameters

	Robust Fuel Assembly (RFA-2)	**V+/P+ Fuel Assembly**
Fuel Assembly Overall Length, inch	159.975	159.975
Fuel Rod Overall Length, inch	152.8	152.8
Assembly Envelope, inch	8.426	8.426
Fuel Rod Pitch, inch	0.496	0.496
Number of Fuel Rods/Assembly	264	264
Number of Guide Thimbles/Assembly	24	24
Number of Instrumentation Tubes/Assembly	1	1
Fuel Tube Material	ZIRLO™	ZIRLO™
Fuel Tube Clad OD, inch	0.374	0.374
Fuel Rod Clad Thickness, inch	0.0225	0.0225
Fuel Clad Gap, mil	6.5 (uncoated pellets)	6.5 (uncoated pellets)
Fuel Pellet Diameter, inch	0.3225 (uncoated pellets)	0.3225 (uncoated pellets)
Guide Thimble Material	ZIRLO™	ZIRLO™
Guide Thimble OD, inch	**0.482**	0.474
Guide Thimble ID, inch	0.442	0.442
Guide Thimble Wall Thickness, inch	**0.020**	0.016
Thimble Dashpot OD, inch	**0.439**	0.430
Thimble Dashpot ID, inch	0.397	0.397
Instrumentation Tube OD, inch	**0.482**	0.474
Instrumentation Tube ID, inch	0.442	0.442
Grid Material, Inner Mid-Grid (6)	ZIRLO™	ZIRLO™
Grid Types Utilized ZIRLO™ Mid-grids ZIRLO™ Intermediate Flow Mixers Inconel Top & Bottom Grids Inconel Protective Grid	 Yes Yes Yes Yes	 Yes No Yes Yes
Compatible with Fuel Handling Equipment	Yes	Yes

Table 2-2 17x17 Design Changes for RFA-2

Component	Changes from V+/P+
Guide Thimble	0.482 in. OD x 0.442 in. ID above dashpot; 0.439 in. OD x 0.397 in. ID at dashpot. Modified thimble end plug (larger diameter).
Instrument Tube	0.482 in. OD x 0.442 in. ID.
Top Grid	No changes
IFM Grid	Incorporation of 3 IFM Grids
Mid-Grid	Modified the mid-grid design with radiused guide thimble cells to accept larger OD guide thimble and instrumentation tube. Larger sleeve OD/ID. Modified grid spring window. Modified spring contact area and dimple contact area.
Bottom Grid	Modified bottom insert OD/ID to accept thicker guide thimble
Protective Grid	Modified bottom insert OD/ID to accept thicker guide thimble
Top Nozzle	No changes
Bottom Nozzle	Enlarge counter bore for instrumentation tube
Fuel Rod	No changes

3.0 Nuclear Design

3.1 Introduction and Summary

The effects of using RFA-2 fuel features with IFMs on the nuclear design bases and methodologies for Watts Bar Unit 1 are evaluated in this section.

The specific values of core safety parameters, e.g., power distributions, peaking factors, rod worths, are primarily loading pattern dependent. The variations in the loading pattern dependent safety parameters are expected to be typical of the normal cycle-to-cycle variations for the standard fuel reloads. The evaluations have shown that the RFA-2 nuclear design bases are satisfied. Margin to key safety parameter limits remain unchanged for the RFA-2 fuel design. Standard nuclear design analytical models and methods (References 2, 14 and 15) accurately describe the neutronic behavior of the RFA-2 fuel design. In addition, the present Watts Bar Unit 1 fresh fuel rack criticality analysis is applicable to the RFA-2.

In summary, the changes from the current VANTAGE+/PERFORMANCE+ (V+/P+) fuel core to a core containing RFA-2 fuel will not cause changes to the current Watts Bar Unit 1 UFSAR nuclear design bases. Nuclear design methodology is not affected by the use of upgraded fuel features.

3.2 Design Basis

The specific design bases and their relation to the General Design Criteria (GDC) in 10 CFR 50, Appendix A for the RFA-2 with IFMs design are the same as those of the V+/P+ without IFMs. The addition of IFMs does not significantly affect the nuclear design.

There is no change resulting from the RFA-2 transition which would invalidate analyses, direction or prevention mechanisms with regard to a fuel misload event. The analysis reported in the UFSAR with regards to a fuel misload, Section 15.3.3, remains valid.

3.3 Methodology

The purpose of this reload transition core analysis is to determine, prior to the cycle specific reload design, if the previously used values for the key safety parameters remain applicable for the transition to 17x17 RFA-2 fuel. This will allow the majority of any safety analysis re-evaluations/re-analyses to be completed prior to the cycle specific design analysis.

No changes to the standard Westinghouse nuclear design philosophy, methods or models are necessary because of the transition to RFA-2 fuel. The reload design philosophy includes the evaluation of the reload core key safety parameters which comprise the nuclear design dependent input to the UFSAR safety evaluation for each reload cycle (Reference 2). These key safety parameters will be evaluated for each Watts Bar Unit 1 reload cycle. If one or more of the parameters fall outside the bounds assumed in the safety analysis, the affected transients will be re-evaluated/re-analyzed and the results documented in the RSE for that cycle.

3.4 Nuclear Design Evaluation Conclusions

The key safety parameters evaluated for Watts Bar Unit 1 as it transitions to an all 17x17 RFA-2 core show little change relative to the current design. The changes in values of the key safety parameters are typical of the normal cycle-to-cycle variations experienced as loading patterns change. The usual methods of enrichment and burnable absorber usage will be employed in the transition and full 17x17 RFA-2 cores to ensure compliance with the Peaking Factor Technical Specifications.

4.0 Thermal and Hydraulic Design

4.1 Introduction and Summary

This section describes the calculational methods used for the thermal-hydraulic analysis, the DNB performance, and the hydraulic compatibility during the transition from a full core of V+/P+ fuel (without IFMs), through mixed-fuel cores, to a full core of RFA-2 fuel with IFMs. Based on the design differences and prototype hydraulic testing of the fuel assemblies, it is concluded that the V+/P+ and RFA-2 fuel assembly designs are hydraulically compatible. Table 4-1 summarizes the thermal-hydraulic design parameters for Watts Bar Unit 1 that were used in this analysis. The thermal-hydraulic design criteria and methods remain the same as those presented in the Watts Bar Unit 1 UFSAR with the exceptions noted in the following sections. All of the current UFSAR thermal-hydraulic design criteria are satisfied.

4.2 DNB Methodology and Analyses

As described in the Watts Bar Nuclear Plant UFSAR, the existing thermal-hydraulic analysis of the V+/P+ fuel is based on the Revised Thermal Design Procedure (RTDP), Reference 16, the WRB-1 DNB correlation, Reference 17, and the VIPRE computer code (as licensed by Westinghouse in Reference 18). For the RFA-2 fuel upgrade, the primary DNB correlation used in the analysis of the RFA-2 fuel is the WRB-2M DNB correlation (Reference 19). The RFA-2 mid-grid design was evaluated by means of the NRC-approved Fuel Criteria Evaluation Process (FCEP), Reference 7. By complying with the requirements of FCEP, it has been demonstrated that the new mid-grid design meets all design criteria of existing tested mid-grids that form the basis of the WRB-2M correlation database and that the WRB-2M correlation with a 95/95 correlation limit of 1.14 applies to the new RFA-2 mid-grid. As required by FCEP, the Westinghouse notification to the NRC of the RFA-2 mid-grid design modifications and the validation of the WRB-2M DNB correlation applicability to the RFA-2 mid-grid was provided in Reference 8. The WRB-1 DNB correlation will continue to be used for the V+/P+ fuel. The W-3 correlation is used when conditions are outside the range of the WRB-1 or WRB-2M correlation.

With RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor which is used to define the design limit DNBR that satisfies the DNB design criterion. The DNB design criterion is that there will be at least a 95% probability (at 95% confidence level) that DNB will not occur on the limiting fuel rods for any Condition I or II event. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR, the plant safety analyses are performed using input parameters at their nominal values.

Instrumentation uncertainties are documented in the Watts Bar RTDP Instrument Uncertainty Methodology Report (Reference 20). Conservative values for the uncertainties relative to the above report were used in determining the RTDP DNBR design limits.

Only the random portion of each plant operating parameter uncertainty is included in the statistical combination for RTDP. Any adverse instrumentation bias is treated as a direct DNBR penalty or as a direct analysis input.

For this application, the Design Limit DNBR for the RFA-2 fuel is 1.23 for both typical cells and thimble cells. The Design Limit DNBR values for the V+/P+ fuel remain unchanged with values of 1.25 for typical cells and 1.24 for thimble cells. For use in the DNB safety analyses, the limit DNBR is conservatively increased to provide DNB margin to offset the effect of rod bow, transition core, instrumentation bias and any other DNB penalties that may occur, and to provide flexibility in design and operation of the plant.

To obtain sufficient DNB margin to cover the DNB penalties for the V+/P+ fuel, including the additional penalty from the increased bypass flow associated with the transition to RFA-2 fuel, the design $F_{\Delta H}$ limit for the V+/P+ fuel is reduced from 1.65 to 1.62. For RFA-2 fuel, for which sufficient DNB margin exists due to the use of IFMs and the WRB-2M correlation, the design $F_{\Delta H}$ limit is 1.65.

As noted in the UFSAR, the standard thermal design procedure (STDP) is used for those DNB analyses where RTDP is not applicable. In the STDP method, the parameters used in the analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety analysis input values to give the lowest minimum DNBR. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

4.3 Hydraulic Compatibility

The RFA-2 design for Watts Bar Unit 1 incorporates Intermediate Flow Mixer (IFM) grids. Hydraulic testing was performed to demonstrate compatibility of different fuel assembly designs in mixed core applications. From the test results, it was concluded that the RFA-2 fuel assembly design is hydraulically compatible in mixed core applications with the V5H or V+/P+ fuel assemblies previously used in Watts Bar.

The incorporation of RFA-2 (with IFMs) has an impact on the fuel assembly pressure drop. The RFA-2 fuel results in an increased core pressure drop of approximately 10%. This pressure drop effect impacts the primary loop conditions. As discussed in Section 1.3, the best estimate RCS flow is reduced due to the RFA-2 implementation, whereas the Minimum Measured Flow and Thermal Design Flow remain unchanged. The increased flow resistance of the RFA-2 fuel caused by the IFM grids also results in an increased core bypass flow. To account for this impact, the design core bypass flow is increased from 9.0% to 9.6%. The NSSS design parameters for the RFA-2 fuel upgrade are shown in Table 1-1 in Section 1.0.

4.4 Effects of Fuel Rod Bow on DNBR

The phenomenon of fuel rod bowing must be accounted for in the DNBR safety analysis of Condition I and Condition II events. Currently, the maximum rod bow DNBR penalty is less than 2.0% for a 20.6 inch grid span, at an assembly average burnup of 24,000 MWD/MTU. For burnups greater than 24,000 MWD/MTU, credit is taken for the effect of $F_{\Delta H}$ burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory. Therefore, no additional rod bow penalty is required at burnups greater than 24,000 MWD/MTU. Based on the similarities between V+/P+ and RFA-2 fuel assemblies, (i.e., fuel rod diameter, fuel rod pitch and grid spacing in the non-IFM spans), these rod bow DNBR penalties are also applicable to RFA-2 fuel assemblies in the grid spans without IFMs. No rod bow penalty is required in the 10 inch grid spacing regions resulting from the use of IFM grids (Reference 3).

As discussed previously, the rod bow penalty is offset with retained DNB margin.

4.5 Fuel Temperature Analysis

The fuel rods in the RFA-2 fuel design for Watts Bar Unit 1 Cycle 6 are identical to the V+/P+ fuel rods that were used in Region 7 for Cycle 5. Fuel temperatures for use in the safety analyses were calculated with the PAD 4.0 fuel performance code (Reference 10). These fuel temperatures are applicable for the V+/P+ fuel and the RFA-2 fuel.

4.6 Transition Core DNB Effect

The RFA-2 fuel assembly has IFM grids located in spans between mixing vane grids, whereas no IFM grid exists in the V+/P+ assembly. The additional grids introduce localized flow redistribution from the RFA-2 fuel assembly into the V+/P+ assembly at the axial zones near the IFM grid position in a transition core. Between the IFM grids, flow returns to the RFA-2 fuel assembly. The localized flow redistribution described above actually benefits the V+/P+ assembly. Thus, the analysis for a full core of V+/P+ fuel assemblies remains appropriate for that fuel type in a transition core.

Transition cores are analyzed as full cores of one assembly type (full core of V+/P+ or full core of RFA-2) and the appropriate transition core DNBR penalty is applied to address the effects of the mixed core. Based on the NRC-approved methodology in Reference 21, the transition core DNBR penalty is a function of the number of each type of fuel assembly in the core. The transition core DNBR penalty is zero for the V+/P+ fuel. As discussed previously, the transition core DNB penalty for the RFA-2 fuel is offset with retained DNBR margin. The DNBR penalty on the RFA-2 fuel will be less than 12% for the first transition core. By the methodology in Reference 21, the DNBR penalty is reduced for subsequent transition cores as the fraction of RFA-2 fuel with IFMs increases. For full core of RFA-2 fuel, this DNBR penalty is no longer applicable.

4.7 Conclusion

The thermal hydraulic evaluation of the fuel upgrade for Watts Bar Unit 1 has shown that the V+/P+ and RFA-2 fuel assemblies are hydraulically compatible and that sufficient DNB margin is available to cover the applicable penalties. All current thermal-hydraulic design criteria are satisfied.

Table 4-1 Watts Bar Unit 1 Thermal and Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters (using RTDP)	Design Parameters
Reactor Core Heat Output, MWt	3,459
Reactor Core Heat Output, 10^6, BTU/Hr	11,803
Heat Generated in Fuel, %	97.4
Pressurizer Pressure, Nominal, psia	2250
Radial Power Distribution*	
V+/P+ (w/o IFMs)	1.62[1+0.3(1-P)]
RFA-2 (w/ IFMs)	1.65[1+0.3(1-P)]
	where P = Fraction of Full Power
HFP Nominal Coolant Conditions	
Vessel Thermal Design Flow (TDF)	
Rate (including Bypass), 10^6 lbm/hr	138.5
GPM	372,400
Core Flow Rate (excluding Bypass, based on TDF)	
10^6 lbm/hr	125.2
GPM	336,650
Core Flow Area, ft^2	51.1
Core Inlet Mass Velocity,	
10^6 lbm/hr-ft^2 (Based on TDF)	2.45

* Includes 4% measurement uncertainty.

Table 4-1 (continued) **Watts Bar Unit 1 Thermal and Hydraulic Design Parameters**

Thermal and Hydraulic Design Parameters (based on TDF)	Design Parameters
Nominal Vessel/Core Inlet Temperature, °F	557.3
Vessel Average Temperature, °F	588.2
Core Average Temperature, °F	593.1
Vessel Outlet Temperature, °F	619.1
Average Temperature Rise in Vessel, °F	61.8
Average Temperature Rise in Core, °F	67.5
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	59,742
Average Heat Flux, BTU/hr-ft^2	192,500
Maximum Heat Flux for Normal Operation, ⁺BTU/hr-ft^2	481,300
Average Linear Power, kW/ft	5.52
Peak Linear Power for Normal Operation,⁺ kW/ft	13.8
Peak Linear Power for Prevention of Centerline Melt, kW/ft	22.4
Pressure Drop Across Core,‡ psi	
Full core of V+/P+ (w/o IFMs)	24.7
Full core of RFA-2 (w/ IFMs)	26.1

⁺ Based on Maximum F_Q of 2.50.

