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10 CFR 50.12
10 CFR 50.90

RS-06-116

September 26, 2006

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Subject: License Amendment Request and Exemption Requests to Allow Use of AREVA NP Inc. Advanced Mark-BW(A) Fuel Assemblies

References:

(1) Framatome ANP BAW-10239(P)-A, "Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report," dated July 2004

(2) Letter from M. Chawla (NRC) to J. L. Skolds (Exelon Generation Company, LLC), "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 – Issuance of Amendments," dated April 19, 2002

(3) Letter from S. A. Richards (NRC) to T. A. Coleman (Framatome Cogema Fuels), "Revised Safety Evaluation (SE) for Topical Report BAW-10227P: 'Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel,'" dated February 4, 2000

(4) Letter from S. Monarque (NRC) to D. A. Christian (Virginia Electric and Power Company), "North Anna Power Station, Units 1 and 2, Issuance of Exemption from the Requirements of 10 CFR 50.44, 10 CFR 50.46, and 10 CFR Part 50, Appendix K, to Allow the use of the M5 Alloy for Fuel Cladding Material," dated September 23, 2003

(5) Letter from H. N. Berkow (NRC) to J. F. Mallay (Framatome ANP), " Safety Evaluation of Framatome ANP Topical Report BAW-10186P-A, Revision 1, Supplement 1, 'Extended Burnup Evaluation,'" dated June 18, 2003

(6) Letter from G. F. Dick (NRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron and Braidwood – Issuance of Amendments on Spent Fuel Storage Racks," dated March 1, 2000

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an amendment to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, respectively. The proposed amendment would revise Technical Specification (TS) 4.2.1, "Fuel Assemblies," to allow up to eight AREVA NP Inc. (AREVA), formerly Framatome, modified Advanced Mark-BW fuel assemblies (i.e., Advanced Mark-BW(A) fuel assemblies) containing M5 alloy to be placed in nonlimiting (i.e., for $F_{\Delta H}$, F_Q, and fuel assembly average power at hot full power normal operating conditions) Braidwood Station Unit 1 core regions (i.e., locations) for evaluation during Cycles 14, 15, and 16, and Safety Limit (SL) 2.1.1, "Reactor Core SLs," to incorporate the peak fuel centerline temperature equations associated with the AREVA NP fuel in SL 2.1.1.3. The proposed amendment also revises the existing Amendment 122 Additional Condition in the Operating License, Appendix C, "Additional Conditions," to address operation during Cycles 14, 15, and 16 with up to eight AREVA NP Advanced Mark-BW(A) fuel assemblies containing fuel pellets incorporating homogeneous poisons. The license for Braidwood Station Unit 2 is affected only due to the fact that Unit 1 and Unit 2 use common TS.

The proposed amendment and exemption will permit Braidwood Station Unit 1 to load up to eight AREVA Advanced Mark-BW(A) fuel assemblies in the reactor core for operation in Cycles 14, 15, and 16 for evaluation. The purpose of this evaluation program is to allow EGC to gain an understanding of the behavior of mixed fuel cores prior to a possible transition from Westinghouse fuel to AREVA fuel. The Advanced Mark-BW(A) fuel assemblies use an advanced zirconium-based M5 alloy for the fuel assembly structural tubing, fuel rod cladding, and grids. In addition, the Advanced Mark-BW(A) fuel pellets may contain homogeneous poisons (i.e., gadolinia).

In accordance with 10 CFR 50.12, "Specific exemptions," EGC is also requesting exemption from the requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

Although the M5 alloy has been approved for use in pressurized water reactors (i.e., References 3 and 5), the alloy does not conform to the specifications for either Zircaloy or ZIRLO™. Therefore, exemptions from the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K are required to support the use of the AREVA Advanced Mark-BW(A) fuel.

A similar exemption request was previously granted for North Anna Power Station, Units 1 and 2 (i.e., Reference 4).

In addition to the above changes, EGC is requesting changes to TS 3.7.15, "Spent Fuel Pool Boron Concentration," TS 3.7.16, "Spent Fuel Assembly Storage," and TS 4.3.1, "Criticality." These changes are administrative and remove all references to Joseph Oat spent fuel pool storage racks that have been physically removed from the spent fuel pool. Braidwood Station, Units 1 and 2, were issued Amendment 105 that approved the installation of new Boral high-density spent fuel storage racks (i.e., Holtec storage racks) on March 1, 2000 (i.e., Reference 6). The replacement of Joseph Oat spent fuel storage racks with Holtec spent fuel storage racks at Braidwood Station was completed in December 2001.

The attached amendment request and exemption request are subdivided as shown below.

Attachment 1 provides an evaluation of the proposed changes.

Attachment 2 includes a marked-up copy of the operating license and TS pages with the proposed changes indicated.

Attachment 3 includes the associated revised operating license and TS pages with the proposed changes incorporated.

Attachment 4 includes the revised TS Bases pages with the proposed changes indicated. Note that the TS Bases pages are provided for information only and do not require prior NRC approval.

Attachment 5 provides the justification for the exemption request. EGC has concluded that special circumstances, as defined in 10 CFR 50.12 exist; that the granting of the requested exemption will not present an undue risk to the health and safety of the public; and that the granting of the requested exemption is consistent with the common defense and security.

Attachment 6 provides a list of regulatory commitments that were made in this submittal.

EGC plans to load up to eight AREVA Advanced Mark-BW(A) fuel assemblies in the Braidwood Unit 1 reactor core during the October 2007 refueling outage. EGC requests that approval of this license amendment and exemption request be granted prior to August 15, 2007, to allow sufficient time for core reload contingencies. Following approval, EGC will implement the change within 60 days.

The proposed amendment has been reviewed by the Braidwood Station Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

EGC is notifying the State of Illinois of this application for a change to the TS and proposed exemption requests by sending a copy of this letter and its attachments to the designated State Official.

