ATTACHMENT 2

ANP-2986(NP), Revision 3

Sequoyah HTP Fuel Transition

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(NON-PROPRIETARY VERSION)

AREVA

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Sequoyah HTP Fuel Transition (NP)

AREVA NP Inc.

Nature of Changes

Item	Page	Description and Justification
Section 5.2.2.27.3	129	Updated text to state that Adv. W17 HTP transition does not affect parameters used in the environmental consequence analysis.
Section 5.2.2.27.7	130	Updated text to reference Section 5.2.2.27.3.

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1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

The purpose of this report is to facilitate the transition of the Sequoyah Nuclear Power Plant Units 1 and 2 from the use of Mark-BW fuel to the Advanced W17 HTP (Adv. W17 HTP) fuel design. Due to the Plant Technical Specification changes that are necessary in support of this fuel design change an LAR (License Amendment Request) will be submitted by TVA (Tennessee Valley Authority) to the U.S. NRC (Nuclear Regulatory Commission). Sequoyah plans to refuel and operate with the Adv. W17 HTP fuel in Units 1 and 2 starting with Unit 2 Cycle 19. The Adv. W17 HTP fuel design will consist of a 17 x 17 assembly configuration with M5™ fuel rods, Zircaloy-4 MONOBLOC™ guide tubes, Nickel Alloy 718 High Mechanical Performance (HMP) spacer at the lowermost axial elevation, Zircaloy-4 HTP spacers in all other axial elevations, Zircaloy-4 intermediate flow mixers (IFM) in the spans between spacers 4 and 5, 5 and 6 and 6 and 7, FUELGUARD lower tie plate (LTP) and the AREVA NP reconstitutable top nozzle (TN).

The Adv. W17 HTP fuel assembly design offers design changes relative to the resident Mark-BW fuel assembly design:

- Zircaloy-4 HTP intermediate spacers
- Zircaloy-4 IFMs
- Nickel Alloy 718 HMP lowermost spacer grid[1]
- MONOBLOC™ guide tubes[1]
- ¼ turn modular quick disconnect guide tube attachment to top nozzle[1]
- FUELGUARD lower tie plate[1]

The Zircaloy-4 HTP spacer and the nickel alloy 718 HMP spacer offer improved protection against fuel rod fretting damage. The Zircaloy-4 IFM enhances flow mixing at mid-span of the fuel assembly relative to an HTP FA without IFMs. The MONOBLOC™ guide tube design has increased lateral fuel assembly stiffness. The ¼ turn modular quick disconnect is an improved design which results in a quicker disengagement of the top nozzle with no loose parts. The FUELGUARD lower tie plate provides effective debris resistance with an acceptable pressure drop. All these features have already been exposed to considerable operating experience at other nuclear facilities in the US and world-wide.

The fuel rod design consists of [ ] inch outer diameter M5™clad containing a stainless steel alloy A286 lower plenum spring and a 302 stainless steel upper plenum spring. The fuel rod will contain enriched UO_2 or UO_2-Gd_2O_3 fuel pellets. The fuel rod end caps will be made of M5™ material and will be welded to the fuel rod cladding using the Upset Shape Welding (USW) process. The only difference between the fuel rod mechanical design used in the Adv. W17 HTP design and the current resident Mark-BW fuel rod is the diameter of the tip of the lower end cap due to the differences in the FUELGUARD lower tie plate and the TRAPPER bottom nozzle.

[1] Design Feature included with Advanced Mark-BW(A) Lead Test Assemblies in Sequoyah Unit 1

Section 1.2 of this report provides a more detailed discussion of the design features of the Adv. W17 HTP fuel assembly. Section 2 of the report outlines AREVA NP's mechanical and structural evaluation methodology for the fuel design. This section of the report further performs a review of NRC-approved mechanical design criteria that were utilized to license the lead fuel assemblies and which will be used to license the Adv. W17 HTP batch fuel. In addition, Section 2 offers the basis for demonstrating compatibility of the Adv. W17 HTP fuel design to the Sequoyah reactor internals, control components, and handling and storage equipment. A detailed assessment of AREVA NP's extensive operating experience with the Adv. W17 HTP fuel design features is also provided in Section 2.

Section 3 discusses the nuclear design bases and the methodologies for transitioning from the Mark-BW fuel design to the Adv. W17 HTP fuel for the Sequoyah units. The thermal and hydraulic design of the reactor that ensures the core can meet steady state and transient performance requirements without violating the acceptance criteria is discussed in Section 4. Transition cycles with the resident Mark-BW fuel and subsequent full core cycles with the Adv. W17 HTP fuel are addressed. Section 5 provides information related to assessing the Sequoyah transient and accident analyses for the proposed transition. Also, summary reports of sample analyses for the non-LOCA and realistic large break LOCA (RLBLOCA) analyses methodologies are referenced for application to Sequoyah.

1.2 Fuel Features

AREVA NP has developed the Adv. W17 HTP fuel assembly design for use in Westinghouse (W) three- and four-loop reactors using a 17 x 17 fuel rod array. The Adv. W17 HTP is a natural evolution of the W17 HTP fuel design that offers additional improvements in performance. AREVA's initial W17 HTP design was reviewed and approved by the NRC for generic use in Reference 1. The significant difference between the Adv. W17 HTP fuel assembly and the current resident fuel assembly, the Mark-BW design as described in Reference 2, are summarized below:

- Robust FUELGUARD™ lower tie plate that provides highly effective debris resistance with good flow characteristics and an acceptable pressure drop

- Low pressure drop quick disconnect (QD) top nozzle that uses a leaf spring holddown system and a low pressure drop nozzle structure

- Zircaloy-4 MONOBLOC™ guide tubes with two inside diameters (ID) (for the upper region and the dashpot) and a single outside diameter (OD). This feature of the MONOBLOC™ guide tubes provides a robust lower cross-section to minimize fuel assembly distortion, while also providing rapid insertion of the control rod cluster and a dashpot region that provides rod cluster deceleration and acceptable impact loads on the top nozzle. QD sleeves are attached to the upper end of the guide tubes for connection to the top nozzle.

- Intermediate flow mixers (IFM)

- HTP intermediate spacers – Zirconium alloy spacers

- High mechanical performance (HMP) lowermost spacer – Nickel alloy UNS N07718 to reduce cell relaxation during irradiation and maintain strength.

Extensive operating experience and data of AREVA pressurized water reactor (PWR) fuel throughout the world, using the M5™ alloy, provides the design bases for consistent irradiation performance and models used for the Adv. W17 HTP fuel assembly design. Based on comprehensive empirical testing and design evaluation analyses, the Adv. W17 HTP fuel assembly design is demonstrated acceptable for batch and full core implementation in Westinghouse-designed 17 x 17 plants. Figure 2-1 is a schematic of the Adv. W17 HTP fuel assembly.

The lower tie plate design is a FUELGUARD structure (See Figure 2-2). This structure uses curved blades to provide non-line-of sight flow paths for the incoming coolant to protect the fuel assembly from debris that may be present. This design is very efficient at preventing debris, including small pieces of wire, from reaching the fuel. The design uses the same blade configuration and spacing that has been used on CE 14x14, CE 15x15, Westinghouse 14x14, Westinghouse 15x15, Westinghouse 17x17 and Babcock & Wilcox (B&W) 15x15 designs in the United States. The FUELGUARD design has been used on reloads in the United States since 1993 and on W17 designs for over a decade.

The top nozzle (TN) is a reconstitutable design. The basic configuration is the same that has been used in the Mark-BW fuel designs. Figure 2-3 shows the Adv. W17 HTP top nozzle. This reconstitutable design uses machined features to engage with the guide tube assembly. The design does not create any loose or disposable parts during reconstitution. The Modular Quick Disconnect (QD) configuration is shown in Figure 2-4. The design has been used in the Adv. Mark-BW fuel designs since the reconstitution capabilities of this top nozzle design have been successfully demonstrated in the North Anna, Shearon Harris, Takahama and Braidwood lead assemblies as well as incorporated in the Adv. Mark BW(A) lead assemblies currently in Sequoyah 1.

The cage or skeleton uses 24 Zircaloy-4 MONOBLOC™ guide tubes, 1 Zircaloy-4 instrument tube, 7 Zircaloy-4 HTP spacers, 3 Zircaloy-4 IFM spacers and 1 alloy 718 HMP spacer at the lowest spacer position. Figure 2-5 shows the cage configuration. The HTP spacers are welded directly to the guide tubes; the HMP spacer is attached to the guide tubes by mechanically capturing the HMP between rings that are welded to the guide tubes. Because the guide tubes are of a zirconium alloy, they cannot be directly welded to the alloy 718 material used in the HMP. The HTP spacer was developed in the late 1980s and has been used on CE 14x14, CE 15x15, Westinghouse 14x14, Westinghouse 15x15, Westinghouse 17x17 and Babcock & Wilcox (B&W) 15x15 reloads in the United States. The initial reloads were in 1991 and the initial W17 reloads were in 1992. The design provides 8-way line contact as the interface between the fuel rod and the spacer grids, and is therefore very resistant to fuel rod failures from flow induced vibration fretting.

The HTP design provides the line contact for the fuel rods but also is configured to facilitate heat transfer. As seen in Figure 2-6, the spring structure forms a channel which provides a flow path. This flow path is set at an angle relative to the rod longitudinal direction, creating turbulent flow around the rod without creating a large pressure drop across the spacer. The HMP has the same line contact configuration but the channel is not angled. Because this spacer is at the lowermost position, the improved heat transfer is not necessary. As stated previously, the HMP material is Alloy 718. This material is very stable in irradiation environments and provides additional assurance that the rod/spacer contact will be maintained throughout the design lifetime. Figures 2-7, 2-8 and 2-9 show the HTP spacer grid assembly, the IFM spacer grid assembly and the HMP spacer grid assembly respectively.

The Adv. W17 HTP fuel bundle assembly uses a MONOBLOC™ guide tube design for the 24 core control component interface locations and a constant outer diameter and wall thickness tube for the center instrument tube (See Figures 2-10 and 2-11). The MONOBLOC™ design maintains the same inner diameters in the dashpot and non-dashpot regions as the Mark-BW fuel, but has a constant outer diameter the full length of the tube. Therefore, the wall thickness in the dashpot region (approximately the lower 20 inches of the guide tube) is increased. The MONOBLOC™ guide tube design has been used for fuel reload batches in Europe and lead assemblies, including the Adv. Mark BW(A), in the United States.

The fuel rod assembly for the Adv. W17 HTP fuel assembly is based on the current fuel rod utilized for the Mark-BW fuel assembly. The fuel rod assembly makes use of M5™ cladding and end caps, an upper and lower plenum spring and a column of UO_2 pellets or UO_2-Gd_2O_3 pellets with axial blankets. The M5™ material has very low corrosion and hydrogen pick-up rates; providing substantial margin for end of life corrosion and hydrogen content. This material has been used extensively both in Europe and the United States for fuel rod cladding and is currently used in the Sequoyah plants. Figure 2-12 provides a schematic of the Adv. W17 HTP fuel rod design.

1.3 ADVANCED Mark-BW(A) Lead Test Assemblies

The AREVA fuel design (Adv. W17 HTP) planned for introduction on a batch basis at Sequoyah is similar to the AREVA NP lead fuel assemblies that were introduced at Sequoyah Unit 1 in Cycle 16 which are currently operating in their third cycle with an expected discharge pin burnup of approximately 53 MWd/kgU.

The Adv. W17 HTP fuel assembly design offers improvements relative to the lead fuel assembly design –

- Zircaloy-4 HTP intermediate spacers
- Zircaloy-4 IFMs

The Zircaloy-4 HTP spacers offer improved protection against fuel rod fretting damage, and reduced slip loads between fuel rods and the upper end grid. Lower slip loads are designed to reduce the propensity for fuel assembly and fuel rod bow. The Zircaloy-4 IFM provides enhanced flow mixing at mid-span elevations. These features have already been exposed to considerable operating experience at other nuclear facilities in the US and world-wide.

1.4 References for Section 1.0

1. EMF-92-116(P)(A), Generic Mechanical Design Criteria for PWR Fuel Designs, February 1999
2. BAW-10172(P)(A), Mark-BW Mechanical Design Report, July 1988
3. BAW-10239(P)(A), ADVANCED Mark-BW Fuel Assembly Mechanical Design Topical Report, July 1, 2004

2.0 ADVANCED W17 HTP MECHANICAL DESIGN FEATURES

2.1 Introduction and Summary

This section evaluates the mechanical design of the Adv. W17 HTP fuel bundle assembly intended for batch implementation at Sequoyah Units 1 & 2 and its compatibility with the Mark-BW fuel during the transition from mixed-fuel type core populations to cores with only Adv. W17 HTP fuel.

The Adv. W17 HTP fuel assembly uses a 17x17 fuel rod array. Figure 2-1 highlights the primary design features of the Adv. W17 HTP fuel assembly. Table 2-1 provides comparisons of basic fuel assembly parameters of the Adv. W17 HTP fuel assembly to the Mark-BW fuel assembly.

The design changes described in this section are based on design change criteria cited in reference 4.

2.2 Mechanical Compatibility

Currently Sequoyah Unit 1 is running with Mark-BW and four Mark-BW(A) fuel assemblies and Sequoyah Unit 2 is running with a full core of Mark-BW fuel assemblies. The Adv. W17 HTP will be mechanically equivalent to the resident fuel and will continue to be mechanically compatible with the host reactor core internals, handling equipment, storage racks and resident Mark-BW fuel. A comparison of the mechanical design parameters of the Adv. W17 HTP to the resident Mark-BW fuel is presented in Table 2-1.

The hydraulic compatibility is discussed in detail within Section 4 of this report. Hydraulic compatibility analyses for the Adv. W17 HTP fuel assembly design in a transition core with Mark-BW fuel are used to calculate bounding cross-flow velocity profiles by assuming a mixed-core configuration that results in more severe cross-flow velocities than in a realistic mixed-core configuration. Preliminary analyses show the cross-flow velocity magnitudes are within the AREVA NP experience base of transition cores with fuel designs having HTP spacer grids.

2.2.1 Fuel Assembly

The Adv. W17 HTP fuel assembly is an improved 17 x 17 fuel assembly design specifically for Westinghouse-designed PWRs and utilizing many proven features of both the Mark-BW design and the W17 design. The array type, the number of fuel rods and guide tubes and the fuel rod pitch dimensions are the same as for the current resident Mark-BW fuel. The square and diagonal widths of the fuel assembly at the top nozzle and the lower tie plate and the spacer grids have been confirmed to be compatible with the core internals, storage racks, fuel elevator and the current resident fuel.

The Adv. W17 HTP fuel assembly utilizes 11 spacers that with the 24 guide thimbles, instrument tube, top nozzle and lower tie plate, provide the structural cage for the 264 fuel rod assemblies. The lower most spacer is made from nickel alloy 718 strip material. The upper most spacer, the 6 intermediate spacers and the 3 intermediate flow mixers are constructed from zircaloy-4 strip material. The M5™ clad fuel rods are lifted above the lower tie plate and are laterally supported by the lower most spacer, the 6 intermediate spacers and the upper most spacer. The Adv. W17 HTP upper most spacer and 6 intermediate spacers are welded to the guide thimbles.

The 24 guide tubes are the MONOBLOC™ design using Zircaloy-4 alloy. The MONOBLOC™ design uses a constant OD with the dashpot features integral to the IDs. The intermediate HTP spacers are welded to each guide tube. The nickel alloy HMP lower most spacer is not welded directly to the guide tubes because of the difference in materials. Instead, they are axially constrained by Zircaloy-4 alloy sleeves welded directly to each guide tube above and below the corresponding grid position at all 24 guide tube locations.

The Adv. W17 HTP top nozzle is constructed of 304L stainless steel and accommodates the QD features for the guide tube-to-top nozzle connections, which enable rapid removal and installation during fuel assembly reconstitution. The top nozzle also houses the holddown spring system, which consists of four sets of three-leaf springs made of nickel alloy 718 that are mounted to the top nozzle with nickel alloy 718 screws. The holddown springs maintain positive fuel assembly contact with the core support structure during normal operating conditions and provide positive holddown margins for precluding liftoff due to hydraulic flow forces while accommodating differential thermal expansion and irradiation growth of the fuel assembly.

The FUELGUARD™ lower tie plate is a 304L stainless steel brazement, which incorporates a series of parallel-curved blades that provide debris resistance by virtue of curved flow passages, allowing no direct line of sight through the nozzle, restricting the passage of debris but allowing coolant to pass through freely. The connection of the lower tie plate to the 24 guide tubes is accomplished using 304L stainless steel bolts that incorporate a mechanical locking feature.

Key fuel assembly dimensions establish compatibility with core and component interfaces. Table 2-1 compares key design attributes of the Adv. W17 HTP fuel bundle assembly to the current resident Mark-BW fuel bundle assembly.

2.2.2 Fuel Rod

The Adv. W17 HTP fuel rod design consists of uranium dioxide (UO_2) pellets contained in a seamless M5™ alloy tube with end plugs made from M5™ alloy barstock welded at each end. The design uses a fuel stack height of [ ]. The fuel pellets have a diameter of [ ]. The fuel pellets are a sintered, high density, ceramic. The fuel pellets are cylindrically shaped with a dish at each end. The corners of the pellet have and outward land taper and a chamfer which reduces the propensity for missing pellet surfaces. The pellet end configuration also reduces the tendency for the pellets to assume an hourglass shape during operation. The design density is 96.0% theoretical. Pellet enrichments may be as high as 5.0 w/o of ^{235}U.

The fuel rod cladding has a [ ] OD with a nominal [ ] wall thickness. This configuration leaves a small radial clearance of [ ] nominal between the ID of the cladding and the OD of the fuel pellets. The fuel cladding is M5™ alloy. M5™ cladding significantly increases resistance to corrosion and hydrogen uptake associated with longer cycles, high temperatures, and high burnup in comparison to early Zircaloy constructions. The Adv. W17 HTP fuel rod length and void volume provide adequate margin against failure due to pin internal pressure buildup.

The fuel rod uses a stainless steel spring in the upper plenum to prevent the formation of fuel stack axial gaps during shipping and handling, and which also allows fuel stack expansion during operation. The fuel column rests on a lower spring in the lower rod plenum. The lower plenum spring provides additional total internal rod volume, which results in overall lower pin pressures. The upper end cap has a grippable shape to interface with field service tooling for removal of fuel rods from the fuel assembly, if necessary. The lower end cap is made from M5™ and has a bullet-nose shape to provide a smooth flow transition. In addition, this shape facilitates reinsertion of fuel rods into the assembly if any rods are removed after the assembly has been irradiated (e. g. during fuel examination programs). The diameter of the tip of the Adv. W17 HTP fuel rod lower end cap is slightly less than the current Mark-BW fuel rod due to the interface with the FUELGUARD™ lower tie plate.

The Adv. W17 HTP fuel rod design can utilize axial blanket and gadolinia fuel configurations similar to the standard Mark-BW design. The axial blanket fuel stack contains three zones; a central portion enriched sintered UO_2 pellets and an axial blanket region at each end of the stack. The axial blanket region consists of sintered UO_2 pellets with a ^{235}U enrichment of a lower weight percent. The fuel pellet may also use gadolinium, which serves as a poison to control peaking. Table 2-2 compares the Adv. W17 HTP fuel rod key design attributes with the current Mark-BW resident fuel rod.

2.2.3 Spacer Grid Assemblies

The Adv. W17 HTP fuel assembly uses six Zircaloy-4 HTP flow mixing spacer grids at the intermediate locations and one HMP nickel alloy 718 at the bottom end location of the assembly. In addition to the HTP intermediate spacers, one HTP spacer is used at the top end of the fuel assembly. AREVA NP received NRC approval of HTP spacers constructed of Zircaloy-4 in the Generic Mechanical Design Thermal Performance Spacer and Intermediate Flow Mixer Report (Reference 1). The HTP spacer grid assembly is constructed of pairs of die formed strips that interlock when they are assembled to form the overall HTP structure. Each cross strip is formed by resistance spot welding two stamped halves (singlets) to form a subcomponent called a doublet. The assembled doublets form channels, slanted at the outlets, which induce a swirling pattern in the coolant flow as it passes through the HTP spacer. The channels are arranged so that there is no net torque on the fuel assembly. These channels also provide the integral springs and contact surfaces that hold the fuel rods in place. The channel strips are formed in the axial direction so that they provide a spring contact with the fuel rods in the mid-region of the spacer. At the inlet and outlet of the spacer, the channels (referred to as castellations) provide more rigid lateral constraint at a slight nominal clearance from the fuel rod. Side plates are welded to the ends of the doublets to form the outer envelope of the spacer. The side plates are provided with top and bottom lead-in tabs to avoid assembly hang-up during fuel movement.

In addition to the HTP intermediate spacers, one HMP spacer is used at the bottom end of the fuel assembly to provide additional support of the fuel rods. The HMP spacer is made of low cobalt precipitation-hardened nickel alloy 718 that provides additional strength and reduced cell relaxation due to irradiation. The HMP spacer maintains line contact on the fuel rod similar to the HTP spacer. The lower relaxation provides the fuel rod lateral support during operation for the design burnup range. The HMP spacer design is similar to the HTP spacer except the flow channels created by the doublets are straight. This minimizes the hydraulic resistance of the grid in locations outside of the active fuel region where flow mixing is not needed.

To establish axial alignment of spacers with adjacent fuel assemblies, the HTP spacers are spot welded to the guide tubes. This limits grid axial movement after irradiation relaxation of the spacers. Sleeves of Zircaloy-4 are spot welded to the guide tubes above and below the HMP spacers for axial location and restraint.

The Adv. W17 HTP fuel assembly also includes 3 Intermediate Flow Mixing spacers (IFM). The IFMs are located mid-span between the upper 4 intermediate HTP spacers. The IFMs provide additional flow mixing in the high-heat flux region for improved performance and DNB margin. The IFMs are rigidly attached to the guide thimbles at all 24 locations since they are non-contacting (i.e. no axial support of the fuel rod). The IFM attachment is a spot weld similar to the attachment of HTPs. The IFM spacer is constructed of pairs of die formed strips that interlock when they are assembled to form the overall IFM structure. Each cross strip is formed by resistance spot welding two stamped halves (singlets) to form a subcomponent called a doublet. The assembled doublets form channels, slanted at the outlets, which induce a swirling pattern in the coolant flow as it passes through the IFM spacer.. The IFM spacer is welded on the top side only.

To minimize the effect of the IFMs on bundle pressure drop and to limit the additional material added within the active fuel region, the height of the IFM spacer is less than the HTP intermediate spacers. The side plate design precludes hang-up or damage during handling due to its lead-in feature. A reduced spacer envelope eliminates mechanical interaction with adjacent fuel assemblies. Table 2-3 compares the key design attributes of the Adv. W17 HTP, HMP and IFM spacers with the Mark-BW vaned/vaneless and end grids.

2.2.4 Low Pressure Drop Top Nozzle

The Adv. W17 HTP fuel assembly design incorporates a low pressure drop top nozzle made of stainless steel. The low pressure drop feature is achieved by an optimization of flow path geometry with the nozzle structural integrity that accommodates each required normal and faulted load. The top nozzle design also incorporates a Quick Disconnect (QD) feature to attach the 24 fuel assembly guide tubes.

The primary features of the top nozzle include:

- Three leaf spring holddown system.
- Low pressure drop nozzle structure.
- QD guide tube attachment

The design consists of a double-spline sleeve made of Zircaloy-4 attached to the guide tube via multiple spot welds. The features in the top nozzle machining provide either clearance for removal, or restraint for securing the nozzle based on the orientation of QD features on the guide tube assemblies. The reconstitution tooling rotates the guide tube QD ring 90° to lock or unlock the sleeve splines and provide a positive lock when the ring rotation is complete.

The top nozzle assembly incorporates four sets of formed leaf springs made of nickel alloy 718 fastened to the nozzle with nickel alloy 718 clamp screws captured in the nozzle body. During operation, the springs prevent fuel assembly lift due to hydraulic forces, while accommodating irradiation growth and thermal expansion. The upper leaf contains an extension that engages a cutout in the top plate of the nozzle. This arrangement provides spring leaf retention in the unlikely event of a spring leaf or clamp screw failure.

The top nozzle structure consists of a stainless steel frame that provides interfaces with the reactor upper internals, the core components and fuel assembly handling tooling and equipment while providing coolant flow. The top nozzle flow-hole pattern enables an increased flow area, yielding a reduced pressure drop while satisfying the strength requirements for the top nozzle plate. The strength requirements of maximum primary membrane and membrane plus bending are met for shipping, normal operating, and faulted loading conditions according to the ASME code. Finite element analysis and tensile testing are both used for qualification of the top nozzle.

The pins in the upper core plate mate with the holes in the top nozzle. The diameter and the location of these holes are established to allow sufficient clearance with the upper core plate pins. The top nozzle has been evaluated with respect to compatibility with the fuel grappling for fuel movement. Table 2-4 compares the Adv. W17 HTP and Mark-BW top nozzle key design attributes. The Operating Experience for the design of top nozzle is extensive. The nozzle is very similar to current top nozzle used at Sequoyah Units 1 & 2. The primary changes relate to the machining configuration to accept the modular QD assembly. This exact nozzle has been supplied to Dominion for use at the North Anna plants.

2.2.5 Debris Filter (FUELGUARD™) Lower Tie Plate

The FUELGUARD™ lower tie plate provides a highly effective barrier to debris. The lower tie plate is stainless steel with a frame of deep ribs connecting the guide tube attachment bushings and conventional legs that interface with the lower reactor internals. The frame distributes the primary loads on the fuel assembly through the lower tie plate. The blade spacing enables good flow characteristics while providing enhanced debris filtering. The lower end of the guide tubes contain threaded features that provide rigid connection of the guide tubes to the lower tie plate with stainless steel bolts that incorporate a mechanical locking feature.

The FUELGUARD™ lower tie plate is an effective barrier to debris with acceptable pressure drop. The pressure drop performance is equivalent to conventional debris filter designs. The location and size of the holes which interface with the lower core plate are identical to the resident fuel assembly bottom nozzle. The Adv. Mark BW(A) lead assemblies inserted in SEQ1, Cycle 16 utilized the FUELGUARD™ lower tie plate. Therefore compatibility with the reactor internals has been demonstrated. Table 2-5 compares the Adv. W17 HTP LTP and the Mark-BW TRAPPER key design attributes.

2.2.6 MONOBLOC™ Guide Tube

The MONOBLOC™ guide tubes are fabricated from Zircaloy-4 alloy. The MONOBLOC™ guide tube as shown in Figure 2-10 has two inside diameters (ID) and a single outside diameter (OD). The larger ID at the top provides a relatively large annular clearance that permits rapid insertion of the rod cluster control assembly (RCCA) during a reactor trip and accommodates coolant flow during normal operation. The reduced ID section (i.e., the dashpot located at the lower end of the tube) provides a relatively close fit with the control rods to decelerate toward the end of the control rod travel. This deceleration limits the magnitude of the RCCA impact loads on the fuel assembly top nozzle. The guide tube wall thickness is much greater in the dashpot region than at the upper end of the tube to maintain the same OD with the smaller ID. This design provides a more rigid tube and thus a more robust structure that helps to reduce fuel assembly distortion and bow.

Four small holes in the guide tube located just above the dashpot allow both outflow of water during RCCA insertion, and coolant flow to control components during operation. There is also a small flow hole in the guide tube lower end fitting that enables flow through the reduced diameter section and flow venting during RCCA deceleration.

The modular QD is attached to the upper end of the guide tube for connection to the top nozzle. At the dashpot end of the guide tube assembly, a lower end fitting is welded. The lower end fitting is internally threaded for engagement with the guide tube cap screw that connects the guide tube to the lower tie plate.

The radial locations of the guide tubes within the assembly, the inner diameters of the guide tubes and the weep hole diameters were defined to be the same as the current resident Mark-BW fuel. The axial locations of the transition area and weep holes are similar to the resident fuel. These critical dimensions assure that control element assembly drop times and guide tube cooling are not affected by the introduction of the Adv. W17 HTP fuel assembly. A comparison of key design attributes for the guide tubes is presented in Table 2-6.

2.2.7 Instrument Tube

The Adv. W17 HTP fuel assembly design incorporates a single instrument tube (IT) fabricated from Zircaloy-4 located in the center lattice of the fuel assembly. The OD of the IT is the same as the MONOBLOC™ guide tube and is uniform over the entire length. The ID of the IT is the same as the ID of the MONOBLOC™ guide tube upper section and is uniform over the entire length. The IT provides the path for the movable incore monitoring instrumentation. A comparison of the key design attributes of the IT are noted in Table 2-6.

2.2.8 Materials

Table 2-7 summarizes the materials used in the Adv. W17 HTP fuel assembly design, identifying the alloys and the corresponding components. The specific use of M5™ for fuel rod cladding has been approved by the NRC per References 2 and 4. Low cobalt material requirements are imposed where applicable to reduce worker radiation exposure levels.

Table 2-1: Comparison of Adv. W17 HTP to Mark-BW and W17 Fuel Assembly Parameters

Fuel Assembly Parameter	Adv. W17 HTP	Mark-BW
Fuel assembly overall length, in.	[ ]	[ ]
FA matrix	17x17	17x17
Fuel rod overall length, in.	[ ]	[ ]
Fuel rod pitch, in.	0.496	0.496
Fuel rods/assembly	264	264
Guide tubes/assembly	24	24
Instrument tubes/assembly	1	1
Guide tube material	Zircaloy-4	M5™
Guide tube design	MONOBLOC™	Standard dashpot GT
Top nozzle	Lower pressure drop multi-leaf spring	Lower pressure drop multi-leaf spring
Top nozzle attachment	QD	Crimp
Lower tie plate/Bottom nozzle	FUELGUARD™	TRAPPER™ coarse mesh
End spacers	Lower most – HMP nickel alloy 718, uppermost – HTP zircaloy-4	2 monometallic nickel alloy 718
Intermediate spacer/ guide tube attachment	Spot welded to guide tubes	Swaged, deflection limiting ferrules with initial gap, 8 guide tube locations
Mid-span mixing spacers	3 intermediate flow mixers	N/A

Table 2-2: Comparison of Adv. W17 HTP and Mark-BW Fuel Rod Parameters

Fuel Rod Parameters	Adv. W17 HTP	Mark-BW
Clad material	M5™ Alloy	M5™ Alloy
Fuel rod length, in.	[ ]	[ ]
Fuel column length, in.	[ ]	[ ]
Rod internal plenum volume, in^3	[ ]	[ ]
Fill gas type	Helium	Helium
Fill gas pressure	[ ]	[ ]
Cladding OD, in.	[ ]	[ ]
Cladding thickness, in.	[ ]	[ ]
Cladding ID, in.	[ ]	[ ]
Clad-to-pellet gap, in.	[ ]	[ ]
Fuel pellet OD, in.	[ ]	[ ]
Plenum spring	Top & bottom	Top & bottom

Table 2-3: Comparison of Adv. W17 HTP and Mark-BW Grid Design Attributes

Grid Parameter	Adv. W17 HTP	Mark-BW
Intermediate Spacer		
Material	Fully annealed recrystallized low-tin Zircaloy-4	Fully annealed recrystallized low-tin Zircaloy-4
Mixing Vanes	N/A	Upper 5 intermediate grids
Outer Strip Height, in.	[ ]	[ ]
Outer Strip Thickness, in.	[ ]	[ ]
Inner Strip Height, in.	[ ]	[ ]
Inner Strip Thickness, in.	[ ]	[ ]
Grid Envelope, in.	[ ]	[ ]
End Spacer		
Grid Parameter	**Adv. W17 HTP**	**Mark-BW**
Material	Nickel alloy 718	Nickel alloy 718
Outer Strip Height, in.	[ ]	[ ]
Outer Strip Thickness, in.	[ ]	[ ]
Inner Strip Height, in.	[ ]	[ ]
Inner Strip Thickness, in.	[ ]	[ ]
Grid Envelope, in.	[ ]	[ ]
IFM/MSMG		
Grid Parameter	**Adv. W17 HTP (IFM)**	**Mark-BW**
Material	Fully annealed recrystallized low-tin Zircaloy-4	N/A
Location	Top 3 intermediate spacer spans	N/A
Outer Strip Height, in.	[ ]	N/A
Outer Strip Thickness, in.	[ ]	N/A
Inner Strip Height, in.	[ ]	N/A
Inner Strip Thickness, in.	[ ]	N/A
Grid Envelope, in.	[ ]	N/A

Table 2-4: Comparison of Adv. W17 HTP and Mark-BW Top Nozzle Design Attributes

Design Attribute	Adv. W17 HTP	Mark-BW
Envelope, in.	[ ]	[ ]
Upper core plate pin interface hole dia., in.	[ ]	[ ]
Height, in. (w/o leaf springs)	[ ]	[ ]
GT connection	¼ turn quick disconnect	Crimp upper GT lock nut
No. of leaf springs	Three	Three

Table 2-5: Comparison of Adv. W17 HTP and Mark-BW Lower Tie Plates

Design Attribute	Adv. W17 HTP	Mark-BW
Envelope, in.	[ ]	[ ]
Lower core plate hole diagonal, in.	[ ]	[ ]
Diameter of lower core plate pin hole, in.	[ ]	[ ]
Height, in.	[ ]	[ ]
Debris resistant feature	Curved blades	Coarse mesh filter plate

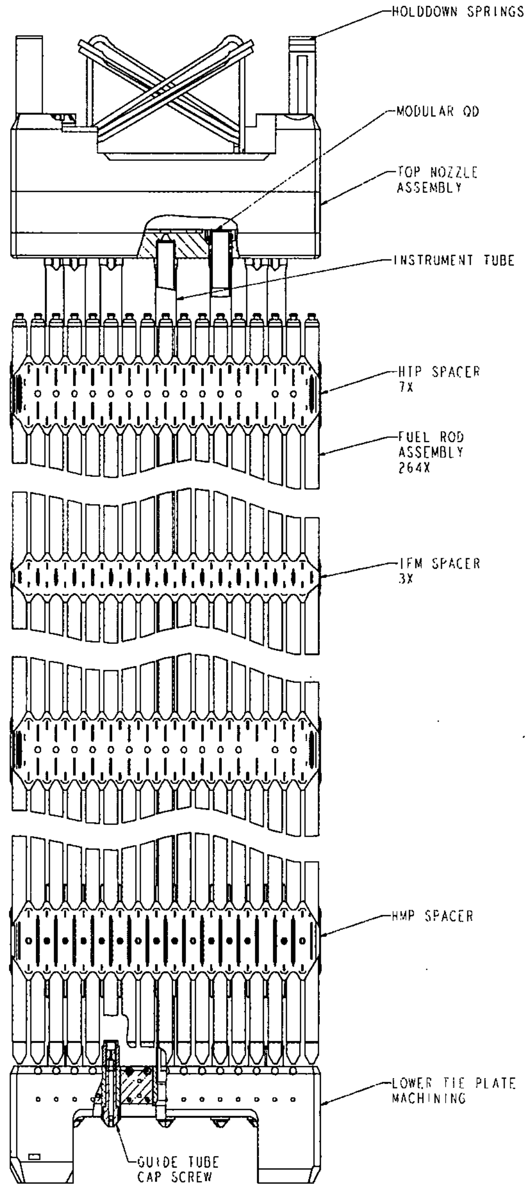
Table 2-6: Comparison of Adv. W17 HTP and Mark-BW Guide Tube and Instrument Tube Parameters

Guide Tube		
Parameter	**Adv. W17 HTP**	**Mark-BW**
Tube Material	Recrystallized Zircaloy-4	M5™
OD (top), in.	[ ]	[ ]
OD (bottom, dashpot), in.	[ ]	[ ]
ID (top), in.	[ ]	[ ]
ID (bottom, dashpot), in.	[ ]	[ ]
Wall thickness, in. Top Bottom	 [ ] [ ]	 [ ] [ ]
No. of weep holes	[ ]	[ ]
Weep hole dia., in.	[ ]	[ ]
Instrument Tube		
Parameter	**Adv. W17 HTP**	**Mark-BW**
Tube material	Recrystallized Zircaloy-4	M5™
OD, in.	[ ]	[ ]
ID, in.	[ ]	[ ]

Table 2-7: Summary of Adv. W17 HTP Component Materials

Alloy	Component
M5™	Fuel rod cladding Fuel rod end caps
Zircaloy-4	Guide tube lower end fitting Guide tubes/Instrument tube HTP spacers/IFM spacers QD retainer sleeve Spacer capture ring QD GT upper sleeve
Z2 CN 19-10	Top nozzle structure
CF3 stainless steel	Lower tie plate structure
Stainless steel	Lock wire
304L stainless steel	Guide tube cap screw LTP grid rods LTP curved blades LTP bushings
302 stainless steel	Fuel rod upper plenum spring
Alloy A286	Fuel rod lower plenum spring
Nickel alloy 718	HMP spacer QD locking ring QD locking lug
NC 19 Fe Nb alloy	Holddown spring clamp screws Holdown spring leaves
Nickel alloy X750	Guide tube locking spring
UO_2 and UO_2-Gd_2O_3	Fuel pellets