‡ Based on total flow rate of 405,200 gpm for V+/P+ core and 402,400 gpm for RFA-2 core.

5.0 Accident Analysis

This report is provided to support an upgrade to RFA-2 fuel with Intermediate Flow Mixing Grids (IFMs). Justification for the RFA-2 with IFMs fuel upgrade program is summarized for the Non-LOCA design basis calculations in Section 5.1 and for the LOCA design basis calculations in Section 5.2.

5.1 Non-LOCA Accidents

5.1.0 Introduction

This section summarizes the Non-LOCA evaluations and analyses performed to support the RFA-2 with IFMs implementation at Watts Bar Unit 1.

5.1.0.1 Fuel Features

The fuel features that were evaluated are:

a. Intermediate Flow Mixing (IFMs) grids,
b. ZIRLO™ guide thimble and instrumentation tube wall thickness increase of 25%,
c. Fuel assembly flow area decrease from 38.2590 in^2 to 38.1089 in^2, and
d. Modified ZIRLO™ mid-grids with slightly increased mass and volume.

5.1.0.2 Other Major Assumptions

The following analysis assumptions have not changed from the values used in the analyses supporting the 1.4% power uprating for Watts Bar Unit 1.

a. An NSSS power level of 3475 MWt
b. A reactor thermal power of 3459 MWt
c. A reactor coolant system Thermal Design Flow (TDF) of 93,100 gpm/loop
d. A reactor coolant system Minimum Measured Flow (MMF) of 94,775 gpm/loop
e. An average vessel average coolant temperature of 588.2°F
f. An average reactor coolant pressure of 2250 psia
g. An average maximum steam generator tube plugging (SGTP) of 10%

For most accidents that are DNB-limited, nominal values of the initial conditions are assumed. The uncertainty allowances on power, temperature, pressure, and RCS flow are included on a statistical basis and are included in the limit DNBR value by using the Revised Thermal Design Procedure (RTDP) (Reference 16).

For accident analyses which are not DNB-limited, or for which RTDP is not used, the initial conditions are obtained by applying the maximum steady-state errors to rated values.

Accidents using RTDP assume a Minimum Measured Flow (MMF), while others assume the Thermal Design Flow (TDF) for total RCS flow. The MMF is bounded by the minimum flow measurement requirement of 380,000 gpm (95,000 gpm/loop) in Technical Specification 3.4.1. The MMF includes an allowance for plant flow measurement uncertainty.

5.1.0.3 Overtemperature-ΔT and Overpower-ΔT

The overtemperature-ΔT (OTΔT) and overpower-ΔT (OPΔT) setpoints remain unchanged for the RFA-2 fuel upgrade program based on the current core thermal limits which are conservative for RFA-2 fuel. These core thermal limits were used to bound mixed cores and full cores with the RFA-2 fuel features analyzed. The core thermal limits on which the OTΔT and OPΔT setpoints are based are provided in Technical Specifications Figure 2.1.1-1. The existing OTΔT and OPΔT setpoints continue to protect the core thermal limits for the RFA-2 fuel for all of the UFSAR events that rely on OTΔT and OPΔT for protection.

5.1.0.4 RCCA Reactivity Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in rod worth as a function of rod position. With respect to the accident analyses, the critical parameter is the time from beginning of RCCA insertion to dashpot entry, or approximately 85% of the RCCA travel. For the accident analyses, the insertion time from fully withdrawn to dashpot entry remains at the Technical Specification limit of 2.7 seconds from the beginning of stationary gripper coil voltage decay.

The rod drop time for the RFA-2 fuel with IFMs remains at 2.7 seconds from fully withdrawn to the dashpot, and the reactivity worth of the control rods remains consistent with that of the existing V+/P+ fuel. The normalized RCCA position (fraction of insertion) versus the normalized time from release remains consistent with UFSAR Figure 15.1-2. The reactivity worth versus rod insertion (fraction) assumed in the safety analyses remains consistent with UFSAR Figure 15.1-3.

5.1.0.5 Reactivity Coefficients

The transient response of the RCS is dependent on reactivity feedback effects, in particular the moderator density coefficient and the Doppler Power Coefficient (DPC). Depending upon event specific characteristics, conservatism dictates use of either large or small reactivity coefficient values. Justification for the use of the reactivity coefficient values is treated on an event-specific basis.

The maximum and minimum integrated DPCs assumed in the safety analyses remain consistent with those presented in UFSAR Figure 15.1-5. Note that the Major Rupture of a Main Steam Line analysis uses a different DPC based on one "stuck" RCCA.

5.1.0.6 Computer Codes Utilized

The principal computer codes used in the Non-LOCA transient analyses are given below.

- LOFTRAN (Reference 22)
- FACTRAN (Reference 23)
- TWINKLE (Reference 24)
- VIPRE (Reference 18)

5.1.0.7 Classification of Events

Each of the events presented in Table 5.1.0-1 is classified by the American Nuclear Society (ANS) according to expected frequency of occurrence and severity of the accident. Each classification has specific acceptance criteria, as presented in Watts Bar UFSAR Section 15 and summarized below. None of the RFA-2 fuel upgrade assumptions described in this report affect the classification of transients discussed in this section.

Condition I transients include steady-state and shutdown operation, as well as operational transients such as heatup, cooldown, and load changes. Occurrences are accommodated with margin to automatic or manual protective action and are not discussed in the safety analyses. Consequently, operational transients are not included in the Non-LOCA section of the report.

Condition II transients are classified as incidents of moderate frequency. Any one of these transients may occur during a calendar year. The design requirements supported by the safety analyses are:

- Occurrences are accommodated with, at most, a reactor trip with the plant capable of returning to operation;

- Release of radioactive materials in effluent to unrestricted areas shall be in conformance with 10CFR20;

- Condition II incidents shall not generate a more serious incident (Condition III or IV) without other incidents occurring independently; and

- There shall be no consequential loss of function of any barrier to the escape of radioactive products (no fuel rod failure or overpressurization).

Condition III transients are classified as infrequent incidents. Any one of these transients may occur during the lifetime of a plant. The design requirements supported by the safety analyses are:

- Occurrences shall not cause more than a small fraction of the fuel elements to be damaged;
- Release of radioactive materials may exceed the guidelines of 10CFR20 but shall be a small fraction of the 10CFR100 limits;
- Radioactive releases shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius (shall remain well below 10CFR100);
- Condition III incidents shall not generate a Condition IV fault; and
- There shall be no consequential loss of function of the RCS or reactor containment barriers.

Condition IV transients are classified as limiting faults. These faults are not expected to occur but are postulated because their consequences would include the potential for radioactive releases. The Condition IV faults are the most drastic design basis events. The design requirements supported by the safety analyses are:

- Release of radioactive material shall not result in any undue risk to public health and safety and do not exceed the guidelines of 10CFR100; and
- There shall be no consequential loss of function of systems needed to cope with the event.

5.1.0.8 Events Evaluated or Analyzed

The effect of the RFA-2 fuel upgrade implementation on each of the Watts Bar Non-LOCA UFSAR transients listed on Table 5.1.0-1 were evaluated with the exception of the Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) event, which was explicitly analyzed. These 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 Watts Bar Unit 1.

Table 5.1.0-1 Non-LOCA Transients Evaluated / Analyzed for the Watts Bar RFA-2 Fuel Upgrade Program

Non-LOCA Transient	*UFSAR Section*	*Report Section*
CONDITION II – FAULTS OF MODERATE FREQUENCY	*15.2*	*5.1.2*
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition	15.2.1	5.1.2.1
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	15.2.2	5.1.2.2
Rod Cluster Control Assembly Misalignment	15.2.3	5.1.2.3
Uncontrolled Boron Dilution	15.2.4	5.1.2.4
Partial Loss of Forced Reactor Coolant Flow	15.2.5	5.1.2.5
Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	15.2.6	5.1.2.6
Loss of External Electrical Load and / or Turbine Trip	15.2.7	5.1.2.7
Loss of Normal Feedwater	15.2.8	5.1.2.8
Coincident Loss of Onsite and External (Offsite) AC Power to the Station – Loss of Offsite Power to the Station Auxiliaries	15.2.9	5.1.2.9
Excessive Heat Removal Due to Feedwater System Malfunctions	15.2.10	5.1.2.10
Excessive Load Increase Incident	15.2.11	5.1.2.11
Accidental Depressurization of the Reactor Coolant System	15.2.12	5.1.2.12
Accidental Depressurization of the Main Steam System	15.2.13	5.1.2.13
Inadvertent Operation of Emergency Core Cooling System	15.2.14	5.1.2.14
CONDITION III – INFREQUENT FAULTS	*15.3*	*5.1.3*
Minor Secondary System Pipe Breaks	15.3.2	5.1.3.1
Complete Loss of Forced Reactor Coolant Flow	15.3.4	5.1.2.5
Single RCCA Withdrawal at Full Power	15.3.6	5.1.3.2
CONDITION IV – LIMITING FAULTS	*15.4*	*5.1.4*
MAJOR SECONDARY SYSTEM PIPE RUPTURES:	*15.4.2*	*N/A*
Major Rupture of a Main Steam Line	15.4.2.1	5.1.4.1
Major Rupture of a Main Feedwater Pipe	15.4.2.2	5.1.4.2
Single Reactor Coolant Pump Locked Rotor	15.4.4	5.1.2.5
Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	15.4.6	5.1.4.3

Note: For the Steam Generator Tube Rupture, Fuel Handling Accident, Waste Gas Decay Tank Rupture, and Main Steam Line Break Events, the radiological effects are TVA-analyzed and are addressed in Section 5.1.6.

5.1.1 Evaluation of Changes Caused by the RFA-2 Fuel Transition

5.1.1.1 RTDP and Non-RTDP Core Bypass Flow Fraction (CBFF)

The best estimate and design core bypass flow fractions (CBFF) have increased due to the fuel change. The current (V+/P+ fuel) CBFF values for the RTDP and non-RTDP methodologies are 7.5% and 9.0%, respectively. These values have increased to 8.6% and 9.6%, respectively, for the RFA-2 fuel upgrade program, due to the addition of the IFM grids.

The best estimate CBFF value is used in the RTDP-based analyses (i.e., events that are DNB-limited) where uncertainties on parameters such as CBFF, nominal power level, and average RCS temperature, are statistically combined via the DNBR correlation applied to the analysis statepoints. The design CBFF is used in the DNB-limited events that initiate from Hot Zero Power and remaining Non-LOCA analyses that are not DNB-limited. Instead, uncertainties on CBFF, nominal power level, and average RCS temperature, are explicitly modeled in the transient analyses. See Section 5.1.0.2 for further details.

In general, CBFF increases of this magnitude (approximately 1%) do not significantly affect the Non-LOCA analyses. Because the overall system parameters in LOFTRAN are not sensitive to the increase in CBFF caused by the RFA-2 fuel, reactor coolant loop system results, such as peak RCS pressure, pressurizer water volume, or secondary system parameters, are not adversely impacted. As such, there is no adverse impact on the Non-LOCA events driven by overall system performance, such as Loss of Load / Turbine Trip, RCS Depressurization, LONF, LOOP, and Feedline Break (FLB). In addition, the "system" portion of the DNB transients is not adversely impacted by such a change in CBFF, i.e., the calculated transient statepoints used to calculate the minimum DNBR do not change.

5.1.1.2 Pressure Drop across the Reactor Vessel

The current V+/P+ fuel product does not contain Intermediate Flow Mixing grids (IFMs), but the RFA-2 fuel product has IFMs primarily to gain DNBR margin in the analyses. One potential impact of the IFM grids is an increased pressure drop across the reactor vessel. For this program, the new reactor vessel pressure drop, at thermal design flow, caused by the RFA-2 fuel was compared to the current V+/P+ fuel values.

The comparison shows that the RFA-2 fuel with IFMs causes the overall reactor vessel pressure drop to increase by a very small amount. An evaluation of reactor vessel pressure drop changes on Non-LOCA analyses has been performed for similar plants. The study demonstrates that changes in reactor vessel pressure drop of similar magnitude do not impact the results of the Non-LOCA safety analyses. Therefore, the small change in reactor vessel pressure drop due to the RFA-2 fuel transition will not impact the Watts Bar safety analyses.

5.1.1.3 Rod Control Cluster Assembly (RCCA) Drop Time

An additional concern with the increased reactor vessel pressure drop and the associated increase in core bypass flow (Section 6.4.1), as well as the increased wall thickness of the guide thimble tubes (see Section 5.1.0.1), is that the time required for the RCCAs to fall into the core on a reactor trip may be impacted.

Therefore, the RCCA drop times associated with the RFA-2 with IFMs fuel were recalculated. The results demonstrate that the RCCA drop time will remain within the Technical Specification limit of 2.7 seconds for the RFA-2 fuel. The RCCA drop curve used in the current Watts Bar analysis of record were compared against the new RCCA drop curve data. Based on the comparison of the old and new RCCA drop curves, the data are in good agreement with one another. Hence, there is no impact on the Non-LOCA analyses.

5.1.1.4 Reactor Coolant System (RCS) Temperature Changes

RCS temperatures, such as core inlet and outlet temperatures, core average temperature, and vessel average temperature, were recalculated considering the upgrade to RFA-2 fuel. A comparison of these with the corresponding V+/P+ fuel-specific parameters shows only a small increase in the core outlet temperature and core average temperature. The nominal vessel average temperature remains unchanged. Further, the vessel inlet and outlet temperatures, along with the thermal design flowrate (TDF) remain unchanged between the current V+/P+ and proposed RFA-2 fuel types. Therefore, while core parameters calculated in the Non-LOCA analyses may change slightly, the overall calculated "system parameters," i.e., peak RCS pressure, peak secondary-side pressure, and peak pressurizer water volume, as well as transient statepoints for events in which DNBR is calculated, remain unchanged.

5.1.1.5 Reload Safety Analysis Checklist (RSAC) Parameter Confirmation

The Watts Bar RSAC limits for the RFA-2 fuel program were confirmed by a process similar to the process employed during each operating cycle. The RSAC limits contain analysis assumptions such as nominal conditions (such as reactor core power, TDF, MMF, CBFF), reactivity coefficients (such as Doppler power and temperature coefficients, moderator temperature coefficient, beta effective), peaking factors, fuel temperatures, RCCA reactivity worth, and event-specific parameters (such as critical boron concentrations for the Boron Dilution accident and maximum ejected RCCA worth for the RCCA Ejection accident). The RSAC also contains analysis statepoints for limiting DNB analyses, such as the Loss of Flow transients (UFSAR 15.2.5, 15.3.4, and 15.4.4) and Main Steam Line Break (UFSAR 15.4.2.1). Currently, these parameters are confirmed to continue to be bounding for each upcoming operating cycle.

The only change to the RSAC parameters required to support this program is the best estimate and design core bypass flow fraction values, which increased by 1% and 0.6%, respectively.

Incorporating the revised CBFF values, the RSAC confirmation process was used to confirm the continued applicability of the RSAC parameters to the RFA-2 fuel upgrade program. No violations of the RSAC limits were calculated, and hence, the current V+/P+ fuel-related parameters continue to apply to the RFA-2 fuel. Therefore, except for the increased CBFF values (evaluated in Section 5.1.1.1), there are no RSAC parameter changes due to the RFA-2 fuel upgrade, and hence, no impact on the Non-LOCA transients.

Further, the analysis statepoints in the RSAC were shown to have resulted in DNBR calculations above the design limit based on the WRB-2M DNBR correlation. Therefore, the DNB-limiting events can support the transition to RFA-2 fuel.