Should you have any questions about this letter, please contact Ms. A. Mackellar at (630) 657-2817.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 26th day of September 2006.

Respectfully,

Keith R. Jury

Keith R. Jury
Director – Licensing and Regulatory Affairs

Attachment 1: Evaluation of Proposed Changes
Attachment 2: Markup of Proposed Operating License and Technical Specifications Page Changes
Attachment 3: Revised Operating License and Technical Specifications Pages

Attachment 4: Revised Technical Specification Bases Pages
Attachment 5: Justification for Exemption from 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models"
Attachment 6: List of Regulatory Commitments

ATTACHMENT 1
Evaluation of Proposed Changes

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ATTACHMENT 1
Evaluation of Proposed Changes

1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an amendment to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, respectively. The proposed amendment would revise Technical Specification (TS) 4.2.1, "Fuel Assemblies," to allow up to eight AREVA NP Inc. (AREVA), formerly Framatome, modified Advanced Mark-BW fuel assemblies (i.e., Advanced Mark-BW(A) fuel assemblies) containing M5 alloy to be placed in nonlimiting (i.e., for $F_{\Delta H}$, F_Q, and fuel assembly average power at hot full power normal operating conditions) Braidwood Station Unit 1 core regions (i.e., locations) for evaluation during Cycles 14, 15, and 16, and Safety Limit (SL) 2.1.1, "Reactor Core SLs," to incorporate the peak fuel centerline temperature equations associated with the AREVA NP fuel in SL 2.1.1.3. The proposed amendment also revises the existing Amendment 122 Additional Condition in the Operating License, Appendix C, "Additional Conditions," to address operation during Cycles 14, 15, and 16 with up to eight AREVA NP Advanced Mark-BW(A) fuel assemblies containing fuel pellets incorporating homogeneous poisons. The license for Braidwood Station Unit 2 is affected only due to the fact that Unit 1 and Unit 2 use common TS.

The proposed amendment and exemption will permit Braidwood Station Unit 1 to load up to eight AREVA Advanced Mark-BW(A) fuel assemblies in the reactor core for operation in Cycles 14, 15, and 16 for evaluation. The purpose of this evaluation program is to allow EGC to gain an understanding of the behavior of mixed fuel cores prior to a possible transition from Westinghouse fuel to AREVA fuel. The Advanced Mark-BW(A) fuel assemblies use an advanced zirconium-based M5 alloy for the fuel assembly structural tubing, fuel rod cladding, and grids. In addition, the Advanced Mark-BW(A) fuel pellets may contain homogeneous poisons (i.e., gadolinia).

In accordance with 10 CFR 50.12, "Specific exemptions," EGC is also requesting exemption from the requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

Although the M5 alloy has been approved for use in pressurized water reactors (i.e., References 3 and 5), the alloy does not conform to the specifications for either Zircaloy or ZIRLO™. Therefore, exemptions from the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K are required to support the use of the AREVA Advanced Mark-BW(A) fuel. The justification for the exemption request is included as Attachment 5 to this submittal.

EGC has concluded that special circumstances, as defined in 10 CFR 50.12 exist; that the granting of the requested exemption will not present an undue risk to the health and safety of the public; and that the granting of the requested exemption is consistent with the common defense and security.

In addition to the above changes, EGC is requesting changes to TS 3.7.15, "Spent Fuel Pool Boron Concentration," TS 3.7.16, "Spent Fuel Assembly Storage, " and TS 4.3.1, "Criticality." These changes are administrative and remove all references to Joseph Oat spent fuel pool storage racks that have been physically removed from the spent fuel pool. Braidwood Station, Units 1 and 2, were issued Amendment 105 that approved the installation of new Boral high-density spent fuel storage racks (i.e., Holtec storage racks) on March 1, 2000 (i.e., Reference

6). The replacement of Joseph Oat spent fuel storage racks with Holtec spent fuel storage racks at Braidwood Station was completed in December 2001.

2.0 PROPOSED CHANGE

SL 2.1.1.3

EGC proposes to revise the existing wording of SL 2.1.1.3 by adding the fuel centerline temperature equations for the AREVA fuel. The revised SL 2.1.1.3, with changes italicized, will be as follows.

> " In MODES 1 and 2, the peak fuel centerline temperature shall be maintained *as follows:*
>
> *a. < 5080°F, decreasing by 58°F per 10,000 MWD/MTU burnup for Westinghouse fuel,*
>
> *b. < 5173°F decreasing by 65°F per 10,000 MWD/MTU burnup for AREVA NP fuel (Unit 1 only), and*
>
> *c. < 5189°F decreasing by 65°F per 10,000 MWD/MTU burnup for AREVA NP fuel containing Gadolinia (Unit 1 only)."*

TS 4.2.1

EGC proposes to revise the existing wording of TS 4.2.1 by noting the exception of the AREVA fuel and by adding the following statement:

> "Up to 8 AREVA NP Advanced Mark-BW(A) fuel assemblies containing M5 alloy may be placed in nonlimiting Unit 1 core regions for evaluation during Cycles 14, 15, and 16."

Operating License, Appendix C

EGC proposes to revise the existing Amendment 122 Additional Condition by adding the following statement:

> "During operation in Cycles 14, 15, and 16, up to eight (8) AREVA NP Advanced Mark-BW(A) fuel assemblies containing fuel pellets incorporating homogeneous poisons may be placed in nonlimiting Unit 1 core regions (i.e., locations). The design basis for the AREVA NP fuel rod centerline melt follows that given in BAW-10162P-A, "TACO3 – Fuel Pin Thermal Analysis Computer Code," October 1989, and BAW-10184P-A, "GDTACO – Urania Gadolinia Fuel Pin Thermal Analysis Code," February 1995."