Figure 2-1: Adv. W17 HTP Fuel Bundle



Assembly

Figure 2-2: Adv. W17 HTP FUELGUARD™ Lower Tie Plate

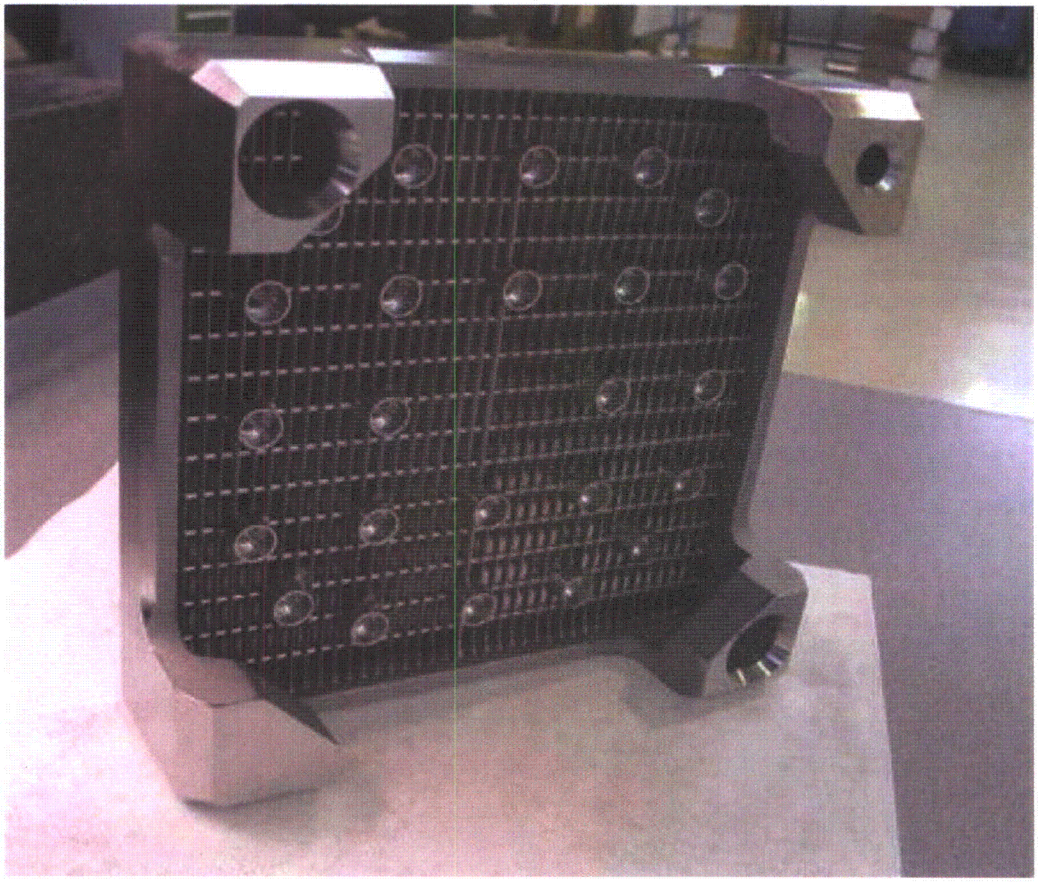


Figure 2-3: Adv. W17 HTP QD Top Nozzle Assembly

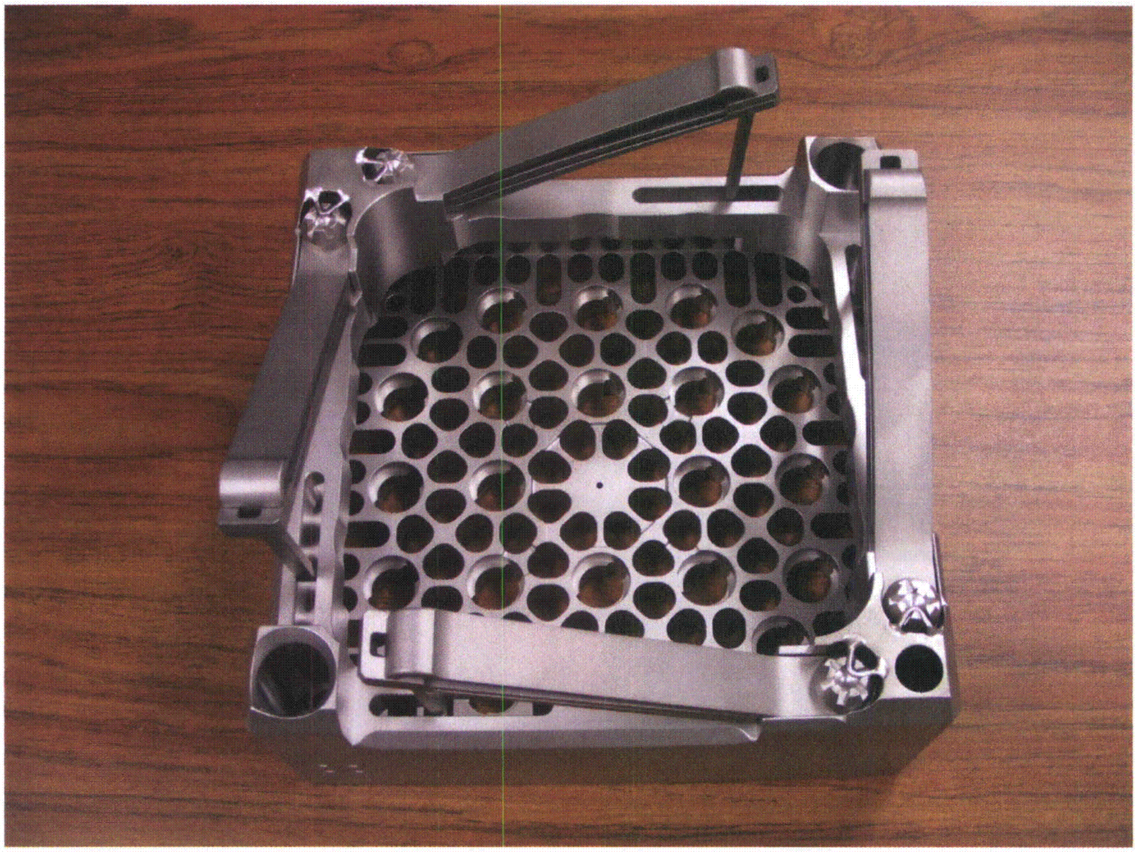


Figure 2-4: Adv. W17 HTP QD Modular Assembly

Figure 2-5: Advanced W17 HTP Cage Assembly

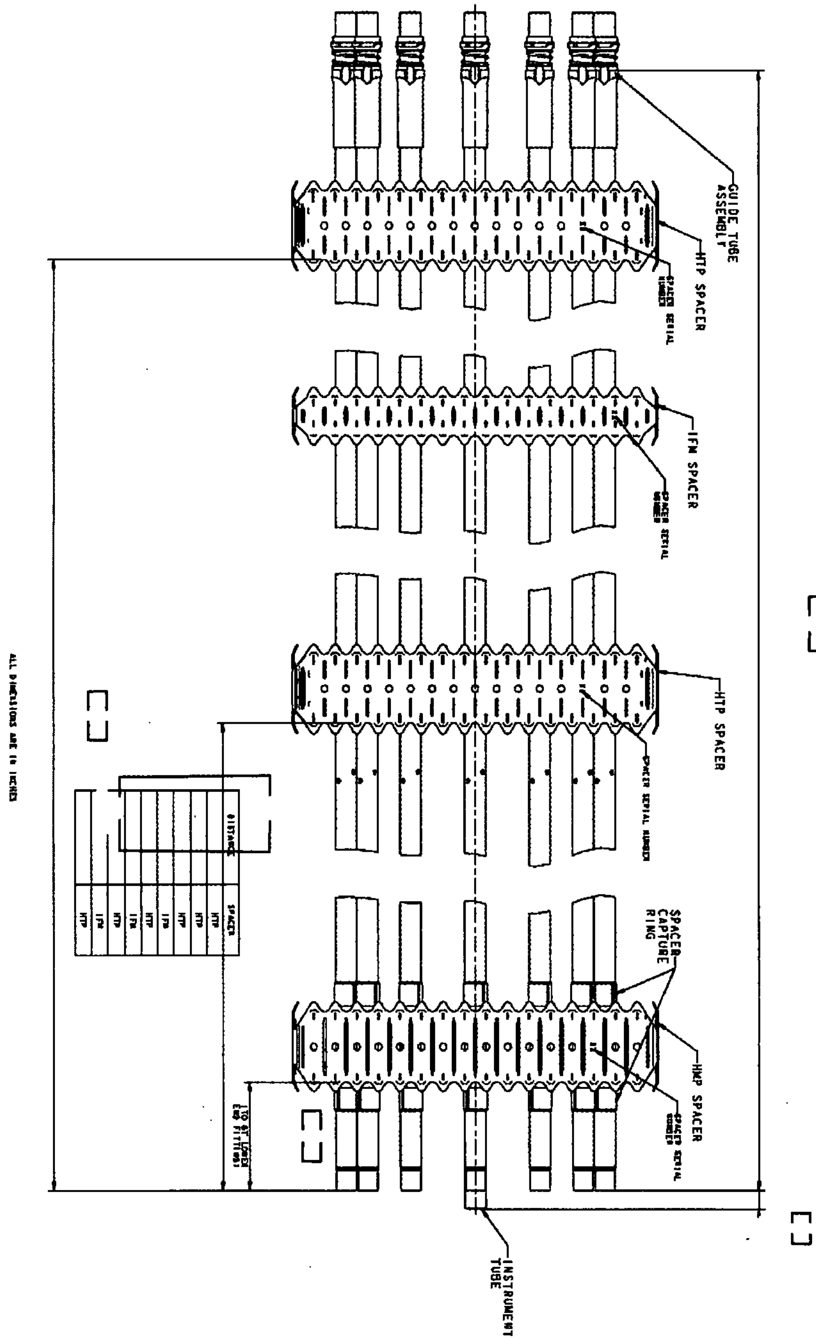


Figure 2-6: HTP Spacer Characteristics

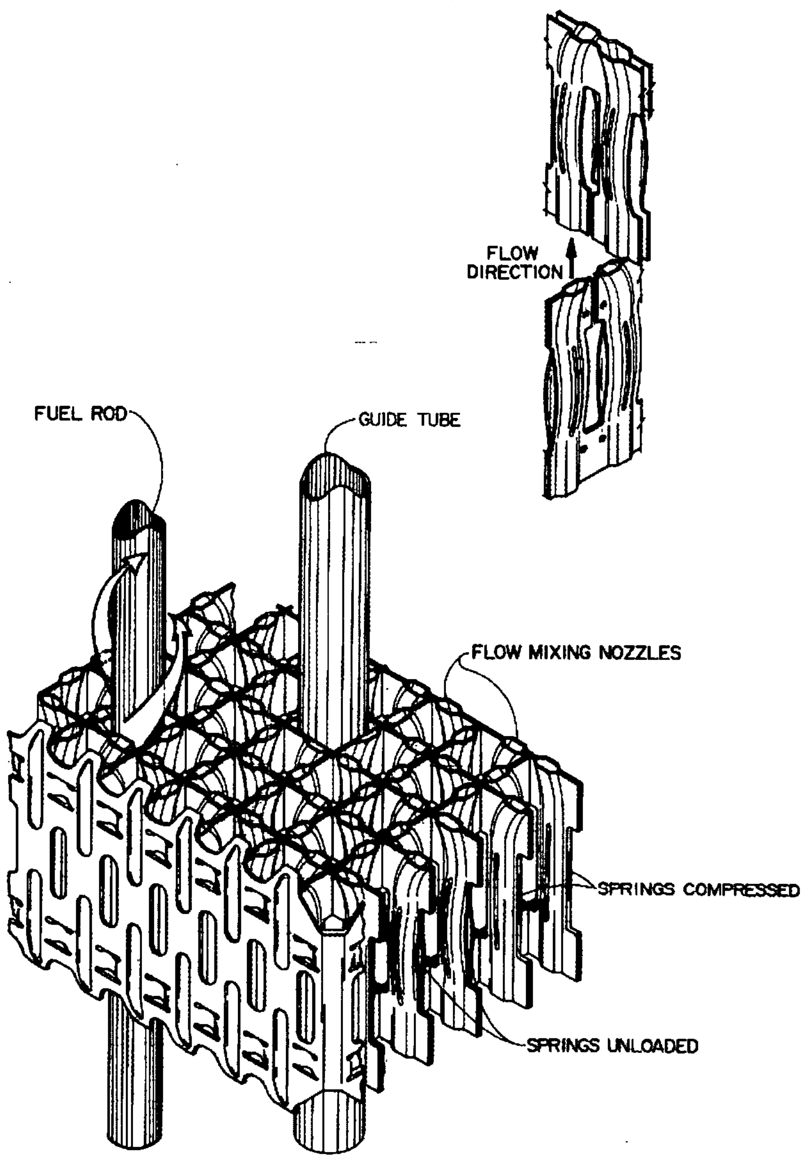


Figure 2-7: Adv. W17 HTP Spacer Grid Assembly

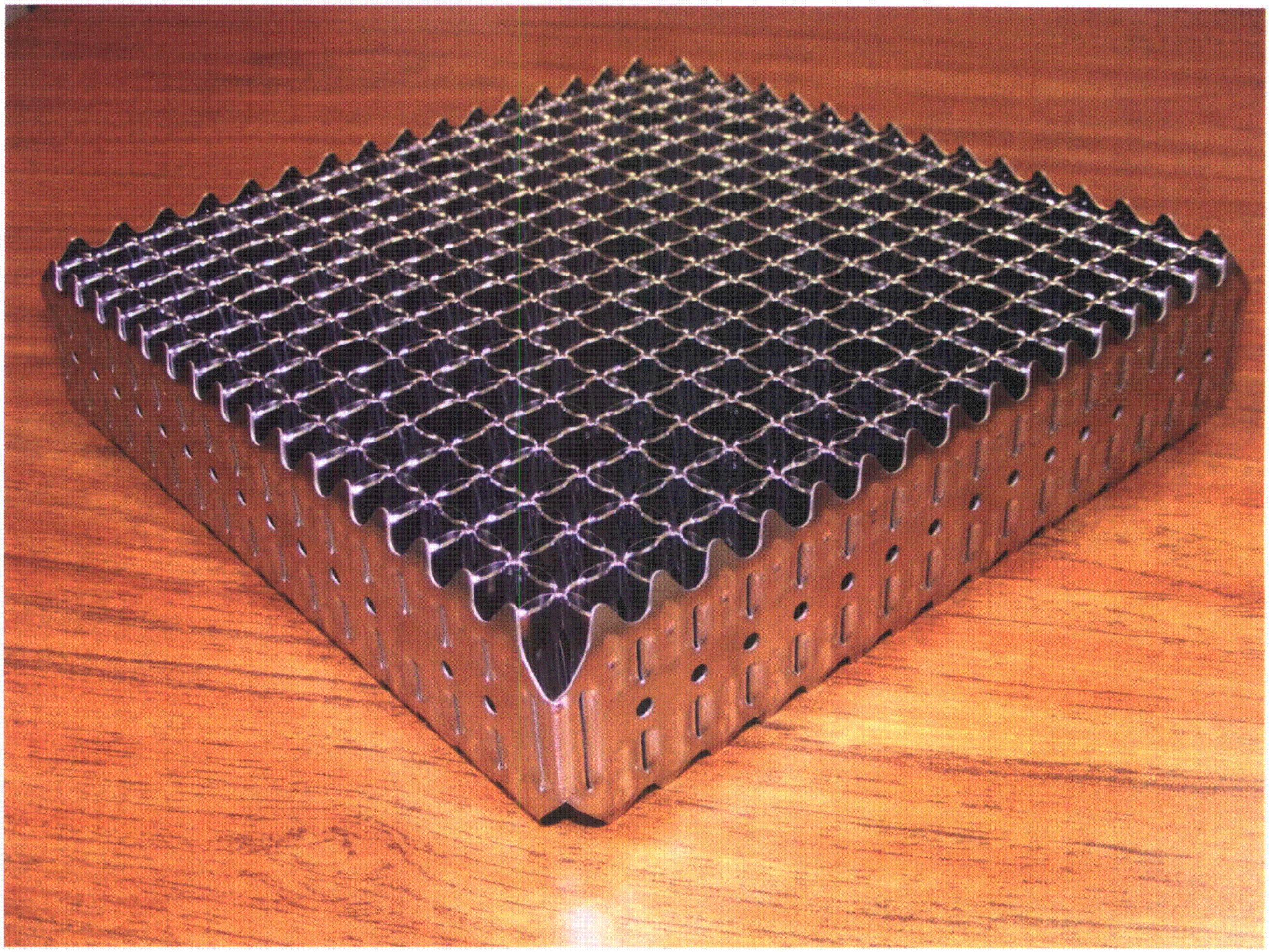


Figure 2-8: Adv. W17 HTP Intermediate Flow Mixer (IFM) Grid Assembly

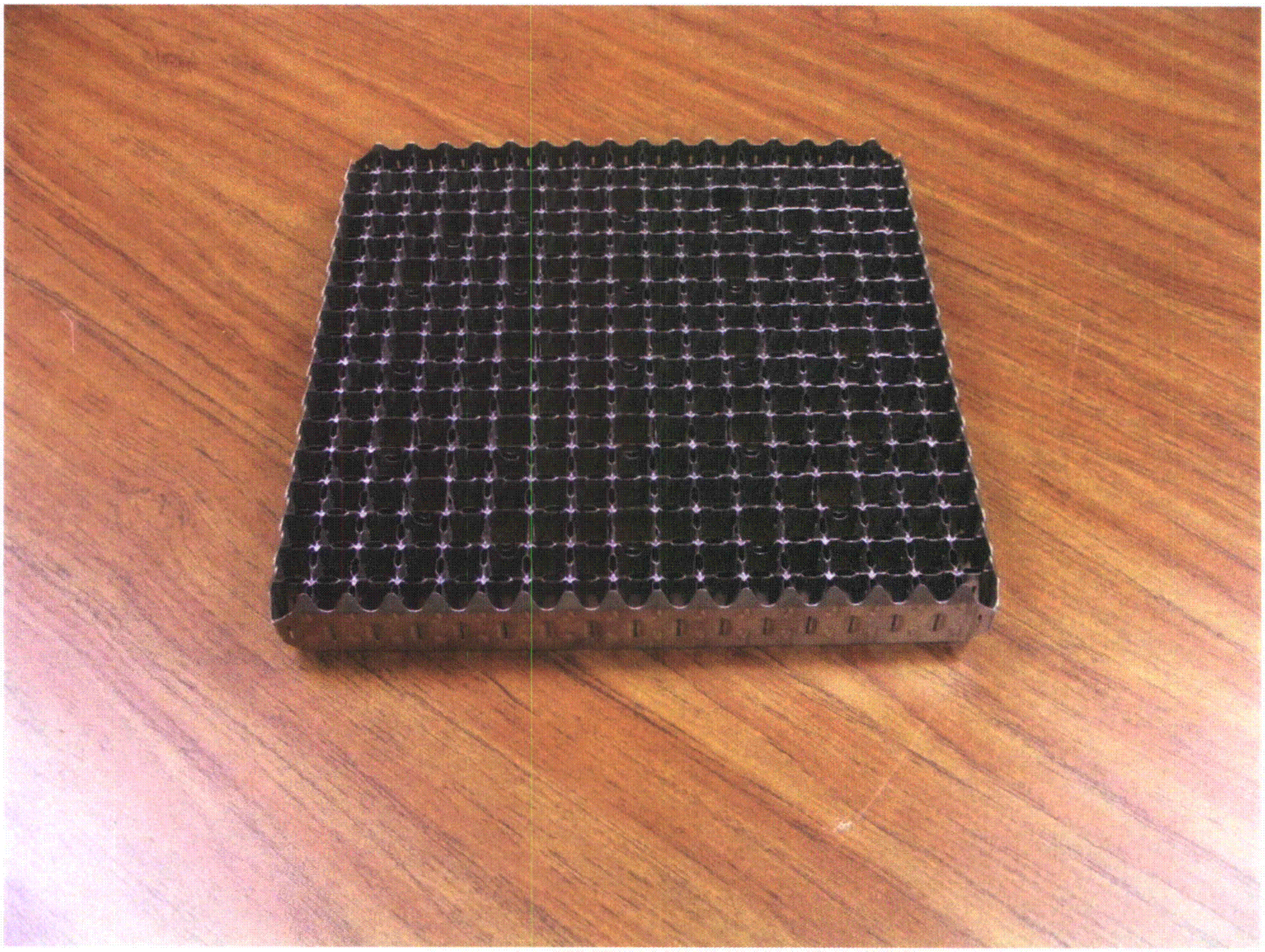


Figure 2-9: Adv. W17 HTP HMP Spacer Grid Assembly

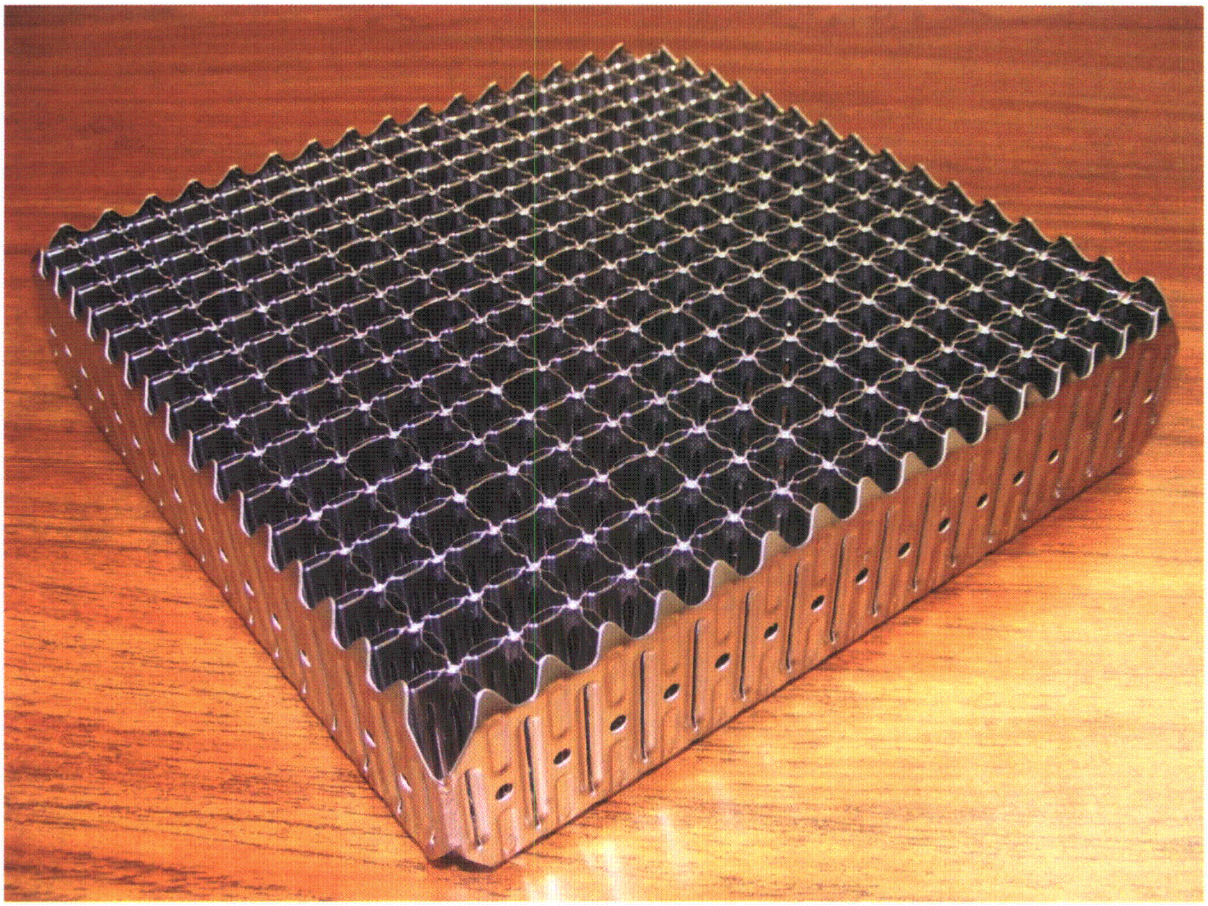


Figure 2-10: Adv. W17 HTP MONOBLOC™ Guide Tube

Figure 2-11: Adv. W17 HTP Instrument Tube

Figure 2-12: Adv. W17 HTP Fuel Rod Assembly

2.3 Mechanical Performance

The AREVA fuel design planned for introduction on a batch basis at Sequoyah is similar to the AREVA NP lead fuel assemblies that were introduced at Sequoyah Unit 1 in Cycle 16 (Reference 6) which are currently operating in their third cycle. The lead fuel assemblies were analyzed in accordance with the NRC-approved generic mechanical design criteria contained in BAW-10239PA (Reference 4). All the mechanical design criteria were shown to be met up to the licensed fuel rod burnup limit of 62 MWd/kgU (justified in Reference 8). The design improvements that are mentioned in Section 1.1 relative to the lead fuel assembly design do not have a large influence on the fuel assembly structural characteristics, such as axial and lateral stiffness, and seismic response.

As stated in section 1.1, the following features of the Adv. W17 HTP fuel assembly are embodied in the Adv. Mark-BW(A) LTAs, with reference to the performance of the resident Mark-BW fuel:

1. Welded cage: Improved lateral stiffness, seismic response as measured by free and forced vibration.

2. Nickel alloy lower HMP spacer grid: Improved rod contact, reduced fuel rod fretting potential.

3. MONOBLOC guide tubes: Improved cage rigidity.

4. FUELGUARD lower tie plate: Improved debris resistance.

The relevance of the lead test assembly program and operating experience to the licensing of the Adv. W17 HTP fuel design is based on the NRC design review criteria stated in Reference 5, Section 3.

Mechanical Prototype testing of the batch assemblies will be performed and the batch fuel design is anticipated to meet the applicable design requirements throughout the life of the fuel.

The NRC-approved generic design criteria used to assess the performance of the lead fuel assemblies were developed to satisfy certain objectives (Reference 4). These objectives are used for designing fuel assemblies so as to provide the following assurances:

- The fuel assembly (system) shall not fail as a result of normal operation and anticipated operational occurrences. The fuel assembly (system) dimensions shall be designed to remain within operational tolerances and the functional capabilities of the fuels shall be established to either meet, or exceed those assumed in the safety analysis.

- Fuel assembly (system) damage shall never prevent control rod insertion when it is required.

- The number of fuel rod failures shall be conservatively estimated for postulated accidents.

- Fuel coolability shall always be maintained.

- The mechanical design of fuel assemblies shall be compatible with co-resident fuel and the reactor core internals.

- Fuel assemblies shall be designed to withstand the loads from in-plant handling and shipping.

The generic criteria are applied to the fuel rod and fuel assembly designs. These criteria are listed in Table 2-8 along with the corresponding section number from Reference 4. As noted in the specific items, some of the criteria specified below are for analyses other than the mechanical design evaluations.

Table 2-8: Generic Mechanical Design Criteria

Criteria Section	Description	Criteria
5.1	**Fuel System Damage Criteria**	
5.1.1	Stress	Stress intensities for the fuel assembly components shall be less than the stress limits based on American Society of Mechanical Engineers (ASME) Code, Section III criteria (Reference 10).
5.1.1.1	Guide Thimble Buckling	Buckling of the guide thimbles shall not occur during normal operation (Condition I) or any other transient where control rod insertion is required. In addition, the primary and primary + secondary stresses shall be lower than the material allowable stresses (Reference 11).
5.1.1.2 and 5.1.1.3	Top and Bottom Nozzles, and Connections	The top and bottom nozzle and connections design criterion is the same as that given in Reference 11, which is based on the ASME Boiler and Pressure Vessel (B&PV) Code, Section III limits and meets the requirements of Section 4.2 of the SRP (Reference 5).
5.1.1.4	Spacer Grids	No grid crushing deformations occur for normal operation and Operational Base Earthquake (OBE) conditions. The grids shall also provide adequate support to maintain the fuel rods in a coolable configuration for all conditions (References 11 and 7).
5.1.1.5	Cladding Stress	Fuel rod cladding stress shall not exceed stress limits established in Reference 2 and are provided below: • Pm < 1.5 Sm in compression and < Sm in tension • Pm + Pb < 1.5 Sm • Pm + Pb + Pl < 1.5 Sm • Pm + Pb + Pl + Q < 3.0 Sm Pm=Primary Membrane, Pb=Bending, Pl=Local, Q=Secondary
5.1.2	Cladding Strain	The fuel rod transient strain limit is 1% for Conditions I and II events per Reference 2.
5.1.3	Cladding Fatigue	The maximum fuel rod fatigue usage factor is 0.9.
5.1.4	Fretting	Span average cross-flow velocities shall be less than 2 ft/sec. The fuel assembly design shall be shown to provide sufficient support to limit fuel rod vibration and clad fretting wear.
5.1.5	Oxidation, Hydriding, and Crud Buildup	The fuel rod cladding best-estimate corrosion shall not exceed 100 microns, per Reference 8. Hydrogen pickup is controlled by the corrosion limit.
5.1.6	Fuel Rod Bow	Fuel rod bowing is evaluated with respect to the mechanical and thermal-hydraulic performance of the fuel assembly. There is no specific design criterion for fuel rod bow.
5.1.7	Axial Growth	The fuel assembly-to-reactor internals gap allowance shall be designed to provide positive clearance during the assembly lifetime (Reference 11). The fuel assembly top nozzle-to-fuel rod gap allowance shall be designed to provide positive clearance during the assembly lifetime.

Criteria Section	Description	Criteria
5.1.8	Fuel Rod Internal Pressure	Fuel rod internal pressure limits are established in Reference 9. The design basis is that the fuel system will not be damaged due to excessive internal pressure. Fuel rod internal pressure is limited to that which would cause 1) the diametral gap to increase due to outward creep during steady-state operation and 2) extensive DNB propagation to occur.
5.1.9	Assembly Liftoff	The fuel assembly holddown springs must be capable of maintaining fuel assembly contact with the lower support plate during normal operating, Conditions I and II events, except for the pump overspeed transient. The fuel assembly shall not compress the holddown spring to solid height for any Conditions I and II event. The fuel assembly top and bottom nozzles shall maintain engagement with reactor internals for all Conditions I through IV events (Reference 11).
5.2	**Fuel Rod Failure Criteria**	
5.2.1	Internal Hydriding	Internal hydriding shall be precluded by appropriate manufacturing controls.
5.2.2	Cladding Collapse	The predicted creep collapse life of the fuel rod must exceed the maximum expected in-core life.
5.2.3	Overheating of Cladding	For a 95% probability at a 95% confidence level, DNB will not occur on a fuel rod during normal operation and anticipated operational occurrences (AOOs).
5.2.4	Overheating of Fuel Pellets	For a 95% probability at a 95% confidence level, fuel pellet centerline melting shall not occur during normal operation and AOOs.
5.2.5	Pellet / Cladding Interaction	Clad strain and fuel melt criteria are used to ensure that the fuel rod design is acceptable.
5.2.6	Cladding Rupture	Addressed in the plant-specific loss of coolant (LOCA) analyses.
5.3	**Fuel Coolability**	
5.3.1	Cladding Embrittlement	Requirements are to be addressed in plant-specific LOCA analyses.
5.3.2	Violent Expulsion of Fuel	Requirements are to be addressed in the plant-specific safety analyses.
5.3.3	Fuel Rod Ballooning	The requirements on fuel rod ballooning are addressed in the plant-specific LOCA analyses.
5.3.4	Fuel Assembly Structural Damage from External Forces	• **OBE** - Allow continued safe operation of the fuel assembly following an OBE event by ensuring the fuel assembly components do not violate their dimensional requirements. • **Safe Shutdown Earthquake (SSE)** - Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies, control rod insertibility, and a coolable geometry within the deformation limits consistent with the Emergency Core Cooling System (ECCS) and safety analysis. • **LOCA or SSE+LOCA** - Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies and a coolable geometry within deformation limits consistent with the ECCS and safety analysis.

AREVA NP intends to apply the generic mechanical design criteria contained in BAW-10239PA (Reference 4) and also listed in Table 2-8 to evaluate the design improvements to the lead fuel assembly design already operating at Sequoyah. AREVA NP will document the design evaluation process demonstrating compliance to the generic criteria and prepare a summary of the evaluation for possible use in an audit to confirm that AREVA NP is in compliance with these design criteria. Per the generic mechanical design criteria topical BAW-10239PA (Reference 4), this is an allowable approach for AREVA NP to proceed with changes or improvements to its existing PWR fuel designs without requiring prior NRC staff review and approval.

2.4 Fuel Rod Thermal-Mechanical Performance

The fuel rod design criteria that will be utilized for the evaluation of the Adv. W17 HTP fuel rod are listed in Table 2-8 of this report along with the rest of the mechanical design criteria. The fuel rod design for the Adv. W17 HTP fuel assembly is maintained the same as the current Mark-BW fuel rod design used for the Sequoyah units. However, the evaluation of the Adv. W17 HTP fuel rod will be performed using the modern NRC-approved COPERNIC fuel performance code (Reference 12) which includes degradation of fuel thermal conductivity with burnup. Therefore, use of the COPERNIC code will result in different transient cladding strain and centerline fuel melt limits relative to the values that support current plant operation for the Sequoyah units. Due to this reason, the cladding transient strain and centerline fuel melt limits have been generated for the Adv. W17 HTP fuel rod using a representative core design.

2.4.1 Cladding Transient Strain

The design criterion for cladding transient strain is that the total cladding strain shall not exceed 1% during Condition I and II transients. Adherence to this criterion is demonstrated by cycle specific calculation of LHGR limits that protect the 1% cladding strain criterion as a function of burnup and use of these limits in a maneuvering analysis to verify that positive margin is available at the core offset limits that provide LHGR protection in the Reactor Protection System (RPS) on a cycle-specific basis. AREVA has performed calculation of the cladding transient strain limits for the Adv. W17 HTP UO_2 and Gadolinia fuel rods for the Sequoyah units. The NRC-approved COPERNIC code along with its associated methodology (Reference 12) was utilized for this calculation.

2.4.2 Centerline Fuel Melt

The design criterion for centerline fuel melt is that the fuel pellet centerline temperature shall not exceed its melting point. Adherence to this criterion is demonstrated by cycle specific calculation of burnup dependent LHGR limits that provide fuel melt protection and use of these limits in a maneuvering analysis to verify that positive margin is available at the core offset limits that provide LHGR protection in the Reactor Protection System (RPS) on a cycle-specific basis. AREVA has performed calculation of the centerline fuel melt limits for the Adv. W17 HTP UO_2 and Gadolinia fuel rods for the Sequoyah units. The NRC-approved COPERNIC code along with its associated methodology (Reference 12) was utilized for this calculation.

2.4.3 Fuel Rod Bow

The design criterion for fuel rod bow is that the fuel rod bowing shall be evaluated with respect to the mechanical and thermal-hydraulic performance of the fuel assembly. Fuel rod bowing is not accounted for within fuel rod thermal-mechanical performance evaluations. Instead, the core protective and operating limits include a peaking uncertainty for capturing the effects of local power changes due to fuel rod bowing.

2.5 Operating Experience

Operational experience (OE) is an indispensable knowledge base to demonstrate the reliability and the performance of a fuel assembly design. The relevance of such OE increases all the more in the case of a design with technical features significantly different from all designs.

The HTP fuel assembly represents such a design. Whereas fuel assemblies equipped with traditional spacers employ springs and dimples to support each fuel rod in its spacer cell, and have mixing vanes along the top edges of the spacer strips which significantly enhance thermal hydraulic performance, the HTP spacer represents an entirely different concept in spacer design for pressurized water reactor (PWR) fuel. The HTP spacer features strip doublets which are shaped such that they not only serve as spring elements to firmly hold the fuel rods in radial alignment but also produce curved internal flow channels to achieve the desired thermal hydraulic performance.

HTP is primarily the designation of a special type of spacer but is also used to denote a fuel assembly design in which this type of spacer is the major component. The first insertion was into a U.S. plant in 1988; the HTP design now possesses over 20 years of operational experience.

The Adv. W17 HTP fuel assembly design for Sequoyah is an HTP-type fuel assembly design with M5 fuel rod cladding, HMP lowermost spacer grid, FUELGUARD lower tie plate, and MONOBLOC guide tubes. An overview of both the overall operating experience gained with the various components of the fuel assembly design as well as the specific operating experience in Westinghouse-17 plants in this section.

2.5.1 Operating Experience with HTP Fuel Assemblies

As of December 2009, the operational experience with HTP fuel assemblies (FA) comprises a total of 11,710 fuel assemblies irradiated in 47 nuclear power plants (NPP). From these, 7,215 are in 27 European plants (Belgium, France, Germany, Spain, Sweden, Switzerland, UK, The Netherlands), 4,355 assemblies in 17 U.S. plants, 80 assemblies in 2 Japanese plants and 60 assemblies in a Brazilian plant.

This experience spans the entire range of fuel rod arrays from 14x14 to 18x18, as well as reactors supplied by various vendors, such as Combustion Engineering (CE), Framatome, Westinghouse, Siemens and Babcock & Wilcox (B&W). The largest share, 4,765 FAs has been loaded into 12 ft Framatome/Westinghouse plants with a 17x17 array, followed by the 16x16 array for Siemens plants with 1,516 assemblies. Table 2-9 provides an overview.

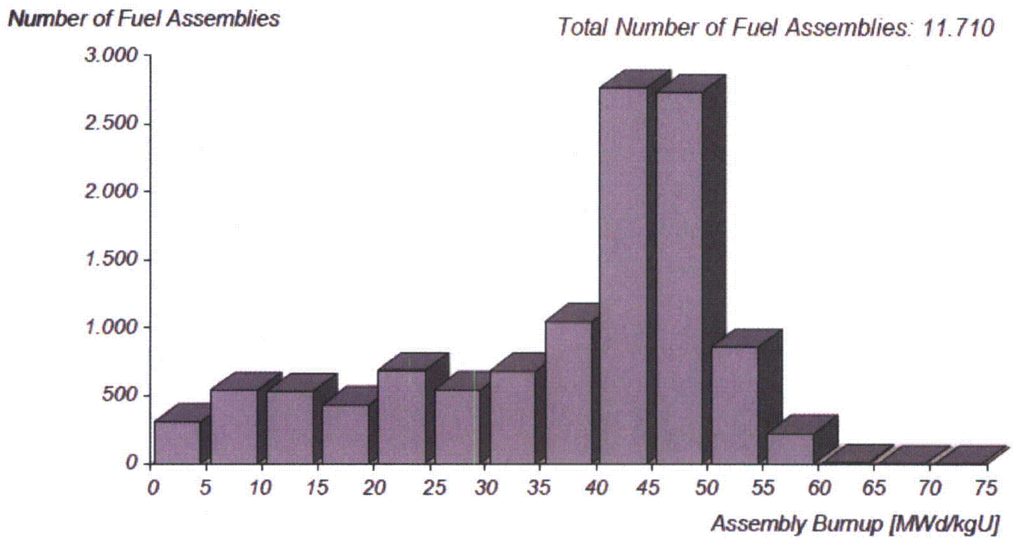
As of December 2009, more than 5,400 HTP FAs equipped with Gadolinia rods have been loaded worldwide into 29 NPPs. The number of Gadolinia rods within an assembly varied between 4 and 28 with Gd_2O_3 concentrations from 2 up to 8 w/o. 15x15 and 17x17 HTP FAs with configurations ranging from 4 Gadolinia rods of 2 w/o to 24 Gadolinia rods of 8 w/o have been prepared for Westinghouse type plants. A maximum fuel assembly average burnup of 67 MWd/kgU has been achieved with HTP assemblies containing Gadolinium poisoned rods.

Table 2-9: Operational Experience with HTP (Status December 2009)

Plant type	# of plants	First Insertion	# of FAs in operation	# of FAs accumulated	Maximum FA burn-up [ ]	# of defective rods (accumulated)
CE-14x14	5	1988	556	1,163	60	12
CE-15x15	1	1988	204	784	53	14
CE-16x16	1	2008	8	8	8	0
Westinghouse-14x14	3	1994	221	837	54	0
Westinghouse-15x15	1	1991	157	702	58	1
Westinghouse 17x17, 12ft	6	1994	726	1,971	57	4
Framatome 17x17, 12ft	8	1993	467	2,794	67	8
B&W- 15x15	7	2003	774	839	50	0
Siemens-15x15	3	2001	357	448	70	2
Siemens-16x16	9	1989	1,047	1,516	59	5
Siemens-18x18	3	1992	468	648	61	1
Total	**47**		**4,985**	**11.710**	**70**	**47**

With 6,593 fuel assemblies, more than half of all inserted HTP FAs have achieved a burnup of higher than 40 MWd/kgU. The maximum assembly burnup is 70 MWd/kgU. The burnup distribution of the HTP fuel assemblies as of December 2009 is shown in Figure 2-13. The extent of OE of welded-cage HTP fuel assembly designs of varying configurations, which include the 17x17 fuel pin array similar to the Adv. W17 HTP design that for Sequoyah, provides assurance that the design is suitable for batch implementation.