5.1.1.6 Fuel Temperatures

The fuel temperatures used in the Non-LOCA analyses have been recalculated to apply to the RFA-2 fuel. The results of these calculations demonstrate that the current licensing basis values calculated for the V+/P+ fuel are limiting with respect to the Non-LOCA analyses. Therefore, the existing fuel temperatures are bounding, and there is no impact on the Non-LOCA analyses.

5.1.1.7 Other Changes Introduced by the RFA-2 with IFMs Fuel

As noted in Section 5.1.0.1, item b, the guide thimble and instrumentation tube wall thickness increases with the RFA-2 fuel. Relative to Non-LOCA analyses, only the change in the guide thimble tubes could potentially impact the analyses by affecting the RCCA drop time curve. However, as evaluated in Section 5.1.1.3, the RCCA drop curve is not impacted. The instrumentation tubes are not modeled in the Non-LOCA analyses. Therefore, these changes do not impact the Non-LOCA analyses.

Also noted in Section 5.1.0.1 in item c is the fuel assembly flow area decrease, which is a decrease of less than 0.4%. This is used to calculate the total core flow area (a LOFTRAN code input) and the coolant core average mass flux (a FACTRAN code input).

With respect to the mass flux, a decrease in the flow area will increase the calculated mass flux, which will result in a slight increase in the rod heat transfer and slight decrease in the cladding and pellet temperatures. The change expected from less than a 0.4% decrease in the fuel assembly flow area is negligible and will not adversely impact the results of FACTRAN analyses.

The total core flow area is used by the LOFTRAN code in the vessel flow calculations to calculate a fluid velocity in reactor coolant pump coastdown, locked rotor, and natural circulation flow situations. The fuel assembly flow area change associated with the RFA-2 fuel is not large enough to adversely affect the results for these flow situations. Therefore, there is no significant impact on the LOFTRAN analyses.

Finally, as noted in Section 5.1.0.3, the overtemperature-ΔT (OTΔT) and overpower-ΔT (OPΔT) setpoints remain unchanged for the RFA-2 fuel upgrade program based on the most conservative core thermal limits. These core thermal limits were used to bound mixed cores and full cores with the RFA-2 fuel features analyzed. Since there are no changes to the OTΔT and OPΔT setpoints or the core thermal limits, the Non-LOCA analyses are not impacted relative to these parameters.

5.1.2 Event-specific Evaluations and Analyses – Condition II

The events in this section are evaluated or reanalyzed for the RFA-2 fuel upgrade program. All of the events listed in Table 5.1.0-1 are evaluation with the exception of the Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) accident, which is explicitly reanalyzed.

5.1.2.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition [RWFS]

An RCCA withdrawal from a subcritical condition accident is defined as an uncontrolled addition of reactivity to the reactor core during startup. This addition of reactivity to the core is caused by withdrawal of one or more RCCA banks, resulting in a rapid power excursion. Since this accident specifically involves a change in the reactivity of the core, a verification of the reactivity parameters for this event is performed through the Reload Safety Analysis Checklist (RSAC) process. As noted in Section 3.0 these RSAC parameters will continue to be bounding for the RFA-2 fuel. The evaluations in Section 5.1.1 demonstrate that the "system" portion of the transient, i.e., the transient statepoints, would be insignificantly affected by the fuel change. Since it has been shown that the DNBR limit to continues to be met, the results of the analysis are unaffected by the proposed change and the results and conclusions presented in the UFSAR remain valid.

5.1.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power [RWAP]

An uncontrolled RCCA withdrawal at power condition accident is defined as an uncontrolled addition of reactivity to the reactor core during at-power conditions. This event is analyzed to demonstrate that the DNBR design basis is met. The RWAP transient is analyzed at various power levels, reactivity feedback conditions, and reactivity insertion rates (caused by the RCCA bank withdrawal) to capture the limiting case relative to minimum DNBR. To protect against this condition, two reactor trip functions are credited in the analysis: overtemperature ΔT (OTΔT) for lower reactivity insertion rates and high neutron flux for larger insertion rates. The high neutron flux setpoint is not changed for the RFA-2 program. Per Section 5.1.0.3, the OTΔT and OPΔT trip setpoints also remain unchanged since the core thermal limits remain unchanged for this program. Therefore, the reactor trip times remain unchanged from those in the current analysis.

It has been demonstrated that the RSAC parameters used in the current RWAP analysis will continue to be bounding for the RFA-2 fuel. Therefore, the reactivity parameters are confirmed for continued applicability for this event.

Since the reactivity parameters, trip functions, and the "system" response are not adversely impacted, there is no adverse impact of the RFA-2 fuel on this transient and the results and conclusions presented in the UFSAR remain valid.

5.1.2.3 Rod Cluster Control Assembly Misalignment [Dropped Rod]

For the Dropped Rod event, the generic WOG Dropped Rod methodology is applied for Watts Bar using the WOG 4-Loop Dropped Rod statepoints (including appropriate power and temperature penalties prescribed for the rod control optimization effort performed in 1997). The "system" portion of the transient (i.e., statepoints) would be insignificantly affected by the changes introduced by the RFA-2 fuel, as documented in Section 5.1.1. Hence, the generic WOG statepoints with added penalties from the rod control optimization effort remain applicable for the RFA-2 fuel transition program.

DNBR is evaluated for this event on a cycle-specific basis. Hence, in confirming the RSAC limits for this program, it has been shown that the DNBR limit was met considering the conditions caused by the transition to RFA-2 fuel. Therefore, the results and conclusions of the Dropped Rod event presented in the UFSAR remain valid.

5.1.2.4 Uncontrolled Boron Dilution

The Boron Dilution event is analyzed to ensure that the operator has sufficient time to terminate the dilution before such time that the shutdown margin is lost and resultant core damage could occur due to a return to criticality. Only a dilution during power operation (Mode 1) and startup (Mode 2) must be considered. For Watts Bar, a dilution during refueling is precluded via administrative controls at the plant. Dilutions during Modes 3, 4, and 5 (Hot Standby, Hot Shutdown, and Cold Shutdown, respectively) are covered by the interim operating procedures.

The Boron Dilution-related RSAC parameters, such as critical boron concentration, are confirmed to be applicable to the RFA-2 program as described in Section 5.1.1.5. Also, the OTΔT trip time taken from the RWAP analysis remains unchanged since core thermal limits are unchanged for this program. This trip time is used in the overall operator action time calculation for this event. Parameters such as core flow, core ΔP, and the trip reactivity curve, are not modeled, and there is no concern relative to the overall system response for this event. Therefore, the results and conclusions of the Boron Dilution event presented in the UFSAR remain valid.

5.1.2.5 Loss of Reactor Coolant Flow Transients

This section addresses the Partial Loss of Forced Reactor Coolant Flow [PLOF], Complete Loss of Forced Reactor Coolant Flow [CLOF], and Single Reactor Coolant Pump Locked Rotor [LR].

The partial or complete loss of forced reactor coolant flow events may result from mechanical or electrical failure(s) in the Reactor Coolant Pump(s) (RCPs). These failures may occur from an undervoltage condition in the electrical supply to the RCPs or from a reduction in motor supply frequency to the RCPs due to a frequency disturbance on the power grid.

The Loss of Flow analysis includes the following events:

- Partial Loss of Flow
- Complete Loss of Flow - Underfrequency Trip
- Complete Loss of Flow - Undervoltage Trip
- Locked Rotor / Shaft Break

The first three events listed above must demonstrate that the minimum DNBR remains above the limit value. The DNB analysis for these events is performed by analyzing statepoints provided in the RSAC. Since the DNB analysis specifically involves the cycle-specific core design, a verification of the reactivity and DNB parameters for this event is performed through the RSAC process, as described in Section 5.1.1.5. In addition, the "system" portion of the transients (i.e., statepoints) is insignificantly affected by the changes introduced by the RFA-2 fuel. Since the DNBR limit is shown to continue to be met and the statepoints are insignificantly impacted, the results and conclusions presented in the UFSAR remain valid.

With respect to the Locked Rotor event, RCS overpressurization, number of fuel rods experiencing DNB, peak cladding temperature, and zirconium-water reaction are of primary concern. Like the loss of flow events, a rods-in-DNB analysis for this event is performed on a cycle-specific basis corresponding to statepoints provided in the RSAC. Again, since the DNB analysis specifically involves the cycle specific core design, a verification of the reactivity and DNB parameters for this event is performed through the RSAC process. The parameters provided in the RSAC bound the corresponding parameters for the reload for RFA-2 fuel. The "system" portion of the transients is insignificantly affected by the changes introduced by the RFA-2 fuel. Therefore, as with the PLOF/CLOF cases, the results of the analysis are not adversely affected by the proposed change and the results and conclusions presented in the UFSAR remain valid.

5.1.2.6 Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature [SUIL]

Watts Bar Technical Specification 3.4.4 requires that all four loops be operable or in operation when the plant is in Modes 1 and 2. Thus, no analysis is necessary to demonstrate that the minimum DNBR is satisfied because the startup of an idle pump in Mode 3 (or lower) is considered an acceptable normal operational occurrence, and will not result in a reactivity insertion or power increase because the loop is at a constant temperature. Therefore, since the technical specifications associated with the operation of the RCPs are unaffected by the proposed fuel change, the conclusions of the evaluation presented in the UFSAR remain valid.

5.1.2.7 Loss of External Electrical Load and / or Turbine Trip [LOL/TT]

The LOL/TT event is initiated by a sudden loss of secondary system steam flow that results in a rapid primary system heat-up. The sudden loss of steam flow and rapid primary system heat-up result in primary and secondary system overpressurization concerns. Thus, this transient is analyzed for both minimum DNBR and peak RCS and Main Steam System (MSS) pressures. The results are not driven by fuel- or core-related parameters (i.e., reactivity parameters), but rather the response of the RCS to the complete load rejection on the secondary side. Therefore, as documented in Sections 5.1.1, the "system" response is not impacted by the changes in core bypass flow fraction or core inlet/outlet temperatures, since overall RCS flow and vessel inlet/outlet temperatures are unchanged. Hence, the RCS and MSS pressure transients, as well as the statepoints for this event are not significantly impacted. Based on this discussion the results and conclusions presented in the UFSAR for the LOL/TT event remain valid.

5.1.2.8 Loss of Normal Feedwater [LONF]

See Section 5.1.2.9

5.1.2.9 Coincident Loss of Onsite and External (Offsite) AC Power to the Station – Loss of Offsite Power to the Station Auxiliaries [LOOP]

The Loss of Normal Feedwater (LONF) event is analyzed, both with- and without offsite power available, to demonstrate that the auxiliary feedwater system has a sufficient capacity to remove core decay heat, stored energy, and RCS pump heat following a reactor trip. This is demonstrated by showing that the pressurizer does not become water solid throughout the duration of the transient. As such, the analysis conservatively assumes the operation of the pressurizer pressure control systems since modeling the pressurizer heaters and sprays conservatively maximize the calculated peak pressurizer inventory.

The LONF/LOOP events are "system" transients; they are not dependent on fuel or core-related parameters, core bypass flow fraction, or core thermal limits. Therefore, as documented in Section 5.1.1, the transient pressurizer water inventory response is insignificantly affected by the proposed fuel change, and the results and conclusions in the UFSAR remain valid.

5.1.2.10 Excessive Heat Removal Due to Feedwater System Malfunctions [FWM]

The feedwater system malfunction accident is defined as an increase in the feedwater to the steam generators or a decrease in feedwater temperature, resulting in excessive heat removal from the plant primary coolant system. The resultant decrease in the average temperature of the core causes an increase in core power due to moderator and control systems feedback. As documented in Section 5.1.1, the "system" portion of the transient would be insignificantly affected by the input changes introduced by the RFA-2 fuel type. Further, the reactivity-related RSAC parameters assumed in the analysis have been confirmed to be applicable to the RFA-2 program. Therefore, the results of the analysis are insignificantly affected by the proposed change and the results and conclusions presented in the UFSAR remain valid.

5.1.2.11 Excessive Load Increase Incident [ELI]

The excessive load increase event is defined as a rapid increase in steam flow, causing a power mismatch between the reactor core power and the steam generator load demand. However, the increase in steam flow also causes core cooling, resulting in an increase in core power due to moderator and control systems feedback. Since feedback causes the power to increase, the power mismatch is reduced and the reactor is able to stabilize without the occurrence of a trip. Thus, DNB is not a concern and no analysis is required to demonstrate that the minimum DNBR is satisfied. The core parameters assumed in the evaluation of this event remain applicable to the corresponding parameters considering the RFA-2 fuel. The system-driven response to the ELI would be insignificantly affected by the changes introduced by the RFA-2 fuel, such as CBFF and reactor vessel ΔP. Therefore, the results of the evaluation are unaffected by the proposed fuel change and the results and conclusions presented in the UFSAR remain valid.

5.1.2.12 Accidental Depressurization of the Reactor Coolant System [RCS Depressurization]

The RCS depressurization event is defined as a rapid depressurization of the reactor coolant system caused by the failure of a power operated relief valve or spray valve. This transient is analyzed to demonstrate that the DNBR design basis is met by conservatively assuming the failure of a pressurizer safety valve. It should be noted that this analysis does not challenge the DNBR limit and is not considered limiting relative to DNBR concerns.

The RCS Depressurization transient is not specifically driven by or dependent on fuel or core-related (i.e., reactivity) parameters, core bypass flow, or core thermal limits, but rather is a system-driven transient. Therefore, as documented previously, the results of the analysis are insignificantly impacted by the proposed fuel change and the results and conclusions presented in the UFSAR remain valid.

5.1.2.13 Accidental Depressurization of the Main Steam System

The Accidental Depressurization of the Main Steam System event is defined as the failure of an atmospheric relief valve, steam dump valve, or steam generator safety valve. The main steam system depressurization event is bounded by the analysis of the Major Rupture of a Main Steam Line event [HZP SLB] and, therefore, no specific main steam system depressurization analysis is performed for Watts Bar Unit 1. An evaluation of the main steam line break event is provided in Section 5.1.4.1.

5.1.2.14 Inadvertent Operation of Emergency Core Cooling System [Inadvertent ECCS]

The Inadvertent ECCS event is analyzed to demonstrate that 1) the calculated DNBR remains above the limit value for the DNB case, and 2) at no time throughout the period of the transient does the pressurizer become water-solid and discharge subcooled water through the pressurizer safety valves for the pressurizer overfill case.

Based on historical precedence, this event does not lead to a serious challenge of the DNB design basis. The decrease in core power and RCS average temperature more than offset the decrease in RCS pressure such that the minimum calculated DNBR occurs at the start of the transient. As such, the DNB case is non-limiting, and the RFA-2 fuel transition does not change this conclusion.

The pressurizer overfill case is driven by the addition of cold ECCS flow being added to the RCS in a tripped condition. Since this is a "system" transient, it is not dependent on fuel or core-related parameters, core bypass flow, or core thermal limits. Therefore, per Section 5.1.1, the results of the analysis are not significantly impacted by the proposed fuel change and the results and conclusions presented in the UFSAR for the Inadvertent ECCS event remain valid.