TS 3.7.15, TS 3.7.16 and TS 4.3.1

EGC proposes to revise TS 3.7.15, TS 3.7.16 and TS 4.3.1 to remove all references to Joseph Oat spent fuel pool storage racks. The removal of these references is editorial only and will delete wording that is no longer required. The revisions will result in various editorial and formatting change requirements and as a result of the proposed revision to TS 3.7.16,

Surveillance Requirement (SR) 3.7.16.3, and existing Figures 3.7.16-1, 3.7.16-2, and 3.7.16-3 will be deleted in their entirety.

3.0 BACKGROUND

The Braidwood Station Unit 1 core consists of 193 fuel assemblies. The core may consist of any combination of Westinghouse VANTAGE 5 and VANTAGE+ fuel assemblies arranged in a checkered low-leakage pattern. Each fuel assembly consists of 264 fuel rods arranged in a 17 x 17 array. The VANTAGE+ fuel assembly design includes the following features: ZIRLO™ clad fuel rods, ZIRLO™ thimble and instrumentation tubes, and a variable pitch plenum spring. The VANTAGE 5 design has added features, known as PERFORMANCE+ design features, which are: ZIRLO™ intermediate grids and flow mixer grids, an oxide protective coating at the lower end of the fuel rod cladding, and a protective bottom grid.

EGC intends to place up to eight AREVA Advanced Mark-BW(A) fuel assemblies in nonlimiting regions (i.e., locations) of the core for evaluation during Cycles 14, 15, and 16. The Advanced Mark-BW(A) fuel assemblies are similar in design to the Advanced Mark-BW assemblies generically approved for use in Westinghouse 3- and 4-loop designed pressurized water reactors with 17 x 17 fuel rod arrays (i.e., Reference 1). The Advanced Mark-BW(A) fuel assemblies incorporate the following minor modifications relative to previously irradiated Advanced Mark-BW assemblies: removable upper end fitting with quarter-turn quick-disconnect feature, M5 MONOBLOC™ guide tubes (and corresponding diameter changes to the central instrument tube), welded connections between the M5 intermediate spacer grids and guide tubes, use of a standard M5 vaned mixing grid in the second position from the bottom of the assembly replacing the otherwise M5 non-vaned grid, use of Alloy 718 HMP (High Mechanical Performance) spacer grids at both the top and bottom positions of the assembly, and a FUELGUARD™ lower end fitting. The purpose of this evaluation program is to allow EGC to gain an understanding of the behavior of mixed fuel cores prior to a possible transition from Westinghouse fuel to AREVA fuel.

The AREVA Advanced Mark-BW(A) fuel assemblies use an advanced zirconium-based M5 alloy for the fuel assembly structural tubing, fuel rod cladding, and grids. The NRC has previously approved the use of the M5 alloy in References 3 and 5. Existing TS 4.2.1 does allow a limited number of lead test assemblies that have not completed representative testing to be placed in nonlimiting core regions (i.e., locations), however, the current TS 4.2.1 restricts fuel rod cladding materials to Zircaloy or ZIRLO™. Representative testing of Advanced Mark-BW lead test assemblies has been completed, as described in Reference 1. Changes to TS 4.2.1 are therefore required to allow the use of fuel assemblies containing M5 alloy as a cladding and structural material.

The Advanced Mark-BW(A) fuel pellets may contain homogeneous poisons (i.e., gadolinia). The current Operating License, Appendix C, Amendment 122 Additional Condition, states in part:

> "If fuel pellets incorporating homogenous poisons are used, the topical report documenting the fuel centerline melt temperature basis must be reviewed and approved by the NRC and referenced in this license condition. TS 2.1.1.3 must be modified to also include the fuel centerline melt temperature limit for the fuel with homogeneous poison."

The proposed change to the Appendix C additional conditions incorporates the topical report documenting the fuel centerline melt basis for the AREVA fuel. As described in the following section, the fuel cycle design for Cycles 14, 15, and 16 will be developed such that the peak fuel centerline temperature of the AREVA Advanced Mark-BW(A) fuel assemblies will be bounded by the safety limit currently described in TS 2.1.1.3 for Westinghouse fuel.

4.0 TECHNICAL ANALYSIS

The NRC has previously reviewed the use of M5 alloy as a cladding and structural material. References 3 and 5 provided the NRC's acceptance of the use of M5 alloy for reload licensing applications up to rod average burnup levels of 62,000 MWD/MTU for Mark B and Mark-BW fuel designs.

The NRC has also previously reviewed the performance of the AREVA Advanced Mark-BW fuel assembly design against the relevant design criteria. The Advanced Mark-BW fuel design is an evolution of the Mark-BW design and includes new design features including: the TRAPPER™ bottom nozzle, mid-span mixing grids, a floating intermediate grid design, a quick connect/disconnect upper end fitting, and use of M5 material for the cladding, structural tubing, and grids. Reference 10 provides the NRC approval for licensees with Westinghouse three- and four-loop reactors that use a 17 X 17 fuel rod array to reference the generic topical report (i.e., Reference 1) for use of the Advanced Mark-BW fuel assemblies, subject to the following two conditions:

1) The 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, and

2) The Advanced Mark-BW fuel assembly design is licensed for a maximum fuel rod burnup of 62,000 MWD/MTU.

Advanced Mark-BW(A) Fuel Assembly Design Features

The AREVA Advanced Mark-BW(A) fuel assembly design proposed to be used in Braidwood Station Unit 1 Cycles 14, 15, and 16 incorporates several minor modifications to the Advanced Mark-BW fuel assemblies. Reference 1 contains a discussion of the design change process under which the Advanced Mark-BW fuel design may be changed without requiring NRC review and approval. The minor modifications incorporated in the Advanced Mark-BW(A) design were performed in accordance with this process. Specific differences include a removable upper end fitting with quarter-turn quick-disconnect feature, M5 MONOBLOC™ guide tubes (and corresponding diameter changes to the central instrument tube), welded connections between the M5 intermediate spacer grids and guide tubes, use of a standard M5 vaned mixing grid in the second position from the bottom of the assembly replacing the otherwise M5 non-vaned grid, use of Alloy 718 HMP spacer grids at both the top and bottom positions of the assembly, and a FUELGUARD™ lower end fitting. A description of each of these specific differences follows. Because these minor modifications are allowed by Reference 1, the Advanced Mark-BW(A) fuel assemblies are still subject to the two conditions for use of the Advanced Mark-BW fuel assemblies provided in Reference 10.