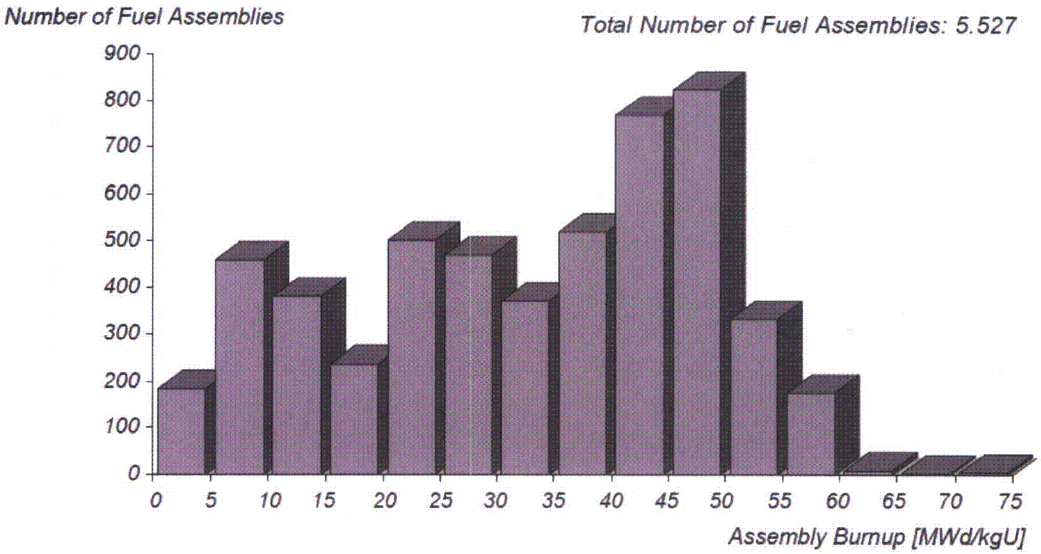
Figure 2-13: Burnup Distribution of the HTP FA (Status December 2009)



2.5.2 HTP Fuel Assemblies Equipped with an HMP Spacer at Lowermost Position

The first insertion of the HTP fuel design with High Mechanical Performance (HMP) Alloy 718 grids (straight flow channels) at the lower grid position was in 1998. Today, significant operational experience with the HTP FA featuring an HMP spacer is available. Altogether, 5,527 such HTP FAs have been loaded worldwide into 33 plants. Figure 2-14 shows the burnup distribution of HTP FAs featuring an HMP at the lowermost position as of December 2009. A maximum assembly burnup of 70 MWd/kgU has been achieved. The above OE representing HTP fuel assembly designs with a lower HMP grid, together with the irradiation of the four Adv. Mark-BW(A) lead test assemblies in Sequoyah Unit 1 which have an HMP grid, demonstrates the acceptability of the HMP spacer grid for use at Sequoyah.

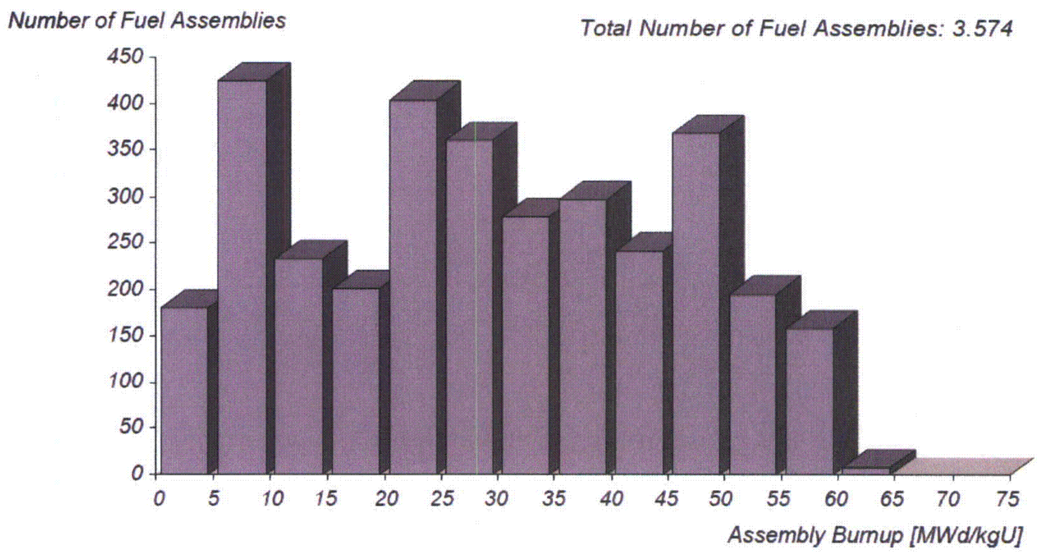
Figure 2-14: Burnup Distribution of FA Featuring an HMP at Lowermost Position (Status December 2009)



2.5.3 HTP Fuel Assemblies with M5 Cladding

The first HTP fuel assemblies equipped with M5 fuel rod cladding were inserted into four plants in 2003 - four LTAs into a South American plant, four LTAs into a US plant (Ft. Calhoun), a reload consisting of 36 assemblies into a German plant with a 16x16 array, and one reload with 85 assemblies into a US plant of a 15x15 B&W design (Crystal River 3). As of December 2009, 3,574 HTP fuel assemblies with M5 cladding have been irradiated in 28 plants in Brazil, Germany, the Netherlands, Sweden, Switzerland, South-America, the UK and in the US. The operational experience of the combination HTP fuel assembly and M5 cladding covers all arrays from 14x14 up to 18x18. Up to now, a maximum assembly average burnup of 61 MWd/kgU has been achieved. Figure 2-15 shows the burnup distribution of HTP fuel assemblies equipped with M5 cladding material as of December 2009. The above summary of OE showing compatibility of M5 cladding for use in HTP fuel assembly designs demonstrated acceptability of M5 cladding for the Adv. W17 HTP fuel assembly design.

Figure 2-15: Burnup Distribution of HTP FA having Fuel Rods with M5 Cladding Material (Status December 2009)



2.5.4 Overall Operating Experience with M5 Cladding

The M5 alloy is the reference alloy of AREVA NP for fuel rod cladding material. M5 is the result of a vast program of optimization and industrial development which started at the end of the 1980's and reached completion at the beginning of this millennium.

Since 1993, more than three million fuel rods having M5 cladding have completed their operation or are operating in 12,528 fuel assemblies in 79 commercial reactors worldwide. These include 53 reactors in Europe (Belgium, France, Germany, Netherlands, Spain, Sweden, Switzerland and UK), 17 in the US, 6 in China, 2 in South-Africa and 1 in Brazil (Table 2-10).

The irradiation experience covers all fuel assembly arrays ranging from 14x14 to 18x18, and different fuel assembly designs as AFA3G, HTP, Mark-B and Mark-BW. It includes enriched natural uranium and enriched reprocessed uranium fuel, both with and without Gadolinium. The range of enrichment extends at present from 3.2 to 4.95 w/o U235. Mixed Oxide fuels are also included, particularly in Germany and in France.

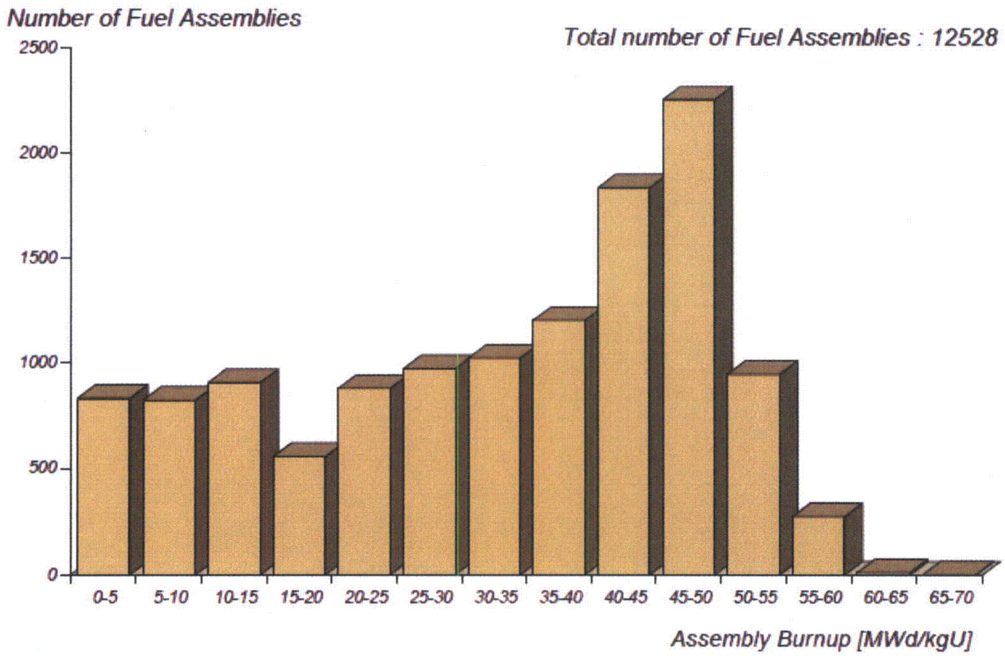
Table 2-10: Operational Experience with M5 Cladding Material (Status December 2009)

Status 12/2009	Fuel Array	Number of Reactors	First Irradiation	Number of FAs	Maximum F/R Burnup (MWd/kgU)	Maximum FA Burnup (MWd/kgU)
Belgium	14x14	1	1993	2	54	49
	15x15	1	1998	476	55	50
	17x17	3	2000	436	59	53
Brazil	16x16	1	2003	60	49	44
China	17x17	6	1999	1704	54	49
France 900MWe	17x17	18	1993	378	80	57
France 1300MWe	17x17	8	1997	905	65	59
France N4	17x17	4	2005	964	51	46
Germany	15x15	1	2004	200	65	59
	16x16	7	1993	1497	65	59
	18x18	3	1993	611	67	61
Netherlands	15x15	1	2004	144	59	54
South Africa	17x17	2	2002	416	63	57
Spain	17x17	1	1999	4	51	46
Sweden	15x15	1	2000	232	67	61
	17x17	2	1998	506	64	58
Switzerland	15x15	1	2005	5	64	58
UK	17x17	1	2008	168	31	28
USA	14x14	2	2003	128	67	45
	15x15	8	1995	2205	68	56
	16x16	1	2008	8	-	-
	17x17	6	1997	1479	72	68
TOTAL		**79**		**12,528**		

Figure 2-16 shows the fuel assembly burnup distribution with status as of December 2009. More than half of the assemblies have achieved burnups in excess of 30 MWd/kgU, while 40 percent have achieved burnups in excess of 40 MWd/kgU. Thus far, the maximum fuel assembly average burnup achieved is 68 MWd/kgU while the maximum fuel rod burnup achieved is 80 MWd/kgU.

M5 cladding material has been successfully irradiated at Sequoyah in batch quantities since 2001 will continue to be used for the Adv. W17 HTP fuel assembly.

Figure 2-16: Burnup Distribution of AREVA NP FA Featuring M5 Fuel Rod Cladding Material (Status December 2009)



2.5.5 Operating Experience with FUELGUARD Lower Tie Plate

Table 2-11 summarizes the total number of fuel assemblies in the U.S. using the Robust FUELGUARD (FG) as an anti-debris filter, capturing significant debris, thereby reducing the potential for fretting failures. First introduced in 1993 in the U.S. at Robinson Unit 2, the FUELGUARD debris filter design has now been used at sixteen U.S. plants in batch quantities, and at another six U.S. plants as lead test assemblies. Over four-thousand FG anti-debris filters have been delivered to date in the U.S. as shown in Table 2-11 below. Worldwide, 11,745 PWR fuel assemblies have been irradiated with the FG anti-debris filter. Such OE is the basis for continued use of the FUELGUARD lower tie plate for the Adv. W17 HTP fuel assembly design.

Table 2-11: US Operational Experience with FUELGUARD Lower Tie Plate (Status Nov 2010)

U.S. Power Plant	Array	# HTP FAs
Kewaunee	W14	172
Robinson 2	W15	662
Comanche Peak 1	W17	173
Comanche Peak 2		266
Shearon Harris 1		756
Braidwood 1		8
Sequoyah 1		4
Palisades	CE15	536
Millstone 2	CE14	420
St. Lucie 1		508
Ft. Calhoun		305
Calvert Cliffs 1		2
Calvert Cliffs 2		2
Palo Verde 1	CE16	8
SONGS		8
ANO1	B&W15	237
Crystal River 3		242
Davis Besse		228
Oconee 1		60
Oconee 2		136
Oconee 3		132
TMI1		161
Total		**5026**

2.5.6 Operating Experience with MONOBLOC Guide Tubes

The MONOBLOC guide tube represents a new design feature for Sequoyah, incorporating a solid tube design that features a constant outer diameter for the full length of the guide tube, and two inner diameters. Worldwide as of December 2010, 22,623 fuel assemblies have been irradiated with MONOBLOC™ guide tubes made from Zircaloy-4 material, and an additional 3,209 fuel assemblies from M5 material. The MONOBLOC™ tube design has also been utilized for guide tubes in multiple lead assembly programs in the U.S. (four at Sequoyah unit 1) and is used for instrument tubes in all seven Babcock & Wilcox plants in the U.S. Implementation of the MONOBLOC™ guide tube at Sequoyah will be the first batch application for Westinghouse 17x17 fuel in the U.S. This extensive OE, and the specific use of MONOBLOC™ guide tubes with the Adv. Mark-BW(A) lead test assemblies in Sequoyah Unit 1 demonstrates acceptable performance at Sequoyah

2.5.7 Overall HTP Fuel Reliability

Over the time period of more than 20 years, during which altogether approximately 2.7 million fuel rods among the 11,710 HTP fuel assemblies irradiated worldwide, a total of 64 fuel rod failures have been reported. The defective fuel rods were found in 50 separate fuel assemblies in 18 different plants. Table 2-12 summarizes the fuel rod failures associated with HTP fuel designs.

Of the 64 defective fuel rods, fretting accounts for 26 of the total number of fuel rod failures. The most prevalent case involves a CE 14x14 design in the U.S. which accounts for over half of the total fretting failures. The CE 14x14 design incorporated an all-HTP spacer grid bundle design, and the failure investigation concluded that irradiation relaxation of the zirconium alloy grids, coupled with baffle flow interaction, led to spinning fuel rods that ultimately led to through-wall failures. The addition of the Inconel HMP spacer grid at the bottom grid location eliminated this failure type, due to improved resistance of the Inconel material to irradiation relaxation. To date there have been no fuel rod failures attributed to Grid-to-Rod-Fretting (GTRF) associated with HTP bundle designs that feature a lower HMP grid design.

Another 8 fuel rods failed due to fretting with the bi-metallic end grid. The so-called bi-met grids feature a dimpled spring design. The eight-way line contact of the HTP and lower HMP spacer grid has shown significant improvement to resistance to GTRF.

The remaining 6 fretting failures are attributed to corner-to-corner contact between adjacent fuel assemblies. The failure mechanism was determined to be a result of excessive fuel assembly bow, where the corner of the spacers were worn through prior to fuel rod-to-fuel rod contact.

Table 2-12: Fuel Rod Reliability with HTP Designs

Plant Type	F/R Fretting	Baffle Interaction	Debris	Handling	Contamination	PCI	Unknown
B&W15	-	-	1	-	1	-	-
CE15	5 (bimet)	1	7	-	3	-	-
W15	-	-	-	-	1	-	-
W17	-	-	1	-	-	-	3
FRA17	3 (bimet)	-	1	-	1	-	3
CE14	12 (spinning rods)	-	-	-	-	-	-
Siemens15	-	-	1	-	-	7	-
Siemens16	6 (corner- to-corner)	-	1	2	-	-	2
Siemens18	-	-	2	-	-	-	-
Totals	**26**	**1**	**14**	**2**	**6**	**7**	**8**

Of the 14 debris failures, one occurred in a fuel assembly without a debris filter. Seven other failures were in a CE15x15 plant that utilizes control blade between bundles. The plant is unique for a PWR in that it utilizes control blades and has wide gaps between fuel assemblies, through which debris can pass easily. The remaining six debris failures were found in the peripheral rods beneath grids in designs with lower debris filters, further suggesting the debris bypassed the debris filters.

Five of the six contamination failures are attributed to internal hydrogenous contamination. All of these rods were manufactured at the same manufacturing facility within a specific time frame spanning 27 months. Causal analysis investigations, process reviews, and internal audits revealed weaknesses in cleanliness and contamination control. Corrective/preventative actions and best practices including an improved FME (foreign material exclusion) program and training were implemented, and no failures with similar characteristics have occurred since then. One fuel rod has been determined to be failed due to contamination of a seal weld at the end of the upper end cap. The seal weld became contaminated during a re-weld process.

There have been seven HTP failures that have been classified at Pellet Clad Interaction at Missing Pellet Surfaces. These failures have occurred at one Siemens 15x15 plant for three cycles in a row. This non-classical PCI failure mechanism is characterized by failures occurring during power maneuvers at startup. The mechanism involves cracks which initiate on the inside of the cladding at locations where pellet chipping has resulted in a large enough missing pellet surface to cause a stress riser at the unsupported cladding. This mechanism has been verified by hot cell examinations. Tighter pellet acceptance standards and improved inspection methods have been implemented at the manufacturing facility along with more conservative maneuvering limits. The affected plant started their last cycle without failures for the first time in four cycles. Recent improvements in manufacturing for AREVA fuel including a tighter pellet chip specification, improved pellet design and manufacturing process, and an improved inspection process. Zero PCI failures have occurred in AREVA fuel (PWR or BWR) built since 2004 when these improvements (with the exception of the improved pellet design) were implemented.

The single baffle interaction failure occurred after the spacer grid adjacent to the baffle plate worn away which allowed direct contact of the fuel rod with the baffle wall.

Of the eight unknown failures, one FA had reached its targeted burnup and was discharged/reprocessed; other attempts to extract defective rod(s) resulted in additional rod damage, rendering further examination not feasible, whereas other more recent failed rods have yet to be evaluated.

Design features of the Adv. W17 HTP fuel assembly, such as the lower HMP spacer grid and FUELGUARD lower tie plate, eliminates the majority of identified causes of fuel rods failures associated with fuel rod fretting at bi-met grid locations and spinning rods as shown in Table 2-12. AREVA's design control and fuel reliability program continues to evaluate all fuel rod failure mechanisms to eliminate such failures from reactor operation.

2.6 SER Restrictions and Limitations

1. BAW-10231(P)(A), Revision 1, "COPERNIC Fuel Rod Design Computer Code"

    <u>Purpose</u>:
    License the COPERNIC fuel performance code for fuel rod design and analysis of natural, slightly enriched (up to 5 percent) uranium dioxide fuels and urania-gadolinia fuels with the Advanced cladding material M5.

SER Restrictions:

- o Valid for up to 5% enriched uranium dioxide fuel
- o Approved only for M5 cladding
- o Approved for the fuel rod designs considered within the topical (includes the Mark-BW type fuel rod also used in the Adv. W17 HTP design)
- o Valid up to a UO2 fuel rod average burnup of 62 GWd/mtU
- o Valid for up to 8 wt% Gd_2O_3
- o Valid up to a maximum rod power of up to 80 kW/m
- o Valid up to a UO2-Gd2O3 fuel rod average burnup of 55 GWd/mtU

2. BAW-10239P-A, Revision 0, *Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report*, July 2004.

- This fuel assembly design is approved for use with low enrichment uranium (LEU) fuel, which has been enriched to less than or equal to 5 percent.
- The Adv. Mark-BW fuel assembly design is licensed for a maximum fuel rod burnup of 62,000 Megawatt-days/metric ton (MWD/MT).

2.7 References for Section 2.0

1. ANF-89-060PA and Supplement 1, *Generic Mechanical Design Report High Thermal Performance Spacer and Intermediate Flow Mixer*, February 1991
2. BAW-10227PA, Revision 1, *Evaluation of Advanced Cladding and Structural Material (M5™) in PWR Reactor Fuel*, June 2003
3. deleted
4. BAW-10239P-A, Revision 0, *Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report*, July 2004.
5. *Standard Review Plan*, Section 4.2, NUREG-0800 Revision 2, U.S. Nuclear Regulatory Commission, July 1981.
6. ANP-2666-001, *Sequoyah Unit 1 Cycle 16 Reload Safety Evaluation Report*, October 2007.
7. BAW-10133P-A Revision 1 Addendum 1, *Mark-C Fuel Assembly LOCA-Seismic Analyses*, October 2000.
8. BAW-10186P-A Revision 2 (Includes Revision 1, Supplement 1), *Extended Burnup Evaluation*, June 2003.
9. BAW-10183P-A Revision 0, *Fuel Rod Gas Pressure Criterion (FRGPC)*, July 1995.
10. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Nuclear Power Plant Components, 1992 Edition.

11. BAW-10172P-A Revision 0, *Mark-BW Mechanical Design Report*, December 1989.

12. BAW-10231PA Revision 01, “COPERNIC Fuel Rod Design Computer Code,” January 2004.

3.0 NEUTRONICS

3.1 Introduction and Summary

The effects of transitioning from AREVA Mark-BW fuel to Adv. W17 HTP fuel on the nuclear design bases and the methodologies for the Sequoyah Nuclear Station Units 1 and 2 are evaluated in this section. The design change to the Adv. W17 HTP assembly has little direct effect on the neutronics model since the fuel rod design is unchanged. There is a small effect from the grid design change. The increased mass (volume) of the HTP grids is captured in the cross-section model. The grid effect on peaking due to a higher grid flux depression will be on the order of <0.5% and is not a significant change. As discussed in Section 4, the thermal performance of the Adv. W17 HTP design is lower than that of the Mark-BW design. In addition, during the transition cycles mixed core effects will further impact thermal margin. This is typically accommodated by adding more feed assemblies to lower the peaking. As a result of these changes, the specific values of core safety parameters, such as power distributions, peaking factors, reactivity coefficients and critical boron concentrations, are primarily loading-pattern dependent. The variations in the loading-pattern dependent safety parameters are expected to be typical of normal cycle-to-cycle variations that occur as a result of variations in cycle length and thus feed enrichment in a standard reload core design. The same methodology and codes currently in place will be used to model the transition cores. The standard AREVA NP codes and methodologies (References 1, 2, 3, and 4), accurately predict the neutronics behavior of the resident Mark-BW fuel and Adv. W17 HTP fuel during the transition effort. AREVA fuel designs with HTP grids have significant nuclear design and operating experience in the AREVA 17x17 fleet, including the Harris plant in the USA and in Japan. Further discussion of operation experience is provided in Section 2.5.

The transition to Adv. W17 HTP fuel from the current Mark-BW design will occur over three reload cycles. Representative reload cycles of 88 feed the first cycle, 85 feed the second cycle, and 81 feed the third cycle were evaluated. The current Sequoyah Units 1 & 2 cores employ 81 and 85 feed assemblies respectively. The cores modeled for the transition to all Adv. W17 HTP fuel vary from the typical reload pattern only in the first transition cycle model, and then only slightly. The higher feed batches in the two transition cycles are intended to accommodate transition peaking penalties that apply for any significant change in fuel design. Note that the data presented here is representative. Actual reload core designs will be addressed in the standard reload licensing process.

3.2 Neutronics Acceptance Criteria

The objective of the nuclear design of the reactor is to ensure that fuel design limits will not be exceeded during normal operation or anticipated operational transients and the effects of reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core and to assure conformance with the requirements of General Design Criteria (GDC).

The following GDC apply to the transition to Adv. W17 HTP fuel described in this section:

- GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences (AOOs).
- GDC 11 requires that, in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.
- GDC 12 requires that power oscillations that could result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- GDC 28 requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to impair significantly the capability to cool the core.

To meet the GDC requirements the following Acceptance Criteria are established (Reference 3, 4, 5, and 6):

1. Power distributions (Axial Flux Difference, Heat Flux Hot Channel Factor, and Nuclear Enthalpy Rise Hot Channel Factor) shall be in accordance with the plant Technical Specifications/Core Operating Limits Report [ ] (GDC 10).
2. Doppler Coefficients shall be negative at all operating conditions (GDC 11).
3. Power Coefficient shall be negative at all operating power levels relative to hot zero power (GDC 11).
4. Moderator Temperature Coefficient shall be in accordance with the plant specific Technical Specifications /COLR (GDC 11).
5. The fuel design and loading shall be such that uncharacteristic power oscillations due to fuel design and loading do not occur (GDC 12).
6. Margin to the Technical Specification/COLR value for minimum shutdown margin, with an allowance for a stuck most reactive rod (MRR), shall be maintained throughout the cycle (GDC 28).

3.3 Methodology

The submittal core designs were developed to provide verification that selected key safety parameters currently in place for Mark-BW fuel (see Table 3-1) would be unaffected by new fuel cycles designs with Adv. W17 HTP. The selection is based upon those parameters that have proven challenging in past fuel designs. These safety parameters are from the analyses of record for the reload specific analyses in order to assure that these cycle specific core designs are bounded by the current key parameters. Reanalysis would be required in the event the current key parameters are exceeded. These designs also provide assurance that the plant licensing basis in the Technical Specifications, COLR and Updated Final Safety Analysis Report (UFSAR) are met for the anticipated operation of the Adv. W17 HTP fuel during transition and future cycles.

The nuclear design methodology and codes will continue to apply to the standard AREVA methodology and code package for the transition and future operation of AREVA fuel. References 1, 3, and 4, are the NRC approved topical reports outlining the approved AREVA neutronics methodology and codes. With respect to the neutronics model of Adv. W 17 HTP, the above SER constraints are met for Sequoyah Nuclear Station Units 1 & 2 transition to Adv. W17 HTP fuel.

- Fuel or core designs with significant differences that might be introduced must be further validated.
- The validation will be maintained by AREVA NP and be available for NRC audit.

The above SER constraints have been met for Sequoyah Nuclear Station Units 1 & 2 transition to Adv. W17 HTP fuel.

Benchmarking of the AREVA neutronics methodology and codes has been performed and demonstrated acceptable for Sequoyah Nuclear Station Units 1 & 2 for the last six cycles of operation at each unit, including startup testing. These confirm accurate predictions by the AREVA code package. AREVA predicts critical boron concentrations based on raw code predictions with an additional boron bias based on the difference between raw code predictions and core follow data from the previous cycles.

Key parameters will be calculated as part of the submittal neutronics analysis. These parameters will then be biased in the safety analysis in order to create an analysis of record for the reload cycles. Key neutronics parameters will then be calculated for the cycle specific reload and compared with the values used in the analysis of record by safety analysis. If the key parameters are not within the analysis of record, then the transient will be re-analyzed or re-evaluated on a cycle-to-cycle basis using the stated methods. The results will be reported in the Safety Analysis Report (SAR) for that cycle.

The thermal hydraulic limits resulting from the analysis described in section 4 are verified as being met using the maneuvering analysis described in reference BAW-10163P-A. The maneuvering analysis demonstrates the existence of adequate margin between these DNB related limits and the predicted power distribution. The analysis evaluates predicted power distribution dependence on core loading, reactor core thermal power level, fuel burnup, control rod insertion, and xenon spatial distribution. The analysis compares simulated power distributions, including those power distributions possible during normal operation and anticipated operational occurrences to power peaking or linear heat rate limits (SAFDLs) based upon criteria related to both the core safety limits (centerline fuel melt, steady-state DNB, transient cladding strain) and accident initial conditions (LOCA and initial-condition DNB). Based upon the analysis, peaking margins to each peaking limit are calculated and evaluated with respect to the appropriate Reactor Trip System LSSS and LCO limits. When peaking margins are calculated, the simulated peaking factors are augmented to account for uncertainties such as the nuclear reliability factor, local engineering hot channel factor, and other uncertainties, and further augmented to accommodate effects that are real but are not explicitly modeled (such as the effect of the variation of the axial power shape due to the presence of spacer grids). The results of the peaking margin calculations are used to either validate the appropriate limits specified in the Core Operating Limits Report (COLR) or to update them for the reload cycle.

The exposure dependency of the core power distributions is determined by explicitly simulating limiting power distributions at several burnup points during the steady-state depletion of the fuel. Typically, eight to thirteen times in life are evaluated for this purpose. At each of the these burnups, limiting power distributions are generated with simulated xenon transients and control rod re-positioning to determine the variation of peaking margin with fuel burnup.

Operation at power levels intermediate to HZP and HFP is accommodated by explicit simulation of core power distributions at several intermediate thermal power levels between design overpower and 50% of rated thermal power at each of the selected burnup steps.

Control rod positions starting at ARO and ending at an insertion deeper than the Rod Insertion Limit are simulated during generation of the RTS LSSS limits. Therefore, the simulated power distributions reflect more severe axial and radial peaking factors that could occur during rod withdrawal, boron dilution, or overcooling accidents. Other accident specific checks that are performed are discussed below.

3.4 Nuclear Design Evaluation

Two transition core designs and an additional follow on core design have been developed for Sequoyah Unit 1 to model the transition to Adv. W17 HTP fuel.

The loading patterns were developed based on projected cycle energy requirements for Sequoyah Units 1 & 2. The loading patterns have incorporated the current rated power of 3455 MWt. These cycles were developed to be representative of future cycle designs to demonstrate acceptable margins. Figure 3-3 below may also be considered representative of the current Sequoyah loading patterns. The first transition cycle contains fresh Adv. W17 HTP fuel with once-burnt and twice-burnt Mark-BW fuel. The second transition cycle contains fresh and once-burnt Adv. W17 HTP fuel with twice-burnt Mark-BW fuel. The third transition cycle contains only Adv. W17 HTP fuel. These cycles were developed to be representative of future cycle designs.

Key parameters were verified for the submittal core design in Table 3-1. These are discussed in section 3.3.

Figure 3-1 shows a possible 1st transition core with 88 Adv. W17 HTP feed assemblies and 105 reinsert Mark-BW assemblies. This core has a higher than usual feed batch with the intention of maximizing the population of Adv. W17 HTP assemblies in the transition cores.

Figure 3-2 shows a possible 2nd transition core with 173 Adv. W17 HTP assemblies and 20 reinsert Mark-BW assemblies. This core pattern is similar to that currently employed in Sequoyah Unit 2 Cycle 17.

Figure 3-3 shows a possible first All Adv. W17 HTP. With 81 feed, this pattern is similar to that currently employed in the Sequoyah Unit 1 Cycle 18.

Figure 3-4 shows the critical boron concentrations for the HTP cycles. The current BOC HFP Equilibrium Xenon boron concentration target (at 4 EFPD) is <1200 ppmb. All the cores modeled here meet this goal. Note that the very flat boron letdown for the 2nd transition cycle is not a concern since the maximum concentration occurs at 4 EFPD. Should the maximum occur after BOC, then additional safety parameter checks are performed to assure BOC analyses remain bounding.

Figure 3-5 shows the full power axial offset for the HTP cycles. All of these cycles are well within the current operating envelope, Typically +7% to -13% at HFP.

Figure 3-6 and Figure 3-7 show the radial peaking (FΔH) and local peaking (FQ) for the HTP cycles. For FΔH, the peaking is well below the normal target of <1.50, and is meant to address the anticipated reduction in thermal margin discussed in Section 4. For FQ, these values are also a bit lower than current cycles.

Figure 3-8 through 3-10 show the BOC and EOC assembly burnup maps for the HTP cycles. The higher feed batches give a lower EOC burnup, and are well within the maximum licensed rod burnup of 62 GWd/mtU.

Figure 3-11 through 3-19 show the BOC, MOC, and EOC power distribution maps for the HTP cycles. Consistent with Figures 3-6 and 3-7 the peak powers are similar to, and particularly for the feed assemblies, somewhat lower than for current cores.

The standard AREVA methods of fresh fuel enrichment loading and integrated burnable poisons will be applied to control the peaking and maintain compliance with the Technical Specifications and COLR. Changes in boron concentration and axial offset are typical of normal cycle-to-cycle variations in the core design.

3.5 BLEU Fuel

The compatibility of Commercial Grade Uranium Fuel (CGU) with Blended Low Enriched Uranium fuel (BLEU) in a transition core environment was shown in report ANP-2692P, "BATCH IMPLEMENTATION OF BLENDED LOW ENRICHED URANIUM FUEL AT SEQUOYAH NUCLEAR PLANT". The transition to

Adv. W17 HTP fuel with CGU is compatible with the Mark-BW assemblies containing BLEU fuel in Sequoyah Unit 2.

3.6 Conclusions

The nuclear core design analysis of the submittal core design for the transition from AREVA Mark-BW fuel to AREVA Adv. W17 HTP fuel has confirmed peaking factor and key safety parameters can be maintained within their specified limits using AREVA methodologies and codes. The key safety parameters generated with the submittal core design were used in the applicable analyses and evaluated to meet the acceptance criteria.

Table 3-1: Key Parameters

Coefficient	Power (%)	Burnup	Design value	Limit
Reactivity Coefficients				
MTC (pcm/°F)	0	BOC	-0.86	<0
MTC (pcm/°F)	100	BOC	-11.78	<0
MTC (pcm/°F)	100	EOC	-36.01	>-45
Power Doppler (pcm/%)	100	BOC	-9.26	>-12.5
Power Doppler (pcm/%)	100	EOC	-7.12	<-6.5
Ejected Rod Parameters				
Coefficient	Power (%)	Burnup	Design value	Limit
Rod Worth (pcm)	0	BOC	511	≤750
FQ	0	BOC	8.74	≤14.05
Rod Worth (pcm)	0	EOC	763	≤910
FQ	0	EOC	17.86	≤24.8
Rod Worth (pcm)	100	EOC	28	≤210
FQ	100	EOC	1.96	≤7.88
Safety Analysis Parameters				
Coefficient	Power (%)	Burnup	Design value	Limit
Shutdown Margin (pcm)	0	EOC	2574	>1600
β_{eff}	100 & 0	BOC	0.0065 (HFP) 0.0064 (HZP)	$0.0044 < \beta_{eff} < 0.0075$
β_{eff}	100	EOC	0.0053	$\beta_{eff} > 0.0044$

Note:

The reload limit remains at 0 pcm/deg-F. Some of the accident analyses, i.e. in Section 5.2.2.24 were performed at +7 in order to maximize core response. AREVA's reload licensing documents will continue to specify 0 pcm/deg-F as the reload limit.

Figure 3-1: 1st Transition HTP Quarter Core Loading Pattern

	H	G	F	E	D	C	B	A
8	20A H08 0 /	20D B08 180 /	21D F	20A G13 270 /	21B F	20A C09 180 /	21G F	20C F10 180 /
9	20D B08 270 /	21C F	20A C11 180 /	21D F	20B D08 90 /	21E F	21G F	20B E09 90 /
10	21D F	20A E13 180 /	21C F	20E D10 270 /	21D F	20F C12 270 /	20F B10 90 /	19G B09 270 /
11	20A G13 0 /	21D F	20E F12 90 /	21C F	20G B11 0 /	21F F	21H F	19D2 F11 90 /
12	21B F	20B G09 270 /	21D F	20G E14 0 /	21A F	20A E11 270 /	21J F	
13	20A C09 270 /	21E F	20F D13 90 /	21F F	20B F08 90 /	21I F	19E2 D11 90 /	
14	21G F	21G F	20F F14 270 /	21H F	21J F	19E2 E12 90 /		
15	20C F10 270 /	20B G11 270 /	19G G14 90 /	19D2 E10 180 /	Batch ID Previous Cycle Location Degrees clockwise rotation/Prev. Cycle Number			

Note: Batch 21 is HTP fuel.

Figure 3-2: 2nd Transition HTP Quarter Core Loading Pattern

	H	G	F	E	D	C	B	A
8	22B F	21D G11 180 /	22B F	21D E09 270 /	22B F	21G B08 180 /	22D F	19D3 G08 180 / 18
9	21D G11 270 /	22A F	21F C11 0 /	22E F	21E C09 90 /	22E F	21J B12 180 /	21C E11 90 /
10	22B F	21F E13 0 /	22B F	21A D12 270 /	22A F	21G B09 0 /	22G F	21B D08 0 /
11	21D E09 0 /	22E F	21C G09 90 /	22E F	21D F08 180 /	22E F	22F F	19A2 F09 270 / 18
12	22B F	21E G13 270 /	22A F	21C F10 180 /	21I C13 0 /	22C F	21H B11 180 /	
13	21G B08 270 /	22E F	21G G14 0 /	22E F	22C F	22H F	19C2 D09 90 / 18	
14	22D F	21J D14 180 /	22G F	22F F	21H E14 180 /	19C2 G12 90 / 18		
15	19D3 G08 270 / 18	21D F12 270 /	21D D10 0 /	19A2 G10 180 / 18	Batch ID Previous Cycle Location Degrees clockwise rotation/Prev. Cycle Number			

Note: Batches 21 & 22 are HTP fuel.

Figure 3-3: All-HTP Quarter Core Loading Pattern

	H	G	F	E	D	C	B	A
8	23B F	22B2 F10 270 /	23C F	22A G09 0 /	23D F	22D B08 0 /	23F F	21D2 F15 90 /
9	22B2 F10 0 /	23B F	22A D10 180 /	23C F	22E E11 90 /	22G B10 270 /	23F F	22B2 D08 180 /
10	23C F	22A F12 180 /	23C F	22C C12 270 /	23D F	22F B11 270 /	23E F	21H B12 90 /
11	22A G09 90 /	23C F	22C D13 90 /	22H C13 0 /	22E E09 270 /	23A F	23G F	21J B09 0 /
12	23D F	22E C09 0 /	23D F	22E G11 270 /	22E G13 90 /	23F F	22E C11 180 /	
13	22D B08 90 /	22G F14 90 /	22F E14 90 /	23A F	23F F	23H F	21C2 A09 180 /	
14	23F F	23F F	23E F	23G F	22E E13 180 /	21D2 G15 180 /		
15	21D2 F15 180 /	22B2 F08 0 /	21H D14 270 /	21J G14 0 /	Batch ID Previous Cycle Location Degrees clockwise rotation/Prev. Cycle Number			

Note: All batches are HTP fuel.