5.1.3 Event-specific Evaluations and Analyses – Condition III

5.1.3.1 Minor Secondary System Pipe Breaks

Per the Watts Bar UFSAR, included in this grouping are ruptures of secondary system lines that would result in steam release rates equivalent to a 6-inch diameter break or smaller. Events in UFSAR Section 15.4.2, "Major Secondary System Pipe Ruptures," (i.e., Major Rupture of a Main Steam Line and Major Rupture of a Main Feedwater Pipe) bound the transients that could occur in this category. Therefore, no specific analyses are required. See the Major Rupture of a Main Steam Line and Major Rupture of a Main Feedwater Pipe sections (Section 5.1.4.1 and Section 5.1.4.3, respectively) for the specific evaluations that bound cases in this category.

5.1.3.2 Single RCCA Withdrawal at Full Power [Single RWAP]

The single RCCA withdrawal at power event is defined as a misoperation of the rod control system, due to either operator action or multiple failures, resulting in the accidental withdrawal of a single RCCA at full power operation. Withdrawal of a single RCCA will result in the addition of reactivity to the reactor core, causing an increase in core power and an increase in local power density in the area of the core associated with the RCCA. Analysis of this event is based on a generically bounding set of core thermal limits, without direct input from a Non-LOCA analysis. Since this accident specifically involves a change in the reactivity of the core, a verification of the reactivity parameters for this event is performed through the RSAC process. The current RSAC parameters have been shown to bound the corresponding parameters for the reload using RFA-2 with IFMs fuel. Therefore, the results of the analysis are unaffected by the proposed change and the results and conclusions presented in the UFSAR remain valid.

5.1.4 Event-specific Evaluations and Analyses – Condition IV

5.1.4.1 Major Rupture of a Main Steam Line [HZP SLB]

The major rupture of a main steam line event is characterized by a the rupture in the steam line causing a secondary-side depressurization resulting in a cooldown in the RCS. The cooldown results in a return to power, which could challenge the DNB limits. The main steam line break analysis is analyzed at Hot Zero Power (HZP) conditions to demonstrate that the DNB licensing basis is met. The DNB analysis for this event is performed on a cycle-specific basis corresponding to statepoints provided in the RSAC. The system portion of the transient (i.e., the statepoints) would be insignificantly impacted by the changes introduced by the RFA-2 fuel (i.e., CBFF, reactor vessel ΔP), per the discussions in Section 5.1.1. Since the DNB analysis specifically involves the cycle specific core design, a verification of the reactivity parameters for this event is performed through the RSAC process. The parameters provided in the RSAC have been shown to bound the corresponding parameters for the reload with RFA-2 fuel. Therefore, the results of the analysis are unaffected by the proposed change and the results and conclusions presented in the UFSAR remain valid.

5.1.4.2 Major Rupture of a Main Feedwater Pipe [FLB]

The main feedwater pipe rupture event is analyzed to demonstrate that the core maintains a coolable geometry throughout the transient. This is ensured by demonstrating that no hot leg bulk boiling occurs prior to the time at which the auxiliary feedwater system heat removal capability exceeds the stored energy, decay heat, and RCS pump heat (for the case with offsite power available) in the RCS. Since this accident does not specifically involve a change in the reactivity of the core, the event is not evaluated in the RSAC. However, the parameters that are common to a number of the analyses bound the corresponding parameters of the reload with the RFA-2 fuel. Also, from Section 5.1.1, the "system" portion of the FLB analysis is not significantly impacted by the input changes caused by the RFA-2 fuel product. Therefore, the results of the analysis are unaffected by the proposed change and the results and conclusions presented in the UFSAR remain valid.

5.1.4.3 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly (RCCA) and drive shaft. The consequences of this mechanical failure are a rapid positive reactivity insertion and an adverse core power distribution, possibly leading to localized fuel rod damage.

If an RCCA Ejection accident were to occur, a fuel rod thermal transient that could cause DNB may occur together with limited fuel damage. The amount of fuel damage that can result from such an accident will be governed mainly by the worth of the ejected RCCA and the power distribution attained with the remaining control rod pattern. The transient is limited by the Doppler reactivity effects caused by the fuel temperature increase. The event is terminated by reactor trip actuated by neutron flux signals before conditions are reached that can result in damage to the reactor coolant pressure boundary or significant disturbances in the core, its support structures, or other reactor pressure vessel internals, which would impair core cooling capability.

This event is classified as a Condition IV event as defined by the American Nuclear Society Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants (ANSI N18.2-1973). Due to the extremely low probability of an RCCA ejection accident, some fuel damage is considered an acceptable consequence.

The ultimate acceptance criteria for this event is that any consequential damage to either the core or the RCS must not prevent long-term core cooling, and that any offsite dose consequences must be within the guidelines of 10 CFR 100. To demonstrate compliance with these requirements, it is sufficient to show that the RCS pressure boundary remains intact, and that no fuel dispersal in the coolant, gross lattice distortions, or severe shock waves will occur in the core. Therefore, the following acceptance criteria are applied to the RCCA Ejection accident:

1. Average fuel pellet enthalpy at hot spot below 225 cal/g for non-irradiated fuel and 200 cal/g for irradiated fuel.

2. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits. This criterion is generically addressed in the Westinghouse Rod Ejection Topical Report, WCAP-7588 (Reference 25).

3. Fuel melting will be limited to less than the innermost 10% of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy is below the limits of Criterion 1, above.

It should be noted that, for informational purposes, the peak clad temperature at the hot spot during the RCCA Ejection transient is shown to remain less than 3000 °F and the zirconium-water reaction at the hot spot is shown to remain less than 16%. However, these items are excluded as explicit acceptance criteria for this event, consistent with the revised Westinghouse acceptance criteria for the RCCA Ejection accident (Reference 26).

The calculation of the RCCA ejection transient is performed in two stages: a neutron kinetics analysis and a hot spot fuel heat transfer analysis. A description of the Westinghouse methodology is provided in Reference 25. The overpressurization of the RCS and number of rods-in-DNB as a result of a postulated ejected rod have been analyzed on a generic basis for Westinghouse PWRs as detailed in Reference 25.

If the safety limits for fuel damage are not exceeded, there is little likelihood of fuel dispersal into the coolant or a sudden pressure increase from thermal-to-kinetic energy conversion. The pressure surge for this analysis can, therefore, be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

A generic calculation of the pressure surge for an ejection worth of one dollar at Beginning-of-Life (BOL), hot full power, indicates that the peak pressure does not exceed that which would cause stresses in the RCS to exceed their Faulted Condition stress limits. Since the severity of the Watts Bar analysis does not exceed this worst case analysis, the RCCA Ejection accident will not result in an excessive pressure rise or further damage to the RCS.

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. Selected important parameters are discussed below. In calculating the nuclear power and hot spot fuel rod transients following RCCA Ejection, the following conservative assumptions are made:

a. The RTDP is not used for the RCCA Ejection analysis. Instead, the STDP (maximum uncertainties in initial conditions) is used.

b. A minimum value for the delayed neutron fraction for Beginning-of-Cycle (BOC) and End-of-Cycle (EOC) conditions is assumed, which increases the rate at which the nuclear power increases following RCCA Ejection.

c. A minimum value of the Doppler power defect is assumed which conservatively results in the maximum amount of energy deposited in the fuel following RCCA Ejection. A positive moderator temperature coefficient is assumed for the beginning of cycle, full- and zero power cases.

d. Maximum values of ejected RCCA worth and post-ejection total hot channel factors are assumed for all cases considered. No credit is taken for the flux flattening effects of reactivity feedback.

e. The start of rod motion occurs 0.50 seconds after the high neutron flux trip point is reached.

The following four cases were analyzed for both beginning- and end-of-life at zero- and full-power.

- Beginning-of-Cycle, Full Power
- Beginning-of-Cycle, Zero Power
- End-of-Cycle, Full Power
- End-of-Cycle, Zero Power

In all cases, the maximum fuel pellet average enthalpy is well below that which could cause sudden cladding failure, the maximum clad average temperature is below the point of clad embrittlement, and fuel melting, if any, is limited to less than 10% of the fuel cross-section at the hot-spot.

For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. The reactor will remain subcritical following reactor trip.

Even on a worst-case basis, the RCCA Ejection analysis results show that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the reactor coolant system. The analyses have demonstrated the fission product release as a result of fuel rods entering DNB is limited to less than 10% of the fuel rods in the core. All event acceptance criteria are satisfied.

The ejection of an RCCA constitutes a break in the RCS located in the reactor pressure vessel head. Following the RCCA ejection and reactor trip, the operator action requirements and protection system operation are similar to those presented in the analysis of a small loss of coolant event. Additionally, the environmental consequences of the Rod Ejection event are bounded by the loss of coolant accident.

5.1.5 MSLB Evaluation for Containment Integrity

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. There are many factors that influence the quantity and rate of the mass and energy release from the steamline, such as the configuration of the plant steam system, feedwater system, protection logic and setpoints. In addition, the power level at which the plant is operating when the steamline break is postulated can cause different competing effects that make it difficult to predetermine a single limiting case.

Main Steamline Break (MSLB) mass and energy release analyses inside/outside containment predict a return to power based on the known reactivity of the core design. These accidents are not affected by the RFA-2 fuel implementation if the shutdown margin or the end of cycle moderator density coefficient do not change as a result of the RFA-2 fuel.

5.1.6 Radiological Effects

The Main Steam Line Break and Steam Generator Tube Rupture thermal hydraulic analyses used in the evaluation of the radiological consequences are not affected by the RFA-2 fuel upgrade program. Therefore, the radiological consequences of these accidents are not affected by the change in fuel. The design features in the RFA-2 fuel rod are identical to the V+/P+ fuel rod. There is no difference in the UO_2 weight, enrichment, or burnup limits between RFA-2 and V+/P+ fuel and therefore there are no impacts to the radiological consequences of the Fuel Handling Accident and Waste Gas Decay Tank Rupture events as a result of the RFA-2 fuel upgrade program.

5.2 LOCA Accidents

This section summarizes the LOCA related evaluations performed for the Watts Bar upgrade to the 17x17 RFA-2 fuel assembly with IFMs.

5.2.1 Large Break LOCA

The Large Break LOCA (LBLOCA) Analysis of Record (AOR) for Watts Bar Unit 1 (Reference 27) is based on the Best Estimate Large Break Evaluation Model (EM) (Reference 28). The analysis therein is applicable for VANTAGE+ (w/o IFMs) fuel with ZIRLO™ cladding. The LBLOCA is limited by a break discharge coefficient (Cd) = 1.0, Minimum Safeguards Injections (SI) and Beginning-of-Life (BOL) fuel conditions. The resultant AOR 95th Percentile Peak Clad Temperature (PCT) was 1892°F for non-IFBA ZIRLO™ fuel, which bounds both IFBA ZIRLO™ fuel and Zirc-4 clad fuel.

The RFA-2 fuel scheduled to be included in Watts Bar Cycle 6 is similar to the V+/P+ fuel that is currently being used, the only significant differences being the addition of Intermediate Flow Mixer grids and thimble tubes with increased wall thickness. The effect of Intermediate Flow Mixer grids is considerable to Best Estimate LBLOCA Analysis due to the importance of grids in the fuel cladding heatup calculation. In the grid model used there is no heat generated in the grid, which makes grid rewet a possibility. The effect of grid rewet is increased heat transfer at the grid elevation and therefore, a decrease in the temperature of the vapor traveling through the rest of the core. The Intermediate Flow Mixer grids also cause an increase in flow resistance through the core, which tends to mitigate the effect of grid rewet. The effect of the thicker thimble tubes is to reduce the flow area of the core, which also tends to mitigate the effects of grid rewet.

The Best Estimate LBLOCA evaluation for the new RFA-2 fuel was performed as a change to the AOR using the same version of WCOBRA/TRAC as used in the AOR with the current version of PAD (Reference 10) for stored energy and rod internal pressures. The limiting time in life (BOL), break size (C_d = 1.0) and SI parameters (minimum) from the AOR were used as the base case conditions for this evaluation. Other important analysis assumptions, consistent with the AOR, are as follows: nominal core power as in the AOR (3411 MWt which supports a core power of 3459 MWt used on uncertainty analysis), 10% uniform SGTP, maximum peaking factor $F_Q(Z)$ envelope of 2.50, a hot channel enthalpy rise factor $F_{\Delta H}^N$ of 1.65, T_{avg} of 588.2 °F, and Thermal Design Flow of 372,400 gpm.

Three cases were considered and analyzed using the AOR reference calculation as the base case input to the analysis. 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, the remainder of the core being V+/P+ fuel with 8,000 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 guide tubes, with the remainder of the core being V+/P+ fuel with 8,000 MWD/MTU minimum burnup. A fourth transition case with V+/P+ fuel with 8,000 MWD/MTU minimum burnup in the hot assembly position and RFA-2 in the remainder of the core is not considered limiting. The lower flow resistance in an assembly without IFMs coupled with the higher resistance of the RFA-2 fuel will cause the V+/P+ fuel to receive more flow during reflood. This coupled with the lower stored energy of burned fuel assures that Case 4 is non-limiting.

The new fuel PCT calculations used current methodology and the same version of the WCOBRA/TRAC computer code such that all of the previously reported Best Estimate LBLOCA PCT model assessments were not considered. Consequently the previously reported Best Estimate LBLOCA PCT model assessments (-115°F net, bounding PCT is 1777 °F) continue to be part of the 10 CFR 50.46 PCT licensing basis. The assessments related to plant safety evaluations will remain. This Best Estimate LBLOCA evaluation concludes that, with the new fuel, and limiting transition core Watts Bar remains in compliance with the requirements of 10CFR50.46 based on the AOR for both the transition from the current fuel to the new fuel and for a full core of the new fuel.

5.2.2 Small Break LOCA

The non-TPBAR small break LOCA analysis-of-record for Watts Bar Unit 1 was completed using the 1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP (NOTRUMP-EM). Small break LOCA transients are characterized by a gradual top-down draining of the reactor coolant system, with low flow rates in the core relative to those occurring at steady-state or for large break LOCA transients. Hydraulic losses in the core due to frictional drag, form loss, and acceleration are small, and reasonable variations in flow resistance would be expected to have a negligible effect on the analysis results. Given this, and since the other changes resulting from the introduction of RFA-2 are expected to have a negligible effect on the analysis results (e.g., decreased assembly flow area and hydraulic diameter), no changes to the small break LOCA analysis-of-record are required to address the introduction of RFA-2, and no mixed-core PCT penalty is necessary during the transition cycles.

5.2.3 Blowdown Reactor Vessel and Loop Forces

The forces created by a hypothetical break in the RCS piping are principally caused by the motion of the decompression wave through the RCS. The strength of the decompression wave is primarily a function of the assumed break opening time, break area and RCS operating conditions of power, temperature and pressure. Thermal/hydraulic data which modeled the 17x17 RFA-2 fuel was used in generating vessel LOCA forcing functions for Watts Bar Unit 1. Operating conditions were based on a Core Power of 3459 MWt, a Vessel Average Temperature of 588.2°F, 10% Steam Generator Tube Plugging and an RCS pressure of 2250 psia. Leak-Before-Break methodology was used to allow consideration of branch line breaks. Therefore, the forcing functions generated were based on breaks of the accumulator line (0.4176 ft^2 area) and the Residual Heat Removal (RHR) line (0.7213 ft^2 area), which have smaller areas than the postulated breaks in the main RCS loop piping. The analyses assume a conservative 1 millisecond break opening time. The LOCA hydraulic forces generated for use in the reactor vessel and internals analysis modeled 17x17 RFA-2 fuel in the Watts Bar Unit 1 vessel. These LOCA forces remain conservative for assemblies containing Tritium Producing Burnable Absorbers (TPBARs).