Upper End Fitting

The Advanced Mark-BW upper end fitting incorporated a quick-disconnect feature, a low pressure drop grillage and a three-leaf hold-down spring system. The 304 stainless steel upper end fitting incorporates four sets of three-leaf hold-down springs made of Alloy-718 fastened to the end fitting with Alloy-718 clamp screws. The upper leaf has an extended tang that engages a cutout in the top plate of the end fitting.

The Advanced Mark-BW(A) upper end fitting is very similar to the upper end fitting of the Advanced Mark-BW fuel assembly design. The upper end fitting assembly for the Advanced Mark-BW(A) incorporates a similar grillage design and the same spring system. There will, however, be some minor dimensional differences such as a thicker grillage to increase the design stress margin and greater lead-in for the machined features at the bottom of the grillage for reduced pressure drop. All critical interface dimensions will be verified to ensure the fit-up with the upper core plate is acceptable. The quick-disconnect feature is also modified with a new attachment that has been used extensively in other AREVA fuel designs.

Guide Tubes

As in the Advanced Mark-BW fuel assemblies, the Advanced Mark-BW(A) guide tubes are fabricated from M5 alloy. Both versions incorporate two inner diameters. The larger diameter at the top provides a relatively large annular clearance that permits rapid insertion of the rod cluster control assembly during a reactor trip and accommodates coolant flow during normal operation. A reduced diameter section, at the lower end of the tube provides a dashpot action that decelerates the control rods near the end of the control rod travel during a reactor trip. This deceleration limits the magnitude of the rod cluster control assembly impact loads on the top of the end fitting. Four small holes located just above the dashpot allow both outflow of water during rod cluster control assembly insertion and coolant flow to components during operation. The Advanced Mark-BW(A) MONOBLOC™ guide tubes differ in that the outside diameter is the same over the entire length of the guide tube to provide additional lateral stiffness to reduce fuel assembly twist and bow. Corresponding with this change to MONOBLOC™ guide tubes, the central instrument tube diameters are changed to match those of the upper region of the MONOBLOC™ guide tubes.

Welded Structure

To ensure axial alignment of intermediate spacer grids with adjacent fuel assemblies, the Advanced Mark-BW design incorporates stops on selected guide tubes that limit grid movement. In contrast, the Advanced Mark-BW(A) assembly is a welded cage design, which is based on the successful experience of many other AREVA fuel assembly designs. The design utilizes spot-welded connections between weld tabs extending from the edges of the M5 grid strips and the M5 guide tubes.

Intermediate Spacer Grids

As in the Advanced Mark-BW fuel assemblies previously irradiated, the Advanced Mark-BW(A) intermediate spacer grids in the active fuel region are made of M5 alloy. For the previous Advanced Mark-BW design, the upper five (5) intermediate spacer grids included mixing vanes on the strips, projecting from the trailing (upper) edges into the coolant; however, the lowermost

intermediate spacer grid was a non-vaned type. The Advanced Mark-BW(A) design utilizes the same vaned grid type in all six locations for additional thermal performance.

Top and Bottom End Grids

The Advanced Mark-BW end grids utilize Alloy 718 to ensure proper gripping of the fuel rod through end of life. The Advanced Mark-BW(A) fuel assembly also utilizes Alloy 718 end grids; however, they are the HMP type which are similar in design to the HTP (High Thermal Performance) type grid and which has been successfully used in numerous other AREVA fuel assembly designs.

Lower End Fitting

The Advanced Mark-BW fuel assembly design incorporated the TRAPPER™ lower end fitting, which was designed with debris-resistant features. The stainless steel bottom nozzle consisted of a frame of deep ribs connecting the guide thimble locations and conventional legs that interface with the reactor internals. A high strength A-286 alloy filter plate was attached to the top of the frame. The filter plate served two functions. First, it provided the axial restraint for fuel rods, which were seated on the filter plate, by distributing these loads to the structural frame. Secondly, it provided an effective barrier to debris while maintaining an acceptable pressure drop.

The Advanced Mark-BW(A) fuel assembly design incorporates the FUELGUARD™ lower end fitting. This end fitting is a cast, machined and brazed assembly that is fastened to the guide tubes with mechanically captured stainless steel screws. A combination of curved blades and perpendicular bars, which provides a "no-line-of-sight" flow path for the coolant, minimizes debris entering from the bottom of the fuel assembly.

Comparison of AREVA Advanced Mark-BW(A) fuel assemblies to Westinghouse Resident Fuel Assemblies

The AREVA Advanced Mark-BW(A) fuel assemblies are mechanically similar to, and fully compatible with the resident Westinghouse fuel. The primary differences between the resident Westinghouse fuel design and the AREVA fuel design include the use of the different zirconium-based alloys for fuel rod cladding, fuel assembly structural tubing, and spacer grids; use of higher nominal fuel pellet density in the Advanced Mark-BW(A) fuel assemblies; use of a different burnable absorber (i.e., gadolinia) and larger diameters for fuel pellets, fuel rods, guide tubes, and instrument tubes (see following table). Note that the differences in density and in fuel pellet diameter will result in a higher uranium loading than in the Westinghouse fuel design. For the core physics model and most other areas, fuel geometry information for the AREVA Advanced Mark-BW(A) fuel assemblies will be used to ensure modeling fidelity.