Figure 3-4: Critical Boron Concentrations for the HTP Transition Cycles

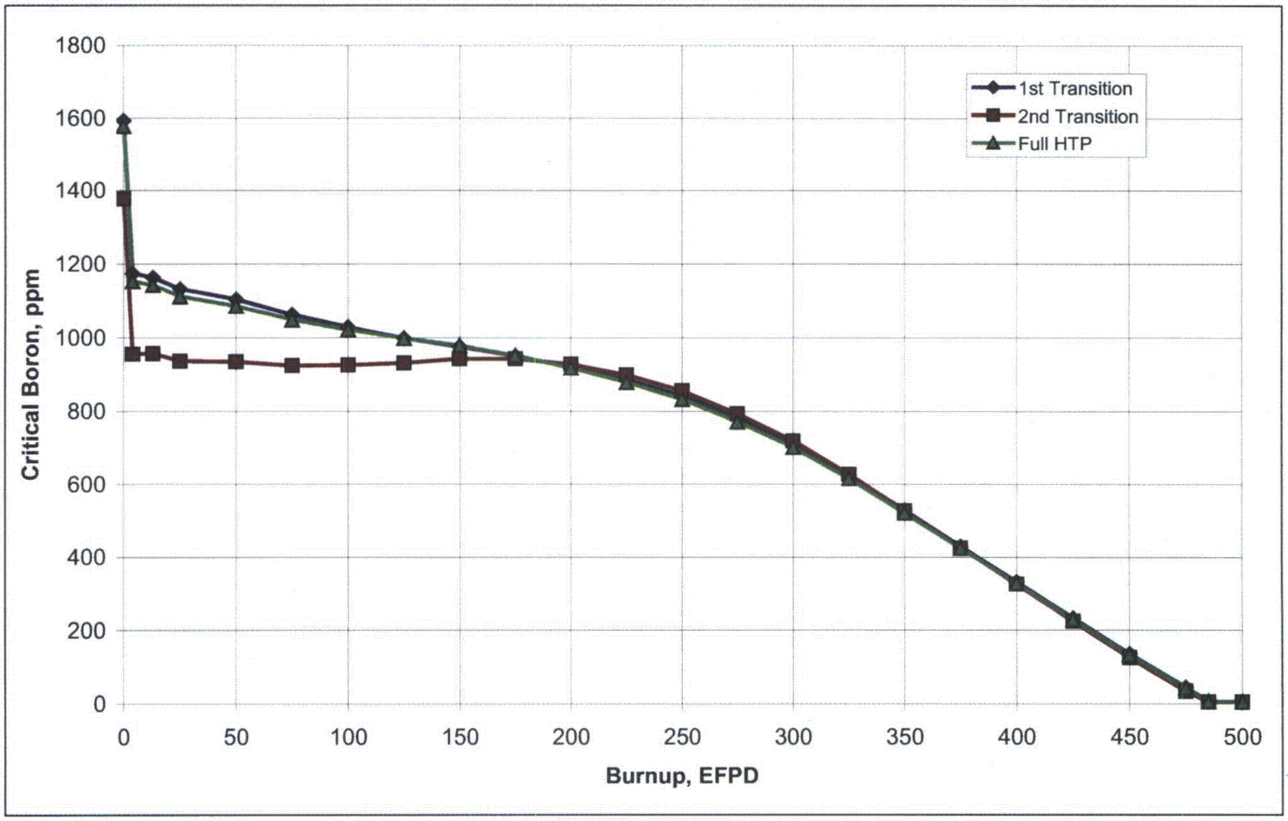


Figure 3-5: Full Power Axial Offset for the HTP Transition Cycles

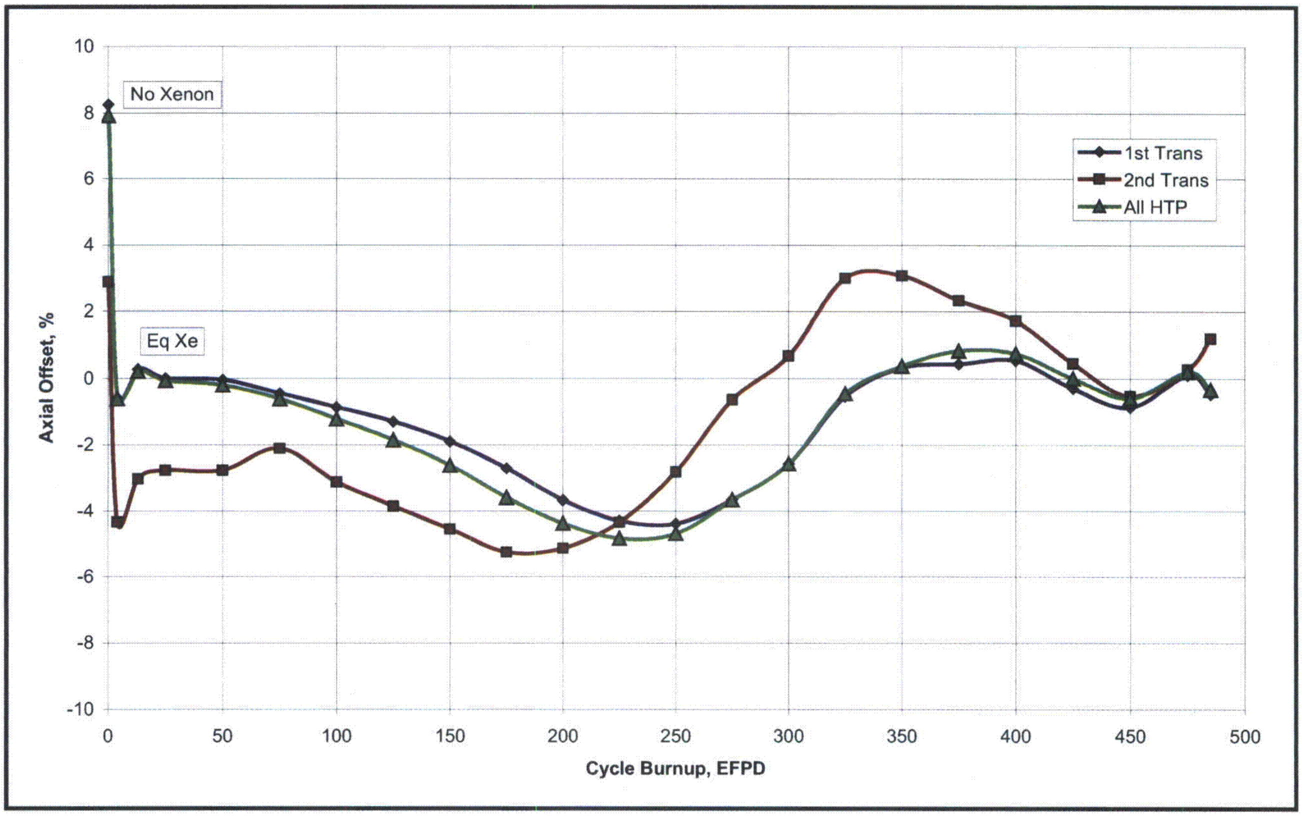


Figure 3-6: $F_{\Delta H}$ Comparisons for the HTP Transition Cycles

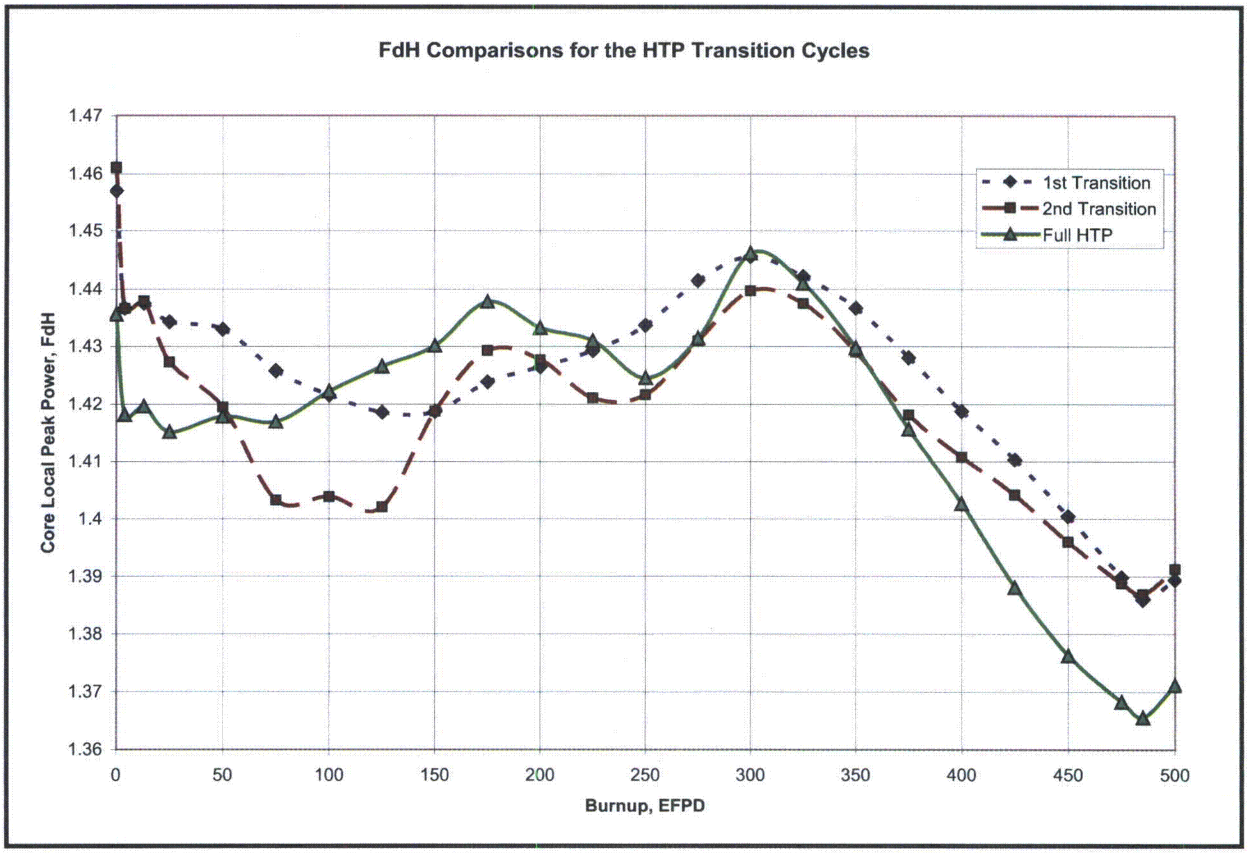


Figure 3-7: F_Q Comparisons for the HTP Transition Cycles

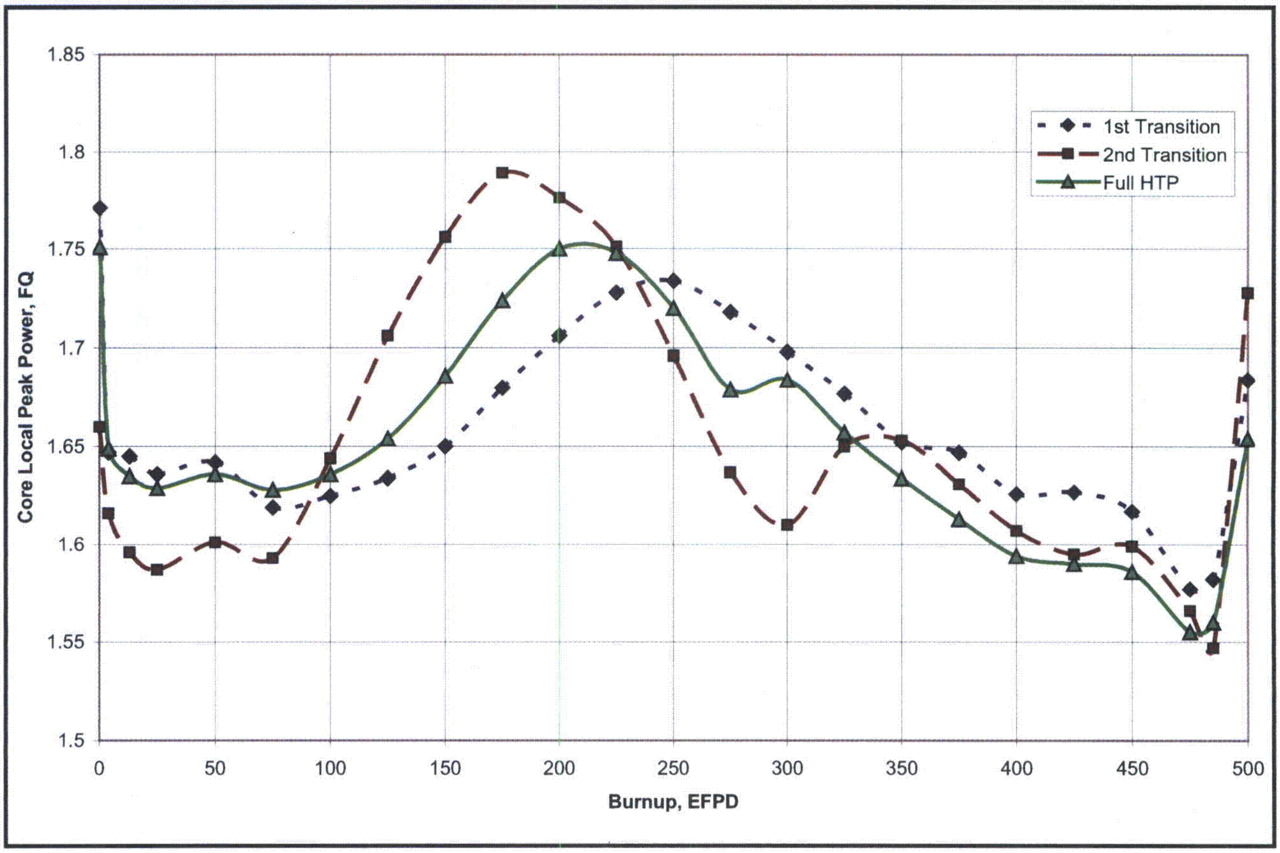


Figure 3-8: 1st Transition-HTP Core Assembly Burnup Distribution (BOC & EOC)

	H	G	F	E	D	C	B	A
8	20A 24.907 45.157	20D 21.008 42.878	21D 0.000 25.545	20A 24.232 46.040	21B 0.000 25.086	20A 24.202 45.894	21G 0.000 23.982	20C 24.908 35.811
9	20D 21.008 42.878	21C 0.000 25.142	20A 24.142 45.933	21D 0.000 25.550	20B 24.728 46.386	21E 0.000 25.423	21G 0.000 23.054	20B 24.777 34.992
10	21D 0.000 25.545	20A 24.055 45.868	21C 0.000 25.346	20E 24.509 46.347	21D 0.000 25.739	20F 21.954 43.771	20F 21.001 38.883	19G 38.424 46.018
11	20A 24.232 46.040	21D 0.000 25.557	20E 24.524 46.360	21C 0.000 25.729	20G 19.924 43.182	21F 0.000 24.665	21H 0.000 20.648	19D2 46.712 52.189
12	21B 0.000 25.086	20B 24.799 46.442	21D 0.000 25.743	20G 19.927 43.178	21A 0.000 24.344	20A 24.453 43.118	21J 0.000 16.964	
13	20A 24.202 45.894	21E 0.000 25.437	20F 21.759 43.618	21F 0.000 24.639	20B 24.783 43.354	21I 0.000 18.483	19E2 45.419 51.589	
14	21G 0.000 23.982	21G 0.000 23.059	20F 21.060 38.931	21H 0.000 20.621	21J 0.000 16.920	19E2 45.500 51.642		
15	20C 24.908 35.811	20B 24.783 34.997	19G 38.462 46.051	19D2 46.698 52.168	Batch ID 0 Gwd/mtU 19.715 Gwd/mtU			

Figure 3-9: 2nd Transition-HTP Core Assembly Burnup Distribution (BOC & EOC)

	H	G	F	E	D	C	B	A
8	22B 0.000 25.158	21D 25.557 46.948	22B 0.000 25.510	21D 25.550 46.999	22B 0.000 25.341	21G 23.982 45.691	22D 0.000 21.631	19D3 46.805 54.193
9	21D 25.557 46.948	22A 0.000 25.495	21F 24.665 46.728	22E 0.000 25.682	21E 25.423 47.179	22E 0.000 25.274	21J 16.964 37.116	21C 25.729 35.001
10	22B 0.000 25.510	21F 24.639 46.704	22B 0.000 25.500	21A 24.344 45.669	22A 0.000 25.346	21G 23.054 45.084	22G 0.000 22.721	21B 25.086 34.331
11	21D 25.550 46.999	22E 0.000 25.653	21C 25.142 46.520	22E 0.000 25.164	21D 25.545 46.503	22E 0.000 25.269	22F 0.000 21.060	19A2 46.019 51.750
12	22B 0.000 25.341	21E 25.437 47.174	22A 0.000 25.318	21C 25.346 46.326	21I 18.483 41.502	22C 0.000 23.652	21H 20.648 34.421	
13	21G 23.982 45.691	22E 0.000 25.276	21G 23.059 45.092	22E 0.000 25.278	22C 0.000 23.658	22H 0.000 19.476	19C2 46.801 52.753	
14	22D 0.000 21.631	21J 16.920 37.099	22G 0.000 22.747	22F 0.000 21.079	21H 20.621 34.405	19C2 46.819 52.772		
15	19D3 46.805 54.193	21D 25.743 35.041	21D 25.739 35.080	19A2 46.033 51.772	Batch ID 0 Gwd/mtU 19.727 Gwd/mtU			

Figure 3-10: All-HTP Core Assembly Burnup Distribution (BOC & EOC)

	H	G	F	E	D	C	B	A
8	23B 0.000 25.676	22B2 25.500 46.842	23C 0.000 25.764	22A 25.495 46.705	23D 0.000 25.475	22D 21.631 42.687	23F 0.000 23.464	21D2 35.080 44.488
9	22B2 25.500 46.842	23B 0.000 25.758	22A 25.346 46.631	23C 0.000 25.614	22E 25.164 46.252	22G 22.721 44.119	23F 0.000 23.419	22B2 25.341 35.667
10	23C 0.000 25.764	22A 25.318 46.611	23C 0.000 25.316	22C 23.652 44.564	23D 0.000 25.542	22F 21.060 43.039	23E 0.000 22.994	21H 34.421 43.783
11	22A 25.495 46.705	23C 0.000 25.612	22C 23.658 44.569	22H 19.476 40.656	22E 25.682 45.960	23A 0.000 25.124	23G 0.000 21.971	21J 37.116 44.262
12	23D 0.000 25.475	22E 25.274 46.351	23D 0.000 25.541	22E 25.653 45.943	22E 25.276 45.438	23F 0.000 24.222	22E 25.269 38.077	
13	22D 21.631 42.687	22G 22.747 44.134	22F 21.079 43.051	23A 0.000 25.130	23F 0.000 24.231	23H 0.000 20.610	21C2 35.001 42.026	
14	23F 0.000 23.464	23F 0.000 23.404	23E 0.000 22.987	23G 0.000 21.972	22E 25.278 38.089	21D2 35.041 42.075		
15	21D2 35.080 44.488	22B2 25.510 35.801	21H 34.405 43.761	21J 37.099 44.245	Batch ID 0 Gwd/mtU 19.730 Gwd/mtU			

Figure 3-11: 1st Transition HTP core BOC Power Distribution

	H	G	F	E	D	C	B	A
8	1.215 1.167 1.041	1.297 1.228 1.056	1.413 1.242 1.138	1.178 1.133 1.040	1.313 1.159 1.133	1.142 1.098 1.041	1.360 1.167 1.166	0.827 0.527 1.568
9	1.297 1.228 1.056	1.432 1.254 1.142	1.224 1.158 1.058	1.374 1.213 1.133	1.172 1.109 1.057	1.342 1.168 1.149	1.355 1.123 1.206	0.813 0.495 1.643
10	1.413 1.242 1.138	1.226 1.159 1.058	1.381 1.220 1.132	1.209 1.148 1.053	1.418 1.236 1.147	1.220 1.158 1.053	1.164 0.942 1.236	0.692 0.372 1.861
11	1.178 1.133 1.040	1.375 1.213 1.133	1.208 1.148 1.053	1.437 1.254 1.146	1.361 1.269 1.073	1.433 1.226 1.169	1.363 1.051 1.297	0.525 0.265 1.980
12	1.313 1.159 1.133	1.169 1.108 1.054	1.417 1.236 1.146	1.360 1.269 1.072	1.409 1.207 1.167	1.179 1.036 1.138	1.297 0.925 1.403	
13	1.142 1.098 1.041	1.343 1.169 1.149	1.222 1.161 1.053	1.432 1.223 1.170	1.168 1.029 1.136	1.316 1.042 1.262	0.699 0.329 2.123	
14	1.360 1.167 1.166	1.355 1.123 1.207	1.162 0.941 1.236	1.359 1.048 1.297	1.291 0.921 1.403	0.690 0.327 2.108		
15	0.827 0.527 1.568	0.813 0.495 1.643	0.693 0.371 1.866	0.525 0.264 1.984	Peak Pin Power Average Assembly Power Peak to Average Power			

Figure 3-12: 1st Transition core HTP MOC Power Distribution

	H	G	F	E	D	C	B	A
8	1.024 0.993 1.032	1.129 1.093 1.033	1.425 1.340 1.063	1.164 1.118 1.041	1.410 1.332 1.059	1.158 1.109 1.044	1.423 1.244 1.144	0.817 0.543 1.505
9	1.129 1.093 1.033	1.387 1.294 1.072	1.150 1.106 1.040	1.431 1.349 1.061	1.153 1.108 1.041	1.433 1.329 1.078	1.422 1.187 1.198	0.802 0.504 1.590
10	1.425 1.340 1.063	1.153 1.107 1.041	1.402 1.319 1.063	1.155 1.107 1.043	1.431 1.344 1.064	1.171 1.092 1.073	1.049 0.881 1.190	0.677 0.371 1.823
11	1.164 1.118 1.041	1.432 1.350 1.061	1.155 1.107 1.043	1.416 1.331 1.064	1.227 1.164 1.055	1.381 1.241 1.113	1.274 1.029 1.238	0.501 0.264 1.894
12	1.410 1.332 1.059	1.152 1.107 1.041	1.431 1.345 1.064	1.226 1.163 1.054	1.374 1.243 1.105	1.034 0.899 1.150	1.149 0.816 1.407	
13	1.158 1.109 1.044	1.434 1.330 1.078	1.175 1.094 1.074	1.380 1.239 1.114	1.026 0.895 1.147	1.136 0.881 1.289	0.578 0.289 2.003	
14	1.423 1.244 1.144	1.423 1.187 1.198	1.048 0.881 1.189	1.273 1.028 1.238	1.146 0.815 1.407	0.572 0.287 1.990		
15	0.817 0.543 1.505	0.802 0.504 1.590	0.678 0.371 1.827	0.501 0.264 1.898	Peak Pin Power Average Assembly Power Peak to Average Power			

Figure 3-13: 1st Transition core HTP EOC Power Distribution

	H	G	F	E	D	C	B	A
8	1.006 0.976 1.031	1.082 1.051 1.029	1.324 1.266 1.045	1.100 1.059 1.039	1.335 1.277 1.046	1.118 1.079 1.036	1.349 1.220 1.106	0.845 0.601 1.406
9	1.082 1.051 1.029	1.318 1.255 1.051	1.091 1.055 1.033	1.333 1.275 1.045	1.109 1.066 1.040	1.386 1.311 1.057	1.355 1.187 1.141	0.832 0.569 1.463
10	1.324 1.266 1.045	1.092 1.056 1.034	1.333 1.274 1.046	1.102 1.060 1.039	1.349 1.288 1.047	1.137 1.081 1.051	1.061 0.921 1.152	0.740 0.440 1.682
11	1.100 1.059 1.039	1.333 1.275 1.045	1.102 1.060 1.039	1.351 1.289 1.048	1.199 1.129 1.062	1.355 1.268 1.069	1.267 1.072 1.182	0.566 0.322 1.762
12	1.335 1.277 1.046	1.108 1.065 1.041	1.349 1.288 1.047	1.199 1.129 1.061	1.323 1.243 1.065	1.042 0.943 1.105	1.175 0.883 1.331	
13	1.118 1.079 1.036	1.386 1.311 1.057	1.139 1.083 1.052	1.355 1.268 1.069	1.040 0.939 1.108	1.171 0.957 1.224	0.644 0.350 1.840	
14	1.349 1.220 1.106	1.355 1.187 1.141	1.060 0.921 1.151	1.267 1.072 1.182	1.174 0.882 1.331	0.638 0.349 1.828		
15	0.845 0.601 1.406	0.832 0.569 1.463	0.741 0.440 1.685	0.567 0.321 1.764	Peak Pin Power Average Assembly Power Peak to Average Power			

Figure 3-14: 2nd Transition core HTP BOC Power Distribution

	H	G	F	E	D	C	B	A
8	1.405 1.271 1.106	1.220 1.180 1.034	1.437 1.294 1.110	1.222 1.180 1.036	1.424 1.279 1.114	1.302 1.215 1.072	1.394 1.124 1.240	0.637 0.387 1.647
9	1.220 1.180 1.034	1.411 1.257 1.122	1.269 1.219 1.041	1.427 1.262 1.131	1.234 1.180 1.045	1.414 1.245 1.136	1.413 1.123 1.259	0.751 0.487 1.543
10	1.437 1.294 1.110	1.269 1.219 1.041	1.431 1.283 1.115	1.205 1.151 1.047	1.361 1.208 1.126	1.242 1.172 1.060	1.364 1.091 1.250	0.744 0.457 1.626
11	1.222 1.180 1.036	1.426 1.260 1.132	1.210 1.153 1.050	1.384 1.219 1.136	1.161 1.117 1.039	1.344 1.165 1.153	1.320 0.998 1.323	0.563 0.266 2.115
12	1.424 1.279 1.114	1.233 1.179 1.045	1.359 1.207 1.126	1.165 1.118 1.042	1.340 1.250 1.072	1.329 1.091 1.218	1.115 0.668 1.669	
13	1.302 1.215 1.072	1.414 1.245 1.136	1.243 1.173 1.060	1.345 1.166 1.153	1.330 1.092 1.218	1.340 0.906 1.479	0.597 0.271 2.206	
14	1.394 1.124 1.240	1.414 1.125 1.257	1.366 1.093 1.249	1.322 1.000 1.322	1.116 0.669 1.669	0.598 0.271 2.209		
15	0.637 0.387 1.647	0.753 0.489 1.541	0.753 0.463 1.627	0.564 0.267 2.111	Peak Pin Power Average Assembly Power Peak to Average Power			

Figure 3-15: 2nd Transition core HTP MOC Power Distribution

	H	G	F	E	D	C	B	A
8	1.361 1.287 1.057	1.095 1.062 1.031	1.388 1.307 1.062	1.100 1.067 1.031	1.372 1.295 1.059	1.147 1.068 1.074	1.265 1.077 1.175	0.563 0.351 1.604
9	1.095 1.062 1.031	1.399 1.311 1.067	1.141 1.096 1.041	1.419 1.324 1.071	1.124 1.080 1.041	1.388 1.290 1.076	1.201 0.985 1.219	0.691 0.440 1.569
10	1.388 1.307 1.062	1.141 1.097 1.041	1.392 1.308 1.064	1.109 1.064 1.042	1.398 1.309 1.068	1.175 1.103 1.065	1.383 1.168 1.184	0.709 0.451 1.572
11	1.100 1.067 1.031	1.418 1.323 1.072	1.103 1.066 1.034	1.389 1.296 1.072	1.093 1.047 1.044	1.422 1.322 1.075	1.371 1.091 1.256	0.578 0.284 2.032
12	1.372 1.295 1.059	1.122 1.079 1.040	1.397 1.308 1.068	1.094 1.047 1.045	1.267 1.159 1.093	1.417 1.246 1.137	1.113 0.699 1.592	
13	1.147 1.068 1.074	1.388 1.290 1.076	1.175 1.103 1.065	1.422 1.322 1.075	1.418 1.247 1.137	1.346 1.016 1.324	0.635 0.299 2.122	
14	1.265 1.077 1.175	1.201 0.986 1.218	1.383 1.169 1.183	1.372 1.092 1.256	1.113 0.699 1.592	0.636 0.299 2.124		
15	0.563 0.351 1.604	0.695 0.442 1.572	0.716 0.455 1.572	0.578 0.285 2.028	Peak Pin Power Average Assembly Power Peak to Average Power			

Figure 3-16: 2nd Transition core HTP EOC Power Distribution

	H	G	F	E	D	C	B	A
8	1.322 1.256 1.053	1.062 1.032 1.029	1.336 1.267 1.055	1.078 1.048 1.029	1.351 1.281 1.055	1.129 1.063 1.062	1.238 1.095 1.130	0.622 0.416 1.497
9	1.062 1.032 1.029	1.331 1.260 1.056	1.107 1.064 1.041	1.368 1.296 1.056	1.101 1.064 1.035	1.364 1.283 1.064	1.184 1.006 1.178	0.745 0.511 1.459
10	1.336 1.267 1.055	1.108 1.064 1.041	1.341 1.274 1.053	1.089 1.048 1.039	1.349 1.279 1.055	1.144 1.084 1.056	1.315 1.160 1.134	0.752 0.514 1.463
11	1.078 1.048 1.029	1.367 1.295 1.056	1.081 1.049 1.030	1.360 1.284 1.059	1.076 1.036 1.038	1.387 1.300 1.067	1.301 1.089 1.194	0.623 0.334 1.866
12	1.351 1.281 1.055	1.100 1.063 1.034	1.348 1.278 1.055	1.078 1.036 1.040	1.222 1.132 1.079	1.343 1.220 1.101	1.082 0.738 1.466	
13	1.129 1.063 1.062	1.364 1.282 1.064	1.144 1.084 1.056	1.387 1.300 1.067	1.343 1.220 1.101	1.282 1.033 1.241	0.675 0.348 1.939	
14	1.238 1.095 1.130	1.184 1.006 1.177	1.315 1.160 1.134	1.301 1.089 1.194	1.082 0.739 1.465	0.676 0.348 1.941		
15	0.622 0.416 1.497	0.748 0.512 1.461	0.758 0.518 1.464	0.622 0.334 1.865	Peak Pin Power Average Assembly Power Peak to Average Power			

Figure 3-17: All HTP core BOC Power Distribution

	H	G	F	E	D	C	B	A
8	1.398 1.245 1.123	1.176 1.126 1.044	1.380 1.249 1.105	1.171 1.119 1.046	1.399 1.243 1.125	1.249 1.173 1.065	1.418 1.152 1.231	0.742 0.443 1.675
9	1.176 1.126 1.044	1.411 1.253 1.126	1.180 1.129 1.045	1.399 1.258 1.112	1.214 1.145 1.060	1.274 1.189 1.071	1.410 1.133 1.245	0.775 0.482 1.609
10	1.380 1.249 1.105	1.181 1.129 1.046	1.393 1.260 1.106	1.234 1.154 1.070	1.411 1.259 1.121	1.273 1.184 1.075	1.355 1.060 1.278	0.713 0.426 1.675
11	1.171 1.119 1.046	1.399 1.258 1.112	1.234 1.154 1.070	1.346 1.230 1.094	1.196 1.127 1.061	1.391 1.210 1.150	1.372 1.038 1.322	0.664 0.322 2.066
12	1.399 1.243 1.125	1.213 1.144 1.060	1.411 1.259 1.121	1.196 1.128 1.060	1.169 1.111 1.052	1.416 1.191 1.188	1.012 0.624 1.622	
13	1.249 1.173 1.065	1.273 1.189 1.071	1.272 1.184 1.075	1.391 1.210 1.150	1.417 1.192 1.189	1.408 1.050 1.342	0.727 0.337 2.159	
14	1.418 1.152 1.231	1.409 1.132 1.245	1.354 1.060 1.278	1.372 1.038 1.322	1.012 0.624 1.622	0.731 0.338 2.165		
15	0.742 0.443 1.675	0.771 0.480 1.606	0.712 0.425 1.675	0.664 0.322 2.066	Peak Pin Power Average Assembly Power Peak to Average Power			

Figure 3-18: All HTP core MOC Power Distribution

	H	G	F	E	D	C	B	A
8	1.397 1.317 1.061	1.114 1.070 1.042	1.411 1.333 1.058	1.113 1.066 1.044	1.408 1.308 1.077	1.109 1.043 1.064	1.377 1.208 1.140	0.750 0.471 1.594
9	1.114 1.070 1.042	1.406 1.322 1.063	1.115 1.068 1.044	1.404 1.319 1.064	1.113 1.045 1.065	1.125 1.059 1.062	1.380 1.210 1.140	0.786 0.518 1.518
10	1.411 1.333 1.058	1.115 1.068 1.044	1.391 1.296 1.073	1.123 1.034 1.087	1.404 1.304 1.077	1.174 1.101 1.066	1.398 1.199 1.166	0.749 0.474 1.580
11	1.113 1.066 1.044	1.404 1.319 1.064	1.123 1.034 1.087	1.095 1.029 1.064	1.077 0.993 1.085	1.399 1.291 1.083	1.425 1.145 1.245	0.715 0.363 1.970
12	1.408 1.308 1.077	1.112 1.044 1.065	1.404 1.304 1.077	1.077 0.993 1.084	1.064 0.993 1.071	1.402 1.251 1.121	0.982 0.644 1.524	
13	1.109 1.043 1.064	1.124 1.058 1.062	1.173 1.101 1.066	1.399 1.292 1.083	1.403 1.252 1.121	1.337 1.050 1.273	0.716 0.349 2.050	
14	1.377 1.208 1.140	1.379 1.209 1.140	1.398 1.199 1.166	1.425 1.145 1.245	0.982 0.644 1.524	0.720 0.350 2.056		
15	0.750 0.471 1.594	0.784 0.516 1.519	0.748 0.474 1.579	0.714 0.363 1.969	Peak Pin Power Average Assembly Power Peak to Average Power			

Figure 3-19: All HTP core HTP EOC Power Distribution

	H	G	F	E	D	C	B	A
8	1.355 1.297 1.045	1.082 1.044 1.037	1.359 1.289 1.054	1.073 1.035 1.037	1.360 1.276 1.066	1.085 1.025 1.058	1.330 1.211 1.098	0.796 0.534 1.491
9	1.082 1.044 1.037	1.350 1.292 1.045	1.078 1.038 1.039	1.350 1.279 1.056	1.078 1.021 1.056	1.099 1.040 1.057	1.337 1.215 1.100	0.832 0.584 1.426
10	1.359 1.289 1.054	1.078 1.038 1.038	1.345 1.266 1.063	1.088 1.014 1.073	1.358 1.273 1.067	1.144 1.079 1.060	1.358 1.213 1.120	0.796 0.539 1.477
11	1.073 1.035 1.037	1.350 1.279 1.056	1.088 1.014 1.073	1.080 1.019 1.060	1.062 0.989 1.074	1.366 1.288 1.060	1.351 1.141 1.184	0.758 0.417 1.820
12	1.360 1.276 1.066	1.077 1.021 1.055	1.358 1.273 1.067	1.062 0.990 1.074	1.055 0.994 1.061	1.344 1.233 1.090	0.968 0.685 1.413	
13	1.085 1.025 1.058	1.099 1.040 1.057	1.143 1.079 1.060	1.366 1.288 1.060	1.344 1.233 1.090	1.275 1.054 1.209	0.747 0.397 1.880	
14	1.330 1.211 1.098	1.337 1.215 1.100	1.358 1.212 1.120	1.351 1.141 1.184	0.968 0.685 1.413	0.750 0.398 1.885		
15	0.796 0.534 1.491	0.830 0.582 1.427	0.796 0.539 1.477	0.758 0.417 1.820	Peak Pin Power Average Assembly Power Peak to Average Power			

3.7 References for Section 3.0

1. BAW-10180A, Rev. 1, "NEMO - Nodal Expansion Method Optimized", B&W Fuel Company, Lynchburg, Virginia, March 1993, released 7/22/1993.
2. M. Edenius, et al., "CASMO-3 — A Fuel Assembly Burnup Program", STUDS VIK/NFA-89/3, Studsvik AB, Nykoping, Sweden, November 1989.
3. BAW-10163A, "Core Operating Limit Methodology for Westinghouse-Designed PWRs", B&W Fuel Company, Lynchburg, Virginia, June 1989.
4. BAW-10220P, Rev. 0, "MARK-BW FUEL ASSEMBLY APPLICATION FOR SEQUOYAH NUCLEAR UNITS 1 AND 2", Framatome Cogema Fuels, Lynchburg, Virginia, March 1996.
5. BAW-10227P-A, Rev. 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", Framatome Cogema Fuels, Lynchburg, Virginia, June 2003.
6. BAW-10186PA, Rev. 2 (Includes Revision 1, Supplement 1), "Extended Burnup Evaluation", June 2003.
7. ANP-2692P, Revision 001, "Batch Implementation of Blended Low Enriched Uranium Fuel at Sequoyah Nuclear Plant, April 2008.

4.0 THERMAL HYDRAULICS

4.1 Introduction

This section provides a summary of the thermal-hydraulic analysis methods and models used by AREVA to support the licensing of the Adv. W17 HTP fuel design for operation at the Tennessee Valley Authority's (TVA) Sequoyah Nuclear Plant (SQN). The Adv. W17 HTP fuel design is a new fuel design for Sequoyah, which consists of a 17 x 17 assembly configuration with M5™ fuel rods, Zircaloy-4 MONOBLOC™ guide tubes, a High Mechanical Performance (HMP) spacer at the lowermost axial elevation, HTP spacers in all other axial elevations with intermediate flow mixers (IFM) in the spans between spacers 4 and 5, 5 and 6 and 6 and 7, a FUELGUARD lower tie plate (LTP) and the AREVA NP reconstitutable top nozzle (TN). The thermal-hydraulic licensing of the Adv. W17 HTP will be performed with NRC approved codes and methods.

The purpose of this section is to demonstrate how the requirements as set forth in the Standard Review Plan are met with respect to thermal-hydraulic performance.

4.2 Methodology

AREVA uses the LYNXT thermal-hydraulic analysis code to perform the various thermal-hydraulic analyses needed to license the Adv. W17 HTP design. LYNXT, a single-pass code, employs crossflow methodologies to evaluate subchannel thermal-hydraulic conditions for both steady-state and transient conditions. A more complete description of LYNXT is provided in the NRC-approved topical report BAW-10156A (Reference TH-1).

4.2.1 Form Loss Coefficients

Component and subchannel form loss coefficients for the Adv. W17 HTP have been developed from pressure drop test data acquired from AREVA's Portable Hydraulic Test Facility (PHTF) in Richland, Washington, for the similar Kansai Ohi and Shearon Harris HTP fuel designs.

4.2.2 CHF Correlation

The applicable critical heat flux (CHF) correlation for the axial regions above the lowermost HTP spacer grid on the Adv. W17 HTP fuel assembly design is the BHTP correlation as documented in the NRC-approved topical report BAW-10241P-A Rev. 01 (Reference TH-2). Revision 00 of the topical report contains the recorrelation of the extensive HTP spacer grid CHF test data base using the LYNXT thermal-hydraulic subchannel code which led to the unique correlation name of BHTP. Revision 01 justifies the BHTP correlation to extended application ranges. The BHTP correlation is applicable exclusively with the LYNXT code for HTP type spacer grid fuel designs. For the axial region below the lowermost HTP grid, the BWU-N CHF correlation (Reference TH-15) is applied.