Forces acting on the RCS loop piping as a result of the hypothesized LOCA are not significantly influenced by changes in fuel assembly design. Thus, the use of 17x17 RFA-2 at Watts Bar Unit 1 will not result in an increase of the calculated consequences of a hypothesized LOCA on the RCS piping. The current analysis for forces on RCS piping resulting from a hypothesized LOCA are considered to be bounding to the application of RFA-2 fuel at Watts Bar Unit 1. These forces are calculated through use of the MULTIFLEX 3.0 (Reference 41) computer code, which has been accepted by the US-NRC for several different applications (References 42 - 44).

5.2.4 Post-LOCA Long-Term Cooling, Subcriticality Evaluation

The Westinghouse licensing position for satisfying the requirements of 10 CFR Part 50.46 (b)(5) "Long-Term Cooling" is defined in WCAP-8339-NP-A (Reference 29), WCAP-8472-A (Reference 30), and Technical Bulletin NSID-TB-86-08 (Reference 31). The Westinghouse commitment is that the reactor will remain shutdown by borated ECCS water residing in the sump following a LOCA. Since credit for the control rods is not taken for a LBLOCA, the borated ECCS water provided by the accumulators and the RWST must have a concentration that, when mixed with other sources of borated and non-borated water, will result in the reactor core remaining subcritical assuming all control rods out.

The primary factors affecting the calculations for the Post-LOCA long term core cooling subcriticality requirement in the Westinghouse methodologies are the RCS, RWST and accumulator water volumes and boron concentrations. The implementation of IFMs will have no affect on the volumes and boron concentrations of the RWST and accumulators used in the calculation. Therefore, implementation of IFMs will not adversely affect the assumptions made when verifying Post-LOCA long term core subcriticality for the Westinghouse ECCS Evaluation Model. The conclusions made regarding Post-LOCA subcriticality and long term cooling would remain unchanged due to implementation RFA-2 for the Westinghouse ECCS Evaluation Model.

5.2.5 Hot Leg Switchover to Prevent Potential Boron Precipitation

The Post-LOCA Hot Leg Switchover time is determined for inclusion in the emergency procedures to preclude boron precipitation in the reactor vessel due to boiling in the core and to ensure that the core remains amenable to cooling in the long term. The Hot Leg Switchover time is dependent on power level and the RCS, Refueling Water Storage Tank (RWST) and accumulator water volumes and boron concentrations. The implementation of IFMs will have no affect on the power level and volumes and boron concentrations of the RWST and accumulators. The RCS volume will decrease by an insignificant amount due to the additional grids and increased outside diameter of the thimble tubes. Therefore, there would be no adverse effect on the Post-LOCA Hot Leg Switchover time for implementation of RFA-2 for the Westinghouse ECCS Evaluation Model.

5.2.6 LOCA Evaluation for Containment Integrity

5.2.6.1 Introduction

Several analyses are performed in order to demonstrate the adequacy of the containment building and the internal walls, and to qualify the equipment inside containment for a LOCA. A long-term LOCA analysis is performed in order to determine the containment pressure and temperature response. A maximum reverse pressure differential calculation is conducted in order to predict the downward load acting on the operating deck, and a short-term LOCA analysis is performed in order to determine compartment pressurization of subcompartments located inside containment.

5.2.6.2 Long-Term LOCA/Containment Integrity Analysis

This analysis demonstrates the ability of the containment safeguards systems to mitigate the consequences of a hypothetical large break LOCA. The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure. Analysis results are also used to support environment qualification.

The methodology for the licensing basis analysis is contained in WCAP-10326-A (Reference 32). Based on this methodology, the analysis of record (Reference 33) presently assumes a thermal power of 3459 MWt based on the recent 1.4% uprate program. In addition, the analysis applies an extra 0.6% calorimetric uncertainly to the 3459 MWt value to account for power measurement uncertainty (Reference 33).

The principal parameters of RCS temperature, SG secondary pressure, core pressure drop and core stored energy were compared to the Reference 33 analysis of record assumptions.

The analysis of record bounds the Watts Bar RFA-2 implementation program parameters. The higher core pressure drop for the RFA w/IFM fuel would be an insignificant benefit, since a higher resistance slows the rate of mass and energy release to containment. Thus, implementation of RFA-2 fuel at Watts Bar falls within the bounds of the current analysis of record.

5.2.6.3 Short-Term LOCA Mass and Energy Release Analysis

Several evaluations are performed to support the loop subcompartment, reactor cavity and pressurizer enclosure analysis. The analysis input that has the potential to change with this program are the initial RCS fluid temperatures. Since this event lasts for approximately 3 seconds, the single effects of 1) core power, 2) core pressure drop and 3) core store energy are not significant.

The short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition. The Zaloudek correlation from Reference 34, which models this condition, is currently used in the short-term LOCA mass and energy release analyses. This correlation was used to evaluate the impact of the deviations in the RCS inlet and outlet temperatures for the 1.4% power uprate program. Since the Reference 33 parameters bound the RFA-2 fuel parameters, the conclusions from Reference 33 can be directly applied to the RFA-2 fuel implementation program.

5.2.6.4 Maximum Reverse Pressure Differential Analysis

The results and conclusions from the Reference 33 evaluation bound the RFA-2 implementation parameters and can be applied to the RFA-2 implementation program.

5.2.6.5 Conclusions

For the changes that affect the LOCA analysis for containment integrity, subcompartment analyses and equipment qualification, it has been determined that:

1. The existing long-term LOCA mass and energy releases from Reference 33 bound the RFA-2 parameters. Thus, Reference 33 is still valid for the RFA-2 w/IFM implementation program.
2. The existing mass and energy releases for the loop subcompartment, reactor cavity and pressurizer enclosure are still more limiting and the existing licensing basis analyses are still valid for the RFA-2 implementation program.
3. The existing analysis for the maximum reverse differential pressure across the operating deck during a LOCA is still more limiting and the existing licensing basis analysis is still valid for application to RFA-2 implementation program.

6.0 Reactor Vessel and Internals Evaluation

6.1 Purpose of Evaluation

Reloading a reactor with fuel other than that for which a plant was originally designed, requires that the reactor vessel system/fuel interface be thoroughly addressed in order to assure compatibility of the replacement fuel/core and to assure that the structural integrity of the reactor vessel system is not adversely affected. In addition, thermal-hydraulic analyses are required to determine plant specific core bypass flows, pressure drops, upper head temperatures, and hydraulic lift forces.

6.2 Assumptions

The reactor pressure vessel (RPV) system consists of the reactor vessel, reactor internals, fuel and control rod drive mechanism. Since the detailed design responsibility for these components is contained within various, distinct organizations, a separate function exists which is used to coordinate the analysis of the combined system for normal and abnormal conditions and to perform analyses for the RPV system. This ensures that consistent loads and other design inputs are applied in the detailed evaluations of the various components. For the Watts Bar Unit 1 proposed fuel upgrade program, the Westinghouse scope is to determine the various pressure drop data to be used in the safety analysis and to generate the time history core plate motions for the seismic and LOCA evaluations of the fuel. The following analyses will be performed as part of this fuel upgrade program;

a) Reactor internals system thermal/hydraulic performance
b) Reactor internals system structural response due to Seismic and LOCA conditions, and
c) Rod control cluster assembly (RCCA) scram performance

The results of these evaluations, documented herein, provides input to various safety analyses and the structural integrity evaluations of the proposed fuel upgrade to 17x17 RFA-2 fuel at Watts Bar Unit 1.

6.3 Discussion of Evaluations

6.3.1 Determination of Hydraulic Characteristics

A key area in evaluation of core performance is the determination of hydraulic behavior of coolant flow within the reactor internals system, i.e., core bypass flow, pressure drops, upper head temperatures, and hydraulic lift forces. This data is provided for input to the ECCS and non-LOCA accident analyses and to overall NSSS performance calculations. The hydraulic forces are critical in the assessment of the structural integrity of the reactor internals.

a) Thermal-Hydraulic Evaluation

Westinghouse methodology determines the distribution of pressure, temperature, and flow within the reactor vessel, internals, and the reactor core, and is based on hydraulic resistances for the various flow paths within the reactor pressure vessel system. The reactor internals system is divided into six flow regions: inlet nozzle, downcomer with neutron panels, lower plenum, core, outlet plenum, outlet nozzle, and the closure head plenum. The results of this analysis are used in subsequent licensing and safety analyses and for verification that design and safety criteria are met.

b) Bypass Flow Evaluation

Bypass flow is the total amount of reactor coolant flow bypassing the core region, and is not considered effective in the core heat transfer process. Since variations in the size of some of the bypass flow paths, such as gaps at the outlet nozzles and the core cavity, occur during manufacturing, plant specific as-built dimensions are used in order to demonstrate that the bypass flow limits are not violated. Therefore, analyses are performed to estimate core bypass flow and to either ensure that the design bypass flow limit for the plant will not be exceeded or to determine a revised design core bypass flow. The conclusions of these analyses are shown in Section 6.4.1.

The core bypass flow paths are:
- Baffle-Barrel Region
- Head Cooling Spray Nozzles
- Outlet Nozzle Gaps
- Baffle Plate-Core Cavity Gap
- Fuel Assembly Thimble Tubes

Since flows within these paths are affected by core hydraulic characteristics, it is essential that the proper fuel characteristics for a particular fuel type are used in their determination.

The methods used to perform the various thermal and hydraulic calculations are the same methods which were used in the initial design and qualification of the plants, so licensing risks associated with new methods are not an issue. In addition, the analyses require the use of empirical data (such as some of the hydraulic resistances of the reactor vessel and internals) and this data has been developed over time as the Westinghouse reactors have evolved.

For the core barrel-to-reactor vessel outlet nozzle gap, and across the core cavity, actual as-built measurements are used as input to the bypass flow calculations.

6.3.2 Reactor Vessel and Internals Structural Integrity

Structural evaluations are required to demonstrate that the structural integrity of the reactor components are not adversely affected directly by the change in fuel/core design, or indirectly by secondary effects of the change on reactor thermal hydraulic or structural performance. Changing the fuel/core design can affect the deadweight, and hydraulic loads during all modes of plant operation. The effects of the change in fuel assembly structural characteristics on the system dynamic response of the reactor pressure vessel and its internals are also addressed. Typical system dynamic analyses address seismic events, and Loss-of-Coolant Accident (LOCA) events due to postulated pipe ruptures.

6.3.3 Control Rod Scram Performance

The Rod Control Cluster Assemblies (RCCAs) represent perhaps the most critical interface between the replacement fuel and the other internals components. It is imperative to ensure that the proposed fuel/core changes for the Watts Bar Unit 1 will not adversely impact the operation of the control rods, either during accident conditions or normal operation. An analysis is performed to provide a basis for implementation of a new fuel/core design and their potential effects on control rod scram performance.

6.4 Thermal-Hydraulic System Evaluation

The thermal-hydraulic evaluations included investigating the effects of fuel upgrade to 17x17 RFA-2 fuel at Watts Bar Unit 1. The hydraulic characteristics within the system requires qualification of important reactor vessel, internals, and fuel system performance parameters. Specifically the internal fluid system pressure losses, core bypass flow, closure head fluid volume average temperature, lift forces on internal components, and RCCA scram performance were examined to provide confidence that system design requirements are not violated.

The following RCS conditions were used in the thermal hydraulic evaluation;

Fuel type	RFA-2 fuel with IFMs
Core inlet temperature	557.3°F
Thermal design flow	93,100 gpm/loop
Core power	3459 MWt
Core bypass flow	9.6%.

A thermal hydraulic evaluation was performed to determine the distribution of the pressure and flow within the reactor vessel internals and the reactor core. The comparison of the values for RFA-2 fuel to the current V+/P+ fuel shows that the overall reactor vessel pressure drop increases by a small amount.

6.4.1 Core Bypass Flow Evaluation

Core bypass flow is the total amount of reactor coolant flow that bypasses the core region and is not considered effective in the core heat transfer process. The design core bypass flow for the Watts Bar Unit 1 with RFA-2 is 9.6% of the total vessel flow. The purpose of this evaluation is to ensure that the design value of 9.6% is acceptable when the fuel upgrade to RFA-2 takes place at Watts Bar Unit 1.

The following flow paths have been identified as bypass:

- Baffle-Barrel Region
- Head Cooling Spray Nozzles
- Outlet Nozzles Gaps
- Baffle Plate-Core Cavity Gap
- Fuel Assembly Thimble Tubes

Fuel assembly hydraulic characteristics, system parameters, such as inlet temperature, reactor coolant pressure and flow were used to determine the impact of the new fuel upgrade on the total core bypass flow. The results of the core bypass flow evaluation for Watts Bar Unit 1 with RFA-2 indicate that the maximum bypass flow that can occur in the plant is bounded by the design core bypass flow limit of 9.6% of the total vessel flow.

6.4.2 System Fluid Temperatures Evaluation

The average temperature of the primary coolant fluid that occupies the reactor vessel closure head volume is an important initial condition for certain dynamic loss-of-coolant accident (LOCA) analyses. Therefore, it was necessary to confirm the use of RFA-2 will not alter this average closure head temperature for Watts Bar Unit 1. On a best estimate basis, the upper head region bulk fluid temperature can be considered to be at T_{cold}. Thus, there is no change from the existing analysis since Watts Bar Unit 1 is designed to be a T_{cold} head plant.

6.4.3 Hydraulic Lift Forces

An evaluation of the hydraulic lift forces on various reactor internal components is necessary to ensure that the reactor internals assembly remains seated and stable. An evaluation was performed to show that the lower internals holddown capability was maintained for the slight increase to the hydraulic lift loads on the lower internals package.

6.5 Mechanical System Evaluation

Changes in fuel assembly properties generally impact the performance of the reactor pressure vessel and its internals under all modes of operation. It is, therefore, important that with a change of fuel, the mechanical response of the reactor pressure vessel and it's internals be evaluated. The mechanical system evaluations, in general, consist of: (a) response due to seismic excitations; (b) response due to Loss-of-Coolant Accident (LOCA).

6.5.1 Seismic Evaluation

The non-linear dynamic seismic analysis of the reactor pressure vessel system includes the development of the system finite element model of the Reactor Pressure Vessel and Internals and the synthesized time history accelerations.

For a time history response of the reactor pressure vessel and its internals under seismic excitations, synthesized time history accelerations are required. The synthesized time history accelerations used in Watts Bar Unit 1 RPV system analysis were based on the plant specific design spectra. Spectrum amplification and suppression techniques are used to modify the initial transients supplied as input to the code. The records of a real earthquake are the basis for the synthesized time history accelerations. The spectral characteristics of the synthesized time histories are similar to the original earthquake records. The spectrum ordinates are computed using suggested frequency intervals given in Regulatory Guide 1.122 Revision 1. Note that the input excitations which were developed are for seismic events ten (10) seconds in duration.

The results of system seismic analysis include time history displacements and impact forces for all major components. The time history displacements of upper core plate, lower core plate and core barrel at the upper core plate elevation were used as input for the reactor core evaluations. The time history motions for the lower core plate, upper core plate, and the core barrel at upper core plate elevation are addressed in Section 2.5 for fuel/grid impact.

6.5.2 LOCA Evaluations

The breaks considered for the dynamic analysis of Watts Bar Unit 1 reactor pressure vessel system was the Accumulator Line and Residual Heat Removal Line nozzle breaks.

Following a postulated LOCA pipe rupture, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant. The release of pressurized coolant results in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wavefront with low pressure on one side and high pressure on the other. The wavefront translates and reflects throughout the primary system until the system is completely depressurized. The rapid depressurization results in transient hydraulic loads on the mechanical equipment of the system.