Incorporation of the quick connect/disconnect upper end fitting with quarter-turn quick-disconnect feature, the use of the FUELGUARD™ lower end fitting, and the use of the MONOBLOC™ guide tubes, the welded structure, the vaned mixing grid at the lower intermediate position, and the HMP end grids are not expected to affect the compatibility of the AREVA Advanced Mark-BW(A) fuel assemblies with the resident fuel.

Comparison of Selected AREVA Advanced Mark-BW(A) and Westinghouse OFA Fuel Parameters

Parameter	AREVA Advanced Mark-BW(A)	Westinghouse OFA	Westinghouse RFA
Fuel Pellet Diameter (in)	0.3225	0.3088	0.3225
Fuel Rod Inside Diameter (in)	0.329	0.315	0.329
Fuel Rod Outside Diameter (in)	0.374	0.360	0.374
Fuel Pellet Density (%)	96.0	95	95.5
Clad Material	M5	ZIRLO™	ZIRLO™
End Grid Material	Inconel 718	Inconel 718	Inconel
Mid Grid Material	M5	ZIRLO™	ZIRLO™
Burnable Absorbers	Gadolinia	IFBA/WABA	IFBA/WABA

Evaluations

Braidwood Station Unit 1 reloads are performed using the analytical methods specified in TS 5.6.5, "Core Operating Limits Report (COLR)." TS 5.6.5.c requires that core operating limits be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as shutdown margin (SDM), transient analysis limits, and accident analysis limits) of the safety analysis are met. To provide assurance that the current bounding evaluations performed for Braidwood Station Unit 1 reloads will remain valid, the AREVA fuel assemblies being used for Braidwood Station Unit 1 Cycles 14, 15, and 16 will be placed in nonlimiting (i.e., for $F_{\Delta H}$, F_Q, and fuel assembly average power at hot full power normal operating conditions) core regions (i.e., locations) and the nuclear design of the Braidwood Station Unit 1 Cycles 14, 15, and 16 cores performed by Westinghouse will ensure sufficient margin between the lead Westinghouse fuel assembly and the AREVA Advanced Mark-BW(A) assemblies for $F_{\Delta H}$, F_Q, and for fuel assembly average power; these margins will be a minimum of 5%. The reload analysis will ensure that the applicable acceptance criteria continue to be met. In addition, the AREVA Advanced Mark-BW(A) fuel assemblies will not be placed in locations containing rod cluster control assemblies.

Evaluations will be performed to ensure that the AREVA Advanced Mark-BW(A) fuel assemblies do not have an adverse impact on the co-resident Westinghouse fuel. Confirmatory evaluations will be performed to demonstrate that the AREVA Advanced Mark-BW(A) fuel assemblies will satisfy the inputs and assumptions of the current Westinghouse Analysis of Record (AOR). The Advanced Mark-BW(A) fuel assemblies will meet AREVA's own mechanical and thermal-hydraulic limits per Topical Report BAW-10239(P)-A and other approved methodologies as discussed in this submittal. Therefore, the list of approved methodologies in TS 5.6.5 is not required to be updated to include the AREVA methodologies.

The Westinghouse Robust Fuel Assembly (RFA) being used by Westinghouse to model the AREVA Advanced Mark-BW(A) assemblies for LOCA and seismic purposes, as discussed below, differs from the AREVA Advanced Mark-BW(A) assemblies in the use of the different zirconium-based alloys for fuel rod cladding, fuel assembly structural tubing, and spacer grids; use of higher nominal fuel pellet density in the AREVA Advanced Mark-BW(A) fuel assemblies; use of a different burnable absorber (i.e., gadolinia); and larger diameters for the guide tubes and instrument tubes. However, the Westinghouse RFA assemblies have the same diameters

for fuel pellets and fuel rods as the AREVA Advanced Mark-BW(A) assemblies and have similar axial locations for the spacer grids; therefore, they provide a good model for determining the effects of the AREVA Advanced Mark-BW(A) assemblies on these two analyses. Note that the core physics evaluations discussed below will model the AREVA Advanced Mark-BW(A) fuel assemblies to ensure fidelity.

Mechanical Design Methodology

AREVA will evaluate the Advanced Mark-BW(A) fuel assembly mechanical performance using the methods outlined in Topical Report BAW-10239(P)-A (i.e., Reference 1). The mechanical analyses will take into consideration the changes in the fuel assembly structure relative to the Advanced Mark-BW design (e.g., the use of MONOBLOC™ guide tubes, the welded structure, the application of different spacer grid types, and different upper and lower end fittings). Also, the Advanced Mark-BW(A) fuel assemblies will be evaluated with respect to Braidwood Station Unit 1 specific operating conditions. The mechanical analyses will evaluate the following:

1. The fuel assembly will be evaluated for axial growth of both the fuel bundle and the fuel rods. Growth models and methods for the M5 fuel rod cladding and guide tube material described in Topical Report BAW-10227P-A (i.e., Reference 11) will be utilized, in accordance with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 4.2, "Fuel System Design," to show acceptable fuel rod shoulder gap (i.e., the axial spacing to allow fuel rod growth) at the design burnup and margin to prevent the fuel assembly from going solid between the upper and lower core plate.

2. Fuel assembly lift off will be evaluated using the LYNXT code in conjunction with the NRC approved statistical fuel assembly hold down methodology described in BAW-10243(P)(A), (i.e., Reference 23). This methodology statistically treats mechanical and thermal-hydraulic uncertainties while maintaining compliance to NUREG-0800, "Standard Review Plan for the review of Safety Analysis Reports for Nuclear Power Plants," Section 4.2, "Fuel System Design and Section 4.4, "Thermal and Hydraulic Design." The analysis will show that the fuel assembly does not lift off from the lower core plate under mechanical design flow conditions.

3. The AREVA Advanced Mark-BW(A) fuel assembly components will be evaluated for stress and fatigue as appropriate for normal operating conditions. This includes the M5 fuel rod assembly which be evaluated for cladding stress, fatigue, creep collapse and transient strain performance using the methods described in BAW-10227P-A (i.e., Reference 11).