4.2.3 LYNXT Modeling

As part of the Adv. W17 HTP thermal-hydraulic analysis task, LYNXT models of the Adv. W17 HTP assembly and the SQN core have been developed. The methods that are used to define the core and assembly geometries in the LYNXT code are documented in Reference TH-1. LYNXT models the core with a group of channels of varying sizes. These channels increase in size from individual subchannels, to a group of subchannels, to a group of bundles. By using this variable-scaling method it is possible to model the entire core, while having a detailed subchannel model of the area around the hot subchannel. Typical models being used in the evaluations of the Adv. W17 HTP fuel include detailed subchannel models ranging in size for 12 to 75 channels for DNB analysis and local crossflow velocities, and more coarse bundle-by-bundle models for determining general crossflow velocities between assemblies and hydraulic lift forces.

4.2.4 Application of Statistical Core Design

The DNB analyses of the AREVA fuel designs in the SQN cores utilize AREVA's Statistical Core Design (SCD) thermal-hydraulic analysis methodology. The following section provides a brief overview of the SCD approach and analysis method. A more in-depth discussion of the SCD method can be found in BAW-10170P-A (Reference TH-3). The general use of SCD methodology does not preclude the use of the conservative deterministic application of uncertainties if it is deemed necessary.

4.2.4.1 Statistical Design Limit - SDL

The purpose of the core DNB analysis is to insure that a 95 percent probability exists, with a 95 percent confidence level, that the hot pin will not experience a departure from nucleate boiling (DNB) during normal operation or during transients of moderate frequency. In addition, the criterion states that when the limiting pin in the core is at the DNB design limit, no more than 0.01% of the pins in the core will experience a DNB. In the SCD method, described in BAW-10170P-A, the uncertainties on specific input variables are subjected to a statistical treatment to determine an overall DNBR uncertainty. This uncertainty is then used to establish a DNBR design limit known as the Statistical Design Limit (SDL). All input variables treated in the development of the SDL are then input into the thermal-hydraulic analysis computer codes at their nominal level. In the application documented in BAW-10170P-A, the SCD method was developed for the 17 x 17 Mark-BW fuel design, using the BWCMV-A CHF correlation. For this Sequoyah application, the SCD method is applied to the Adv. W17 HTP fuel design, using the BHTP CHF correlation, with statepoints and uncertainties applicable to the Sequoyah core design and licensing basis. Since the resident AREVA fuel design utilizes the BWCMV-A CHF correlation, there is a respective BWCMV-A SDL that is applicable to the Mark-BW fuel.

4.2.4.2 Thermal Design Limit – TDL

The application of the SCD method to SQN includes the addition of margin to the SDL which defines an analysis limit known as the Thermal Design Limit (TDL). The TDL becomes the DNBR design criterion for a fuel reload with flexibility to accommodate cycle-specific analysis needs. The difference between the TDL and the SDL is known as the retained thermal margin (RTM) and is calculated using the following formula (where 1 DNB point = 0.01 in absolute DNBR).

$$\text{Retained Thermal Margin (in DNB points)} = (\text{TDL} - \text{SDL}) \times 100$$

The retained thermal margin is used to provide analysis flexibility to accommodate cycle specific needs. A summary of plant specific penalties, if any, to be assessed against the retained margin is included in the cycle-specific reload analysis documents.

4.2.5 Core Power Distribution

A reference design core power distribution, a radial and axial peak combination, is used to determine a DNBR performance that conservatively bounds the DNBR performance of an actual power distribution occurring during normal operation. Peaking conditions for the core power distribution are defined by the use of a hot pin design radial peak of 1.64 ($F^N_{\Delta H}$) and a design axial peak (F_z) and peak location (x/L) selected to ensure bounding DNB performance. It should be noted that the radial peak of 1.64 corresponds to a maximum allowable radial peak of 1.70 when a 4% total rod power uncertainty factor is included.

4.2.6 Core Conditions

A summary of general core conditions used in the SQN thermal-hydraulic analyses is provided in Table 4-1.

4.2.7 Engineering Hot Channel Factors

Engineering hot channel factors (HCFs) are penalty factors that are used to account for the effects of manufacturing variations on the linear heat generation rate and enthalpy rise.

4.2.7.1 Local Heat Flux Engineering Hot Channel Factor

The local heat flux engineering hot channel factor, F^E_Q, is used in the evaluation of the maximum linear heat generation rate. This factor is determined by statistically combining manufacturing variances for pellet enrichment and weight at the 95% probability level with 95% confidence. As discussed in References TH-5 and TH-6, relatively small heat flux spikes such as those represented by F^E_Q have no effect on DNB, therefore this factor is not used in DNBR calculations.

4.2.7.2 Average Pin Power Engineering Hot Channel Factor

The average pin power factor, $F^E_{\Delta H}$, accounts for the effects of variations in fuel stack weight, enrichment, fuel rod diameter, and pin pitch on hot pin average power. This factor is combined statistically with other uncertainties to establish the statistical design limit (SDL) DNBR used with the statistical core design method (discussed in Section 4.2.4).

Since $F^E_{\Delta H}$ is incorporated into the statistical design limit (SDL), this factor is not included in the LYNXT model used for SCD analyses. For non-SCD analyses, $F^E_{\Delta H}$ is incorporated into the LYNXT model as a multiplier on the hot pin average power.

4.2.7.3 Densification Power Spike Factors

The peaking increase due to the power spike that results from a gap between UO_2 pellets has been analyzed and documented in topical report BAW-10054, Rev. 2 (Reference TH-13). These gaps may occur when pellet-cladding interaction causes a pellet to stick to the cladding. The underlying pellets densify and a gap beneath the stuck pellet is formed. Gap measurements have been performed on modern irradiated AREVA fuel rods, and only very small-gaps have been observed (≤0.1 inch)(Reference TH-12). The reported gap measurements were performed on fuel at cold temperature conditions. Since the fuel rod stack increases in length during heatup at a rate greater than the cladding (0.5 to 1 inch), the gaps are eliminated or reduced to less than 0.1 inch at power operation. Any remaining gaps during power operation will produce negligible power peaking effects. Therefore, no explicit penalty is included to account for densification spike effects (Reference TH-9).

4.2.8 Fuel Rod Bowing

The bowing of fuel rods during reactor operation has the potential to affect both local power peaking and the margin to DNB. The impact of fuel rod bowing on DNB performance is addressed in the NRC approved methodology in BAW-10147PA-Rev.1 (Reference TH-7). The effect of fuel rod bow is manifest as a DNBR penalty. However, as discussed in Reference TH-7, a 1% DNBR credit is accounted for by the flow area (pitch) reduction allowance that is incorporated into the engineering hot channel factor on hot pin average power, discussed in Section 4.2.7.2 of this report. In the event that limiting peaking occurs beyond the 24,000 MWd/mtU threshold (i.e. at the point where the 1% DNBR credit is exhausted), an additional DNBR penalty or an assessment of offsetting conservatisms, performed in accordance with Reference TH-7, may be necessary. Both the Adv. W17 HTP fuel design and the resident Mark-BW fuel design will be used in compliance with the requirements and accommodations specified in Reference TH-7.

4.2.9 Reactor Coolant Flow Rate and Bypass

An analysis was performed to assess the change in reactor coolant system loop flow attributed to the fuel transition. The analysis indicates that the transition from a full core of Mark-BW fuel to a full core of Adv. W17 HTP fuel results in a small increase in bypass flow and a small decrease in the RCS loop flow due to the higher pressure drop of the Adv. W17 HTP fuel. However, coincident with the fuel transition, steam generators will be replaced at Sequoyah Unit 2. Steam generators were previously replaced in Sequoyah Unit 1 in 2003 prior to Unit 1 Cycle 13 startup. The combined effect of the fuel transition and the steam generator replacement is a small net increase in RCS loop flow. Given this beneficial increase in RCS flow, the new replacement steam generators with minimal tube plugging, and favorable historical measured flow data, the Technical Specification minimum loop flow rate requirement (T.S. 3.2.5) is being increased for 360,100 gpm to 378,400 gpm. An evaluation of DNBR margin for reduced flow conditions showed that for a 5% reduction in flow a 10% reduction in power would preserve the DNBR margin at 100% power and 100% flow. This evaluation supports the modification to Technical Specification Figure 3.2-1, which shows the flow versus power for four loops in operation relationship varying from an allowable flow of 378,400 gpm at 100% power to an allowable flow of 359,400 gpm at 90% power.

The statistical core design (SCD) method, discussed in Section 4.2.4, incorporates uncertainties associated with the reactor core coolant flow into the overall DNBR uncertainty, as represented by the SDL. Calculations performed with the SCD method therefore use a core coolant flow rate that is equal to the nominal thermal design flow rate, less the core bypass flow fraction. Non-SCD calculations account for the flow measurement uncertainty by using the minimum thermal design flow rate and the maximum core bypass flow fraction. In addition to the conservative treatment of the flow measurement uncertainty, an inlet flow distribution factor is also applied when performing design basis DNB analyses. The basis for the Sequoyah inlet flow distribution is provided in Section 4.4.3.1.2 of the Sequoyah FSAR. As discussed in that section, the core inlet flow distribution is based on several 1/7 scale model tests that determined a 5% reduction in flow to the hot fuel bundle results in a conservative design basis. The same section identifies that no significant variation could be found in inlet velocity distribution with reduced flow rate, and that the use of a 5% reduction in inlet flow to the hot assembly for a loop out of service is adequate. Consistent with that discussion, the AREVA DNB licensing basis applies a 5% reduction in flow to the limiting hot bundle.

4.2.10 Full Core DNB Performance

Sections 7.3, 7.4, and 7.5 of BAW-10220P (Reference TH-11) provide a general description of the processes used to develop or validate Core Safety Limit (CSL) Lines, the processes used to perform transient DNB analyses, and the processes used to develop Maximum Allowable Peaking (MAP) limits. For the transition to Adv. W17 HTP fuel at Sequoyah, these base T-H analyses are performed using the LYNXT thermal-hydraulic analysis code (Reference TH-1), the Statistical Core Design method (Reference TH-3), and the BHTP and BWU-N CHF correlations (References TH-2 and TH-15, respectively). Using these methods and the full core Adv. W17 HTP model, evaluations of the Core Safety Limit Lines have been performed. Using the BHTP and BWU-N CHF correlations and the SCD method, it was shown that the existing Core Safety Limit Lines needed to be tightened to maintain adequate DNB protection at the limits. This has resulted in a revision of Technical Specification Figure 2.1-1. An evaluation of the existing Overtemperature ΔT and Overpower ΔT functions showed that even with the reduced CSL lines adequate protection is being provided by the existing trip function, so no change to the trip function definitions is required. A comparison of the existing Core Safety Limits to the new Adv. W17 HTP based Core Safety Limits is provided in Figure 4-1. In addition to the steady-state Core Safety Limit evaluation, both generic and cycle specific transient DNB analyses are performed which provide the basis for peaking limits that are validated in the cycle specific Maneuvering Analysis and the cycle specific Nuclear Analysis checks. Section 3.3 provides a general overview of the cycle specific Maneuvering Analysis, while the event by event dispositions in Sections 5.2.2.1 through 5.2.2.27 illustrate that cycle-specific reload checks are designed to verify acceptable margin to event specific DNB based peaking limits.

Figure 4-1: Sequoyah Core Safety Limit Comparison for Advanced W17 HTP Transition

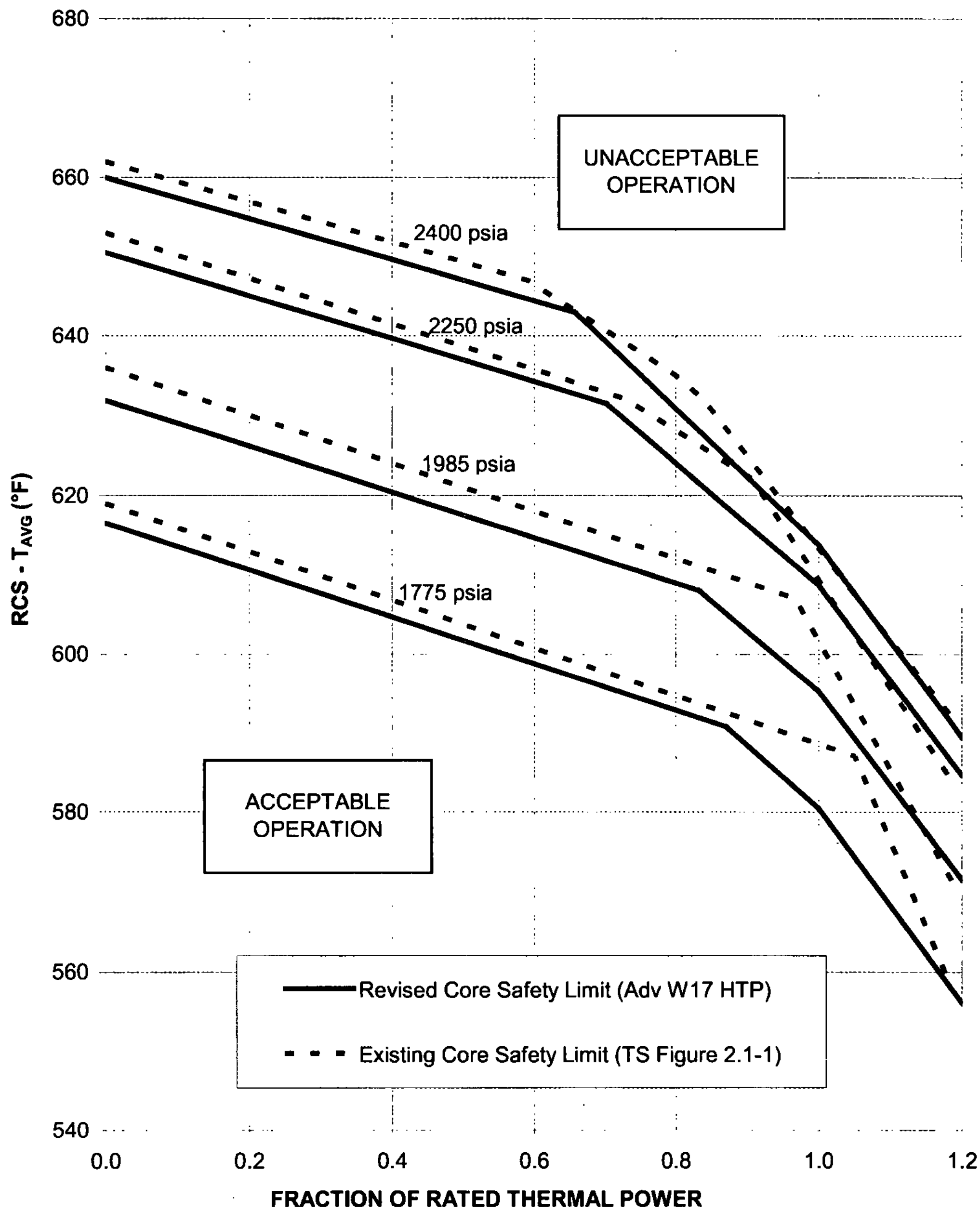


Table 4-1: Thermal-Hydraulic Analysis Design Parameters

Design Parameter	Value
Core Configuration:	
Number of Fuel Assemblies	193
Fuel Assembly Type	17x17
Number of Fuel Rods Per Assembly	264
Number of Control Clusters	53
Number of Guide Tubes per Assembly	24
Number of Instrument Tubes per Assembly	1
Reactor Coolant System:	
Rated Thermal Power, MWt	3455
Heat Generated In Fuel, %	97.4
Nominal System Pressure, psia	2280
Nominal Thermal Design Flow, gpm	378,400
Flow Fraction Effective for Heat Transfer (9.0% Bypass)	0.91
Minimum Thermal Design Flow, gpm	365,600
Average Vessel Coolant Temperature (nominal) at 100%RTP, °F	578.2
Vessel Coolant Inlet Temperature (nominal) at 100%RTP, °F	547.3
DNBR Calculations:	
CHF Correlations	BHTP and BWU-N for Adv. W17 HTP BWCMV-A and BWU-N for Mark-BW

4.3 Hydraulic Compatibility

This section documents the results of the hydraulic compatibility analysis of AREVA's Adv. W17 HTP fuel assemblies with AREVA's Mark-BW resident fuel in Sequoyah Units 1 and 2.

4.3.1 Core Pressure Drop

The Adv. W17 HTP fuel assemblies have a higher overall hydraulic resistance to flow than the Mark-BW fuel assemblies primarily resulting from the presence of the three intermediate flow mixing (IFM) grids on the Adv. W17 HTP design. As the core transitions from a full core of Mark-BW fuel to a full core of Adv. W17 HTP fuel, the core pressure drop will increase. During the transition cycles, core flow redistribution will take place driven by axial pressure drop differences between the Mark-BW and Adv. W17 HTP fuel designs. These pressure drop differences and flow diversion effects are evaluated and accommodated in the transition core analysis.

4.3.2 Hydraulic Lift

The Adv. W17 HTP and the Mark-BW fuel assemblies are both equipped with a three-leaf spring system to counteract the vertical hydraulic lift force created by the core flow rate. The Standard Review Plan requires that the fuel assembly remain in a seated position during Condition I and II events. AREVA uses the NRC approved Statistical Hold Down methodology (Reference TH-4) to demonstrate that the hold down force provided by the spring system is sufficient to prevent lift in both the full core or mixed core configuration.

4.3.3 Impact of Crud on Core Pressure Drop

Plant chemistry is maintained in a manner to control crud deposition such that the impact on pressure loss is negligible.

4.3.4 Crossflow

One of the key design concerns in any mixed core configuration is the magnitude of crossflow that occurs in the mixed core, and the impact of that crossflow on the thermal-hydraulic performance of the core. Assessing this concern for the Mark-BW to Adv. W17 HTP transition requires the performance of a full spectrum of transition core DNB evaluations. Those studies are discussed in Sections 4.4.1 and 4.4.2.

4.3.5 Guide Tube Heating

Coolant boiling within the guide tubes has the potential to increase corrosion rates and be detrimental for neutron moderation. Generic boiling analyses for the Mark-BW fuel design, which consider conservative core conditions and limiting control component heating rates, have demonstrated that long term bulk boiling will not occur within the guide tubes of the Mark-BW fuel assemblies. These analyses set bounding peaking limits that are validated on a cycle specific basis. AREVA's analysis of RCS flow rate and bypass for the Adv. W17 HTP transition cycles has demonstrated that the guide tubes of the Adv. W17 HTP are hydraulically similar to those of the resident Mark-BW fuel design. As is the case of the Mark-BW fuel design, bounding fuel rod peaking limits are used for the Adv. W17 HTP fuel assemblies to assure that long term guide tube bulk boiling will not occur.

4.3.6 Control Rod Drop Time

The control rod drop time is primarily dependent on the number, size, and location of the guide tube weep holes, as well as the inner diameter and height of the guide tube dashpot region. As discussed in Section 2.2.6 and shown in Table 2-6, in order to ensure the control rod drop times would not be impacted, the Adv. W17 HTP guide tubes were defined and designed to be similar to the Mark-BW for these critical parameters. Due to these similarities between the Mark-BW and Adv. W17 HTP guide tube designs, the control rod drop times are not significantly impacted by the fuel transition.

4.3.7 Thermo-Hydrodynamic Instability

Flow in heated boiling channels is susceptible to several forms of thermo-hydrodynamic instability. These instabilities are undesirable because they may cause thermal hydraulic conditions that reduce the margin to CHF during steady-state flow conditions or induce the vibration of core components.

Sequoyah was evaluated for its susceptibility to a wide range of potential thermo-hydrodynamic instabilities as outlined in Section 4.4.3.5 of the Sequoyah FSAR. The features that enhance stable fluid flow conditions include:

- Rod bundle core configuration – resists parallel channel instability.
- Highly subcooled operation – a power/flow margin to saturation avoids bulk boiling, thus preventing two-phase driven dynamic instabilities.

- High pressure operation – reduces density-driven effects associated with localized steam formation.
- Core channel pressure drop-flow curve has a positive slope while the reactor coolant system pump head-flow curve is negative – prevents Ledinegg flow excursion instability.
- Margin to CHF – avoids boiling crisis and film-boiling induced instabilities.

The transition from Mark-BW fuel to Adv. W17 HTP fuel will not adversely impact any of these features. Consequently, the thermo-hydrodynamic stability of the core will not be affected by the transition to the Adv. W17 HTP fuel assemblies.

4.4 Transition Core DNB Performance

For any new fuel design, such as the Adv. W17 HTP, that is being introduced on a reload basis, hydraulic compatibility must be demonstrated with the existing, or resident, fuel in the core. Therefore, when the Adv. W17 HTP fuel design, having different hydraulic characteristics from the resident fuel is introduced, a transition core analysis is performed. For each mixed core configuration during the transition cycles, DNB performance of each fuel type is evaluated relative to a reference analysis. This reference analysis is typically based on a full core of the new fuel. To determine the performance of each fuel type relative to the reference analysis, each fuel type is modeled in the actual core configuration or a conservatively bounding mixed core configuration. Section 3.1 and Figures 3-1, 3-2, and 3-3 describe demonstration configurations that could potentially be used for the first three cycles of transition from Mark-BW to Adv. W17 HTP. These demonstration configurations depict a typical pattern for the fresh fuel, with feed batches of 88 assemblies for the first transition cycle, 85 assemblies for the second transition cycle, and a full complement of Adv. W17 HTP assemblies by the third cycle. A conservatively bounding mixed core model would preferentially arrange the mixed core configuration, both in terms of location and number, in such a way as to impose mixed core hydraulic effects on the limiting hot bundle that were more limiting than the actual configuration. These mixed core configurations and analyses are used to demonstrate that the requirements for DNB performance are met for both fuel types.

During the transition to the Adv. W17 HTP fuel at Sequoyah, the resident fuel (i.e. the fuel being displaced by Adv. W17 HTP fuel assemblies) will be the Mark-BW fuel design. The Adv. W17 HTP fuel assembly, described in Section 2.0, is hydraulically compatible with the resident Mark-BW fuel. Tables 2-1 through 2-7 provide a comparison of the key design differences between the two assembly types.

4.4.1 Mixed Core DNB Analysis – Relative to the Advanced W17 HTP

For transition cycles in which the resident Mark-BW fuel is being displaced by Adv. W17 HTP fuel, core DNB safety and operating limits and DNB margin during transients are based on analysis of the full-core Adv. W17 HTP configuration. However, the transition core effects of the differing fuel types must also be evaluated. This is accomplished by performing a mixed core analysis. This mixed core analysis quantifies the transition cycle penalty that must be applied to either the resident or the new fuel design or to both fuel designs. The applicability of the full-core analyses is maintained by applying the transition core DNB penalty either as an assessment against retained thermal margin that is incorporated in the DNB analysis through the use of the Thermal Design Limit (TDL, per BAW-10170P-A, Reference TH-3), or by the identification of an offsetting conservatism.

The transition core DNB penalty is determined by modeling the actual configuration or by a bounding mixed core configuration. The magnitude of the DNB penalty is determined by assessing the change in minimum DNBR due to mixed core effects. The retained thermal margin (RTM), or an offsetting conservatism will be used to accommodate the transition core penalty (RTM is outlined in Section 4.2.4.2).

As the Sequoyah cores move toward fewer Mark-BW assemblies, AREVA's experience base with similar transitions shows that the mixed core effects become less pronounced. For applications where it is desirable to reduce the penalty to a value less than the generic value, the actual cycle specific configuration is analyzed, using a model that represents the actual transition cycle core geometry.

4.4.2 Mixed Core DNB Analyses - Relative to the Mark-BW

Core safety limits, Maximum Allowable Peaking limits, and transient DNB analyses for the resident Mark-BW fuel are performed using the LYNXT thermal-hydraulic analysis code (Reference TH-1), the Statistical Core Design method (Reference TH-3), the BWCMV-A CHF correlation (Reference TH-8, Reference TH-10, Reference TH-11 -Section 7.1.1), the BWU-N CHF correlation (Reference TH-15), and a full core Mark-BW model. As summarized in Section 7.1.1 of Reference TH-11, when applying the BWCMV correlation to the Mark-BW assembly using the equivalent grid spacing defined in BAW-10189P-A (Reference TH-10), it is referred to as BWCMV-A. The BWCMV-A CHF correlation is applied for the axial regions above the lowermost Mark-BW mixing vane spacer grid, while the non-mixing based BWU-N CHF correlation is applied below the lowermost Mark-BW mixing vane grid, in the region that contains the lower end grid and the non-mixing Mark-BW spacer grid.

The Mark-BW fuel will be protected against DNB failure. This can be shown by demonstrating that the Mark-BW is non-limiting relative to the Adv. W17 HTP using offsetting conservatisms or, alternatively, by using a process similar to that applied to the Adv. W17 HTP to determine the Mark-BW transition core DNB penalty.

4.4.3 DNB Propagation

The propagation of DNB failures is considered for PWRs when two conditions exist simultaneously:

- the DNB limiting rod of a bundle is calculated to have a MDNBR below the 95/95 limit value of the CHF correlation being used, and

- the internal pressure of the DNB limiting rod exceeds core pressure at the time of MDNBR.

DNB propagation is addressed by AREVA in the NRC approved methodology in (Reference TH-14). Using a process known as the Statistical Calculation of Core Protection, the maximum burnup for 99.99% corewide protection is determined. This generic analysis is performed using inputs of core power, design overpower, fuel pin peak to average power, the low pressure trip setpoint, and the DNB performance of the fuel design of interest. In this case, corresponding analyses will be performed for both the Mark-BW and Adv. W17 HTP fuel designs.

4.4.4 Impact of Crud on DNB Performance

The BHTP and the BWU-N critical heat flux (CHF) correlations (Reference TH-2 and TH-15, respectively), for application with the Adv. W17 HTP fuel design in the Sequoyah core, have been developed from CHF testing of electrically heated rods with no simulation of crud deposition. This is standard procedure for PWR CHF testing. The BHTP and BWU-N CHF correlations are applied in DNB analyses with no adjustment for the possible presence of crud since crud will result in a slightly rougher and larger surface area that improves CHF.

4.5 Thermal-Hydraulic SER Restrictions / Limitations

BAW-10220P Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 and 2
Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment 223 to Facility Operating License No. DPR-77 and Amendment 214 to Facility License No. DPR-79, April 21, 1997

Purpose
To present a complete LOCA, non-LOCA, mechanical, nuclear, thermal-hydraulic, and containment evaluation for operation of the Sequoyah Nuclear Units with Mark-BW reload fuel.

SER Restrictions

License Conditions for Sequoyah Unit 1 and Unit 2 [ ] stated:

TVA will obtain NRC approval prior to startup for any cycle's core that involves a reduction in the departure from nucleate boiling ratio initial transition core penalty below that value stated in TVA's submittal on Framatome fuel conversion dated April 6, 1997.

Implementation of SER Restrictions

A bounding transition core DNBR penalty to be taken against the Mark-BW assembly's retained thermal margin was developed and submitted to the NRC in response to Question 27 from the NRC's Request for Additional Information on TVA's 1996 LAR submittal related to the transition to Mark-BW17 fuel. The Mark-BW17 fuel design was co-resident with the Westinghouse Vantage 5H fuel design for SQN-1 Cycle 9, SQN-1 Cycle 10, SQN-2 Cycle 9, and SQN-2 Cycle 10. The transition core DNBR penalty was maintained at the bounding generic value throughout each of these four cycles, never encroaching on the License Condition imposed in the April 21, 1997 SER. The bounding transition core DNBR penalty submitted in the April 6, 1997 response was specifically calculated for the Vantage 5H to Mark-BW17 transition. Therefore, the April 21, 1997 License Condition is no longer applicable to the Sequoyah cores.

BAW-10170P-A Statistical Core Design for Mixing Vane Core

Purpose
To develop a thermal-hydraulic analysis technique that provides an increase in core thermal (DNB) margin by treating core state and bundle uncertainties statistically.

SER Restrictions

1. The component uncertainties and their distributions are to be reviewed on a plant-specific basis to determine their applicability.
2. The "bounding" assembly-wise power distribution assumed in the core-wide SDL calculation should be shown to bound the expected operating power distributions on a cycle-specific basis.
3. The response surface model should be validated and revised (as necessary) when applied to new fuel assembly designs and extended operating conditions, and with new codes and DNB correlations. The approved codes are LYNXT, LYNX1, and LYNX2, and the approved correlation is BWCMV.

Implementation of SER Restrictions

1. Component uncertainties and their distributions are reviewed and validated each reload cycle as part of the reload licensing process.
2. A core-wide protection calculation is performed each cycle using the cycle specific core power distribution.
3. For this application, a new response surface model basis has been established for the Adv. W17 HTP fuel design based on the BHTP CHF correlation.

BAW-10241P-A Rev. 01, BHTP DNB Correlation Applied with LYNXT

Purpose
BAW-10241P-A Rev. 01 documents development of the BHTP DNB correlation for application with the LYNXT thermal-hydraulic analysis code for use in the DNB analysis of the HTP fuel design.

SER Restrictions
Application of the BHTP DNB correlation with LYNXT is limited to the following ranges of local conditions and fuel design parameters:

Range of Coolant Conditions for BHTP Correlation	
Independent Variable	**Range**
Pressure (psia)	**1385 to 2425**
Local Mass Flux (Mlb/hr-ft^2)	**0.492 to 3.549**
Inlet Enthalpy (BTU/lb)	**383.9 to 644.3**
Local Quality	***no lower limit* to 0.512**

Range of Fuel Design Parameters for BHTP Correlation	
Design Parameter	**Range**
Fuel Rod Diameter (in)	**0.360 to 0.440**
Fuel Rod Pitch (in)	**0.496 to 0.580**
Axial Spacer Span (in)	**10.5 to 26.2**
Hydraulic Diameter (in)	**0.4517 to 0.5334**
Heated Length (ft)	**9.8 to 14.0**

Actions for analyzing the operating conditions outside the approved ranges of the maximum pressure (2425 psia) but less than 2600 psia are stated below.

- When pressures greater than the pressure limit of 2425 psia but less than 2600 psia are encountered, all of the local coolant conditions are calculated at the upper pressure limit of 2425 psia using the NRC-approved LYNXT thermal-hydraulic code and then used in the calculation of the BHTP CHF.
- Extrapolations below the minimum quality range are performed with no lower limit, consistent with EMF-92-153(P) (A) Revision 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel."

Implementation of SER Restrictions

1. A local conditions check has been programmed into the LYNXT code logic allowing for an automated confirmation that local conditions are within the approved range of applicability within each LYNXT DNB calculation.

2. The fuel design parameters of the Adv. W17 HTP fuel design are within the fuel design parameter range supported by the BHTP DNB correlation.

4.6 Thermal-Hydraulic Technical Specification Changes

This section summarizes the Thermal-Hydraulic Technical Specification changes being implemented with the Sequoyah Adv. W17 HTP LAR submittal.

Technical Specification 2.1 Safety Limits
Text is being revised to a form similar to the Improved Standard Technical Specifications. Note: the modification in the Technical Specification format is not related to the fuel transition. The following DNBR design limits and applicable CHF correlations are added to Technical Specification 2.1.1.1:

> *For the Advanced W17 HTP fuel design, 1.132 for the BHTP correlation (Reference* TH-2, *pg. 4 - 1) and 1.21 for the BWU-N correlation (Reference* TH-15 *pg. v); for the Mark-BW fuel design, 1.21 for the BWCMV correlation (References* TH-8 *pg. xviii and* TH-10 *pg. iv) and 1.21 for the BWU-N correlation (Reference* TH-15 *pg. v).*

Technical Specification Figure 2.1-1
Core Safety Limit Lines are being modified as a result of the transition to the Adv. W17 HTP design and the implementation of the BHTP DNB correlation. As noted in Section 4.2.10 of this document, Section 7.3 of BAW-10220P (Reference TH-11) provides a general description of the processes used to develop or validate Core Safety Limit (CSL) Lines. For the transition to Advanced W17 HTP fuel at Sequoyah, the revised CSL lines are developed using the LYNXT thermal-hydraulic analysis code (ReferenceTH-1), the Statistical Core Design method (Reference TH-3), and the BHTP and BWU-N CHF correlations (References TH-2 and TH-15, respectively). Using these methods and the full core Adv. W17 HTP model, it was shown that the existing Core Safety Limit Lines needed to be tightened to maintain adequate DNB protection at the limits. This has resulted in a revision of Technical Specification Figure 2.1-1. An evaluation of the existing Overtemperature ΔT and Overpower ΔT functions showed that even with the reduced CSL lines adequate protection is being provided by the existing trip function, therefore, no change to the trip function definitions is required. A comparison of the existing Core Safety Limits to the new Adv. W17 HTP based Core Safety Limits is provided in Figure 4-1.

Technical Specification Table 2.2-1
Revise footnote to reflect an increase in thermal design flow.

Technical Specification Figure 3.2-1
Revise figure to reflect an increase in thermal design flow.

Technical Specification 6.9.1.14.a Core Operating Limits Report
The following topical reports, which present reviewed and approved T-H analytical methods, are added:

BAW-10241P-A Rev. 01, “BHTP DNB Correlation Applied with LYNXT”

BAW-10199P-A “The BWU Critical Heat Flux Correlations”

BAW-10159P-A, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies.

BAW-10189P-A, CHF Testing and Analysis of the Mark-BW Fuel Assembly Design.

These applicable and previously approved Critical Heat Flux (CHF) reports are added to provide a more comprehensive COLR reference list. These approved methods support the following Technical Specification limits: IC-DNB MAP limits, AFD Limits, f1(ΔI) Limits.

Technical Specification 2.1 Safety Limits Bases
Text is being revised to a form similar to the Improved Standard Technical Specifications. Note: the modification in the Technical Specification format is not related to the fuel transition.

Technical Specification 3/4.2.5 Bases
Delete last sentence which is no longer needed with the increase in thermal design flow.

4.7 Thermal-Hydraulic Conclusions

This section has outlined the thermal-hydraulic licensing methods and models that are used to justify the transition from the AREVA's Mark-BW fuel design to AREVA's Adv. W17 HTP fuel design at the Sequoyah Nuclear Plant. As this section has shown, during the transition to Adv. W17 HTP fuel, thermal-hydraulic safety and operating limits will be defined using a full core Adv. W17 HTP analysis, with transition core effects evaluated with appropriately bounding mixed core models. The thermal-hydraulic methods, models, and assessments discussed in this section form the basis by which AREVA will demonstrate that the Adv. W17 HTP is compatible with the Mark-BW in the Sequoyah core and that all operational design requirements will be met during transition cycles and during full core operation.

4.8 REFERENCES For Section 4.0

TH-1 BAW-10156-A Revision 1, LYNXT: Core Transient Thermal-Hydraulic Program, August 1993.

TH-2 BAW-10241P-A Rev. 01, BHTP DNB Correlation Applied with LYNXT, July 2005.

TH-3 BAW-10170P-A, Statistical Core Design For Mixing Vane Cores, December 1988.

TH-4 BAW-10243P-A, Statistical Fuel Assembly Hold Down Methodology, September 2005.

TH-5 Letter, K.E. Suhrke (B&W) to Mr. S.A. Varga (NRC), December 6, 1976.

TH-6 Letter, S.A. Varga to J.H. Taylor, Update of BAW-10055, "Fuel Densification Report," December 5, 1977.

TH-7 BAW-10147PA-Rev. 1, Fuel Rod Bowing in Babcock & Wilcox Fuel Designs - Revision 1, May 1983.

TH-8 BAW-10159P-A, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, July 1990.

TH-9 BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWR's, June 1989.

TH-10 BAW-10189P-A, CHF Testing and Analysis of the Mark-BW Fuel Assembly Design, January 1996.

TH-11 BAW-10220P Rev. 0, Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 and 2, March 1996.

TH-12 Letter, D. M. Crutchfield (NRC) to James H. Taylor (B&W), "Acceptance for Referencing of a Special Licensing Report," December 5, 1986.

TH-13 BAW-10054, Rev. 2, Fuel Densification Report, May 1973.

TH-14 BAW-10183P-A, Fuel Rod Gas Pressure Criterion (FRGPC), July 1995.

TH-15 BAW-10199P-A The BWU Critical Heat Flux Correlations, August 1996.

5.0 ACCIDENT ANALYSES

The LOCA and non-LOCA events and accident system analyses are discussed in this section. A description of the analytical methodology, computer codes, and event scenarios and results/dispositions are presented.

5.1 Introduction

This section provides information related to assessing the Sequoyah nuclear power plants transient and accident analyses for the proposed transition from AREVA Mark-BW fuel to AREVA Adv. W17 HTP fuel. It includes a brief description of methodology used in the past by AREVA to evaluate the original transition to AREVA fuel. Also, a discussion from the standpoint of UFSAR Chapter 15 is included which concludes that the safety analysis for Sequoyah remains valid for the transition to AREVA Adv. W17 HTP fuel with the proposed changes.

5.2 Non-LOCA Transients

5.2.1 Analysis Methodology and Computer Codes

The AREVA NP methodology for evaluating non-LOCA transients is described in (Reference 3). The non-LOCA analysis methodologies to be applied for the Sequoyah fuel transition have been previously reviewed and approved by the NRC (Reference 9).

Other than the codes mentioned in section 2, 3, and 4, the principal computer code used by AREVA to perform non-LOCA transient analyses for Sequoyah is RELAP5/MOD2-B&W. A description of the code is provided below.

RELAP5/MOD2-B&W

RELAP5/MOD2-B&W is an adaptation of the Idaho National Engineering Laboratory code RELAP5/MOD2. The code, developed for best-estimate transient simulation of pressurized water reactors, has been modified to include models required for licensing analysis. Modeling capabilities encompass the analysis of operational transients defining the safety envelope of a reactor. The code has been benchmarked extensively to existing experimental data for regulatory approval of its use in analyzing Non-LOCA transients. RELAP5/MOD2-B&W is documented in topical BAW-10164P-A (Reference 2).

5.2.2 Event Dispositions

The Sequoyah UFSAR Chapter 15 event analyses are listed in Table 5-1 with a cross-reference to the corresponding SRP section. A review of each event was conducted relative to the transition from AREVA Mark-BW fuel to AREVA Adv. W17 HTP fuel, which determined that no plant simulations are needed to be re-analyzed as a result of the transition. An evaluation of acceptable margins to specified acceptance fuel design limits (SAFDLs) is performed on a reload cycle basis. The events are listed below with a more detailed event-by-event disposition of the challenge to the design change.

Several of the arguments needed in the evaluation of the non-LOCA safety analyses for the Adv. W17 HTP fuel assemblies are generically applicable - independent of the class of transient. These discussions are listed below.

<u>System Modeling Inputs</u>

Non-LOCA accident analysis sequences and system responses are largely a function of system configuration, plant control, and design. Input assumptions regarding:

- initial system conditions such as core power, core flow rate, RCS temperature and inventory
- system geometry inputs, including component sizes, pipe diameters, system pressure drops
- automated and operational controls like trip setpoints, electronic signal compensation, valve and pump ratings

have the potential to affect the results of the non-LOCA transients and the margin to acceptance criteria for a given event. The introduction of Adv. W17 HTP fuel assemblies has the following effects on these system parameters.