The loads on the RPV and internals that result from the depressurization of the system may be categorized as (1) reactor internal hydraulic loads (vertical and horizontal), and (2) reactor coolant loop mechanical loads. All the loads are calculated individually and combined in a time-history manner.

The severity of a postulated break in a reactor vessel is related to two factors: the distance from the reactor vessel to the break location, and the break opening area. The nature of the reactor vessel decompression following a LOCA, as controlled by the internals structural configuration, results in larger reactor internal hydraulic forces for pipe breaks in the cold leg than in the hot leg (for breaks of similar area and distance from the RPV). Pipe breaks farther away from the reactor vessel are less severe because the pressure wave attenuates as it propagates toward the reactor vessel.

The loads described in this section were applied to the model of the reactor pressure vessel system and the input to the analysis was specifically applicable to Watts Bar Unit 1. The time history displacements of the lower core plate, upper core plate, and the upper core barrel at the upper core plate elevation for this analysis are addressed in Section 2.5 for fuel/grid impact.

6.5.3 Structural Evaluation of Reactor Internals Components

From the RPV System LOCA/seismic time history analyses the component impact loads were used to assess the integrity of the reactor internals during faulted conditions. Based on this assessment the structural integrity of critical reactor internals components is maintained during faulted conditions (LOCA/seismic).

6.6 RCCA Scram Performance Evaluation

The Rod Control Cluster Assemblies (RCCAs) represent perhaps the most critical interface between the fuel assemblies and the other internals components. It is imperative to ensure that the impact of the RFA-2 will not adversely impact the operation of the control rods during normal operation.

An analysis was performed to determine this impact on the control rod scram performance. The results of this analysis indicated that the currently accepted 2.7 second RCCA drop time to dashpot entry (from gripper release of the drive rod) remained limiting.

7.0 Tritium Producing Burnable Absorber Rods (TPBAR) Evaluation

7.1 Introduction and Summary

The use of TPBARs (Tritium Producing Burnable Absorber Rods) in conjunction with RFA-2 fuel has been evaluated. This section will discuss the results of the evaluation. Specifically, Section 7.2, Mechanical Design Features, discusses the mechanical compatibility of TPBARs and RFA-2 fuel. Section 7.3, Nuclear Design, discusses the use and implications of RFA-2 fuel in Watts Bar tritium production core (TPC) designs. Section 7.4, Thermal and Hydraulic Design, discusses the thermal-hydraulic aspects of RFA-2 fuel and TPBARs. Finally, Section 7.5, Accident Analysis, provides an evaluation of TPBARs and RFA-2 fuel with respect to the UFSAR Chapter 15 safety analyses.

As the discussion below demonstrates, the use of RFA-2 fuel will have only minor effects on Tritium Production Core designs relative to TPC designs with all V+/P+ fuel. The power distributions, peaking factors, and TPBAR performance (tritium production) for TPC designs with RFA-2 fuel will be very similar to core designs utilizing V+/P+ fuel. The primary impact of RFA-2 fuel is a small decrease in core reactivity (~28 ppm boron in the TPC equilibrium cycle). Tritium production is only minimally affected. Furthermore, the evaluations of key safety parameters for these designs confirm that the conclusions of the Reference 1, "Implementation and Utilization of Tritium Producing Burnable Absorber Rods in Watts Bar Unit 1," with respect to UFSAR Chapter 15 events remain valid.

7.2 Mechanical Design Features

The new mechanical features of the RFA-2 fuel assembly described in Section 2.0 have been designed to be compatible with existing core components including the TPBAR design. The RFA-2 thimble tubes retain the same ID as the V+/P+ thimble tubes and, therefore, provide sufficient clearance for insertion of TPBARs and assure the proper operation of these core components.

The RFA-2 and V+/P+ fuel assemblies have the same BOL cold distance between the top nozzle adapter plate and thimble screw. Through this comparison, it was determined that there will continue to be a sufficient EOL gap between the bottom tip of the TPBAR and top of the thimble screw. The BOL, hot distance from the bottom core plate to the center of the fuel stack does not change.

The evaluation for the TPBAR holddown assembly was reviewed and remains applicable for RFA-2 implementation. The holddown spring of the TPBAR has sufficient holddown force at beginning and end of life conditions.

As discussed in Reference 1, the additional weight per assembly for 24 TPBARs plus the holddown assembly is approximately 4% of the weight of a typical fuel assembly. The added TPBAR assembly weight, together with the rodlet stiffness, has an insignificant effect on the RFA-2 fuel assembly dynamic characteristics. The grid impact load results for RFA-2 implementation were reviewed. The maximum combined seismic and LOCA grid impact force is less than 46% of the allowable grid strength. With a conservative load increase penalty of 20% due to the added TPBAR assembly, there is still sufficient grid load margin. Thus, the use of TPBAR assemblies in conjunction with RFA-2 fuel has no impact on the fuel assembly structural integrity.

Therefore, it is concluded that the RFA-2 implementation will not affect the use of a full complement of TPBARs in the Watts Bar plant. The design basis analyses and evaluations performed for Watts Bar TPBAR assessment in Reference 1 remain applicable.

7.3 Nuclear Design

To assess the implications of RFA-2 fuel on Watts Bar TPC designs, the conceptual first transition, second transition, and equilibrium cycle TPC designs developed for Reference 1 were re-modeled with RFA-2 using the same loading patterns and TPBAR inventories. These core designs each used 96 feed assemblies at 4.95 w/o U-235. No changes were made to the assembly loading patterns, TPBAR patterns, or TPBAR Li-6 loadings.

The RFA-2 has more structural material than the V+/P+ fuel currently used in Watts Bar due to the presence of IFM grids, slightly thicker guide thimble and instrumentation thimble material, and slightly increased grid masses and volumes. The neutronic effect this additional material is to displace a small amount moderator from the assembly and slightly increase parasitic neutron absorption. As will be discussed below, however, the implications of these differences are small.

The use of IFM grids also results in an increase in core hydraulic resistance as discussed in Section 1.3.2. This increased resistance causes the bypass flow to increase from 9.0% to 9.6%. The reduced core flow leads to an increase in the core average moderator temperature of 0.3 °F. The RFA-2 core models used in this evaluation include this moderator temperature difference. The effect of this small temperature change on core neutronics is minimal.

Table 7-1 provides a depletion summary comparison of the RFA-2 and V+/P+ equilibrium cycle TPC models. The cycle depletion parameters compared are critical boron concentration, steady-state F_Q, $F_{\Delta H}$, and axial offset. As the table shows, the effect of RFA-2 on these key core depletion parameters is very minor. The F_Q and $F_{\Delta H}$ peaking factors exhibit differences of less than 1%. Comparison of the axial offset values gives comparable results. Critical boron concentration differences of up to 28 ppm were observed, with the RFA-2 designs exhibiting lower critical boron concentrations. The largest critical boron differences occurred in the equilibrium cycle model since this design has the largest inventory of RFA-2 fuel. As mentioned above, the critical boron concentration differences are due to the additional structural material employed in the RFA-2 fuel. This additional material is a small reactivity penalty, resulting in slightly lower critical boron concentrations and slightly longer power coastdowns to achieve the target cycle energy of 510 EFPD. While Table 7-1 provides comparisons for only the equilibrium cycle models, comparisons for the transition models gave similar results.

Comparison of the RFA-2 and V+/P+ core power distributions demonstrates that the transition from V+/P+ to RFA-2 fuel will have only minimal impacts on radial power distribution. The largest effect is seen in the first transition cycle. In the first transition cycle RFA-2 core model, only the 96 feed assemblies are RFA-2 fuel. At BOL, the relative powers in these assemblies are up to 0.5% lower than the powers in the corresponding feed assemblies in the V+/P+ model. This is consistent with the slightly lower reactivity of these assemblies due to the additional structural material. The assembly percent differences in the second transition and equilibrium cycle models are even smaller than 0.5%, as expected since in these designs, except for the second transition cycle center assembly, all of the fuel is RFA-2.

These comparisons demonstrate that the effect of RFA-2 fuel on core power distributions and peaking factors will be minimal. As such, key safety parameters for RFA-2 TPC designs are expected to be comparable to those for V+/P+ TPC designs. The primary effect of RFA-2 fuel is a small decrease in excess reactivity that results in slightly lower critical boron concentrations (~28 ppm).

Tritium production is only minimally affected by the use of RFA-2 fuel. In the RFA-2 equilibrium cycle model, tritium production decreased by 9 grams or ~0.5%. This can likely be attributed to the slightly thicker RFA-2 guide thimbles which displace some moderator in the TPBAR cells, slightly reducing the thermal flux and, therefore, the Li-6 reaction rate.

The core design related key safety parameter evaluations performed as part of Reference 1 were reviewed with respect to the use of RFA-2 fuel in the tritium production core (TPC) designs. Given the small differences in power distributions, peaking factors, and critical boron concentrations, the result of this evaluation was that the conclusions of Reference 1 remain valid for RFA-2 for the fuel management scheme employed in the TPC designs. As part of this evaluation, the key safety parameters associated with the Steamline Break with Coincident Rod Withdrawal at Power event were re-calculated since this event showed very little margin in the Reference 1 analysis. Small changes of previously assumed values for the Doppler-Only Power Coefficient (DPC) and control rod worth were noted and, as a result, the event was evaluated using the recalculated DPC and control rod worth. The evaluation of the Steamline Break with Coincident Rod Withdrawal at Power event demonstrated that the DNB design basis would be met for the use of TPBARs in conjunction with RFA-2 fuel.

The TPBAR design inputs of Reference 1 were reviewed to determine whether the introduction of RFA-2 fuel would result in significant changes to these inputs. Since RFA-2 fuel causes only minor changes to core power distributions and peaking factors, the introduction of RFA-2 fuel will have only minimal impacts on the TPBAR designs inputs as well. In the original evaluation of the TPBAR design inputs, a 12.5% uncertainty was employed when calculating tritium production in the peak TPBAR. This uncertainty included components of 5% for local power and flux, 2% for PHOENIX-L TPBAR reaction rate uncertainties, 3.5% for TPBAR loading tolerances, and a 2% bias term for intra-assembly gradients. The 12.5% total uncertainty was conservatively determined by adding these components. However, convolution of the first three of these terms, which are independent, is justified. With the addition of the intra-assembly bias, this convolution results in a lower uncertainty of 8.4%. This value could be modified further depending upon the as-built tolerance data of the $LiAlO_2$ pellet Li-6 loadings.

7.4 Thermal and Hydraulic Design

As discussed in Section 4.3, the increased flow resistance of the RFA-2 fuel caused by the IFM grids results in an increased core bypass flow. To account for this impact, the core bypass flow limits were increased. An evaluation of the core bypass flow demonstrated that the new core bypass flow limits would be met for the use of TPBARs in conjunction with RFA-2 fuel.

The TPBAR guide thimble boiling analysis was reviewed for RFA-2 implementation. The acceptance criteria for this analysis are: 1) no surface boiling from the TPBAR within the dashpot region, and 2) no bulk boiling in the thimble along its length. The analysis performed demonstrated that these criteria will be met for RFA-2 fuel.

As discussed in Section 7.3, the evaluations of the tritium production core designs with RFA-2 fuel resulted in small violations of previously assumed values for the Doppler-Only Power Coefficient and control rod worth. As a result, the Steamline Break with Coincident Rod Withdrawal at Power event was evaluated using a recalculated DPC and control rod worth. The DNB evaluation for the Steamline Break with Coincident Rod Withdrawal at Power event showed that the Safety Analysis Limit DNBR was met for the RFA-2 fuel. The results of the DNB analysis for the V+/P+ fuel required the allocation of a small amount of available DNBR margin to demonstrate that the DNB design criterion was met. The DNBR margin allocated for this event for the V+/P+ fuel is less than the limiting event (Dropped Rod) listed in Table 4-2. As a result, the total DNBR penalties and net unused DNBR margin listed in Table 4-2 are unaffected by the Steamline Break with Coincident Rod Withdrawal at Power event. Therefore, the evaluation of the Steamline Break with Coincident Rod Withdrawal at Power event demonstrated that the DNB design basis would be met for the use of TPBARs in conjunction with RFA-2 fuel.

7.5 Accident Analyses

7.5.1 Non-LOCA Accidents

The Tritium producing burnable absorber rods may be added to the Watts Bar reactor core in a future reload cycle with the RFA-2 fuel. In evaluating the impact that a Tritium Production Core, or any other reload, has on the Non-LOCA licensing basis safety analyses, the Reload Safety Analysis Checklist (RSAC) is used, as described in Section 5.1.1.5. In using the RSAC, reactivity parameters for the reload core are compared to the parameters that are assumed in the safety analyses. If the parameters for the reload, in this case the reload with TPBARs, are bounded by those assumed in the safety analyses, it is concluded that the reload has no impact on the licensing basis safety analyses.

The TPBARs do not affect any parameters assumed in the Non-LOCA analyses with the exception of a potential impact on core reactivity characteristics. As discussed in Section 7.3, the evaluations of the tritium production core designs with RFA-2 fuel resulted in small changes to previously assumed values for the Doppler-Only Power Coefficient and control rod worth. Therefore, the Steamline Break with Coincident Rod Withdrawal at Power event was evaluated using a recalculated DPC and control rod worth. The resultant DNB evaluation for the Steamline Break with Coincident Rod Withdrawal at Power event showed that the DNB design basis would be met for the use of TPBARs in conjunction with RFA-2 fuel.

The analysis previously performed to address adding TPBARs to the Watts Bar core demonstrated that, relative to Non-LOCA analyses, so long as the RSAC parameters are not violated on a cycle-to-cycle basis, the results of the analyses are determined to be unaffected by the proposed reload and the results/conclusions presented in the UFSAR remain valid. Thus, there is no impact on the Non-LOCA analyses regardless of fuel type so long as the RSAC parameters continue to be met on a cycle-to-cycle basis. Therefore, the addition of TPBARs can be supported for the RFA-2 fuel upgrade program.

7.5.2 LOCA Accidents

This section summarizes the Tritium-producing burnable absorber rods LOCA related evaluations performed for the Watts Bar upgrade to the 17x17 RFA-2 fuel assembly.

7.5.2.1 Large Break LOCA

The TPBAR large break LOCA analysis for Watts Bar Unit 1 was completed using the LOCTA_JR computer code, with thermal-hydraulic boundary conditions derived from the non-TPBAR large break LOCA analysis-of-record. (Note that it was also concluded previously that the non-TPBAR large break LOCA analysis applies to Tritium Production Cores.) The thermal-hydraulic boundary conditions used in LOCTA_JR were based on a calculation with a PCT above the 95th-percentile value that is reported for licensing purposes, and the heat transfer coefficients were adjusted in an extremely conservative manner to ensure a bounding prediction of the maximum TPBAR temperature. Even with this highly conservative formulation, significant margin was demonstrated to the pertinent 10 CFR 50.46 limits.

For this program, a partial reanalysis using the 1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model has concluded that the analysis-of-record bounds the introduction of RFA-2 fuel. Since the thermal-hydraulic boundary conditions used in LOCTA_JR therefore remain bounding, and since the small increase in the thickness of the guide thimbles would be expected to have a negligible effect on results, no changes to the current TPBAR large break LOCA analysis are required to address the introduction of RFA-2.