4. An evaluation of AREVA Advanced Mark-BW(A) fuel rod oxide and corrosion levels using the COROS02 code as described in BAW-10227P-A (i.e., Reference 11) assuming a bounding power history along with a bounding fuel rod lifetime will be performed.

5. The fuel rod internal gas pressure predictions for the AREVA Advanced Mark-BW(A) fuel assemblies will be made using the TACO3 code (i.e., Reference 12) for the UO_2 fuel and the GDTACO code (i.e., Reference 13) for the gadolinia-bearing fuel. Note that for assemblies with gadolinia, only a relatively small number of the fuel rods have gadolinia, and the rest are UO_2 only. Fuel rod internal gas pressures will be permitted to exceed the reactor coolant system pressure according to the approved methodology contained in BAW-10183P-A (i.e., Reference 14). The TACO3 and GDTACO codes are approved for a maximum fuel rod

burnup of 62,000 MWD/MTU, which is also the approved burnup for the Advanced Mark-BW(A) fuel (i.e., Reference 10).

Seismic

Westinghouse will perform a set of evaluations to confirm that the seismic impacts of up to eight Westinghouse RFA-2 fuel assemblies (i.e., the fuel assemblies used by Westinghouse to model the AREVA Advanced Mark-BW(A) fuel assemblies) would not invalidate the Westinghouse grid deformation analysis. EGC will review grid strength information for the Westinghouse analysis and the Advanced Mark-BW(A) design, and independently confirm that the analysis maintains adequate margin to limits with the AREVA designed fuel.

Core Physics

The AREVA Advanced Mark-BW(A) fuel assemblies will be modeled by Westinghouse, using current methodologies as described in TS 5.6.5.b, using AREVA fuel geometry information provided by AREVA. The nuclear design of the Braidwood Station Unit 1 Cycles 14, 15, and 16 cores by Westinghouse will ensure sufficient margin between the lead Westinghouse fuel assembly and the AREVA Advanced Mark-BW(A) assemblies for $F_{\Delta H}$, F_Q, and for fuel assembly average power. The amount of margin required will be determined by the amount needed to show that the AREVA Advanced Mark-BW(A) fuel satisfies the inputs and assumptions of the current Westinghouse AOR; these margins will be a minimum of 5%.

As stated previously, the NRC approval of the AREVA Advanced Mark-BW topical report BAW-10239(P)-A (i.e., Reference 1) contained two conditions:

1. The 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, and

2. The Advanced Mark-BW fuel assembly design is licensed for a maximum fuel rod burnup of 62,000 MWD/MTU.

The nuclear designs for the Braidwood Station Unit 1 Cycles 14, 15, and 16 cores will ensure that these two conditions are met.

In addition, the Westinghouse nuclear design will generate and provide the data required by AREVA to perform confirmatory analyses as described in the summaries below.

Loss-of-Coolant Accidents

Of potential concern to the loss-of-coolant accident (LOCA) evaluation is mixed core effects; the AREVA Advanced Mark-BW(A) assemblies are expected to have a higher axial pressure drop than the resident Westinghouse fuel. The pressure drop difference between the two fuel designs is expected to be small. Ultimately, the peak cladding temperature (PCT) for the AREVA Advanced Mark-BW(A) fuel assemblies will be qualified to a net value (i.e., the F_Q reduction plus any potential mixed core increase) that is lower than the Westinghouse AOR value for the resident Westinghouse fuel. This PCT qualification fixes the magnitude of the peaking reduction required; the F_Q reduction will be a minimum of 5%. This assessment will also be used to show that the AREVA Advanced Mark-BW(A) fuel assemblies meet the 17% fuel rod cladding oxidation limit.

Small break LOCA (SBLOCA) is assumed to be bounded by LBLOCA because the resultant PCT associated with SBLOCA is significantly less than for LBLOCA. SBLOCA is mostly plant system determinant and is not dependent on fuel assembly design for reasonably equivalent designs. The SBLOCA event is driven by decay heat, safety injection flow rates and break size. Small differences in fuel rod characteristics have little effect on the event. The initial stored energy of the fuel is of little concern in SBLOCA since it is released quickly following reactor scram. SBLOCA is characterized by fuel rod heatup after core uncovery due to decay heat. Fuel thermal conductivity and gap conductance are of little importance during the heatup phase since the temperature distribution across the fuel rod is fairly uniform. The dominant effect is the reduced decay heat in the AREVA Advanced Mark-BW(A) fuel integrated over a long time period. Therefore, the fuel assembly average power reduction in the AREVA Advanced Mark-BW(A) fuel should be sufficient to assure it will not be limiting relative to the co-resident Westinghouse fuel for SBLOCA. Hence, LBLOCA will be the area of interest and concentration.

Westinghouse will also evaluate Westinghouse RFA-2 assemblies without gadolinia in the AREVA Advanced Mark-BW(A) fuel assembly core regions (i.e., locations) for their impact on the co-resident Westinghouse fuel. Westinghouse will use an evaluation method that relies on an enrichment cutback to ensure that the gadolinia rods do not lead the core. The LOCA evaluation will quantify and evaluate the RFA-2 assembly for PCT impacts on the resident Westinghouse fuel, as applicable. Westinghouse will also evaluate the resident Westinghouse fuel assemblies and the AREVA Advanced Mark-BW(A) fuel assemblies with respect to maximum hydrogen generation, coolable geometry, and long-term cooling.

Non-LOCA Events

Departure from Nucleate Boiling (DNB) calculations will be performed to ensure that the AREVA Advanced Mark-BW(A) fuel assemblies satisfy the core safety limit lines, the limiting transients, and the core operating limits. Evaluations will show that the AREVA Advanced Mark-BW(A) fuel has more margin to its applicable DNB limit than the Westinghouse fuel has to its applicable limit. Non-LOCA analyses with DNB acceptance criteria described in the Updated Final Safety Analysis Report (UFSAR), which assume a full core of Westinghouse fuel, will therefore be conservative relative to a Braidwood Unit 1 core containing the AREVA Advanced Mark-BW(A) assemblies.