The Adv. W17 HTP fuel design has no effect on core power, Reactor Coolant System (RCS) temperature, or system geometry and inventory outside the core region. The Adv. W17 HTP fuel has a higher pressure drop due to the presence of flow mixing grids and differences in grid design and end fittings. The higher pressure drop could impact total core flow rate, core bypass flow rate, and flow coastdown characteristics when the primary coolant pumps lose power. The effect on total core flow has been examined (Section 4.3.5) and flow is expected to remain above the minimum initial core flow rate assumed in the current safety analyses. The effect on core bypass flow rate has been examined (Section 4.2.9) and the increase determined to be minor [ ]. The increased bypass flow will result in negligibly higher core coolant and fuel temperatures, but no change to coolant loop temperatures. The effect of the increase in core moderator and fuel temperatures will be reflected in the neutronics calculations used to verify that cycle specific reactivity feedback parameters fall within the current non-LOCA analysis assumptions. The effect of the additional core bypass on DNB will also be considered in the cycle specific analyses performed to address DNB criteria.

System analyses predict core power dynamics, and system flow, temperature, and pressure responses subsequent to event initiation. Although the system models may include core bypass components, the change proposed - about a [ ] increase in bypass flow relative to the [ ] value assumed in the system analyses is not sufficient to perturb these models to a significant extent. Any small changes in reactor vessel flow, pressure, and temperature distribution interior to the reactor vessel associated with the transition to Adv. W17 HTP fuel are well within the expected calculational accuracy of the system models. Since setpoints are normalized to ΔT_0, plant simulations are more independent of core flow. Since the RCS flows are not significantly affected by the change, there is no change to the heat transfer characteristics of the system model. Therefore transient responses generated by the existing system models are equally applicable to SQN operation with Adv. W17 HTP fuel.

The Adv. W17 HTP fuel has identical fuel rod cladding dimensions and identical fuel rod arrangement, so the open core system geometry is identical to the current core design. Minor geometry changes occur in grids and end fittings, as well as in the exterior dimensions of the guide tube dashpot region. These dimensional differences are negligible with respect to overall system inventory and for non-LOCA system transient models and would not influence the previously calculated transient response.

The Adv. W17 HTP fuel has no direct effect on automated and operational controls like trip setpoints, electronic signal compensation, valve and pump ratings.

The minimum RCS coolant flow assumed in the analyses is expected to remain unaffected by the introduction of Adv. W17 HTP fuel. Relative to the current analysis of record, coincident with the introduction of Adv. W17 HTP fuel, the plant will be operating with replacement steam generators (RSGs) with a reduced pressure drop relative to the original steam generators. Thus the loop flow resistance is reduced relative to the current UFSAR analysis assumptions. The transition to Adv. W17 HTP fuel has a small effect of additional core bypass and higher core pressure drop. The effect of the increased reactor vessel pressure drop due to the Adv. W17 HTP fuel on RCP flow coastdown characteristics is expected to be negligible when considered in conjunction with the installation of the RSGs and the attendant reduction in steam generator pressure drop.

Thus, the general system response to a non-LOCA accident is unaffected by the Adv. W17 HTP fuel.

Reactivity Feedback Inputs

Adv. W17 HTP fuel rods are essentially identical to the Mark-BW fuel rods, with the exception of cladding material. Fuel pellet fabrication and the mode and method of operating the Adv. W17 HTP fuel are the same as current Mark-BW fuel. Nuclear analyses performed as part of reload licensing requirements verify that the current limits in reactivity feedback parameters - rod worth, boron worth, moderator and Doppler feedback – used in Mark-BW safety analyses bound the behavior of the reload core design. This validation will ensure that Mark-BW safety analysis remains applicable for the core design containing Adv. W17 HTP fuel.

The Adv. W17 HTP fuel assembly design employs MONOBLOC™ guide tubes. As noted in Section 2.2.6, key guide tube dimensions are similar to the Mark-BW fuel and the difference in rod drop times is negligible. Post-trip rod drop times used in current Sequoyah safety analyses are, therefore, equally applicable to the Adv. W17 HTP fuel assembly.

Fuel Initial Stored Energy

Initial fuel stored energy - fuel and clad temperature - is an important input to non-LOCA safety calculations. Transients initiated from zero power assume fuel temperatures that are initially in equilibrium with the RCS temperature independent of fuel type. Transients initiated at power, however, require an estimate of the initial fuel temperature based on power, fuel pin dimension, and material properties.

The initial stored energy at power for the two assembly designs is assessed by considering cladding characteristics and fuel rod power density. There is no difference in fuel rod dimensions (Table 2-2) or material, thus there is no effect on the energy present in the Adv. W17 HTP fuel rods relative to the current design. Regarding fuel power density, the fuel pellet radius (and hence, assembly loading) are identical in the Adv. W17 HTP fuel relative to the Mark-BW assembly, thus there is no difference in power density when operated at the same power output. Consequently, there is no significant change in the amount of stored energy in both the clad and fuel for the Adv. W17 HTP fuel assembly. Minor differences in fuel assembly end fittings and spacer grids (including the presence of IFM grids) have negligible effect on fuel stored energy for non-LOCA events. Thus, the fuel initial stored energy for the Mark-BW assembly remains applicable to the Adv. W17 HTP fuel assembly design.

Decay Heat

Long-term events are typically analyzed to assure the plant cooling capacity - e.g., secondary liquid inventories and feedwater flow - is sufficient to remove core decay heat. Fuel pellet material and operational characteristics - uranium enrichment, fuel cycle length, linear heat rates - are, by design, identical for both Adv. W17 HTP and Mark-BW fuels. Therefore, the decay heat models used in Mark-BW safety analyses remain applicable to the Adv. W17 HTP fuel assemblies.

Fuel Transient Thermal Response

Fuel material properties dictate transient fuel mechanical and thermal behavior during an accident. Clad and fuel material properties remain the same in the Adv. W17 HTP and the Mark-BW assembly fuel rods. The Adv. W17 HTP fuel has essentially identical heat capacitance/thermal inertia as the Mark-BW fuel since the dimensions of the fuel rod cladding and fuel loading are similar, the densities of M5 and Zirc-4 cladding are similar, and differences between the M5 and Zirc-4 cladding thermal properties are relatively small.

Therefore, there is negligible change in the transient core thermal response to a non-LOCA transient analysis with the introduction of Adv. W17 HTP fuel.

DNB

Thermal hydraulic and nuclear analyses are performed to confirm compliance with DNB criteria on a cycle specific basis through validation and adherence to radial and total power peaking limits.

Peak Clad Temperature

Thermal hydraulic and nuclear analyses are performed to address peak clad temperature criteria on a cycle specific basis through validation and adherence to local radial and total power peaking limits.

5.2.2.1 Event Disposition for Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From A Subcritical Condition (UFSAR § 15.2.1)

5.2.2.1.1 Event Description

The uncontrolled withdrawal of a rod cluster control assembly (RCCA) bank could be caused by a malfunction in the reactor control or rod control systems. This could occur with the reactor either subcritical, at hot zero power or at power. The "at power" case is discussed in Section 5.2.2.2. The malfunction could lead to a large and rapid positive reactivity addition, resulting in a power transient which challenges the DNBR and fuel centerline melt (FCM). The event was assumed to be initiated from hot zero power.

The rapid increase of the neutron flux which results from the bank withdrawal is countered by the reactivity feedback effect of the negative Doppler coefficient. This inherent self-limitation of the power excursion is of primary importance, because it limits the power to a tolerable level during the delay time for protective action. Although the nuclear power peaks at a very high level during the rapid excursion, the duration is short enough to preclude significant energy deposition. The fuel rod surface heat flux lags behind the nuclear power level but still peaks at a significant fraction of the rated-power value. The increase in the primary coolant temperatures, in turn, lags behind the increase in the fuel rod heat flux.

The reactor protection system (RPS) is designed to terminate the transient before the DNBR limit is reached. The principal protective trip for this event is the power range high neutron flux (low setting).

5.2.2.1.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions

- Maximum differential worth for RCCAs moving in sequence
- Maximum RCCA withdrawal rate
- Doppler reactivity feedback
- Trip setpoint(s), uncertainty and delay time
- Number of reactor coolant pumps (RCPs) running
- Fuel rod gap conductance
- Maximum FQ predicted for the purpose of calculating the peak (hot spot) fuel centerline temperature

5.2.2.1.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. This event does not provide a significant challenge to peak pressure. Therefore, the principally challenged acceptance criteria for this event are:

1. Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.
2. Fuel centerline melting shall not occur.

5.2.2.1.4 Event Disposition

System modeling inputs such as the initial operating conditions, trip setpoint(s), uncertainty and delay time, and number of RCPs running are not affected by the Adv. W17 HTP fuel for this event. The initial fuel and clad temperatures for the Mark-BW fuel remains applicable to the Adv. W17 HTP fuel due to identical fuel rod dimensions and material for both Mark-BW fuel and Adv. W17 HTP fuel (Table 2-2).

The modeling of decay heat is of secondary importance in analyzing this reactivity anomaly event. Decay heat models are equally applicable to Adv. W17 HTP fuel and Mark-BW fuel. Fuel transient thermal responses predicted in response to reactivity anomaly remain unchanged by the Adv. W17 HTP fuel because of the limited power responses for this event.

Limiting reactivity feedback inputs such as maximum differential RCCA worth and Doppler reactivity feedback are used to maximize power response following the initiation of this event. These inputs are expected to bound the Adv. W17 HTP fuel. Moreover, nuclear analyses will be performed as part of reload licensing requirements to verify that the current limits in the parameters - trip worth delayed neutron fraction, moderator and Doppler feedback – used in Mark-BW safety analyses bound the behavior of the reload core design. Margin to DNB will also be demonstrated as part of reload licensing based on the peaking for the core design with Adv. W17 HTP fuel.

Therefore, the analysis of the uncontrolled withdrawal of a rod cluster control assembly bank from a subcritical condition remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.2 Event Disposition for Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power (UFSAR § 15.2.2)

5.2.2.2.1 Event Description

The uncontrolled withdrawal of a rod cluster control assembly (RCCA) bank could be caused by a malfunction in the reactor control or rod control systems. This could occur with the reactor either subcritical, at hot zero power or at power. The "subcritical" case is discussed in Section 5.2.2.1. The malfunction could lead to a large and rapid positive reactivity addition, resulting in a power transient which challenges the DNBR and fuel centerline melt (FCM). The event was assumed to be initiated at hot full power.

The increase of the neutron flux resulting from the bank withdrawal is followed by a rise in thermal power, with the thermal power lag determined by the reactivity insertion rate of the RCCA withdrawal. The positive reactivity addition results in a power transient, increasing the primary coolant temperatures and core heat flux and decreasing the margin to the DNB and FCM.

The reactor protection system (RPS) is designed to terminate the transient before the DNBR limit is reached. The principal protective trip for this event is the power range high neutron flux and overtemperature ΔT trip.

5.2.2.2.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- Maximum differential worth for RCCAs moving in sequence
- Maximum RCCA withdrawal rate
- Doppler reactivity feedback
- Moderator reactivity feedback
- Trip setpoint(s), uncertainty and delay time

5.2.2.2.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. This event does not provide a significant challenge to peak pressure. Therefore, the principally challenged acceptance criteria for this event are:

1. Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.

2. Fuel centerline melting shall not occur.

5.2.2.2.4 Event Disposition

System modeling inputs such as the initial operating conditions, trip setpoint(s), uncertainty and delay time are not affected by the Adv. W17 HTP fuel for this event. The initial fuel and clad temperatures for the Mark-BW fuel remains applicable to the Adv. W17 HTP fuel due to identical fuel rod dimensions and material for both Mark-BW fuel and Adv. W17 HTP fuel (Table 2-2).

The modeling of decay heat is of secondary importance in analyzing this reactivity anomaly event. Decay heat models are equally applicable to Adv. W17 HTP fuel and Mark-BW fuel. Fuel transient thermal responses predicted in response to reactivity anomaly remain unchanged by the Adv. W17 HTP fuel because of the limited power responses for this event.

Limiting reactivity feedback inputs such as maximum differential RCCA worth and Doppler reactivity feedback are used to maximize power response following the initiation of this event. These inputs are expected to bound the Adv. W17 HTP fuel. Moreover, nuclear analyses will be performed as part of reload licensing requirements to verify that the current limits in reactivity feedback parameters - rod worth, moderator and Doppler feedback – used in Mark-BW safety analyses bound the behavior of the reload core design. As part of the maneuvering analysis, the margin to DNB and fuel centerline melt will also be demonstrated for the reload design.

Therefore, the analysis of the uncontrolled withdrawal of a rod cluster control assembly bank at power condition remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.3 Event Disposition for Rod Cluster Control Assembly Misalignment (UFSAR § 15.2.3)

5.2.2.3.1 Event Description

Rod cluster control assembly misalignment accidents include:

1. A dropped full-length assembly;

2. A dropped full-length assembly bank;

3. Statically misaligned full length assembly.

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion that may be detected by the power range negative neutron flux rate trip circuitry. For those dropped RCCAs that do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power.

A dropped RCCA bank results in a relatively large reactivity insertion which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is tripped fairly quickly. Consequently, the core is not adversely affected.

The maximum statically misaligned RCCA occurs when a single RCCA in a control bank is either fully inserted or fully withdrawn. The resulting increase in local core peaking may challenge the DNB operating limits. Typically, the statically misaligned RCCA event is less limiting than the dropped RCCA event. The analysis assumes the event is initiated from hot full power conditions.

5.2.2.3.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- Moderator reactivity feedback
- Worth of dropped rod

5.2.2.3.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. This event does not provide a significant challenge to peak pressure. Therefore, the principally challenged acceptance criteria for this event are:

1. Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.
2. Fuel centerline melting should not occur.

5.2.2.3.4 Event Disposition

System modeling inputs such as the initial operating conditions, trip setpoint(s), uncertainty and delay time are not affected by the Adv. W17 HTP fuel for this event. The initial fuel and clad temperatures for the Mark-BW fuel remains applicable to the Adv. W17 HTP fuel due to identical fuel rod dimensions and material for both Mark-BW fuel and Adv. W17 HTP fuel (Table 2-2).

The modeling of decay heat is of secondary importance in analyzing this reactivity anomaly event. Decay heat models are equally applicable to Adv. W17 HTP fuel and Mark-BW fuel. Fuel transient thermal responses predicted in response to reactivity anomaly remain unchanged by the Adv. W17 HTP fuel as discussed previously.

Limiting reactivity feedback inputs such as maximum differential RCCA worth and Doppler reactivity feedback are used to maximize power response following the initiation of this event. These inputs are expected to bound the Adv. W17 HTP fuel. Moreover, nuclear analyses will be performed as part of reload licensing requirements to verify that the current limits in the key parameters - rod worth, dropped rod worth, moderator and Doppler feedback – used in Mark-BW safety analyses bound the behavior of the reload core design. Margin to DNB due to the increased peaking from the asymmetry will also be demonstrated as part of reload licensing as discussed in Reference 3.

Therefore, the analysis of the rod cluster control assembly misalignment remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.4 Event Disposition for Uncontrolled Boron Dilution (UFSAR § 15.2.4)

5.2.2.4.1 Event Description

An uncontrolled boron dilution may be caused by a malfunction or an inadvertent operation of the chemical and volume control system (CVCS) that results in a dilution of the active portion of the RCS. A dilution of the RCS can be the result of adding primary grade water into the RCS. Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution.

The analysis of the uncontrolled boron dilution covers refueling, startup, and power operation. In all cases, operator action is required to secure the dilution to prevent a loss of shutdown margin. An uncontrolled boron dilution during full power operation is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power, which can approach the DNBR, FCM, and the RCS pressure limits.

Under the worst conditions, the operator has adequate time from the time of initiation of the event to secure the dilution to prevent losing the minimum shutdown margin. The DNBR, FCM, and the RCS pressure limit criteria will be met if the entire shutdown margin is not lost.

5.2.2.4.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions

- Initial boron concentration
- Critical boron concentration
- Makeup water pump capacity
- RCS water volume

5.2.2.4.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. As long as the reactor remains sub-critical then overpressure and event progression are not limiting. Therefore, the principally challenged acceptance criterion for this event is:

Fuel cladding integrity should be maintained by ensuring that the minimum calculated DNBR is not less than the 95/95 DNB correlation limit.

5.2.2.4.4 Event Disposition

The system simulation of an uncontrolled boron dilution during full power operation is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power. A system analysis of the uncontrolled boron dilution is not required to support the transition to the Adv. W17 HTP fuel. Therefore, an analysis of the uncontrolled boron dilution during full power operation is not required.

The key parameters listed for boron dilution analysis in evaluating shutdown margin and required operator action time are not affected by the transition to the Adv. W17 HTP fuel, because they are not fuel related parameters. Furthermore, a cycle specific check will be performed to demonstrate that adequate time is available for operator action to prevent loss of shutdown margin.

Therefore, an analysis of the uncontrolled boron dilution is not required to support the transition to the Adv. W17 HTP fuel.

5.2.2.5 Event Disposition for Partial Loss of Forced Reactor Coolant Flow (UFSAR § 15.2.5)

5.2.2.5.1 Event Description

A partial loss of coolant flow may be caused by a mechanical or electrical failure in a pump motor, a fault in the power supply to the pump motor, or a pump motor trip caused by such anomalies as over-current or phase imbalance. The partial loss of coolant flow event is a less severe transient than the complete loss of coolant flow (UFSAR § 15.3.4) event due to the smaller flow reduction.

A decrease in reactor coolant flow occurring while a plant is at power results in a degradation of core heat transfer, reduction in DNBR margin, and a challenge to the DNB. The reduction in primary system flow and associated increase in core coolant temperatures result in a reduction in DNBR margin. The increasing primary system coolant temperatures also results in expansion of the primary coolant volume, causing an insurge into the pressurizer and an increase in the pressure of the primary system. However, the overpressure transient response for this event is bounded by the loss of external electrical load and/or turbine trip event (UFSAR § 15.2.7) due to the rapid loss of primary-to-secondary heat transfer.

The MDNBR is controlled by the interaction of the primary coolant flow decay, the trip signal, the trip signal generation delay time, the scram delay time, the core power decrease following reactor trip, and the rod surface heat flux. The power-to-flow ratio initially increases, peaks, and then declines as the challenge to the DNB is mitigated by the decline in core power due to the reactor trip.

5.2.2.5.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- RCP coastdown rate (pump inertia and pump frictional torque)
- Trip setpoint(s), uncertainty and delay time
- Minimum scram worth
- Fraction of scram reactivity versus fraction of control rod insertion distance and delay time
- Fuel rod gap conductance

5.2.2.5.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. This event does not provide a significant challenge to peak pressure and the FCM is not challenged because there is no significant increase in power for this event. Therefore, the principally challenged acceptance criterion for this event is:

Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.

5.2.2.5.4 Event Disposition

System modeling inputs such as the initial operating conditions, RCP trip setpoint(s), uncertainty and delay time are not affected by the Adv. W17 HTP fuel for this event. The RCS flow and RCP coastdown would be slightly affected in an adverse direction due to the increased fuel assembly pressure drop of the Adv. W17 HTP fuel; however, the unit will have replacement steam generators installed prior to HTP reloads, and the combined effect is an overall decrease in RCS loop resistance and a consequent increase in the RCS flow and RCP coastdown. The initial fuel and clad temperatures for the Mark-BW fuel remains applicable to the Adv. W17 HTP fuel due to identical fuel rod dimensions and material for both Mark-BW fuel and Adv. W17 HTP fuel (Table 2-2).

Loss of flow is short in duration and the modeling of decay heat is relatively unimportant. Decay heat models are equally applicable to Adv. W17 HTP fuel and Mark-BW fuel. Fuel transient thermal responses remain unchanged by the Adv. W17 HTP fuel for the flow coastdown transient due to the limited flow responses for this event.

Limiting reactivity feedback inputs such as the minimum scram worth and MTC are used to maximize power response following the initiation of this event. These inputs are expected to bound the Adv. W17 HTP fuel. Moreover, nuclear analyses will be performed as part of reload licensing requirements to verify that the current limits in reactivity feedback parameters - trip worth, moderator and Doppler feedback – used in Mark-BW safety analyses bound the behavior of the reload core design. The maneuvering analysis will verify adequate margin to DNB as part of reload licensing.

Therefore, the analysis of the partial loss of forced reactor coolant flow remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.6 Event Disposition for Startup Of An Inactive Reactor Coolant Loop (UFSAR § 15.2.6)

5.2.2.6.1 Event Description

This event is initiated by starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature. The startup of an inactive reactor coolant loop may be caused by an operational error.

Before the initiation of the startup of an inactive coolant pump, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature, because there is a temperature drop across the steam generator in the inactive loop and with the reverse flow, if the reactor is operated at power. Therefore, this event would result in the injection of cold water into the core which causes a rapid reactivity insertion and subsequent power increase.

The principal protective trip for this event is the low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint, which would have been previously reset for three loop operation.

5.2.2.6.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- RCS pressure drop
- RCS flow distribution
- Trip setpoint(s), uncertainty and delay time
- Minimum scram worth
- Fraction of scram reactivity versus fraction of control rod insertion distance and delay time

5.2.2.6.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. This event does not provide a significant challenge to peak pressure and the FCM is not challenged because there is no significant increase in power for this event. Therefore, the principally challenged acceptance criterion for this event is:

Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.

5.2.2.6.4 Event Disposition

System modeling inputs such as the initial operating conditions, and trip setpoint(s), uncertainty and delay time are not affected by the Adv. W17 HTP fuel for this event. Any small changes in the core pressure drop, core bypass flow fraction, and core inlet flow distribution interior to the reactor vessel associated with the transition to Adv. W17 HTP fuel are well within the expected calculation accuracy of the system models. For this event, small perturbations in those parameters do not affect the analysis of record.

The initial fuel and clad temperatures for the Mark-BW fuel remains applicable to the Adv. W17 HTP fuel due to identical fuel rod dimensions and material for both Mark-BW fuel and Adv. W17 HTP fuel (Table 2-2).

Startup of an inactive reactor coolant loop is short in duration and the modeling of decay heat is relatively unimportant. Decay heat models are equally applicable to Adv. W17 HTP fuel and Mark-BW fuel. Fuel transient thermal responses remain unchanged by the Adv. W17 HTP fuel for the flow transient due to the limited flow responses for this event.

Limiting reactivity feedback inputs such as the minimum scram worth, MTC, Doppler feedback are used to maximize power response following the initiation of this event. These inputs are expected to bound the Adv. W17 HTP fuel.

Therefore, the analysis of the startup of an inactive reactor coolant loop remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.7 Event Disposition for Loss of External Electrical Load and/or Turbine Trip (UFSAR § 15.2.7)

5.2.2.7.1 Event Description

The loss of external electrical load and/or turbine trip are characterized by a decrease in heat removal by the secondary system caused by either a direct turbine trip or following a loss of external electrical load. For either case, off site power remains available for the continued operation of plant components such as the reactor coolant pumps. The case of loss of all AC power (station blackout) is analyzed in UFSAR § 15.2.9. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur.

Steam pressure and temperature increase significantly as the kinetic energy of flowing steam is changed to pressure and internal energy, and as thermal energy from the RCS continues to be transferred to the steam generators. The higher secondary side temperature causes the RCS temperature to increase. Both the maximum steam pressure and its rate of increase are dependent on the amount of steam relief capacity available. For safety analysis, no credit is taken for the non-safety steam dump system, steam pressure is relieved only by the main steam safety valves (MSSVs). The main feedwater (MFW) may be conservatively assumed to be isolated at event initiation. As energy continues to be transferred from the RCS to the secondary side of the steam generators, the steam generator pressure increases rapidly until successive stages of the MSSVs open to mitigate the increase in pressure.

The RCS temperature and pressure increase continues until a RPS setpoint is reached and a reactor trip occurs. Coolant thermal expansion causes a rapid insurge into the pressurizer, increasing pressurizer pressure and level. Reactor trip is actuated by the first RPS trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, high pressurizer water level, low reactor coolant loop flow, reactor coolant pump power supply undervoltage, and low-low steam generator water level.

The minimum DNBR for a total loss of load transient is bounded by the value calculated for a complete loss of forced reactor coolant flow (UFSAR § 15.3.4). Consequently, the analysis of total loss of load is performed to show the adequacy of the pressure relieving devices on the primary and secondary systems. Two loss of load cases are analyzed. These are a loss of load from 102 percent of full power and a total loss of load from 52 percent of full power. The loss of load from 102 percent of full power is more limiting than the total loss of load from 52 percent of full power in terms of overpressure event for the primary and secondary.

5.2.2.7.2 Key Parameters

The key parameters for this event are:

- Initial core power
- Trip setpoint(s), uncertainty and delay time
- Primary safety relief valve setpoint and capacity (for the RCS overpressurization case)
- MSSV setpoints and capacities
- RCP coastdown rate (52% power with fast bus transfer failure)

5.2.2.7.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. The principally challenged acceptance criteria for this event are:

1. The pressures in the reactor coolant and main steam systems should be less than 110% of design values.
2. Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.
3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently (may challenge pressurizer overfill).

5.2.2.7.4 Event Disposition

The event behavior is predominantly a function of the primary-to-secondary heat transfer capability. Therefore, small perturbations in parameters such as the core pressure drop, core bypass flow fraction, core inlet flow distribution, RCP coastdown rate, and reactivity feedback do not impact the parameters of interest in assessing the acceptance criteria. The plant system characteristics that potentially impact the key parameters listed for this event such as the initial operating conditions, trip setpoint(s), uncertainty and delay time, primary safety relief valve setpoint and capacity, MSSV setpoints and capacities remain unchanged for both the Mark-BW fuel and the Adv. W17 HTP fuel.

Fuel initial stored energy, dependent on fuel parameters, is an important modeling characterization for this event. The initial fuel and clad temperatures for the Mark-BW fuel remains applicable to the Adv. W17 HTP fuel due to identical fuel rod dimensions and material for both Mark-BW fuel and Adv. W17 HTP fuel (Table 2-2).

This event is short in duration and the modeling of decay heat is relatively unimportant. Decay heat models are equally applicable to Adv. W17 HTP fuel and Mark-BW fuel. Fuel transient thermal responses predicted for the event remain unchanged by the Adv. W17 HTP fuel because the limited power responses for this event.
Limiting reactivity feedback inputs such as the minimum scram worth and MTC are used to maximize power response following the initiation of this event. These inputs typically bound the Adv. W17 HTP fuel.

The cause of the event and the parameters that control the consequences of the event are unchanged from or bounded by the current analysis of record presented in UFSAR Section 15.2.7. Therefore, an analysis of the loss of external electrical load and/or turbine trip is not required to support the transition to the Adv. W17 HTP fuel.

5.2.2.8 Event Disposition for Loss of Normal Feedwater Flow Event (UFSAR § 15.2.8)

5.2.2.8.1 Event Description

The loss of normal feedwater flow is initiated by the termination of MFW flow which results from pump failures, valve malfunctions, or loss of offsite AC power. The worst postulated loss of normal feedwater event is one initiated by a loss of offsite AC power which is described in Section 5.2.2.9 (UFSAR § 15.2.9). This is due to the decreased capability of the reactor coolant to remove residual core heat as a result of the RCP coastdown.

The sudden loss of subcooled MFW flow, while the plant continues to operate at power, causes steam generator heat removal rates to decrease. This, in turn, causes reactor coolant temperatures to increase. The reactor coolant expands, surging into the pressurizer. The resulting increase in pressure actuates the pressurizer spray system and may cause the pressurizer PORVs to open.

Steam generator liquid levels, which have been steadily dropping since the termination of MFW flow, soon reach the low steam generator level reactor trip setpoint. This initiates a reactor scram, which ends the short-term-heatup phase of the event.

The automatic turbine trip at reactor scram and the continuing primary-to-secondary transfer of the decaying core power and the reactor coolant pump heat cause steam generator pressures to rapidly increase. When steam generator pressures and coolant temperatures have increased to the appropriate values, the steam dump system and/or the MSSVs serve to limit the increase in steam generator pressures. However, credit is typically not taken for the steam dump system since it is not safety grade.

Steam generator levels continue to drop and soon reach the low-low steam generator level AFW actuation setpoint. This initiates the starting sequence for the AFW pumps. When the delivery of AFW begins, the rate of level decrease in the fed steam generators slows.
Eventually, a long-term-heatup phase of the event may begin if primary-to-secondary heat transfer degrades as a result of steam generator tube uncovery. If AFW is not being delivered to one of the steam generators, that steam generator may completely dry out.

As the decay heat level drops, liquid levels in the fed steam generators stabilize and then begin to rise. Also, reactor coolant temperatures stabilize and then begin to decrease. These conditions mark the end of the challenge to the event acceptance criteria.

5.2.2.8.2 Key Parameters

The key parameters for this event are:

- Initial core power (decay heat)
- Initial pressurizer level
- Trip setpoint(s), uncertainty and delay time
- Low SG water level reactor trip setpoint
- AFW actuation setpoint, minimum flow rate and actuation delay time
- RCS pump heat

- MSSV setpoints and capacity
- PSV setpoint and capacity
- Operator response time

5.2.2.8.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. The principally challenged acceptance criteria for this event are:

1. The pressures in the reactor coolant and main steam systems should be less than 110% of design values.
2. Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.
3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently (may challenge pressurizer overfill).

5.2.2.8.4 Event Disposition

The event behavior is predominantly a function of the primary-to-secondary heat transfer capability. Therefore, small perturbations in parameters such as the core pressure drop, core bypass flow fraction, core inlet flow distribution, and reactivity feedback do not impact the parameters of interest in assessing the acceptance criteria. The plant system characteristics that potentially impact the key parameters listed for this event such as the initial operating conditions, initial pressurizer level, trip setpoint(s), uncertainty and delay time, low SG water level reactor trip setpoint, AFW actuation setpoint, minimum flow rate and actuation delay time, RCS pump heat, primary safety relief valve setpoint and capacity, and MSSV setpoints and capacities remain unchanged for both the Mark-BW fuel and the Adv. W17 HTP fuel.

Fuel initial stored energy, dependent on fuel parameters, is an important modeling characterization for this event. The initial fuel and clad temperatures for the Mark-BW fuel remains applicable to the Adv. W17 HTP fuel due to identical fuel rod dimensions and material for both Mark-BW fuel and Adv. W17 HTP fuel (Table 2-2).

Decay heat modeling is important in the analysis of long-term overheating event. Decay heat models are equally applicable to Adv. W17 HTP fuel and Mark-BW fuel. Fuel transient thermal responses predicted for the event remain unchanged by the Adv. W17 HTP fuel because the limited power responses for this event.

Limiting reactivity feedback inputs such as the minimum scram worth and MTC are used to maximize power response following the initiation of this event. These inputs typically bound the Adv. W17 HTP fuel.

The cause of the event and the parameters that control the consequences of the event are unchanged from or bounded by the current analysis of record presented in UFSAR Section 15.2.8. Therefore, an analysis of the loss of normal feedwater is not required to support the transition to the Adv. W17 HTP fuel.

5.2.2.9 Event Disposition for Loss of Off-site Power to the Station Auxiliaries (UFSAR § 15.2.9)

5.2.2.9.1 Event Description

The design basis loss of off-site power to the station auxiliaries event is defined as loss of power to the plant auxiliaries, i.e., the reactor coolant pumps, feedwater pumps, condensate pumps, etc. when the reactor is at full power. The most limiting loss of off-site power event is one in which the reactor continues to operate at power until the steam generator level drops to the low-low level of zero percent of narrow range span, with a loss of power to RCPs at the time of reactor trip. This is an over-heating event. The event behavior is predominantly a function of the primary-to-secondary heat transfer capability.

The following signals provide the necessary protection for the loss of off-site power to the station auxiliaries event:

1. Steam generator low-low water level (0 percent NRS) signal
2. High pressurizer pressure signal.

The low-low level signal from steam generators will occur earlier than the high pressurizer pressure signal since the RCPs are running until the reactor trip and there is adequate secondary inventory initially to keep the primary system cooled. Since this event is not analyzed as an over pressurization event, the high pressurizer pressure signal is not used in the analysis.

This event is bounded in primary and secondary over pressurization by the loss of external electrical load event. The DNB is bounded by the complete loss of forced coolant flow event. The DNB is not evaluated. This event is analyzed to assure that no liquid loss will occur through the primary system relief valves and to assure that the minimum available auxiliary feedwater will cool the primary system, and the primary system will be shown not to saturate in this event. The major hazard associated with a loss of off-site power to station auxiliaries is the possibility of filling the pressurizer, allowing liquid to pass through the PORVs and the pressurizer safety valves, during the overheating phase of the event.
Analysis performed shows that the natural circulation flow in the RCS following a loss of AC power event is sufficient to remove residual heat from the core.

5.2.2.9.2 Key Parameters

The key parameters for this event are:

- Initial core power (decay heat)
- Initial pressurizer level
- Trip setpoint(s), uncertainty and delay time
- Low SG water level reactor trip setpoint
- AFW actuation setpoint, minimum flow rate and actuation delay time
- RCS pump heat
- MSSV setpoints and capacity
- PSV setpoint and capacity
- Condensate inventory
- Reactor Coolant Inventory
- RCPs coastdown rate
- RCS flow resistance
- Operator response time

5.2.2.9.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. The principally challenged acceptance criteria for this event are:

1. The pressures in the reactor coolant and main steam systems should be less than 110% of design values.

2. Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95/ DNB correlation limit.

5.2.2.9.4 Event Disposition

This event is bounded in primary and secondary over pressurization by the loss of external electrical load event. The DNB is bounded by the complete loss of forced coolant flow event. An analysis of either the loss of external electrical load event or the complete loss of forced coolant flow event is not required for the Adv. W17 HTP fuel. Consequently, an analysis of this event is not required for the Adv. W17 HTP fuel either for DNB or for over pressurization.

The event behavior is predominantly a function of the primary-to-secondary heat transfer capability. Therefore, small perturbations in parameters such as the core pressure drop, core bypass flow fraction, core inlet flow distribution, and reactivity feedback do not impact the parameters of interest in assessing other acceptance criteria (i.e., no liquid loss through the primary system relief valves, adequate available auxiliary feedwater, no RCS saturation, and adequate natural circulation flow).

The cause of the event and the parameters that control the consequences of the event are unchanged from or bounded by the current analysis of record presented in UFSAR Section 15.2.9. Therefore, an analysis of the loss of off-site power to the station auxiliaries is not required to support the transition to the Adv. W17 HTP fuel.

5.2.2.10 Event Disposition for Excessive Heat Removal Due to Feedwater System Malfunctions (UFSAR § 15.2.10)

5.2.2.10.1 Event Description

The excess heat removal due to feedwater system malfunctions is defined as an increase in heat removal from the primary side to the steam generator (SG) secondary side due to excessive feedwater flow, or a reduction in feedwater temperature. Excessive feedwater flow could be caused by a full opening of one or more feedwater regulator valves due to a feedwater control system malfunction or an operator error. The excess flow causes a greater demand on the RCS due to increased subcooling in the steam generators.

The system response to this event is that the RCS temperature and pressure will decrease. When there is a negative moderator temperature coefficient, a reactivity insertion occurs in the core in response to the decreasing core average temperature. This increases core power and the core average heat flux. Elevated cladding heat fluxes and fuel temperatures in the hot assembly may result in approaching to the DNBR limit.

The reactor protection system trips that provide the necessary protection for this event include the high neutron flux trip, overtemperature and overpower ΔT trips, and the turbine trip.

This event is not limiting in terms of core response due to overcooling. The accidental depressurization of the main steam safety system (Section 5.2.2.13) is the limiting overcooling event, and at no load conditions, the reactivity insertion rates are bounded by the rod withdrawal from subcritical event.

5.2.2.10.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions

- Moderator reactivity feedback
- Doppler reactivity feedback
- Trip setpoint(s), uncertainty and delay time
- Fuel rod gap conductance (HZP case)
- Maximum FQ predicted for the purpose of calculating the peak (hot spot) fuel centerline temperature (HZP case)

5.2.2.10.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. This event does not provide a significant challenge to peak pressure. Therefore, the principally challenged acceptance criteria for this event are:

1. Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.
2. Fuel centerline melting should not occur.

5.2.2.10.4 Event Disposition

The event is bounded by the accidental depressurization of the main steam safety system (Section 5.2.2.13) in terms of core response due to overcooling. At no load conditions, the reactivity insertion rates are bounded by the rod withdrawal from subcritical event (Section 5.2.2.1). Neither the accidental depressurization of the main steam safety system nor the rod withdrawal requires analysis for the transition to the Adv. W17 HTP fuel, an analysis of this event is therefore not required for the Adv. W17 HTP fuel.

The introduction of the Adv. W17 HTP fuel does not change the conclusion in the analysis of record that for this event, the steam generators do not pass bulk liquid into the steam lines, because steam generator overfill is a strong function of the feedwater flow rate, steam flow rate, and trip setpoints and delays, which are independent from fuel types.

Therefore, an analysis of the excessive heat removal due to feedwater system malfunctions is not required to support the transition to the Adv. W17 HTP fuel.

5.2.2.11 Event Disposition for Excessive Load Increase (UFSAR § 15.2.11)

5.2.2.11.1 Event Description

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

The increase in steam flow creates a mismatch between the energy being generated in the reactor core and the energy being removed by the secondary system and results in a cooldown of the primary system. A power increase will occur if the moderator temperature reactivity feedback coefficient is negative. If the power increase is sufficiently large, a reactor trip will occur. If the power increase is less significant, the reactor will stabilize at an elevated power level without reaching a reactor trip.

The event is protected by the overpower ΔT, overtemperature ΔT, or power range high neutron flux, which terminates the moderator feedback driven power excursion. As the cold water front enters the core, over-moderation will result in the core power distribution shifting towards the bottom of the core.

The event analyzed in the analysis of record is initiated from a 10% step load increase from rated load. As analyzed, no reactor trips occurred.

5.2.2.11.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- Magnitude of the step increase in load (i.e., the event initiator)
- Moderator reactivity feedback
- Trip setpoint(s), uncertainty and delay time

5.2.2.11.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. This event does not provide a significant challenge to peak pressure. Therefore, the principally challenged acceptance criteria for this event are:

1. Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.
2. Fuel centerline melting shall not occur.

5.2.2.11.4 Event Disposition

System modeling inputs such as the initial operating conditions, increase in load, and trip setpoint(s), uncertainty and delay time are not affected by the Adv. W17 HTP fuel for this event. The initial fuel and clad temperatures for the Mark-BW fuel remain applicable to the Adv. W17 HTP fuel due to identical fuel rod dimensions and material for both Mark-BW fuel and Adv. W17 HTP fuel (Table 2-2).

The modeling of decay heat is of secondary importance in analyzing this overcooling event. Decay heat models are equally applicable to Adv. W17 HTP fuel and Mark-BW fuel. Fuel transient thermal responses predicted for the overcooling event remain unchanged by the Adv. W17 HTP fuel because of the limited power responses for this event.