7.5.2.2 Small Break LOCA

The TPBAR small break LOCA analysis for Watts Bar Unit 1 was completed using the LOCTA_JR computer code, with thermal-hydraulic boundary conditions derived from the non-TPBAR small break LOCA analysis-of-record. (Note that it was also concluded previously that the non-TPBAR small break LOCA analysis applies to Tritium Production Cores.) Since the introduction of RFA-2 fuel with IFM grids has been determined to have a negligible effect on the non-TPBAR small break LOCA analysis-of-record, and since the small increase in the thickness of the guide thimbles would be expected to have a negligible effect on results, no changes to the current TPBAR small break LOCA analysis are required to address the introduction of RFA-2.

7.5.2.3 Post-LOCA Long-Term Cooling, Subcriticality Evaluation

The long term core cooling calculation is performed to ensure that the mixed mean sump boron concentration remains adequate to prevent a return to criticality in the long term following a LOCA, assuming no credit for control rod insertion (in the event of a large break LOCA). The key inputs for this calculation include the minimum volumes and boron concentrations of the various sources of borated water and the maximum volumes of various dilution sources that could arrive in the sump post-LOCA.

For the implementation of Tritium Producing Burnable Absorber Rods (TPBARs), the minimum RWST boron concentration was increased from 2500 ppm to 3600 ppm and the minimum accumulator boron concentration was increased from 2400 ppm to 3500 ppm. With the increases in minimum boron concentration in the RWST and accumulators, a new post-LOCA sump boron curve was calculated. This new curve will serve as the reload limit for future Watts Bar core designs.

Testing on TPBARs has indicated that, following a large break LOCA, there is a potential for TPBARs to fail and lose lithium by leaching and/or expulsion of $LiAlO_2$ pellets (Reference 1). This loss of lithium can lead to increased core reactivity, which makes it more difficult to maintain core subcriticality under post-LOCA conditions. For cold leg breaks TPBAR failure is assumed to occur, however control rod insertion is also expected. The Westinghouse Owners Group (WOG) has demonstrated that control rod insertion will occur for these breaks. Therefore, the control rod worth was used to offset the penalty of TPBAR leaching and pellet loss. For hot leg breaks, it is highly unlikely that TPBAR failure will occur since hot leg breaks have more favorable thermal hydraulic conditions, and, therefore the accumulator and RWST boron increase is sufficient to maintain post-LOCA subcriticality for these types of breaks. This evaluation is applicable to both V5H and RFA-2 fuel types.

7.5.2.4 Hot Leg Switchover to Prevent Potential Boron Precipitation

The hot leg switchover (HLSO) procedure is initiated to preclude the possibility of boron precipitation in the reactor vessel core region following a LOCA. The key inputs to this calculation include:

- the maximum volumes and boron concentrations of the various sources of borated water which could arrive in the sump post-LOCA,
- the minimum volumes of various dilution sources which could arrive in the sump post-LOCA,
- the mixing volume of the core region in the vessel, and,
- the maximum core power level (prior to the LOCA).

For the implementation of TPBARs, the maximum RWST and accumulator boron concentrations were increased from 2700 ppm to 3800 ppm. With the increases in maximum boron concentration in the RWST and accumulators, a new HLSO time was calculated. The evaluation, which used the new RWST and accumulator boron concentrations and the Appendix K decay heat assumption, resulted in a HLSO time of 4.16 hours, which was reduced to 3.0 hours for further margin to the boron precipitation limit. The minimum flow requirements were evaluated at the new hot leg switchover time. It was determined that the minimum flow requirements were satisfied for a HLSO time of 3.0 hours. This calculation is applicable to both V5H and RFA-2 fuel types.

Table 7-1 RFA-2 and V+/P+ ANC-L Model Depletion Comparisons for the TPC Equilibrium Cycle

Cycle Burnup (MWD/MTU)	Critical Boron Concentration			Steady State F_Q			$F_{\Delta H}$			Axial Offset (%)		
	RFA-2	V+/P+	Δ ppm	RFA-2	V+/P+	%Δ	RFA-2	V+/P+	%Δ	RFA-2	V+/P+	Δ
0	1693	1708	-15	1.685	1.693	-0.47	1.437	1.442	-0.35	0.75	1.72	-0.97
150	1216	1236	-20	1.661	1.658	0.18	1.433	1.437	-0.28	-3.22	-2.66	-0.56
1000	1202	1222	-20	1.603	1.603	0.00	1.391	1.391	0.00	-3.79	-3.25	-0.54
2000	1217	1237	-20	1.604	1.599	0.31	1.385	1.384	0.07	-4.49	-3.91	-0.58
3000	1221	1241	-20	1.619	1.612	0.43	1.376	1.376	0.00	-5.02	-4.33	-0.69
5000	1164	1185	-21	1.653	1.646	0.43	1.405	1.412	-0.50	-5.05	-4.59	-0.46
7000	1056	1078	-22	1.662	1.667	-0.30	1.426	1.432	-0.42	-4.72	-4.56	-0.16
9000	915	939	-24	1.653	1.659	-0.36	1.432	1.438	-0.42	-4.36	-4.33	-0.03
11000	756	781	-25	1.630	1.633	-0.18	1.425	1.430	-0.35	-3.91	-3.93	0.02
13000	583	609	-26	1.606	1.605	0.06	1.409	1.413	-0.28	-3.91	-3.78	-0.13
15000	404	430	-26	1.568	1.569	-0.06	1.390	1.393	-0.22	-3.24	-3.21	-0.03
17000	219	245	-26	1.548	1.558	-0.64	1.367	1.371	-0.29	-2.79	-3.04	0.25
19000	30	58	-28	1.564	1.560	0.26	1.360	1.360	0.00	-3.19	-3.02	-0.17
19772	10	10	0	1.492	1.493	-0.07	1.357	1.359	-0.15	1.56	-0.36	1.92

*Note: The last row of data is for the EOL burnup step modeled as a power coastdown. The critical boron concentration at EOL was set to 10 ppm in both models.

8.0 SER Conditional Requirements for the DNB Analysis of RFA-2 Fuel

8.1 Introduction

The Watts Bar Unit 1 fuel upgrade to the Robust Fuel Assembly-2 (RFA-2) design includes the use of Intermediate Flow Mixer (IFM) grids and the use of the WRB-2M DNB correlation to provide improved DNB performance. The IFM fuel feature was initially reviewed and approved in WCAP-10444-P-A (Reference 3). The WRB-2M correlation was reviewed and approved in WCAP-15025-P-A (Reference 19). The SERs associated with these topical reports contain conditional requirements for application. In addition, the SERs for the related methodologies used in the DNB analysis of the RFA-2 fuel, i.e., the subchannel analysis code VIPRE-01 (VIPRE) (WCAP-14565-P-A, Reference 18) and the statistical DNB methodology RTDP (WCAP-11397-P-A, Reference 16), also contain conditional requirements. The conditional requirements for these DNB-related topical reports are addressed herein for the Watts Bar Unit 1 fuel upgrade to the RFA-2 design. The SER conditional requirements associated with the topical report on the methodology for calculating transition core DNBR penalties (WCAP-11837-P-A, Reference 21) were addressed in WCAP-14565-P-A (Reference 18) for use with the VIPRE code.

8.2 Responses to NRC Staff Questions on the Proposed Use of IFMs – WCAP-10444

The NRC Staff reviewed Westinghouse WCAP-10444 and concluded in a Staff Safety Evaluation Report (SER), Reference 35, that the generic topical report was an acceptable reference to support plant-specific applications for use of VANTAGE 5 fuel provided thirteen conditions identified in the SER were addressed by the licensees. These thirteen conditions were appropriately considered in the safety evaluation. Each of the thirteen conditions is addressed below, and a reference is provided for the specific section in the safety evaluation or other appropriate documentation where the condition is discussed.

<u>WCAP-10444 SER Condition 1:</u>

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.

<u>Response:</u>

The statistical convolution method was not used for fuel evaluation. To determine the initial rod-to-nozzle growth gap from fuel rod irradiation growth, the worst-case fabrication tolerances are used to evaluate fuel rod performance on a cycle-specific basis as described in Section 2.0 of the safety evaluation. This evaluation was in compliance with Condition 1.

<u>WCAP-10444 SER Condition 2:</u>

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.

<u>Response:</u>

An evaluation of the structural integrity of the RFA-2 fuel assembly with IFMs has been performed considering the lateral effects of a LOCA and seismic accident. The results, as described in Section 2.0 of the safety evaluation show that the RFA-2 fuel assembly is structurally acceptable for an all RFA-2 core and a transition core consisting of both RFA-2 fuel assemblies and VANTAGE+ (w/o IFMs) fuel assemblies.

<u>WCAP-10444 SER Condition 3:</u>

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

<u>Response:</u>

A demonstration program was successfully completed to determine early performance data on the VANTAGE 5 fuel assembly design features. Since then, numerous commercial reactors have operated successfully with VANTAGE 5 fuel assembly design features in transition cores and in full cores. For the

Robust Fuel Assembly design, a demonstration program was also successfully completed. In addition, fuel products with RFA-2 fuel assembly design features (including IFMs) have operated successfully in transition cores and full cores at a number of commercial reactors.

WCAP-10444 SER Condition 4:

For those plants using ITDP, the restrictions numerated in Section 4.1 of this report (WCAP-10444 SER), must be addressed and information regarding measurement uncertainties must be provided,

Response:

Watts Bar Unit 1 is not using ITDP. The Revised Thermal Design Procedure (RTDP), WCAP-11397, is being used and as such, the restrictions given in the RTDP SER are addressed specifically for that topical report in Section 8.5.

WCAP-10444 SER Condition 5:

The WRB-2 correlation with a DNBR 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.

Response:

The WRB-2M DNB correlation (WCAP-15025-P-A) is being applied to the 17X17 RFA-2 fuel with IFMs. The restrictions given in the WRB-2M SER are addressed specifically for that topical report in Section 8.3.

WCAP-10444 SER Condition 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 Technical Specification Bases.

Response:

This condition is not applicable since all fuel in Watts Bar Unit 1 will be 17X17.

WCAP-10444 SER Condition 7:

Plant-specific analysis should be performed to show that the DNBR limit will not be violated with the higher value of $F_{\Delta H}$.

Response:

This condition is not applicable since the $F_{\Delta H}$ limit is not being increased.

WCAP-10444 SER Condition 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.

Response:

For the RFA-2 fuel transition in Watts Bar Unit 1, an evaluation of the steamline break transient was performed as discussed in Section 5.1.4.1 of the safety evaluation. This evaluation concluded that the transient response would be insignificantly impacted by the changes introduced by the RFA-2 with IFMs. The DNB analysis for this event is performed on a cycle-specific basis to confirm that the DNB design basis is met.

WCAP-10444 SER Condition 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.

Response:

The mechanistic approach identified in NUREG-0562 (Reference 3 of WCAP-10444) was not used for the locked rotor accident, which was evaluated in Section 5.1.2.5 of the safety evaluation. For the locked rotor analysis, any fuel rods which violates the 95/95 DNBR limit is assumed to fail.

WCAP-10444 SER Condition 10:

If a positive MTC is intended for VANTAGE 5, the same positive MTC consistent with the plant Technical Specification should be used in the plant-specific safety analysis.

Response:

A positive MTC is not used in the operation of Watts Bar Unit 1. All of the analyses in the safety evaluation were performed consistent with the MTC limits listed in the Core Operating Limits Report.

WCAP-10444 SER Condition 11:

The LOCA analysis performed for the reference plant with higher F_Q of 2.55 has shown that the PCT limit of 2200°F is violated during transitional mixed core. Plant specific LOCA analysis must be done to show that the appropriate value of F_Q, the 2200°F criteria can be met during use of transitional mixed core.

Response:

Watts Bar Unit 1 LOCA analyses for RFA-2 fuel were performed with consideration of transition core effects. The large break LOCA analyses are summarized in Section 5.2.1 of the safety evaluation. Section 5.2.1 presents the results of a transition core study which concludes that Watts Bar Unit 1 remains in compliance with the requirements of 10CFR50.46 based on the Analysis Of Record for both the transition from the current V+/P+ fuel to the RFA-2 fuel with IFMs and for a full core of the RFA-2 fuel with IFMs based on a LOCA F_Q of 2.50 and an $F_{\Delta H}$ of 1.65.

WCAP-10444 SER Condition 12:

Our 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. Our 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.

Response:

WCAP-10125 has been approved (Reference 11). The extended burnup methodology contained in this topical has been applied and is addressed in Section 2.0 of the safety evaluation.

WCAP-10444 SER Condition 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.

Response:

The Watts Bar Unit 1 has 12-foot long fuel assembly bundles. Therefore, the above condition is not applicable.

8.3 Responses to NRC Staff Questions on the Proposed Use of WRB-2M - WCAP-15025

The NRC Staff reviewed Westinghouse WCAP-15025, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Bundles with Modified LPD Mixing Vane Grids," and concluded in the Staff Safety Evaluation Report (SER), Reference 36, that the use of the WRB-2M correlation with a DNBR limit of 1.14 is acceptable for plant safety analyses provided that the following conditions are met:

WCAP-15025 SER Condition 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 the type without further justification. Modified Vantage 5H fuel with or without modified intermediate flow mixer grids may be evaluated with WRB-2M.

Response:

The structural mid-grid design used in the RFA-2 fuel assembly is a minor modification of the Modified Low Pressure Drop mid-grid design that was addressed in WCAP-15025 for use with the WRB-2M DNB correlation. The RFA-2 mid-grid design was evaluated by means of the NRC-approved Fuel Criteria Evaluation Process (FCEP), Reference 7. By complying with the requirements of FCEP, it has been demonstrated that the new mid-grid design meets all design criteria of existing tested mid-grids that form the basis of the WRB-2M correlation database and that the WRB-2M correlation with a 95/95 correlation limit of 1.14 applies to the new RFA-2 mid-grid. As required by FCEP, the Westinghouse notification to the NRC of the RFA-2 mid-grid design modifications and the validation of the WRB-2M DNB correlation applicability to the RFA-2 mid-grid was provided in Reference 8.

WCAP-15025 SER Condition 2:

Since WRB-2M is dependant 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.

Response:

For the RFA-2 fuel upgrade in Watts Bar Unit 1, the analysis of the RFA-2 fuel was based on the VIPRE computer code (as licensed by Westinghouse in Reference 18) and the WRB-2M DNB correlation with a 95/95 correlation limit of 1.14. The use of VIPRE-01 by Westinghouse with WRB-2M was approved by the NRC as part of Reference 18. As discussed for Condition 1, the Westinghouse notification to the NRC of the validation of the WRB-2M DNB correlation applicability to the RFA-2 mid-grid was provided in Reference 8.

WCAP-15025 SER Condition 3:

WRB-2M may be used for PWR 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.

Response:

The WRB-2M correlation is not used for the loss of coolant accident analysis of the RFA-2 fuel in Watts Bar Unit 1.

WCAP-15025 SER Condition 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.

Response:

Application of the WRB-2M correlation to the RFA-2 fuel upgrade in Watts Bar Unit 1 was consistent with the range of parameters specified in Table 4-1 of WCAP-15025-P-A.

8.4 Responses to NRC Staff Questions on the Proposed Use of VIPRE - WCAP-14565

The NRC Staff reviewed Westinghouse WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal/Hydraulic Safety Analysis," and concluded in a Staff Safety Evaluation Report (SER), Reference 37, that the generic topical report was an acceptable reference to support plant specific applications for use of VIPRE-01, provided four conditions identified in the SER were addressed by the licensees. These four conditions were appropriately considered in the safety evaluation. Each of the four conditions is addressed below, and a reference is provided for the specific section in the safety evaluation or other appropriate documentation where the condition is discussed. The original SER conditions on the VIPRE-01 code (Reference 38) have been addressed in Reference 18.