AREVA will perform thermal margin calculations to evaluate the Advanced Mark-BW(A) fuel assemblies for DNB performance using the generically approved XCOBRA-IIIC code (i.e., Reference 16) with the approved Critical Heat Flux (CHF) correlation discussed in BAW-10244P-A (i.e., Reference 15).

Westinghouse will provide axial pressure drop profiles of the resident Westinghouse fuel design to allow AREVA to determine loss coefficients. Given this data, along with fuel assembly geometry data provided by Westinghouse, AREVA will determine the localized flow redistribution occurring in the mixed core environment when assessing the DNB performance. A mixed core DNB penalty for the AREVA Advanced Mark-BW(A) fuel will be determined, if appropriate.

Westinghouse will provide statepoint conditions associated with the TS Safety Limits and the limiting safety analyses for the DNB assessment. These statepoints will reflect the DNB limiting

safety analysis statepoint conditions and will contain the respective operating conditions and axial power shapes.

As previously noted, the nuclear design of the Braidwood Station Unit 1 Cycles 14, 15, and 16 cores by Westinghouse will ensure a sufficient margin between the lead Westinghouse fuel assembly and the AREVA Advanced Mark-BW(A) assemblies for $F_{\Delta H}$, F_Q, and for fuel assembly average power; these margins will be a minimum of 5%. The statepoint conditions will be evaluated by each vendor at the Braidwood Station Unit 1 $F_{\Delta H}$ COLR limit, and EGC will compare the results to quantify the necessary power reduction on the AREVA Advanced Mark-BW(A) fuel assemblies. This will ensure that the AREVA Advanced Mark-BW(A) fuel assemblies have more DNB margin than the resident Westinghouse fuel assemblies, and satisfy the TS Safety Limits and the COLR operating limits.

By demonstrating nonlimiting DNB performance for the AREVA Advanced Mark-BW(A) fuel assemblies in the mixed core environment, the surveillance of the AREVA Advanced Mark-BW(A) fuel assemblies to the Westinghouse $F_{\Delta H}$ limits will assure:

1. The AREVA Advanced Mark-BW(A) fuel assemblies will operate with acceptable DNB performance,

2. The overtemperature differential temperature (OTΔT) trip function, developed for the Westinghouse fuel, will provide DNB protection the AREVA Advanced Mark-BW(A) fuel assemblies, and

3. The DNB analysis is in compliance with the DNB protection defined in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 4.4, "Thermal and Hydraulic Design."

AREVA will also confirm that the AREVA Advanced Mark-BW(A) fuel assembly fuel temperatures do not exceed bounding temperatures (i.e., pellet average and pellet surface) provided by Westinghouse. These temperatures will be calculated by AREVA using the NRC-approved TACO3 and GDTACO codes (i.e., References 12 and 13, respectively).

Westinghouse will perform evaluations of the non-LOCA events to ensure the AREVA Advanced Mark-BW(A) fuel assemblies do not adversely impact the analyses for these transients. The transients will be evaluated based on the imposed power and peaking factor constraints on the AREVA Advanced Mark-BW(A) fuel assemblies.

Westinghouse will use the current AOR for the Westinghouse fuel as the basis for confirming the reload. The thermal-hydraulic mixed core methodology is to evaluate the core as a full core of one fuel type and apply any mixed core penalties, if appropriate.

The impacts on the thermal-hydraulic design will be primarily determined via a flow redistribution analysis. AREVA will provide axial pressure drop profiles for the AREVA Advanced Mark-BW(A) fuel assembly design to allow Westinghouse to determine appropriate loss coefficients. Given this data, along with fuel assembly geometry data provided by AREVA, Westinghouse will determine the localized flow redistribution occurring in the mixed core environment. A DNB penalty will be assessed during the reload design to account for the flow redistribution, if appropriate.

Thermal-Hydraulic

AREVA will perform Advanced Mark-BW(A) fuel thermal hydraulic analyses using the Braidwood Station Unit 1 Cycles 14, 15, and 16 operating conditions.

The AREVA Advanced Mark-BW(A) fuel assemblies will be evaluated by AREVA for fuel rod bow and its impact on mechanical and thermal-hydraulic performance. After the evaluation, it is expected that the standard Mark-BW bow penalty will be applied because the Advanced Mark-BW(A) fuel rod design and the design of the spacer grid fuel rod cells are equivalent to the Advanced Mark-BW design, which also used the standard Mark-BW penalty. The standard Mark-BW bow penalty is based on Zircaloy-4 fuel rod bow, which is expected to be greater than that of M5 fuel rods given the same operating conditions because M5 rods exhibit less axial growth (i.e., Reference 11).

Westinghouse will use guide tube and instrument tube flow data provided by AREVA to confirm whether there is an impact to bypass flow due to the AREVA Advanced Mark-BW(A) assemblies.

In addition, the Westinghouse calculation of core lift will incorporate the results of the Westinghouse analysis on flow redistribution resulting from the mixed core.