Limiting reactivity feedback inputs such as moderator feedback are used to maximize power response following the initiation of this event. These inputs are expected to bound the Adv. W17 HTP fuel. Moreover, nuclear analyses will be performed as part of reload licensing requirements to verify that the current limits in reactivity feedback parameters - moderator and Doppler feedback – used in Mark-BW safety analyses bound the behavior of the reload core design. The maneuvering analysis will verify adequate margin to DNB as part of reload licensing

Therefore, the analysis of the load increase remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.12 Event Disposition for Accidental Depressurization of the Reactor Coolant System (UFSAR § 15.2.12)

5.2.2.12.1 Event Description

The accidental depressurization of the reactor coolant system (RCS) event is initiated by the inadvertent opening of a reactor coolant system relief valve. The most severe core conditions resulting from an accidental depressurization of RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially the event results in a rapidly decreasing RCS pressure until this pressure reaches a value corresponding to the hot leg saturation pressure. At that time, the pressure decrease is slowed considerably. The average coolant temperature decreases slowly, but the pressurizer level increases until the reactor trip.

The RCS pressure decrease increases the potential for DNB because the RCS fluid approaches saturated conditions. The reactor will be tripped by either the pressurizer low pressure or overtemperature ΔT. The event analyzed is initiated at HFP, which bounds all power modes of operation. A zero moderator coefficient of reactivity and most negative Doppler coefficient are conservatively assumed.

5.2.2.12.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- Capacity of the stuck open safety valve
- Trip setpoint(s), uncertainty and delay time

5.2.2.12.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. This event does not provide a significant challenge to peak pressure and the fuel centerline melting is not challenged because there is no significant increase in power for this event. Therefore, the principally challenged acceptance criterion for this event is:

Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.

5.2.2.12.4 Event Disposition

The key parameters listed for this event are not impacted by the transition to the Adv. W17 HTP fuel, because the Adv. W17 HTP fuel does not affect the initial operating conditions, the capacity of the safety valves, or the trip setpoint(s) and delay time.

The initial fuel and clad temperatures for the Mark-BW fuel remains applicable to the Adv. W17 HTP fuel due to identical fuel rod dimensions and material for both Mark-BW fuel and Adv. W17 HTP fuel (Table 2-2).

Limiting reactivity feedback inputs such as moderator feedback and Doppler feedback are used to maximize power response following the initiation of this event. These inputs are expected to bound the Adv. W17 HTP fuel. The maneuvering analysis and setpoint verification will verify adequate margin to DNB for the safety limits and setpoints as part of reload licensing.

Therefore, the analysis of the accidental depressurization of the reactor coolant system remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.13 Event Disposition for Accidental Depressurization of the Main Steam System (UFSAR § 15.2.13)

5.2.2.13.1 Event Description

The accidental depressurization of the main steam system is initiated by an inadvertent opening of a single steam dump, relief or safety valve.
The steam release as a consequence of this accident results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin.
Safety injection, reactor trip, and main feedwater line isolation provide the necessary protection against an accidental depressurization.

5.2.2.13.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- Steam dump or safety valve capacity
- Moderator reactivity feedback
- Doppler reactivity feedback
- Trip setpoint(s), uncertainty and delay time
- Safety Injection flow rate and delay time
- Injected boron concentration

5.2.2.13.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. This event does not provide a significant challenge to peak pressure and the fuel centerline melting is not challenged because there is no significant increase in power for this event. Therefore, the principally challenged acceptance criterion for this event is:

Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.

5.2.2.13.4 Event Disposition

System modeling inputs such as the initial operating conditions, steam release valve capacity, trip setpoint(s), uncertainty and delay time, safety injection flow rate and delay time, and injected boron concentration are not affected by the Adv. W17 HTP fuel for this event

The modeling of decay heat is of secondary importance in analyzing this overcooling event. Decay heat models are equally applicable to Adv. W17 HTP fuel and Mark-BW fuel. Fuel transient thermal responses predicted for the overcooling event remain unchanged by the Adv. W17 HTP fuel because of the limited power responses for this event.

Limiting reactivity feedback inputs such as moderator feedback are used to maximize power response following the initiation of this event. These inputs typically bound the Adv. W17 HTP fuel. Moreover, nuclear analyses will be performed as part of reload licensing requirements to verify that the current limits in reactivity feedback parameters - moderator and Doppler feedback – used in Mark-BW safety analyses bound the behavior of the reload core design. The maneuvering analysis and setpoint verification will verify adequate margin to DNB for the safety limits and setpoints as part of reload licensing.

Therefore, an analysis of the accidental depressurization of the main steam system is not required to support the transition to the Adv. W17 HTP fuel.

5.2.2.14 Event Disposition for Spurious Operation of Safety Injection at Power (UFSAR § 15.2.14)

5.2.2.14.1 Event Description

The spurious operation of safety injection at power is assumed to occur by inadvertent initiation of borated water from the safety-grade emergency core coolant system injection source while the reactor is at rated full power. Spurious SIS operation at power could be caused by operator error or a false electrical actuating signal. An SIS normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the SIS will also produce a reactor trip. If trip does not occur at the same time spurious injection starts, the reactor will be tripped by the low pressurizer pressure later in the transient.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. DNB ratio is never less than the initial value.

5.2.2.14.2 Key Parameters

The key parameters for this event are:

- Charging flow rate
- Letdown flow rate
- Operator response time

5.2.2.14.3 Acceptance Criteria

This event is classified as a Condition II event, which is expected to occur no more often than once per year. The principally challenged acceptance criterion for this event is:

1. Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.

5.2.2.14.4 Event Disposition

The event behavior is predominantly a function of plant system capability, specifically the charging and letdown flow. The plant system characteristics that would affect the key parameters listed above remain unchanged for the Adv. W17 HTP fuel. The cause of the event and the parameters which control the consequences of the event are unchanged from or bounded by previous analysis. Therefore, an analysis of the spurious operation of safety injection at power event is not required for the Adv. W17 HTP fuel.

5.2.2.15 Event Disposition for Minor Secondary System Pipe Breaks (UFSAR § 15.3.2)

5.2.2.15.1 Event Description

Minor secondary system pipe breaks are breaks of a size equivalent to 6 inch diameter or less. These breaks must be accommodated with a limited failure of fuel elements. The effects of "major" secondary pipe breaks are bounding relative to all of the relevant safety margins.

5.2.2.15.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- Initial SG inventory
- Break size and location
- Moderator reactivity feedback
- Doppler reactivity feedback
- Trip setpoint(s), uncertainty and delay time
- Core power (NI & ΔT) signal decalibration
- AFW flow rate and delay time
- Safety Injection flow rate and delay time
- MSIV closure time
- MFW isolation time
- Post-scram radial power peaking factors

5.2.2.15.3 Acceptance Criteria

This is a Condition III event expected to occur infrequently. Condition III events are allowed to have some fuel failures so long as the site dose releases are within the 10CFR100 limits. In addition, these events may not cause failures that could lead to a worse, Condition IV, event.

5.2.2.15.4 Event Disposition

This event is not analyzed for Sequoyah since the response of the plant to these events is bounded by the analysis of "major" secondary system pipe breaks. Although the major secondary breaks are Condition IV events, they are analyzed to Condition II acceptance criteria. Assurance that all of the acceptance criteria for this event are successfully met for operation of Sequoyah with AREVA Adv. W17 HTP fuel is based on the disposition of the major secondary breaks in Sections 5.2.2.19 and 5.2.2.20.

5.2.2.16 Event Disposition for Inadvertent Loading of a Fuel Assembly into an Improper Position (UFSAR § 15.3.3)

5.2.2.16.1 Event Description

The arrangement of assemblies with different fuel enrichments in the core will determine the power distribution of the core during normal operation. The loading of fuel assemblies into improper core positions or the incorrect preparation of the fuel assembly enrichment could alter the power distribution of the core leading to potentially increased power peaking and possible violation of fuel thermal limits. The following fuel misloadings have been considered in the UFSAR:

- Misloading a fuel pellet or pellets with an incorrect enrichment in a fuel rod.
- Misloading a fuel rod with an incorrect enrichment in a fuel assembly.
- Misloading a fuel assembly with an incorrect enrichment or burnable poison rods into the core.

5.2.2.16.2 Key Parameters

The key parameters for this event are:

- Fuel fabrication administrative procedures
- Core loading administrative procedures
- Fuel assembly neutronic characteristics

5.2.2.16.3 Acceptance Criteria

This is a Condition III event expected to occur infrequently. Condition III events are allowed to have some fuel failures so long as the site dose releases are within the 10CFR100 limits. In addition, these events may not cause failures that could lead to a worse, Condition IV, event.

5.2.2.16.4 Event Disposition

The UFSAR contains an evaluation of the inadvertent loading of a fuel assembly into an improper position in Section 15.3.3. The evaluation concludes that:

- Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication.
- In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and cladding temperatures will be limited to the incorrectly loaded pin or pins.
- Fuel assembly loading errors are prevented by administrative procedures during core loading. In the unlikely event that a loading error occurs, resulting power distribution effects will either be readily detected by the incore moveable detector system or will cause a sufficiently small perturbation as to be acceptable within the uncertainties allowed between nominal and design power shapes.

These conclusions are unaffected by the introduction of AREVA Adv. W17 HTP fuel, which is neutronically similar to the AREVA Mark-BW fuel and Westinghouse fuel considered in the UFSAR analysis. The results of the UFSAR analyses are applicable to the HTP fuel. It is, therefore, assured that the acceptance criteria for this event are successfully met for operation with AREVA HTP fuel.

5.2.2.17 Event Disposition for Complete Loss of Forced Reactor Coolant Flow (UFSAR § 15.3.4)

5.2.2.17.1 Event Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. The loss of forced reactor coolant flow event is characterized by a decrease in forced RCS flow.

A decrease in reactor coolant flow occurring while a plant is at power results in a degradation of core heat transfer, reduction in DNBR margin, and a challenge to the DNB. The reduction in primary system flow and associated increase in core coolant temperatures result in a reduction in DNBR margin. The increasing primary system coolant temperatures also results in expansion of the primary coolant volume, causing an insurge into the pressurizer and an increase in the pressure of the primary system. However, the overpressure transient response for this event is bounded by the loss of external electrical load and/or turbine trip event (UFSAR § 15.2.7) due to the rapid loss of primary-to-secondary heat transfer.

The MDNBR is controlled by the interaction of the primary coolant flow decay, the trip signal, the trip signal generation delay time, the scram delay time, the core power decrease following reactor trip, and the rod surface heat flux. The power-to-flow ratio initially increases, peaks, and then declines as the challenge to the DNB is mitigated by the decline in core power due to the reactor trip.

5.2.2.17.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- RCP coastdown rate (pump inertia and pump frictional torque)
- Trip setpoint(s), uncertainty and delay time
- Minimum scram worth
- Fraction of scram reactivity versus fraction of control rod insertion distance and delay time
- Fuel rod gap conductance

5.2.2.17.3 Acceptance Criteria

This event is classified as a Condition III event, an infrequent event. However, it is analyzed as a Condition II event, which is expected to occur no more often than once per year. This event does not provide a significant challenge to peak pressure and the FCM is not challenged because there is no significant increase in power for this event. Therefore, the principally challenged acceptance criterion for this event is:

Fuel cladding integrity shall be maintained by ensuring that the minimum calculated DNBR shall remain above the 95/95 DNB correlation limit.

5.2.2.17.4 Event Disposition

System modeling inputs such as the initial operating conditions, RCP trip setpoint(s), uncertainty and delay time are not affected by the Adv. W17 HTP fuel for this event. The RCS flow and RCP coastdown would be slightly affected in an adverse direction due to the increased fuel assembly pressure drop of the Adv. W17 HTP fuel; however, the unit will have replacement steam generators installed prior to HTP reloads, and the combined effect is an overall decrease in RCS loop resistance and a consequent increase in the RCS flow and RCP coastdown duration. The initial fuel and clad temperatures for the Mark-BW fuel remains applicable to the Adv. W17 HTP fuel due to identical fuel rod dimensions and material for both Mark-BW fuel and Adv. W17 HTP fuel (Table 2-2).

Loss of flow is short in duration and the modeling of decay heat is relatively unimportant. Decay heat models are equally applicable to Adv. W17 HTP fuel and Mark-BW fuel. Fuel transient thermal responses remain unchanged by the Adv. W17 HTP fuel for the flow coastdown transient due to the limited flow responses for this event.

Limiting reactivity feedback inputs such as the minimum scram worth and MTC are used to maximize power response following the initiation of this event. These inputs are expected to bound the Adv. W17 HTP fuel. Moreover, nuclear analyses will be performed as part of reload licensing requirements to verify that the current limits in reactivity feedback parameters - trip worth, moderator and Doppler feedback – used in Mark-BW safety analyses bound the behavior of the reload core design. The maneuvering analysis will verify adequate margin is preserved at the LCO limits so that the DNBR is not exceeded during this event as part of reload licensing.

Therefore, the analysis of the complete loss of forced reactor coolant flow remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.18 Event Disposition for Waste Gas Decay Tank Rupture (UFSAR 15.3.5)

5.2.2.18.1 Event Description

The Gaseous Waste Processing System is designed to remove fission product gases from the reactor coolant. The system consists of a closed loop with waste gas compressors, hydrogen analyzers, waste gas decay tanks for service at power and other waste gas decay tanks for service at shutdown and startup. The most limiting waste gas incident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in one waste gas decay tank.

5.2.2.18.2 Key Parameters

The key parameters for this event are:

- RCS activity concentration
- Meteorology

5.2.2.18.3 Acceptance Criteria

This is a Condition III infrequent event. The event is evaluated to demonstrate the adequacy of the plant design. The principally challenged acceptance criterion for this event is:

1. The radiological consequences must not exceed 10 CFR 100 ("Reactor Site Criteria") limits.
2. The dose to control room personnel shall not exceed 5 rem.

5.2.2.18.4 Event Disposition

The existing analyses for this event are contained in UFSAR Section 15.5.2. The analysis concludes that all doses resulting from a waste gas decay tank rupture are well within the limits and that the acceptance criteria are met. The parameters important to the dose calculations listed above are not affected by the transition to AREVA Adv. W17 HTP fuel. The current UFSAR analysis results, therefore, remain applicable to the transition to AREVA Adv. W17 HTP fuel.

5.2.2.19 Event Disposition for Single RCCA Withdrawal at Full Power (UFSAR § 15.3.6)

5.2.2.19.1 Event Description

This event is the continuous withdrawal of a single RCCA. The withdrawal of a single RCCS from its inserted bank results in both a reactivity increase and increased power peaking in the region of the core surrounding the withdrawn RCCA. The reactivity increase causes the neutron flux to increase and produces a localized increase in peaking. Subsequently, thermal power, coolant and fuel temperature, and system pressure increase. Reactor trip on overtemperature ΔT provides protection for this event. The peaking asymmetry associated with the withdrawn RCCA can, however, lead to localized fuel failures.

5.2.2.19.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- Trip setpoint(s), uncertainty and delay time
- Localized power peaking

5.2.2.19.3 Acceptance Criteria

This event is classified as a Condition III infrequent fault. Condition III events are allowed to have some fuel failures so long as the site dose releases are within the 10CFR100 limits. In addition, these events may not cause failures that could lead to a worse, Condition IV, event.

5.2.2.19.4 Event Disposition

An analysis of the single RCCA withdrawal event is reported in the Section 15.3.6 of the UFSAR. There are two parts of the analysis, the system analysis and the peaking analysis. The system analysis for the RCCA bank withdrawal accident evaluates very low to very high reactivity insertion rates. The reactivity insertion rate of a single rod withdrawal event is within the range of reactivity rates analyzed for the bank withdrawal analysis. Therefore, the core response from a single rod withdrawal event is already analyzed by the RCCA bank withdrawal analysis. The results for the bank withdrawal event demonstrate that no DNB occurs (as it is a Condition II event).

Since the localized peaking for the single RCCA withdrawal event can be higher than the peaking for the RCCA bank withdrawal, the second part of the evaluation of the single RCCA withdrawal event (the peaking analysis) is performed on a cycle by cycle basis. It is conservatively assumed that the minimum DNBR of the bank withdrawal event is at the SAFDL, even though the analysis demonstrates that it only approaches the limit for certain reactivity insertion rates. Thus, any localized peaking increase caused by the single RCCA withdrawal event that is above a limit established by the RCCA bank withdrawal event is assumed result in fuel failure. The number of pins with peaking that exceeds the limit must be less than 5 percent of the total fuel rods in the core to assure that the results of the UFSAR single RCCA withdrawal dose analysis are bounding and remain applicable.

Therefore, the analysis of the single RCCA withdrawal at full power event remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.20 Event Disposition for Steam Line Break Coincident with Rod Withdrawal at Power (UFSAR § 15.3.7)

5.2.2.20.1 Event Description

In this event, a steam line break (SLB) is assumed to occur when the plant is operating at full power, and as a consequence of the steam line break, a malfunction of the automatic rod control system is assumed to occur, resulting in a rod pull. The automatic rod control system derives its signals from excore detectors and turbine impulse pressure transmitters which are susceptible to malfunction due to the adverse environment typical of a SLB event. In addition, it is postulated that, due to harsh environmental conditions, the Power Range Neutron Flux and the overtemperature ΔT (OTΔT) protection trip functions may not be available. The minimum DNBR in the hot fuel pin needs to be calculated for this event to assure that this event poses no unacceptable radiation release. The trip that protects the reactor under the different size SLBs depends on the break size and consequent cooldown of the reactor coolant system. Larger break sizes will result in a reactor trip on safety injection (SI), smaller breaks will result in a reactor trip on overpower ΔT (OPΔT).

5.2.2.20.2 Key Parameters

The key parameters for this event are:

- Initial operating condition
- Limiting SLB break size
- Trip setpoint(s), uncertainty and delay time
- Control rod withdrawal speed
- Maximum differential rod worth
- Localized power peaking

5.2.2.20.3 Acceptance Criteria

This event is classified as a Condition III infrequent fault. Condition III events are allowed to have some fuel failures so long as the site dose releases are within the 10CFR100 limits. In addition, these events may not cause failures that could lead to a worse, Condition IV, event.

5.2.2.20.4 Event Disposition

The steam line break coincident with rod withdrawal at power event was reanalyzed using RELAP5/MOD2-B&W. The results of the analysis were that the minimum value for DNBR occurs for a 0.0 ft^2 break size at BOC conditions. This means that the limiting event is equivalent to a RCCA withdrawal at power event with only the SI and OPΔT trips available for event mitigation. The plant system characteristics that potentially impact the key parameters listed for this event remain unchanged for both the transition fuel cycle, and the full core implementation of AREVA fuel at Sequoyah. The cause of the event and the parameters which control the consequences of the event are unchanged from or bounded by the analysis. A statepoint check of the conservatism of the reactivity addition and feedback assumed in the AREVA analysis of record is performed for each reload cycle, to assure the bounding thermal-hydraulic statepoints remain applicable.

Therefore, an analysis of the MSLB event with coincident RCCA withdrawal is not required to support the transition to Adv. W17 HTP fuel.

5.2.2.21 Event Disposition for Steam Line Break Event (UFSAR § 15.4.2.1)

5.2.2.21.1 Event Description

The Main Steam Line Break (MSLB) event is analyzed for post-scram return-to-power behavior.

The post-scram MSLB event is initiated by a break in a main steam line upstream of the Main Steam Isolation Valve (MSIV). The maximum break size (i.e., a double-ended guillotine break) is limiting for the post-scram return-to-power consequences of an MSLB event because it maximizes the rate of cooldown and positive reactivity feedback.

The rupture of a main steam line will cause the affected SG pressure and temperature to rapidly decrease. This in turn will cause a rapid cooldown in the RCS loop containing the affected steam generator and in the core sector cooled primarily by water from the cold leg of the affected loop. Other loops and related core sectors will cool at a lesser rate, depending on the various mixing and/or cross-flow phenomena present within the reactor vessel. The drop in SG pressure will initiate a steam line isolation signal. Following appropriate delays, the MSIVs on both the affected and unaffected SGs will close and terminate the blowdown from the unaffected SG(s).

Due to cooldown of the RCS, the RCS coolant will contract. This may cause the pressurizer to empty and the RCS pressure to decrease rapidly. Water in the reactor vessel upper head may flash if this region is fairly stagnant. Upper head flashing will act to delay the RCS pressure decay once the saturation pressure of the upper head is reached. This in turn will delay the injection of borated water initiated by the Safety Injection Signal (SIS). The SIS will also cause main feedwater isolation to occur. The accumulators provide an additional source of borated water after the RCS pressure decreases to below 641.5 psig.

The cooldown of the RCS will insert positive reactivity from both moderator and fuel temperature reactivity feedbacks (particularly at EOC conditions with a most-negative MTC). This positive reactivity addition will erode the negative reactivity added by the RCCAs. The magnitude of core subcriticality depends on the scram worth and the moderator and fuel temperature reactivity feedbacks. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the Safety Injection System.

Reactor trip would be expected to occur on one of the following reactor trips: Containment High Pressure (CHP) (for breaks inside containment), Overpower, or Low SG Pressure (the trip from the SI signal based on the lead-lag compensated low steam line pressure is used in the analysis). No credit is taken in the post-scram analysis for reactor trip or MSIV closure on a predicted high containment pressure. The MSLB event is analyzed at end-of-life, no-load conditions, with and without a loss of offsite power and with the most reactive RCCA stuck in its fully withdrawn position.

5.2.2.21.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- Initial SG inventory
- Break size and location

- Moderator reactivity feedback
- Doppler reactivity feedback
- Trip setpoint(s), uncertainty and delay time
- Technical Specifications minimum SDM
- AFW flow rate and delay time
- Safety Injection flow rate and delay time
- MSIV closure time
- MFW isolation time
- Post-scram radial power peaking factors

5.2.2.21.3 Acceptance Criteria

This event is classified as a Condition IV event (or Postulated Accident), which is not expected to occur during the lifetime of the plant, but must be evaluated to demonstrate the adequacy of the plant design. The principally challenged acceptance criterion for this event is:

If fuel failure is predicted, the radiological consequences must not exceed the 10 CFR 100 ("Reactor Site Criteria") limits.

5.2.2.21.4 Event Disposition

Various steam line break scenarios were evaluated based on BAW-10220P. The limiting case was a double-ended rupture of a steam line upstream of the steam measurement device with off-site power available. The event behavior is predominantly a function of the primary-to-secondary heat transfer capability. Therefore, small perturbations in parameters such as the core pressure drop, core bypass flow fraction, core inlet flow distribution, and reactivity feedback do not impact the parameters of interest in assessing the acceptance criteria. The plant system characteristics that potentially impact the key parameters listed for this event remain unchanged by the implementation of AREVA Adv. W17 HTP fuel. A cycle-specific statepoint analysis is performed each cycle to demonstrate that the DNBR limit is not exceeded.

Therefore, the analysis of the MSLB event remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.22 Event Disposition for Major Rupture of a Main Feedwater Pipe (UFSAR § 15.4.2.2)

5.2.2.22.1 Event Description

The Feedwater Line Break (FWLB) event is defined as a major break in a main feedwater line that is sufficiently large to prevent maintaining the SG secondary side water inventory in the affected SG. This event can be considered as a heat-up event, a cool-down event, or a combination of both. There can be an initial, short, heat-up transient when the feedwater flow stops. This phase is terminated by a reactor trip. This heat-up portion of the transient produces the so-called "first peak" RCS response, which may result in a challenge to RCS pressure limits. Following the reactor trip, the RCS begins to cool down as a result of the heat removal from the affected SG.

The RCS pressure may decrease enough to cause HPSI to activate. The cool-down portion of the transient is terminated when the low steamline pressure SI signal actuates isolation of the steam generators from the affected generator, and the primary system heats up. The loss of steam generator inventory and rising steam pressure cause primary temperatures to rise. Successful termination of the transient is achieved when auxiliary feedwater supplied to the steam generators is sufficient to remove core decay heat. The FWLB is analyzed to demonstrate overpressure protection of the RCS and continued capability for core cooling.

5.2.2.22.2 Key Parameters

The key parameters for this event are:

- Break size
- SG liquid inventory at the time of reactor trip
- Trip setpoint(s), uncertainty and delay time
- AFW actuation setpoint, minimum flow rate and actuation delay time
- SG blowdown flow rate and isolation time
- Core decay heat
- RCP heat
- MSSV setpoints and capacities

5.2.2.22.3 Acceptance Criteria

This event is classified as a Condition IV event (or Postulated Accident), which is not expected to occur during the lifetime of the plant, but must be evaluated to demonstrate the adequacy of the plant design. The principally challenged acceptance criteria for this event are:

1. The pressures in the reactor coolant and main steam systems should be less than 110% of design values.
2. Any fuel damage calculated to occur must be sufficiently limited such that the core will remain in place and intact with no loss of core cooling capability. Preclusion of fuel failure is demonstrated by delivering sufficient AFW to remove core decay heat such that there is no significant heatup of the RCS following reactor trip.
3. Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 ("Reactor Site Criteria") guidelines.

5.2.2.22.4 Event Disposition

The event behavior is predominantly a function of the primary-to-secondary heat transfer capability. Therefore, small perturbations in parameters such as the core pressure drop, core bypass flow fraction, core inlet flow distribution, and reactivity feedback do not impact the parameters of interest in assessing the acceptance criteria. The plant system characteristics that potentially impact the key parameters listed for this event remain unchanged by the transition to AREVA Adv. W17 HTP fuel. The cause of the event and the parameters which control the consequences of the event are unchanged from or bounded by the current analysis of record presented in UFSAR Section 15.4.2. Fuel pellet material and operational characteristics - uranium enrichment, fuel cycle length, linear heat rates - are, by design, similar for both Adv. W17 HTP and Mark-BW fuels. Therefore, the decay heat models used in Mark-BW safety analyses remain applicable to the Adv. W17 HTP fuel assemblies. A cycle-specific reload check is performed each cycle to assure that the decay heat assumptions used in the non-LOCA analyses remain applicable for the cycle.

Therefore, the analysis of the FWLB remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.23 Event Disposition for Steam Generator Tube Rupture Event (UFSAR § 15.4.3)

5.2.2.23.1 Event Description

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RCS. In the event of a coincident loss of offsite power, or failure of the condenser dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power operated relief valves.

The operator is expected to determine that a steam generator tube rupture has occurred, and to identify and isolate the faulty steam generator on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. The primary system event is effectively terminated when makeup flow via the safety injection system matches the rate of coolant loss matches the rate of coolant loss through the failed steam generator tube. The tube leakage is terminated when the operator depressurizes the primary system below the steam pressure of the affected steam generator.

The recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the isolation procedure can be completed within 30 minutes of accident initiation. The analysis assumptions lead to a conservative upper limit estimate of 125,000 lbs. for the total amount of reactor coolant transferred to the secondary side of the faulty steam generator as a result of a tube rupture accident.

5.2.2.23.2 Key Parameters

The key parameters for this event are:

- Initial conditions
- SG tube break area
- Primary-to-secondary pressure difference
- Safety injection flow rate
- SG atmospheric relief valve capacity

- Operator actions

5.2.2.23.3 Acceptance Criteria

This event is classified as a Condition IV event (or Postulated Accident), which is not expected to occur during the lifetime of the plant, but must be evaluated to demonstrate the adequacy of the plant design. The principally challenged acceptance criterion for this event is:

The radiological consequences must not exceed the 10 CFR 100 ("Reactor Site Criteria") limits or NRC-accepted dose limits.

5.2.2.23.4 Event Disposition

The event behavior is predominantly a function of the primary-to-secondary pressure differential, break size, atmospheric relief valve capacity and the timing of operator actions. Therefore, small perturbations in parameters such as the core pressure drop, core bypass flow fraction, core inlet flow distribution, and reactivity feedback do not impact the parameters of interest in assessing the acceptance criteria. The plant system characteristics that potentially impact the key parameters listed for this event remain unchanged for the transition to Adv. W17 HTP fuel. The cause of the event and the parameters which control the consequences of the event are unchanged from or bounded by the current analysis of record presented in UFSAR Section 15.4.3.

Therefore, an analysis of the steam generator tube rupture event is not required to support the transition to AREVA Adv. W17 HTP fuel.

5.2.2.24 Event Disposition for Single Reactor Coolant Pump Locked Rotor Event (UFSAR § 15.4.4)

5.2.2.24.1 Event Description

The Reactor Coolant Pump Rotor Seizure event is postulated to be caused by the instantaneous seizure of a reactor coolant pump rotor. The analysis assumes the event is initiated from hot full power conditions. Flow through the faulted RCS loop rapidly decreases, causing a reactor trip on a Low RCS Loop Flow signal within 1 to 2 seconds and a turbine trip on the reactor trip. Loss of off-site power is assumed to occur simultaneously with the reactor trip, causing the remaining reactor coolant pumps to begin to coastdown.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the reactor coolant. The combination of the relatively high fuel rod surface heat fluxes, decreasing core flow, and increasing core coolant temperatures challenges the DNBR safety limit.

At the same time, the steam generator primary-to-secondary heat transfer rate decreases, because (1) the decreasing primary coolant flow degrades the steam generator tube primary-side heat transfer coefficients and (2) the turbine trip causes the secondary-side temperature to increase. The decreasing rate of heat removal in the steam generators and the decreasing flow of coolant removing heat from the reactor core cause the reactor coolant to heat up. The resultant reactor coolant expansion causes fluid to surge into the pressurizer and pressurization of the RCS. Only the primary safety valves are allowed to mitigate the primary pressure increase during the transient, thus maximizing the peak primary pressures. For the DNB calculations, the pressure is assumed constant at the initial value.

To maximize the power response during the event, the least negative Doppler power coefficient and a +7.0 pcm/°F moderator coefficient are assumed.

5.2.2.24.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- RCS coolant inertia
- RCS loop resistance
- Locked rotor pump resistance
- Trip setpoint(s), uncertainty and delay time
- Minimum HFP scram worth
- Fraction of scram reactivity versus fraction of control rod insertion distance at HFP
- Scram delay time
- Fuel rod gap conductance
- Relief valve setpoints and flow capacities

5.2.2.24.3 Acceptance Criteria

This event is classified as a Condition IV event (or Postulated Accident), which is not expected to occur during the lifetime of the plant, but must be evaluated to demonstrate the adequacy of the plant design. The principally challenged acceptance criteria for this event are:

1. The faulted condition stress limits for RCS pressure.
2. The radiological consequences must not exceed the 10 CFR 100 ("Reactor Site Criteria") limits or NRC-accepted dose limits.

5.2.2.24.4 Event Disposition

The event system behavior is predominantly a function of the reactor coolant inertia, loop flow resistance, locked rotor pump resistance, low flow trip setpoint and time delay, scram characteristics, reactivity feedbacks, and relief valve characteristics. Various locked rotor event scenarios were evaluated in BAW-10220P, assuming operation with the original steam generators and Mark-BW fuel. The low flow trip setpoint and delays and relief valve characteristics are not affected by the introduction Adv. W17 HTP fuel. The scram characteristics and assumed reactivity feedbacks are also checked on a cycle-specific basis to assure they remain bounded. The minimum RCS coolant flow assumed in the analyses is expected to remain unaffected by the introduction of Adv. W17 HTP fuel, so the reactor coolant inertia assumed in the analysis is not impacted. The locked rotor pump resistance also is not impacted. Relative to the current analysis of record, coincident with the introduction of Adv. W17 HTP fuel, the plant will be operating with replacement steam generators with a reduced pressure drop relative to the original steam generators. Thus the loop flow resistance is reduced relative to the current UFSAR analysis assumptions. The transition to Adv. W17 HTP fuel has a small effect of additional core bypass and higher core pressure drop. The effect of the increased reactor vessel pressure drop due to the Adv. W17 HTP fuel on flow coastdown characteristics for the limiting complete loss of flow has been examined and determined to be negligible when considered in conjunction with the installation of the replacement steam generators and the attendant reduction in steam generator pressure drop. Therefore, the cause of the event and the parameters which control the consequences of the event are unchanged from or bounded by the current analysis of record presented in UFSAR Section 15.4.4., which remains bounding for operation with Adv. W17 HTP fuel.

The conclusions of the current UFSAR analysis in Section 15.4.4 were that:

1. Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

2. Since the peak fuel temperature is well below the 5080°F fuel temperature limit and the peak cladding temperature is well below the 1800°F cladding temperature limit, the core will remain intact with no consequential loss of core cooling capability.

For the purpose of dose calculations, all pins that experience DNB are assumed to fail. The radiological effects for this accident assumed that 10% of the fuel pins experience DNB. A cycle-specific assessment (pin census) assures that less than 10% of pins fail at the limiting locked rotor transient conditions. This cycle-specific assessment will include or bound the effect on DNB due to the introduction of Adv. W17 HTP fuel.

Therefore, the analysis of the single reactor coolant pump locked rotor event remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.25 Event Disposition for Fuel Handling Accident (UFSAR § 15.4.5)

5.2.2.25.1 Event Description

The accident is defined as dropping of a spent fuel assembly onto the spent fuel pit floor resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations.

5.2.2.25.2 Key Parameters

The key parameters for this event are:

- Core power level
- Activity of highest powered fuel assembly in core region discharged
- Radial peaking factor
- Decay time prior to removal from reactor vessel
- Minimum water depth between damaged fuel rods and pool surface
- Maximum fuel rod pressurization
- Decontamination factors and meteorology

5.2.2.25.3 Acceptance Criteria

This event is evaluated to demonstrate the adequacy of the plant design. The principally challenged acceptance criterion for this event is:

The radiological consequences must be “well within” the 10 CFR 100 ("Reactor Site Criteria") limits.

5.2.2.25.4 Event Disposition

The key parameters for this event are not impacted by the introduction of Adv. W17 HTP fuel. The fuel rod and fuel pellet materials and design are similar to the current Mark-BW fuel. The fuel burnup limits are also similar to the current Mark-BW fuel. The analyses of the consequences of the event in UFSAR Section 15.5.6 remain applicable.

5.2.2.26 Event Disposition for Rod Cluster Control Assembly Ejection (UFSAR § 15.4.6)

5.2.2.26.1 Event Description

The Control Rod Ejection event is initiated by a postulated rupture of a control rod drive mechanism housing. Such a rupture allows the full system pressure to act on the drive shaft, which ejects its control rod from the core. The consequences of the mechanical failure are a rapid positive reactivity insertion and an increase in radial power peaking, which could possibly lead to localized fuel rod damage.

Doppler reactivity feedback mitigates the power excursion as the fuel begins to heat up. Although the initial increase in power occurs too rapidly for the scram rods to have any effect on the power during that portion of the transient, the scram negative reactivity insertion does affect the fuel temperature and fuel rod cladding surface heat flux.

5.2.2.26.2 Key Parameters

The key parameters for this event are:

- Initial operating conditions
- Ejected rod worth
- Doppler reactivity feedback
- Trip setpoint(s), uncertainty and delay time
- Fuel rod gap conductance
- Post ejection FQ predicted for the purpose of calculating the peak (hot spot) fuel centerline temperature

5.2.2.26.3 Acceptance Criteria

This event is classified as a Condition IV event (or Postulated Accident), which is not expected to occur during the lifetime of the plant, but must be evaluated to demonstrate the adequacy of the plant design. The principally challenged acceptance criteria for this event are:

1. The radial-average fuel pellet enthalpy at the hot spot must be ≤ 280 cal/g. (The UFSAR analysis of record employs more conservative criteria of 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel).
2. The maximum RCS pressure during any portion of the transient must remain below the faulted condition stress limits.
3. If fuel failure is predicted, the radiological consequences must not exceed the 10 CFR 100 limits.

5.2.2.26.4 Event Disposition

The key parameters for this event are not impacted by the introduction of Adv. W17 HTP fuel as shown in Section 3. The system response and hot spot analyses are dependent upon the neutronics characteristics and thermal response of the fuel. The fuel properties of the Adv. W17 HTP fuel are similar to the Mark-BW fuel and are very similar to the Westinghouse standard and Vantage 5 fuel upon which the current UFSAR analysis or record is based. The thermal response of Mark-BW fuel to an ejected rod power excursion using a representative core average nuclear power excursion was compared to that of the Westinghouse fuel in BAW-10220P. No discernible differences were found. The steady-state fuel temperatures for the Westinghouse and Mark-BW fuel were also compared in BAW-10220P and no discernible difference observed. Due to the similarity of the Adv. W17 HTP and Mark-BW fuel rod and fuel pellet designs, the results of these comparisons remain applicable to Adv. W17 HTP fuel. Since the thermal responses of the Westinghouse and AREVA fuels has been shown to be similar at steady-state and during an ejected rod accident, the bounding parameters in the current UFSAR RCCA Ejection analysis of record remain applicable for operation with Adv. W17 HTP fuel. The HZP and HFP cases represent the extremes of maximum power pulse and maximum initial thermal conditions, respectively. Since the rod position limits and the core average temperature versus power remain linear, the power relationship is not affected by change to the Adv. W17 HTP fuel. The Adv. W17 HTP fuel does not affect the neutronic properties so that checking the BOC and EOC parameters remains valid for the Rod Ejection Event. The calculated ejected rod worths, post ejected F_q, delayed neutron fraction, least negative Doppler power coefficients, and pin census are evaluated on a cycle-specific basis to assure they are bounded by the current UFSAR analysis of record in Section 15.4.6.

Therefore, the analysis of the RCCA ejection event remains bounding for the transition to the Adv. W17 HTP fuel and the cycle specific checks for the neutronic analysis remain valid.

5.2.2.27 Environmental Consequences (UFSAR 15.5.1, 15.5.2, 15.5.4, 15.5.5, 15.5.6, 15.5.7)

This section summarizes the effects of the implementation of Adv. W17 HTP fuel at Sequoyah on the environmental consequences of non-LOCA radiological accidents reported in Section 15.5 of the FSAR. The dose consequence analyses consist of:

- Loss of AC Power to the Station Auxiliaries (UFSAR 15.5.1)
- Waste Gas Decay Tank Rupture (UFSAR 15.5.2)
- Steam Line Break (UFSAR 15.5.4)
- Steam Generator Tube Rupture (UFSAR 15.5.5)
- Fuel Handling Accident (UFSAR 15.5.6)
- Rod Ejection Accident (UFSAR 15.5.7)

The Adv. W17 HTP fuel pin design is similar, both physically and neutronically, to the Mark-BW fuel pin design. Operational design characteristics – (power and burnup) are unchanged or more restrictive relative to previous fuel cycles. The source terms used in the environmental consequence analyses are, therefore, unaffected by the implementation of Adv. W17 HTP fuel.

No new failure mechanisms are introduced by the use of Adv. W17 HTP fuel. In addition, because the Adv. W17 HTP fuel is thermally similar and hydraulically compatible with the Mark-BW assemblies, the mass and energy releases utilized in environmental consequences remain unaffected by the Adv. W17 HTP fuel.

Section 5.2 indicates that the implementation of Adv. W17 HTP fuel assemblies will not adversely affect the predicted results of a non-LOCA accident analyzed in the Sequoyah licensing basis. That is, all acceptance criteria for non-LOCA Condition II, III, and IV events continue to be met and the inputs regarding failed fuel fraction to the dose consequence analyses remain unchanged.