<u>WCAP-14565 SER Condition 1:</u>

Selection of the appropriate CHF 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.

<u>Response:</u>

The WRB-2M correlation with a 95/95 correlation limit of 1.14 was used in the DNB analyses for the RFA-2 fuel upgrade in Watts Bar Unit 1. As discussed in Section 8.3, the Westinghouse notification to the NRC of the validation of the WRB-2M DNB correlation applicability to the RFA-2 mid-grid was provided in Reference 8.

The use of the plant specific hot channel factors and other fuel dependent parameters in the DNB analysis for the RFA-2 fuel upgrade in Watts Bar Unit 1 have been previously used and approved for the safety evaluation supporting the 1.4% power uprate for Watts Bar (Reference 39).

<u>WCAP-14565 SER Condition 2:</u>

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.

<u>Response:</u>

The core boundary conditions for the VIPRE calculations for the RFA-2 fuel upgrade are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE as discussed in the safety evaluation. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in Reference 2.

WCAP-14565 SER Condition 3:

The NRC Staff's generic SER for VIPRE (Reference 38) 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.

Response:

As discussed in response to Condition 1, the WRB-2M correlation with a limit of 1.14 was used for the DNB analyses of RFA-2 fuel in Watts Bar Unit 1. The WRB-1 correlation with a DNBR limit of 1.17 continues to be used for the DNB analyses of the VANTAGE-5H and VANTAGE+ fuel designs without IFMs that have been previously loaded into Watts Bar Unit 1.

WCAP-14565 SER Condition 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 (Reference 38) did not extent 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.

Response:

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

8.5 Responses to NRC Staff Questions on the Proposed Use of RTDP - WCAP-11397

The NRC Staff reviewed Westinghouse WCAP-11397, "Revised Thermal Design Procedure," and concluded in a Staff Safety Evaluation Report (SER), Reference 40, that the generic topical report was an acceptable reference to support plant specific applications for use of RTDP, provided seven conditions identified in the SER were addressed by the licensees. These seven conditions were appropriately considered in the safety evaluation. Each of the seven conditions is addressed below, and a reference is provided for the specific section in the safety evaluation or other appropriate documentation where the condition is discussed.

<u>WCAP-11397 SER Condition 1:</u>

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

<u>Response:</u>

Sensitivity factors were evaluated using the WRB-2M DNB correlation and the VIPRE code for parameter values applicable to the RFA-2 fuel in Watts Bar Unit 1. These sensitivity factors were used to determine the maximum Design Limit DNBR for the RFA-2 fuel. The resultant Design Limit DNBR is included in the Watts Bar Unit 1 UFSAR.

<u>WCAP-11397 SER Condition 2:</u>

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.

<u>Response:</u>

See Response to Condition 1 above, as well as Section 8.3 for the justification of the WRB-2M correlation for the Watts Bar Unit 1 application and Section 8.4 for the justification of the VIPRE-01 code for the Watts Bar Unit 1 application.

<u>WCAP-11397 SER Condition 3:</u>

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.

<u>Response:</u>

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 core which was used for the parameters, and the VIPRE model used in this application do not differ significantly from those used in WCAP-11397-P-A.

<u>WCAP-11397 SER Condition 4:</u>

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

<u>Response:</u>

The plant specific variances and distributions for this application were justified in the proprietary report supporting the 1.4% power uprate for Watts Bar (Reference 20).

<u>WCAP-11397 SER Condition 5:</u>

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

<u>Response:</u>

Nominal initial conditions were only applied to DNBR analyses which used RTDP.

<u>WCAP-11397 SER Condition 6:</u>

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

<u>Response:</u>

Bounding nominal conditions were used in the DNBR analyses that were based on RTDP.

<u>WCAP-11397 SER Condition 7:</u>

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.

<u>Response:</u>

The code uncertainties specified in Table 3-1 of WCAP-11397-P-A were included in the DNBR analyses using RTDP.

9.0 References

1. NDP-00-0344, Revision 1, "Implementation and Utilization of Tritium Producing Burnable Absorber Rods in Watts Bar Unit 1", Westinghouse Electric Company, July 2001.

2. Davidson, S. L. (Ed.), et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-NP-A, July 1985.

3. Davidson, S. L. (Ed.), et al., "VANTAGE 5 Fuel Assembly Reference Core Report," WCAP-10444-P-A and Appendix A, September 1985; Addendum 1-A, March 1986; Addendum 2-A, April 1988.

4. Davidson, S. L., et al., "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A, April, 1995.

5. Brown, U. L.,"PERFORMANCE+ Fuel Features Safety Evaluation," SECL-92-305, April 5, 1994.

6. Smith, L. D., et al., "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Bundles with Modified LPD Mixing Vane Grids," WCAP-15025-P-A, April 1999.

7. Davidson, S. L. (Ed.), "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A, October 1994.

8. Letter from H. A. Sepp, (Westinghouse) to J. S. Wermiel (NRC), "Fuel Criterion Evaluation Process (FCEP) Notification of the RFA-2 Design, Revision 1 (Proprietary)," LTR-NRC-02-55, November 13, 2002.

9. "Updated Final Safety Analysis Report - Watts Bar Nuclear Plant, Amendment 3," Docket Number 50-390, updated August 30, 2002.

10. Foster, J. P. and Sidener, S., "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A, Revision 1 with Errata, July 2000.

11. Davidson, S. L. (Ed.), et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A (Proprietary), December 1985.

12. Davidson, S. L., et al., "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," WCAP-14297-A, March 1995.

13. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

14. Nguyen, T. Q., et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, June 1988.

15. Liu, Y. S., et al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, September 1986.

16. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.

17. Motley, F. E. et al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8763-A, July 1984.

18. Sung, Y., et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A/15306-NP-A, October 1999.

19. Smith, L. D., et al., "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Bundles with Modified LPD Mixing Vane Grids," WCAP-15025-P-A, April 1999.

20. Camarero, A. and Moomau, W. H., "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Tennessee Valley Authority Watts Bar Unit 1 1.4% Uprate to 3475 MW NSSS Power," WCAP-14738, Revision 1, August 2000.

21. Schueren, P. and McAtee, K. R., "Extension of Methodology for Calculating Transition Core DNBR Penalties," WCAP-11837-P-A, January 1990.

22. WCAP-7907-P-A, "LOFTRAN Code Description," T. W. T. Burnett, et al., April 1984.

23. WCAP-7908-A, "FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," H. G. Hargrove, December 1989.

24. WCAP-7979-A, "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," D. H. Risher and R. F. Barry, January 1975.

25. WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," D. H. Risher, January 1975.

26. Letter from W. J. Johnson (Westinghouse) to R. C. Jones (NRC), "Use of 2700 °F PCT Acceptance Limit in Non-LOCA Accidents," NS-NRC-89-3466, October 1989.

27. WCAP-14839 Rev. 1, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for the Watts Bar Nuclear Plant," September 1998.

28. WCAP-12945-P-A, Rev 1, "Computer Code Qualification Document for Best Estimate LOCA Analysis", S. M. Bajorek, et. al., March 1998.

29. WCAP-8339-NP-A, "Westinghouse ECCAS Evaluation Model Summary," Bordelon, F. M., July 1974.

30. WCAP-8471-NP-A, WCAP-8472-A, "Westinghouse ECCS Evaluation Model: Supplementary Information," Bordelon, F. M., April 1975.

31. NSID-TB-86-08, "Post-LOCA Long-Term Cooling: Boron Requirements," October 31, 1986.

32. WCAP-10325-P-A (Proprietary), WCAP-10326-A (Non-Proprietary), "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," May 1983.

33. WCAP-15699 Rev.1, "Tennessee Valley Authority Watts Bar Nuclear Plant Unit 1 Containment Integrity Analyses for Ice Weight Optimization Engineering Report," August 2001.

34. WCAP-8264-P-A, Rev. 1 (Proprietary), WCAP-8312-A (Non-Proprietary), "Westinghouse Mass and Energy Release Data For Containment Design," August 1975.

35. Letter from C. O. Thomas (NRC) to E. P. Rahe, Jr. (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-10444, 'VANTAGE 5 Fuel Assembly'," undated.

36. Letter from T. H. Essig (NRC) to H. Sepp (W), "Acceptance for Referencing of Licensing Topical Report WCAP-15025-P, 'Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Bundles with Modified LPD Mixing Vane Grids,' (TAC NO. MA1074)," December 1, 1998.

37. Letter from T. H. Essig (NRC) to H. Sepp (W), "Acceptance for Referencing of Licensing Topical Report WCAP-14565, VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal/Hydraulic Safety Analysis (TAC NO. M98666)," January 19, 1999.

38. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA Executive Committee), "Acceptance for Referencing of Licensing Topical Report, EPRI NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores,' Volumes 1, 2, 3 and 4," May 1, 1986.

39. Letter from R. T. Purcell (TVA), to the NRC, "Watts Bar Nuclear Plant (WBN) Unit 1 – Technical Specification (TS) Change No. 00-06 – Increase Unit 1 Reactor Power to 3459 MWt," TVA-WBN-TS-00-06, June 7, 2000.

40. Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-11397, 'Revised Thermal Design Procedure'," undated.

41. WCAP-9735 (proprietary), Rev. 2, WCAP-9736 (non-proprietary), Rev. 1, "MULTIFLEX 3.0, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics (III) -- Advanced Beam Model", February 1998.

42. WCAP-15029-P-A, WCAP-15030-NP-A "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions", December 1998.

43. WCAP-15245 (proprietary), WCAP-15246 (non-proprietary), "Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook, Units 1 and 2", May 28, 1999.

44. Letter from John F. Stang (US-NRC) to Robert P. Powers (Indiana Michigan Power Company), "Issuance of Amendments – Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6473 and MA6474)", December 23, 1999.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES - TS-02-13
MARK-UP

AFFECTED PAGE LIST:

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5.9 Reporting Requirements

5.9.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

5. WCAP-15088-P, Rev. 1, "Safety Evaluation Supporting A More Negative EOL Moderator Temperature Coefficient Technical Specification for the Watts Bar Nuclear Plant," July 1999, (W Proprietary), as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. 20 (Methodology for Specification 3.1.4 - Moderator Temperature Coefficient.).

6. Caldon, Inc. Engineering Report-80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997; and Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™," Revision 0, May 2000; as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. 31.

Insert A

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

INSERT A

7. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

8. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17 x 17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

9. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

PROPOSED TS BASES CHANGES - TS-02-13
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The primary DNB correlations are the WRB-1 correlation (Ref. 7) for VANTAGE 5H and VANTAGE+ fuel and the WRB-2M correlation (Ref. 8) for RFA-2 fuel with IFMs. These DNB correlations take

BASES

BACKGROUND (continued)

DNB is not a directly measurable parameter during operation; therefore, THERMAL POWER, reactor coolant temperature, and pressure are related to DNB through critical heat flux (CHF) correlations. ~~The WRB-1 CHF correlation (Ref. 7) is the primary DNB correlation and takes~~ credit for significant improvement in the accuracy of the CHF predictions. The W-3 CHF correlation ~~(Ref. 8 and 9)~~ is used for conditions outside the range of the WRB-1 correlation.

9 and 10

for VANTAGE 5H and VANTAGE+ fuel or the WRB-2M correlation for RFA-2 fuel with IFMs.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. High pressurizer pressure trip;

b. Low pressurizer pressure trip;

c. Overtemperature ΔT trip;

(continued)

1.25 / 1.24 (typical / thimble) for VANTAGE 5H and VANTAGE+ fuel and the WRB-2M CHF correlation with design limit DNBR values of 1.23 / 1.23 (typical / thimble) for RFA-2 fuel with IFMs.

BASES

SAFETY LIMITS (continued)

To meet the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. The effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit DNBR values that satisfy the DNB design criterion. SL 2.1.1 reflects the use of the WRB-1 CHF correlation with design limit DNBR values of ~~1.25 and 1.24 for the typical and thimble cell, respectively.~~

Additional ~~10%~~ DNBR margin is maintained by performing the safety analyses to ~~a~~ higher DNBR ~~limit of 1.39 and 1.38 for the typical and thimble cell, respectively.~~ limits. This margin between the design and safety analysis limit is more than sufficient to offset known DNBR penalties (e.g., rod bow) and transition core and to provide the DNBR margin for operating and design flexibility.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

(continued)

References (continued)

4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

5. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors."

6. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System."

7. WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1984.

9. $\boxed{8.}$ Tong, L. S., "Boiling Crisis and Critical Heat Flux," AEC Critical Review Series, TID-25887, 1972.

10. $\boxed{9.}$ Tong, L. S., "Critical Heat Fluxes on Rod Bundles," in "Two-Phase Flow and Heat Transfer in Rod Bundles," pages 31 through 41, American Society of Mechanical Engineers, New York, 1969.

8. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17 x 17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 3);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. Refer to the Bases for the Reactor Core Safety Limits, B 2.1.1 for a discussion of the applicable DNBR limits. The W-3 Correlation with a DNBR limit of 1.3 is applied in the heated region below the first mixing vane grid. In addition, the W-3 DNB correlation is applied in the analysis of accident conditions where the system pressure is below the range of the WRB-1 correlation. For system pressures in the range of 500 to 1000 psia, the W-3 correlation DNBR limit is 1.45 instead of 1.3.

for VANTAGE 5H and VANTAGE+ fuel or the WRB-2M correlation for RFA-2 fuel with IFMs.

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ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

PROPOSED TS AND TS BASES CHANGES TS-02-13
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B 2.0-6
B 3.2-13

5.9 Reporting Requirements

5.9.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

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7. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

8. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17 x 17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

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c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

BASES

BACKGROUND (continued)

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The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

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b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. High pressurizer pressure trip;

b. Low pressurizer pressure trip;

c. Overtemperature ΔT trip;

(continued)

BASES

SAFETY LIMITS (continued)

To meet the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. The effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit DNBR values that satisfy the DNB design criterion. SL 2.1.1 reflects the use of the WRB-1 CHF correlation with design limit DNBR values of 1.25/1.24 (typical/thimble) for VANTAGE 5H and VANTAGE+ fuel and the WRB-2M CHF correlation with design limit DNBR values of 1.23/1.23 (typical/thimble) for RFA-2 fuel with IFMs.

Additional DNBR margin is maintained by performing the safety analyses to higher DNBR limits. This margin between the design and safety analysis limit is more than sufficient to offset known DNBR penalties (e.g., rod bow and transition core) and to provide the DNBR margin for operating and design flexibility.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

(continued)

BASES

References (continued)

4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

5. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors."

6. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System."

7. WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1984.

8. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17 x 17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

9. Tong, L. S., "Boiling Crisis and Critical Heat Flux," AEC Critical Review Series, TID-25887, 1972.

10. Tong, L. S., "Critical Heat Fluxes on Rod Bundles," in "Two-Phase Flow and Heat Transfer in Rod Bundles," pages 31 through 41, American Society of Mechanical Engineers, New York, 1969.

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 3);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. Refer to the Bases for the Reactor Core Safety Limits, B 2.1.1 for a discussion of the applicable DNBR limits. The W-3 Correlation with a DNBR limit of 1.3 is applied in the heated region below the first mixing vane grid. In addition, the W-3 DNB correlation is applied in the analysis of accident conditions where the system pressure is below the range of the WRB-1 correlation for VANTAGE 5H and VANTAGE+ fuel or the WRB-2M correlation for RFA-2 fuel with IFMs. For system pressures in the range of 500 to 1000 psia, the W-3 correlation DNBR limit is 1.45 instead of 1.3.

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