Many of the inputs used in dose consequence analyses - fractional fuel failure, primary to secondary leakage, iodine partitioning - are prescribed by the relevant regulatory guidelines and are independent of fuel type.

Plant-specific inputs such as containment parameters - volume, surface area, atmospheric leakage rates - and engineered safeguard feature - containment spray, ice condensers, fans - capabilities in the reduction and/or removal of radionuclides are unaffected by the Adv. W17 HTP fuel.

Atmospheric dispersion factors used in environmental consequence analysis is estimated specifically for the plant, is based on local weather information, and is not affected by Adv. W17 HTP fuel.

All of these arguments, taken together, justify the implementation of Adv. W17 HTP Fuel at Sequoyah with respect to existing licensing basis environmental consequence analyses. Continued compliance with the acceptance criteria for the dose events - 10CFR100 offsite dose limits and 10CFR50 General Design Criteria 19 control room habitability requirements are assured.

5.2.2.27.1 Environmental Consequences of a Postulated Loss of A.C. Power to the Plant Auxiliaries (UFSAR 15.5.1)

The fuel cladding damage is not expected following a loss of A.C. power to the plant auxiliaries. The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the RCS to the secondary system in the steam generator. The primary-to-secondary leak rate primary coolant activity, iodine activity in the secondary side liquid, and iodine partition factor are set by Technical Specification limits and are not affected by fuel design. Also, the steam release to cool the plant is not affected by the fuel design. Since all parameters affecting this event for environmental consequences are not adversely affected by the Adv. W17 HTP fuel, the results of the existing analysis are applicable to the Adv. W17 HTP fuel.

5.2.2.27.2 Environmental Consequences of a Postulated Waste Gas Decay Tank Rupture (UFSAR 15.5.2)

The analysis of this event is performed based on Regulatory Guide 1.24, 1972. The parameters used for the analysis are not affected by the fuel design. The tank activity assumed at the event initiation is conservatively determined based on the reactor coolant system volume. The RCS volume is unaffected by the Adv. W17 HTP fuel, and the assumed tank activity remains bounding. Therefore, an environmental consequences analysis of a postulated waste gas decay tank rupture is not required for the Adv. W17 HTP fuel.

5.2.2.27.3 Environmental Consequences of a Loss of Coolant Accident (UFSAR 15.5.3)

The analysis of this event is performed based on Regulatory Guide 1.4. The parameters used in the environmental consequence analysis are not affected by the Adv. W17 HTP fuel. The results of the existing analysis remain applicable.

5.2.2.27.4 Environmental Consequences of a Postulated Steam Line Break (UFSAR 15.5.4)

The fuel cladding damage is not expected following a steam line break. The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the RCS to the secondary system in the steam generator. The primary-to-secondary leak rate primary coolant activity, iodine activity in the secondary side liquid, and iodine partition factor are set by Technical Specification limits and are not affected by fuel design. Also, the amount of steam released as a result of a steam line break is not affected by fuel design. The environmental consequences of a steam line break are not adversely affected by the Adv. W17 HTP fuel. The results of the existing analysis remain applicable.

5.2.2.27.5 Environmental Consequences of a Postulated Steam Generator Tube Rupture (UFSAR 15.5.5)

The fuel cladding damage is not expected following a steam generator tube rupture. The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the RCS to the secondary system in the steam generator. A conservative analysis of the postulated steam generator tube rupture assumes the loss of offsite power and hence involves the release of steam from the secondary system. A conservative analysis of the potential offsite doses resulting from this accident assuming steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. The primary-to-secondary leak rate primary coolant activity, iodine activity in the secondary side liquid, and iodine partition factor are set by Technical Specification limits and are not affected by fuel design. Also, the amount of steam released to cool the plant is not affected by fuel design. The environmental consequences of a steam generator tube rupture are not adversely affected by the Adv. W17 HTP fuel. The results of the existing analysis remain applicable.

5.2.2.27.6 Environmental Consequences of a Postulated Fuel Handling Accident (UFSAR 15.5.6)

All the parameters used in the environmental consequence analysis are not adversely affected by the Adv. W17 HTP fuel. The key parameters for this event are not impacted by the introduction of Adv. W17 HTP fuel. The fuel rod and fuel pellet materials and design are similar to the current Mark-BW fuel. The fuel burnup limits are also similar to the current Mark-BW fuel. The analyses of the consequences of the event in UFSAR Section 15.5.6 remain applicable. This event is evaluated each cycle as part of the reload licensing process to ensure that the analysis of record remains bounding.

5.2.2.27.7 Environmental Consequences of a Postulated Rod Ejection Accident (UFSAR 15.5.7)

The consequences of a postulated rod ejection accident are bounded by the results of the loss of coolant accident analysis evaluated in Section 5.2.2.27.3.

5.2.2.28 Event Disposition for Containment Response (UFSAR § 6.2.1.3.3. 6.2.1.3.4, 6.2.1.3.11)

5.2.2.28.1 Event Description

The Containment Structure encloses the primary and secondary plant and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The Containment Structure must be capable of withstanding the pressure and temperature conditions resulting from a postulated LOCA or MSLB accident. While other events, such as a feedwater line break also discharge mass and energy to Containment, the LOCA and MSLB have been confirmed to be the two most severe inside containment events with respect to maximizing the peak containment pressure and temperature.

5.2.2.28.2 Key Parameters

The key parameters for the containment response are those related to the LOCA and MSLB design basis events.

- Initial NSSS power and core decay heat
- RCS flow rate
- RCS Pressure (LOCA)
- SG pressure (MSLB)
- Trip setpoint(s), uncertainty and delay time
- MFW flow and temperature
- AFW flow and temperature
- Safety injection flowrates and delay times
- Containment cooling capability and delay times
- Containment heat sinks
- Containment atmosphere pressure, temperature and humidity

5.2.2.28.3 Acceptance Criteria

The Sequoyah containment has a design pressure and temperature of 12 psig and 327°F, respectively. These containment design values were selected as a result of the original analysis of the LOCA. The acceptance criterion for the containment response analysis is that pressure and temperature remain below these limits.

5.2.2.28.4 Event Disposition

In BAW-10220P, the effect of transitioning to and loading AREVA Mark-BW fuel on the UFSAR containment integrity analysis of record, which utilized Westinghouse fuel, was evaluated. The important aspects of the fuel change that had the possibility of impacting the analysis included the changes in the flow characteristics past the fuel, the RCS operating T_{avg}, the fuel-heat capacity and core stored energy, and the decay heat. The effect of including Mark-BW Fuel on the current LOCA M&E and the containment integrity analysis was evaluated therein and it was concluded that the current UFSAR analysis results remain bounding. These same aspects are evaluated for the transition from Mark-BW fuel to Adv. W17 HTP fuel.

There are small deviations in flow characteristics past the fuel between the Mark-BW and Adv. W17 HTP fuel. However, for an ice condenser design, since the peak pressure occurs late in the transient, well after the ice bed has melted out, the single effect of small deviations in flow is insignificant relative to analysis results.

Total energy content, or total energy available for release to containment, is significant, which remains unchanged. The RCS T_{avg} remains at 578.2°F.

For the Adv. W17 HTP fuel, there is negligible difference in the mechanical heat capacity of the fuel relative to the current Mark-BW fuel, which was determined to be negligibly different from that of the Westinghouse fuel assumed in the UFSAR analysis.

Initial fuel stored energy is dependent upon fuel and clad temperature. Transients initiated from zero power assume fuel temperatures that are initially in equilibrium with the RCS temperature independent of fuel type. Transients initiated at power, however, require an estimate of the initial fuel temperature based on power, fuel pin dimension, and material properties. The initial stored energy at power for the two assembly designs (Mark-BW and Adv. W17 HTP) is assessed by considering cladding characteristics and fuel rod power density. There is no difference in fuel rod dimensions or material, thus there is no effect on the energy present in the Adv. W17 HTP fuel rods relative to the current Mark-BW fuel design (Table 2-2). Regarding fuel power density, the fuel pellet radius (and hence, assembly loading) are identical in the Adv. W17 HTP fuel relative to the Mark-BW assembly, thus there is no difference in power density when operated at the same power output. Consequently, there is no significant change in the amount of stored energy in both the clad and fuel for the Adv. W17 HTP fuel assembly. Thus, the fuel initial stored energy for the Mark-BW assembly remains applicable to the Adv. W17 HTP fuel assembly design.

As noted in BAW-10220P, the increased core stored energy of the Mark-BW fuel currently operating represented a slight increase relative to that of the Westinghouse fuel assumed in the UFSAR containment integrity analyses. The increase was evaluated in BAW-10220P, and it was determined to have an increased energy effect of 1.32×10^6 BTUs. It was also determined that there are margins in the current UFSAR containment integrity calculations that offset this small increase. For example, the current UFSAR analysis utilizes the specific TVA Sequoyah Decay Heat Curve (UFSAR Table 6.2.1-8a) until the time of steam generator equilibration (i.e., 1697.2 seconds). The LOTIC code then conservatively determines the decay heat based upon Table 6.2.1-8 after equilibration. If the Sequoyah specific data is also used after steam generator equilibration, it is found that 2.11×10^6 BTUs can be removed from the calculation up to the time of ice bed meltout, and 6.00×10^6 BTUs can be removed up to the time of peak pressure. This conservatism more than offsets the increased core-stored energy effect. The conclusions of this evaluation continue to apply with respect to the use of Adv. W17 HTP fuel.

In summary, the effect of including Adv. W17 HTP Fuel on the current LOCA M&E and the containment integrity analysis has been evaluated. It has been concluded that the current UFSAR analysis results remain
bounding.

Therefore, an analysis of the Containment Integrity is not required to support the transition to AREVA Adv. W17 HTP fuel.

5.2.2.29 Event Disposition for Anticipated Transient Without Scram (UFSAR § 7.7.1.12)

5.2.2.29.1 Event Description

Anticipated Transient Without Scram (ATWS) is defined as a Condition II event followed by the failure of the reactor trip portion of the protection system. The function of ATWS Mitigating System Actuation Circuitry (AMSAC) is to mitigate the effects of an ATWS by providing alternate means of tripping the main turbine and actuating auxiliary feedwater (AFW) flow independent from the reactor protection system (RPS). AMSAC actuation will prevent reactor coolant system (RCS) over-pressurization, maintain fuel integrity, and meet 10 CFR 100 radiation release requirements.

5.2.2.29.2 Acceptance Criteria

The ATWS acceptance criteria below provide assurance that the reactor coolant system (RCS) pressure shall not exceed ASME Service Level C limits, maintain fuel integrity, and meet 10 CFR 100 radiation release requirements.

The ATWS acceptance criteria are:

1. The reactor coolant system (RCS) pressure shall not exceed ASME Service Level C limits.

2. Fuel integrity shall be maintained.

3. Radiation release shall be within 10 CFR 100 requirements.

5.2.2.29.3 Event Disposition

AMSAC is not required to be evaluated within the plant design basis and therefore, is not addressed in UFSAR Chapters 4.0 and 15.0. The cause of the event and the parameters which control the consequences of the event are unchanged with the introduction of the Adv. W17 HTP fuel. Therefore, an analysis of ATWS events is not required for the fuel transition.

5.2.3 Non-LOCA SER Restrictions / Limitations

No new SER restrictions or limitations.

5.2.4 Non-LOCA Technical Specification Changes

None.

Table 5-1 Summary of Event Disposition

SRP Section	UFSAR Section	Event Description	Disposition	Discussion
15.1.1 15.1.2	15.2.10	Excess Feedwater Heat Removal • Decrease in Feedwater Temperature • Increase in Feedwater Flow	No Analysis Required	5.2.2.10
15.1.3	15.2.11	Excess Load	No Analysis Required	5.2.2.11
15.1.4	15.2.13	Inadvertent Opening of a Steam Generator Relief or Safety Valve	No Analysis Required	5.2.2.13
15.1.5	15.3.2 15.3.7 15.4.2.1	Steam Line Break • Minor Secondary System Pipe Breaks • Steam Line Break Coincident with Rod Withdrawal at Power • Rupture of a Main Steam Line	No Analysis Required	5.2.2.15 5.2.2.20 5.2.2.21

SRP Section	UFSAR Section	Event Description	Disposition	Discussion
15.2.1 15.2.2 15.2.3 15.2.5	15.2.7	Loss of Load • Loss of Electric Load • Turbine Trip • Loss of Condenser Vacuum • Steam Pressure Regulator Failure	No Analysis Required	5.2.2.7
15.2.4	---	Closure of Main Steam Isolation Valve	N/A	N/A
15.2.6	15.2.9	Loss of Non-Emergency AC Power	No Analysis Required	5.2.2.9
15.2.7	15.2.8	Loss of Feedwater Flow	No Analysis Required	5.2.2.8
15.2.8	15.4.2.2	Feedwater Line Break	No Analysis Required	5.2.2.22
15.3.1	15.2.5 15.3.4	Loss-of-Coolant Flow	No Analysis Required	5.2.2.5 5.2.2.17
15.3.2	---	RCS Flow Controller Malfunction	N/A	N/A
15.3.3	15.4.4	RCP Seized Rotor	No Analysis Required	5.2.2.24
15.3.4	---	RCP Shaft Break	N/A	N/A
15.4.1 15.4.2	15.2.1 15.2.2 15.3.6	RCCA Withdrawal • RCCA Bank at Subcritical or Low Power Startup Condition • RCCA Bank At Power • Single RCCA Withdrawal at Full Power	No Analysis Required	5.2.2.1 5.2.2.2 5.2.2.19
15.4.3	15.2.3	RCCA Drop	No Analysis Required	5.2.2.3
15.4.4	15.2.6	Startup of an Inactive Loop at an Incorrect Temperature	No Analysis Required	5.2.2.6
15.4.6	15.2.4	Boron Dilution	Assessed on Reload	5.2.2.4
15.4.7	15.3.3	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Location	Assessed on Reload	5.2.2.16
15.4.8	15.4.6	RCCA Ejection	No Analysis Required	5.2.2.26
15.5.1	15.2.14	Inadvertent Operation of ECCS	No Analysis Required	5.2.2.14
15.5.2	---	Excess Charging	N/A	N/A
15.6.1	15.2.12	RCS Depressurization	No Analysis Required	5.2.2.12
15.6.2	---	Radiological Consequences of The Failure of Small Lines Carrying Primary Coolant Outside Containment	N/A	N/A
15.6.3	15.4.3	Steam Generator Tube Rupture	No Analysis Required	5.2.2.23

SRP Section	UFSAR Section	Event Description	Disposition	Discussion
15.6.5	15.3.1 15.4.1	Loss-of-Coolant Accident • Small Break • Large Break	See Section 5.3 of this document	5.3.1 5.3.2
15.7.3	15.3.5	Waste Process System Incident	Assessed on Reload	5.2.2.18
15.7.4	15.4.5	Fuel Handling Incident	Assessed on Reload	5.2.2.25
15.7.5	---	Spent Fuel Cask Drop Accidents	N/A	N/A
---	15.5.1	Environmental Consequences of a Postulated Loss of A.C. Power to the Plant Auxiliaries	No Analysis Required	5.2.2.27.1
15.7.3	15.5.2	Environmental Consequences of a Postulated Waste Gas Decay Tank Rupture	No Analysis Required	5.2.2.27.2
15.6.5	15.5.3	Environmental Consequences of a Loss of Coolant Accident	See Section 5.3 of this document	5.2.2.27.3
15.1.5.A	15.5.4	Environmental Consequences of a Postulated Steam Line Break	No Analysis Required	5.2.2.27.4
15.6.3	15.5.5	Environmental Consequences of a Postulated Steam Generator Tube Rupture	No Analysis Required	5.2.2.27.5
15.7.4	15.5.6	Environmental Consequences of a Postulated Fuel Handling Accident	Assessed on Reload	5.2.2.27.6
15.4.8.A	15.5.7	Environmental Consequences of a Postulated Rod Ejection Accident	No Analysis Required	5.2.2.27.7
15.8	7.7.1.12	Anticipated Transients Without Scram	No Analysis Required	5.2.2.29
6.3	6.2.1.3.3 6.2.1.3.4 6.2.1.3.11	Containment Response	No Analysis Required	5.2.2.28
8.4	15.2.9	Station Blackout	No Analysis Required	5.2.2.9

5.3 Loss of Coolant Accidents (LOCA)

The loss-of-coolant accident is analyzed as required by SRP Section 6.3, Emergency Core Cooling System, to assure that the design bases for the ECCS satisfy the requirements of 10 CFR 50.46 regarding ECCS acceptance criteria, which includes a cross-reference to SRP Section 15.6.5. The small break LOCA (SBLOCA) and realistic large break LOCA (RLBLOCA) analyses are discussed in Sections 5.3.1 and 5.3.2 respectively. Also required in SRP Section 6.3 is a review of the effects of pipe breaks, including containment response. Discussion relative to the containment response is included in Section 5.3.3.

Section 5.3.1 describes the application of NRC-approved methodology for SBLOCA analysis (Reference 3). The method is a change from that used in the analysis of record (AOR) for Sequoyah. Although not required, this new application is included as an attachment to the HTP transition request for NRC review.

The large break analysis was performed with the same methodology as the current Sequoyah RLBLOCA AOR, found in Reference 10 and Reference 14. Any deviations from the method used in the Sequoyah RLBLOCA AOR are discussed in section 5.3.2 and implemented in Reference 8.

5.3.1 Small Break LOCA

The AREVA NP S-RELAP5 SBLOCA evaluation model for event response of the primary and secondary systems and hot fuel rod used in this analysis (Reference 9) consists of two computer codes, S-RELAP5 and RODEX2/2A, described in Section 5.3.1.1. The appropriate conservatisms, as prescribed by Appendix K of 10 CFR 50, are incorporated and specific deviations outlined in Section 3.2 of the SBLOCA Summary Report (Reference 9) attached to the license amendment request. These deviations are implemented in response to recent NRC RAIs.

5.3.1.1 Analysis Methodology and Computer Codes

The Reference 3 methodology has been reviewed and approved by the NRC to perform SBLOCA analyses for Westinghouse 4-loop designed plants and is applicable to Sequoyah Unit 1 and Unit 2. The evaluation model for event response of the primary and secondary systems and hot fuel rod consists of two computer codes. The two AREVA NRC approved computer codes used in this analysis are:

1. RODEX2-2A (References 1 and 2) determines the burnup-dependent initial fuel rod conditions for the system calculations.

2. S-RELAP5 (Reference 7) predicts the thermal-hydraulic response of the primary and secondary sides of the reactor system and the hot rod response.

As a result of the new methodology application for Sequoyah, the analyses supporting the HTP fuel transition require the following TS/COLR change:

- Remove BAW-10168(P)(A) from the Reference list and replace with EMF-2328(P)(A)

Changes made to the methodology pertain to the improved representation of the plant parameters and address recent NRC issues. The changes are discussed in the attached SBLOCA Summary Report (Reference 9).

5.3.1.2 SBLOCA Analysis

The break spectrum calculations were executed for breaks of 1.00, 2.00, 2.75, 3.00, 3.50, 4.00, 4.50, 4.75, 4.90, 4.95, 5.00, 5.05, 5.10, 5.12, 5.13, 5.14, 5.15, 5.20, 5.25, 5.50, 5.75, 6.00, 6.50, 7.00, 8.00, 8.50, 9.00, 9.75, 9.76, 9.77 and 9.78 inch diameter (the 9.76 inch diameter break corresponds to an area equal to 10% of the cold leg area).

The results of the analysis demonstrated that the adequacy of the Emergency Core Cooling System (ECCS) by conformance to the criteria given in 10 CFR 50.46(b) which were proven in the current analysis for Sequoyah Unit 1 and Unit 2 operating with AREVA supplied 17x17 HTP M5 clad fuel, as follows:

(1) *Peak cladding temperature:* The calculated limiting fuel element cladding temperature is 1470°F, less than the 2200° F limit criterion.

(2) *Maximum local cladding oxidation:* The calculated maximum local oxidation of the cladding is 0.17% which is less than the 17% limit of the criterion.

(3) *Maximum core-wide oxidation:* The calculated core-wide total oxidation is less than 0.0013%, which is less than the 1% limit of the criterion.

(4) *Coolable geometry:* The cladding remains amenable to cooling. None of the cases analyzed predicted hot rod rupture, hence no blockage is predicted to occur which would degrade core cooling. Both thermal and mechanical deformations of the fuel assemblies in the core have been assessed and the resultant deformations have been shown to maintain coolable core configurations. Therefore, the coolable geometry requirements of the criterion are met.

5.3.2 Large Break LOCA

The large break analysis was performed with the same methodology as the approved Sequoyah RLBLOCA analysis of record (Transition Program or Transition Package), found in Reference 10. The updates include error corrections / deviations from the approved AOR that were previously reported via 10 CFR 50.46 and changes to address recent issues brought up by the NRC. Specific deviations are outlined in Section 5.3.2.1 and in the attached RLBLOCA Summary Report (Reference 8).

The large break LOCA event is characterized by a postulated large rupture in the reactor coolant system cold leg. Two scenarios are run, both with loss of offsite power and no loss of offsite power. The non-parametric statistical approach of the RLBLOCA analysis samples key plant parameters such as break size and pressurizer pressure through an operational range. A mixed core of AREVA NP HTP 17x17 and AREVA Mk-BW 17x17 fuel is modeled for the analysis. The full list of sampled parameters and their range of values as well as more detailed large break LOCA event description may be found in the Summary Report (Reference 8). The purpose of the analysis is to verify typical technical specification peaking factor limits and the adequacy of the ECCS by demonstrating that the following 10CFR 50.46(b) criteria are met:

- Large break LOCA analysis results show that the limiting PCT occurred for a UO_2 rod in a case with offsite power available conditions. This case yielded a limiting PCT of 1941°F for a fresh UO_2 hot assembly.
- Results from the analysis show that the 10 CFR 50.46(b) Acceptance criteria for PCT, maximum oxide thickness, and hydrogen generation are met with significant margin.

As indicated, the RLBLOCA Summary Report (Reference 8) provides a more detailed summary of the large break LOCA analysis for Sequoyah Unit 1 and Unit 2.

5.3.2.1 Analysis Methodology and Computer Codes

The large break LOCA approach applied for Sequoyah Units 1 and 2 is based on the methodology documented in EMF-2103(P)(A) *Realistic Large Break LOCA Methodology* (Reference 4) with specific deviations outlined in Section 1 of the realistic large break LOCA (RLBLOCA) Summary Report (Reference 8). Supplemental information to address recent NRC RAIs is found in Section 6 of the RLBLOCA Summary Report (Reference 8). This altered methodology is referred to as the "Transition Program or Transition Package". This methodology follows the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation approach (Reference 11), which outlines an approach for defining and qualifying a best-estimate thermal hydraulic code and quantifies the uncertainties for the large break LOCA analysis. The RLBLOCA methodology conforms to the SRP Section 6.3 acceptance criteria for realistic evaluation models as described in Regulatory Guide 1.157.

The RLBLOCA methodology consists of the following computer codes:

- RODEX3A (Reference 5 and 6) for computation of the initial fuel stored energy, fission gas release, and fuel-cladding gap conductance.
- S-RELAP5 (Reference 7) for the system calculation.
- ICECON (Reference 12) for the containment backpressure calculation (coupled in S-RELAP5 code).

The following is a detailed description of the errors / deviations from the approved AOR mentioned above that have been addressed in the new analysis and the supplemental information that is provided to address recent NRC RAIs.

Cold Leg Condensation for RLBLOCA:

AREVA recently (2008) determined that, for analyses assuming a single train of pumped injection due to a single failure assumption, the S-RELAP5 modeling significantly under predicts the condensation in the cold legs and the downcomer during the reflood phase after the accumulators empty. Because of this, the ECCS water entering the downcomer is sufficiently subcooled to absorb the downcomer wall heat release without significant boiling. If the condensation were properly modeled, it is expected that the ECCS water would enter the downcomer in a saturated or only slightly subcooled state and that heat release from the downcomer would lead to boiling and reduction of reflood driving head.

Radiation to Fluid Heat Transfer model for RLBLOCA:

During the development of a new radiation heat transfer model (for Rev. 2 of the RLBLOCA methodology), a significant discrepancy between the currently used model in S-RELAP5 for the RLBLOCA methodology and other published models was discovered. A well known industry model was documented and installed into TRAC-B (sometime in the late 70's, early 80's). Part of the documentation for that model is a figure which shows radiation heat transfer data versus the TRAC-B model and the Thomson model. This figure has also been copied and published in other journals and documents. The radiation to fluid heat transfer model currently employed in S-RELAP5 for the RLBLOCA methodology used the flawed figure as the data basis for determining coefficients for the correlation of emissivity of water vapor. The result is that the S-RELAP5 radiation to fluid correlation under predicts the radiative heat transfer. This issue has been caused by flawed data used within the industrial community.

S-RELAP5 Coding of the point kinetics model:

The corrections were provided by the INL and then installed into S-RELAP5 (2007). Recently (2009), the INL announced that the previous error corrections were incorrect and that the recommended convergence criteria supplied with those corrections should be retained.

Heat conduction solution is incorrectly programmed:

The error is associated with using the incorrect heat capacity when evaluating the right boundary mesh point. Instead of using the last (adjacent) mesh interval heat capacity, the code incorrectly uses the next to last mesh interval heat capacity. The affect of the error is maximized in cylindrical and spherical geometries with few mesh points, which can be minimized with an increased number of mesh points. The effect is further minimized by the S-RELAP5 RLBLOCA, SBLOCA and Non-LOCA methodology guidelines requiring close mesh spacing at the left and right boundaries. This error exists exclusively in the RELAP5 series of codes.

Thermal Conductivity Degradation in Fuel Performance Codes:

The RODEX2 and RODEX3 code series have been questioned as to their ability to account for burnup dependent thermal conductivity. It is considered that they may under-predict the fuel pellet temperatures at burnup near and beyond 20 GWd/mtU and therefore not be appropriate for the initialization of LOCA evaluations.

S-RELAP5 FIJ multiplier and underpredicting liquid entrained to the steam generator tubes:

The impact of not entraining the appropriate amount of liquid into the steam generator tubes during a LBLOCA event. The Realistic Large Break LOCA (RLBLOCA) methodology uses a bias on interphase friction at the steam generator tube sheet entrance to insure an acceptable amount of liquid is entrained into the steam generator tubes during a large break. The bias determination was performed by comparing calculated results from S-RELAP5 with measured data from the Upper Plenum Test Facility (UPTF) Tests 10 and 29. The UPTF test facility represents a full scale, four-loop PWR complete with the necessary hardware that can be used to represent geometry specific phenomena that occurs during a large or small break LOCA. The S-RELAP5 parameter that controls entrainment is interphase friction. The range of interphase friction spans several orders of magnitude between the flow regimes occurring in the hot leg, hot leg riser, steam generator inlet plenum and steam generator tube sheet. Consequently, determining the uncertainty in interphase friction is not feasible; a conservative bias is used instead. The magnitude of the bias is determined by adjusting the S-RELAP5 RLBLOCA Multiplier “FIJ” until S-RELAP5 over-predicts the entrainment observed in UPTF Tests 10 and 29 by an arbitrary amount. Therefore, the FIJ multiplier of 1.75 is invalid and under-predicts the measured entrainment. The re-evaluation of the S-RELAP5 entrainment yielded a value of 5.0 for the FIJ multiplier is appropriate with a modeling change to the steam generator riser angle, greater than 30-degrees, and with the horizontal stratification flag set to off in the hot leg.

S-RELAP5 RLBLOCA Model producing Non-physical Phenomena in Upper Plenum

A recirculation pattern in the upper plenum nodes above the hot channel and surrounding 6 assemblies was producing liquid flowing into the core. This was traced back to the reactor vessel modeling of the 3-loop W plant for the EMF-2103 sample problem. The 3-loop sample problem had a geometric feature known as flow mixers (or standpipes). This feature split the upper plenum into two sections, one to an open hole region and one to a flow mixer region. The modeling in the sample problem blocked the cross flow between radial junctions in the first level of the upper plenum and this blockage was carried forward into plants without flow mixers as a conservatism. The RLBLOCA guideline discusses the top-down quench SER restriction and how the current modeling was setup to prevent the liquid drainback into the hot channel from occurring. Industry experience (SCTF) for LBLOCA has shown that steam velocity profiles during the transient would not allow liquid to fall back into the "hot channel" of the core. Thus, our conclusion that this is a non-physical phenomenon in the S-RELAP5 code. A high reverse loss coefficient was applied to the hot assembly and central core exit junctions to the upper plenum and all radial junction flow paths in the upper plenum were opened.

Recent NRC Generic Issues with RLBLOCA methodology

- The following issues are addressed in the RLBLOCA summary report (Reference 8):

1. Single Failure Assumption
2. Technical Specifications / Sampling Ranges
3. Thermal Conductivity Degradation
4. Fuel Swelling and Rupture, Relocation, and Co-planer Blockage
5. Single-Sided Oxidation
6. Decay Heat Sampling

5.3.2.2 RLBLOCA Analysis

The RLBLOCA analysis was performed in accordance with Reference 4 and to support application of the AREVA NP RLBLOCA analysis methodology to Sequoyah Units 1 and 2. The RLBLOCA analysis summary report (Reference 8) is attached for review for the HTP fuel transition because of to the number of deviations from the NRC approved RLBLOCA analysis (Reference 10). The EMF-2103(P)(A) with Transition Package methodology was used in the current analysis of record for the Sequoyah units with AREVA's Mk-BW fuel.

5.3.3 Containment Response

This section discusses the containment backpressure analysis used in the RLBLOCA analysis to support the HTP fuel transition. The concurrent containment transient pressure calculation is performed by the ICECON module within the NRC approved S-RELAP5 code (Reference 4). For the RLBLOCA analysis the dominant containment parameters, as well as nuclear steam supply system (NSSS) parameters, were established via a Phenomena Identification and Ranking Table (PIRT) process (Reference 4). Other model inputs are generally taken as nominal or conservatively biased. The PIRT outcome yielded two important (relative to peak cladding temperature) containment parameters – containment pressure and temperature. In many instances, the conservative guidance of Containment Systems Branch Technical Position 6-2 (Reference 13) was used in setting the remainder of the containment model input parameters.

Input Parameters

The RLBLOCA summary report (Reference 8) provides the general parameters used in the containment model for RLBLOCA analysis in Table 3-8 and Table 3-9 provides the structural heat sink data used in the containment model for RLBLOCA analysis. The containment pressure as a function of time for the limiting case is shown in Figure 4-31 in the RLBLOCA summary report (Reference 8). Ongoing processes ensure that the values and ranges used in the ECCS containment backpressure analyses for RLBLOCA bound the values and ranges of the plant operational parameters.

Acceptance Criteria

As specified in 10 CFR 50, Appendix K, the containment backpressure boundary condition analysis is acceptable if the containment pressure used for evaluating the cooling effectiveness during reflood is calculated conservatively for this purpose. The calculation includes the effects of all installed pressure reducing systems and processes.

LOCA SER Restrictions / Limitations

EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based"

- Purpose: Provide Pressurized Water Reactor (PWR) Small Break Loss of Coolant Accident (SBLOCA) evaluation methodology based on S-RELAP5 that applies to Westinghouse and Combustion Engineering PWRs with AREVA NP fuel.
- SER Restrictions: S-RELAP5 is acceptable for modeling transients where the break flow area is less than or equal to 10% of the cold leg flow area.
- Implementation of SER Restrictions: SBLOCA analyses performed with S-RELAP5 cover a break spectrum with an upper break size limited through the AREVA NP work flow process to no more that 10% of the cold leg flow area.

EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors"

- Purpose: RLBLOCA complies with the rules issued by the U.S. NRC in 1988 which allow the use of a realistic LOCA evaluation model in place of the prescribed conservative evaluation models as specified by 10 CFR 50 Appendix K. The NRC rule allows the use of realistic LOCA models provided that it can be established with a high probability that the criteria of 10 CFR 50.46 are not violated.
- SER Restrictions:
  1. A CCFL violation warning will be added to alert the analyst to CCFL violation in the downcomer should such occur.
  2. AREVA NP has agreed that it is not to use nodalization with hot leg to downcomer nozzle gaps.
  3. If AREVA NP applies the RLBLOCA methodology to plants using a higher planar linear heat generation rate (PLHGR) than used in the current analysis, or if the methodology is to be applied to an end-of-life analysis for which the pin pressure is significantly higher, then the need for a blowdown clad rupture model will be reevaluated. The evaluation may be based on relevant engineering experience and should be documented in either the RLBLOCA guideline or plant specific calculation file.
  4. Slot breaks on the top of the pipe have not been evaluated. These breaks could cause the loop seals to refill during late reflood and the core to uncover again. These break locations are an oxidation concern as opposed to a PCT concern since the top of the core can remain uncovered for extended periods of time. Should an analysis be performed for a plant with loop seals with bottom elevations that are below the top elevation of the core, AREVA NP will evaluate the effect of the deep loop seal on the slot breaks. The evaluation may be based on relevant engineering

experience and should be documented in either the RLBLOCA guideline or plant-specific calculation file

5. The model applies to 3 and 4 loop Westinghouse- and CE-designed nuclear steam systems.
6. The model applies to bottom reflood plants only (cold side injection into the cold legs at the reactor coolant discharge piping).
7. The model is valid as long as blowdown quench does not occur. If blowdown quench occurs, additional justification for the blowdown heat transfer model and uncertainty are needed or the calculation is corrected. A blowdown quench is characterized by a temperature reduction of the peak cladding temperature (PCT) node to saturation temperature during the blowdown period.
8. The reflood model applies to bottom-up quench behavior. If a top-down quench occurs, the model is to be justified or corrected to remove top quench. A top-down quench is characterized by the quench front moving from the top to the bottom of the hot assembly.
9. The model does not determine whether Criterion 5 of 10 CFR 50.46, long term cooling, has been satisfied. This will be determined by each applicant or licensee as part of its application of this methodology.
10. Specific guidelines must be used to develop the plant-specific nodalization. Deviations from the reference plant must be addressed.
11. A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the topical report approval process must be provided. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effects of that deviation.
12. The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses, including the calculated worst break size, PCT, and local and total oxidation.
13. The licensee or applicant wishing to apply AREVA NP realistic large break loss-of-coolant accident (RLBLOCA) methodology to M5 clad fuel must request an exemption for its use until the planned rulemaking to modify 10 CFR 50.46(a)(i) to include M5 cladding material has been completed.

- Implementation of SER Restrictions:

  See RLBLOCA Summary Report (Reference 8, attached) Section 3.4, Table 3-4 and Table 3-7 for responses to SER restrictions.

5.3.4 LOCA Technical Specification Changes

Remove BAW-10168(P)(A) from the Reference list and replace with EMF-2328(P)(A).

5.4 Conclusions

AREVA NP SBLOCA and RLBLOCA methods were applied in support of Sequoyah fuel transition to HTP fuel. The application demonstrates that all salient acceptance criteria associated with 10 CFR 50.46 with the exception of long-term cooling are met with the fuel change. SBLOCA and RLBLOCA summary reports are attached for NRC review (References 9 and 8, respectively). In these reports, AREVA describes any deviation from approved methods of analysis, demonstrates adherence to relevant SERs related to the methods, and responds to recent NRC questions regarding AREVA methodology applications.

TVA has assessed the impact that the HTP fuel transition would have on the minimum containment pressure analysis and concludes that the impact has been adequately addressed by the AREVA RLBLOCA analyses used to ensure that Sequoyah Unit 1 and Unit 2 will continue to meet its current licensing basis with respect to the requirements in 10 CFR 50.46 regarding ECCS performance following implementation of the HTP fuel. Therefore, TVA finds the proposed HTP fuel transition acceptable with respect to minimum containment pressure analysis for ECCS performance.

5.5 References for Section 5.0

Non-LOCA Transients (Sections 5.1 & 5.2)

1. BAW-10220P-A, Revision 0, Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 and 2, March 1996.

2. BAW-10164P-A, Revision 6, RELAP5/MOD2-B&W, An Advanced. Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis, January 2006.

3. BAW-10169P-A, RSG Plant Safety Analysis, October 1989.

4. BAW-10170P-A, Statistical Core Design for Mixing Vane Cores, December 1988.

5. BAW-10156-A, LYNXT Core Transient Thermal-Hydraulic Program, August 1993.

6. BAW-10159P, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, May 1986.

7. BAW-10180-A, Revision 1, NEMO – Nodal Expansion Method Optimized, March 1993.

8. BAW-10162P-A, TACO-3 – Fuel Pin Thermal Analysis Computer Code, October 1989.

9. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment 223 to Facility Operating License No. DPR-77 and Amendment 214 to Facility Operating License No. DPR-79, dated April 21, 1997.

LOCA (Sections 5.3 & 5.4)

1. XN-NF-81-58(P)(A), Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model", Exxon Nuclear Company Inc, March 1984.

2. ANF-81-58(P)(A), Revision 2 and Supplements 3 and 4, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model", Siemens Power Corporation, April 1990.

3. EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based" March 2001.

4. EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.

5. ANF-90-145(P)(A), Supplements 1, "RODEX3 Fuel Thermal-Mechanical Response Evaluation Model," Advanced Nuclear Fuels, April 1996.

6. ANF-90-145(P)(A), "RODEX3 - Fuel Rod Thermal -Mechanical Response Evaluation Model," Vol. 1, 2, and Supplement 1, April 1996.

7. EMF-2100(P) Rev. 13 "S-RELAP5 Models and Correlations Code Manual," February 2009.

8. AREVA NP Doc. ANP- 2970(P), Revision 0, Sequoyah Units 1 and 2 HTP Fuel Realistic Large Break LOCA Analysis, April 2011.

9. AREVA NP Doc. ANP- 2971(P), Revision 1, Sequoyah Units 1 and 2 HTP Fuel S-RELAP5 Small Break LOCA Analysis, May 2011.

10. AREVA NP Doc. ANP- 2655(P), Revision 1, Sequoyah Nuclear Plant Unit 2 Realistic Large Break LOCA Analysis, February 2008.

11. NUREG/CR-5249, EGG-2552, Technical Program Group, "Quantifying Reactor Safety Margins", October 1989.

12. EMF-CC-039(P), Supplement 1, Revision 4, ICECON Code Users Manual: A Computer Program Used to Calculate Containment Back Pressure for LOCA Analysis (Including Ice Condenser Plants), AREVA NP Inc, March 2006.

13. NUREG-0800 Revision 3 Standard Review Plan, U.S. Nuclear Regulatory Commission, Chapter 6 Engineered Safety Features, Branch Technical Position 6-2, Minimum Containment Pressure Model for PWR ECCS Performance Evaluation, March 2007.

14. AREVA NP Doc. ANP- 2695(P), Revision 0, Sequoyah Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis, February 2008.

ATTACHMENT 3

AREVA NP Affidavit

Attached is the affidavit supporting the request to withhold proprietary information (included in Attachment 1) from the public.

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in ANP-2986(P), Revision 003, entitled "Sequoyah HTP Fuel Transition," dated July 2011 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 22nd

day of July, 2011.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/14
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2014