Fuel Centerline Melt Temperature

Westinghouse will provide a fuel rod linear heat rate limit, along with associated operating conditions, for the AREVA Advanced Mark-BW(A) assemblies. If the limiting (minimum) linear heat rate for the AREVA Advanced Mark-BW(A) fuel is greater than that required by the Westinghouse analysis at the same conditions, then the AREVA Advanced Mark-BW(A) fuel is nonlimiting for fuel centerline melt. If the limiting (minimum) linear heat rate for the AREVA Advanced Mark-BW(A) fuel is less than required by the Westinghouse analysis at the same conditions, then AREVA will determine the amount of $F_{\Delta H}$ cutback that is required in the cycle nuclear design to demonstrate that the AREVA Advanced Mark-BW(A) assemblies have greater UO_2 fuel centerline fuel melt (CFM) margin than the resident Westinghouse UO_2 fuel; this $F_{\Delta H}$ cutback will be a minimum of 5%. For the AREVA Advanced Mark-BW(A) fuel with gadolinia, AREVA will use the enrichment cutback of the gadolinia-bearing fuel rods to demonstrate that the Advanced Mark-BW(A) gadolinia-bearing fuel rods have more CFM margin than the Advanced Mark-BW(A) UO_2 fuel rods. AREVA's linear heat rates will be based on a thermo-mechanical assessment of the AREVA Advanced Mark-BW(A) fuel rods using the NRC-approved best-estimate TACO3 and GDTACO codes (i.e., References 12 and 13, respectively).

These analyses ensure that the Overpower Differential Temperature (OPΔT) trip function provides the necessary protection for the AREVA Advanced Mark-BW(A) fuel assemblies.

Fuel Handling and Fuel Storage

EGC will review the design of the Advanced Mark-BW(A) fuel assemblies to ensure mechanical compatibility with the Braidwood fuel handling and fuel storage systems. The applicable thermo-hydraulic, criticality, and mechanical analyses for the Braidwood fuel handling and fuel storage systems will also be reviewed to ensure compatibility with the Advanced Mark-BW(A) fuel assemblies. This assessment of the Advanced Mark-BW(A) fuel to ensure mechanical

compatibility with the Braidwood fuel handling and fuel storage systems will be documented in the plant modification package for the Advanced Mark-BW(A) fuel in accordance with the EGC configuration control process.

The specific review of the Braidwood spent fuel pool Holtec storage racks thermo-hydraulic and criticality analyses' compatibility with the Advanced Mark-BW(A) fuel assemblies is complete, and found that the Advanced Mark-BW(A) fuel assemblies are compatible with the current analyses of record. Note that Holtec Report HI-982094, Revision 2, "Criticality Evaluation for the Byron/Braidwood Rack Installation Project for ComEd," (i.e., Reference 18) is the current Analysis of Record for Westinghouse OFA fuel in the Holtec storage racks. The Bases for T.S. 3.7.15 (Spent Fuel Pool Boron Concentration) and T.S. 3.7.16 (Spent Fuel Assembly Storage) are being revised to reflect the use of AREVA Advanced Mark-BW(A) fuel assemblies, since these Bases currently specifically reference the use of Westinghouse OFA fuel assemblies only. Attachment 4 provides the revised TS Bases pages with the proposed changes indicated.

Best Estimate Analyzer for Core Operations Nuclear (BEACON™) Core Monitoring System

Westinghouse will determine a set of limits, along with the associated operating conditions, for the AREVA Advanced Mark-BW(A) fuel assemblies. The limits will include a linear heat rate (i.e., kW/ft) limit to preclude fuel melt, a kW/ft limit for clad stress/strain, and a rod internal pressure (RIP) limit. AREVA will evaluate their fuel's performance against these criteria at the given operating conditions to show that their fuel is nonlimiting. Westinghouse will determine the limits such that, if AREVA determines that their fuel meets the criteria, the allowable operating space for the reload will not be adversely impacted by the presence of the Advanced Mark-BW(A) fuel assemblies.

As previously noted, the nuclear design of the Braidwood Station Unit 1 Cycles 14, 15, and 16 cores by Westinghouse will ensure sufficient margin between the lead Westinghouse fuel assembly and the AREVA Advanced Mark-BW(A) assemblies for $F_{\Delta H}$, F_Q, and for fuel assembly average power; these margins will be a minimum of 5%. Westinghouse will provide statepoint conditions to AREVA associated with the Operating Limits for the BEACON™ assessment. These statepoints will reflect the BEACON™ operating limits and will contain the respective operating conditions and axial power shapes. The statepoint conditions will be evaluated by each vendor at the Braidwood Station Unit 1 $F_{\Delta H}$ COLR limit, and EGC will compare the results to quantify the necessary power reduction on the AREVA Advanced Mark-BW(A) fuel assemblies. This will ensure that the AREVA Advanced Mark-BW(A) fuel assemblies have more DNB margin than the resident Westinghouse fuel for the statepoints provided. Westinghouse will then confirm that the allowable operating space for the reload will not be adversely impacted by the presence of the AREVA Advanced Mark-BW(A) fuel assemblies. To account for differences in vendor fuel types, conservatisms may be needed for core monitoring by BEACON™.

Emergency Core Cooling System Sump Screens

As part of the response to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," Braidwood Station is installing new ECCS sump screens. The effect of the new screens on the resident Westinghouse OFA fuel has been evaluated and found to be acceptable. The sump screen evaluation (i.e., Reference 22) was reviewed and found to be applicable, without

change, for the AREVA Advanced Mark-BW(A) assemblies. This assessment of the AREVA Advanced Mark-BW(A) fuel and its applicability to the existing evaluation (i.e., Reference 22) will be documented in the plant modification package for the AREVA Advanced Mark-BW(A) fuel in accordance with the EGC configuration control process.

Alternate Source Term

The AREVA Advanced Mark-BW(A) fuel design has been evaluated for impact on the Fuel Handling Accident (FHA) dose consequences contained in the recently approved License Amendments (i.e., Reference 17) for Braidwood Station that utilized the AST methodology. The AREVA Advanced Mark-BW(A) fuel assemblies do not result in a significant increase in dose consequences and radiological dose limits continue to be met. This evaluation of the AREVA Advanced Mark-BW(A) fuel and its applicability to the AST license amendment safety evaluation will be documented in the plant modification package for the AREVA Advanced Mark-BW(A) fuel in accordance with the EGC configuration control process.

Administrative Change to remove all references to Joseph Oat spent fuel storage racks

TS 3.7.15, "Spent Fuel Pool Boron Concentration," TS 3.7.16, "Spent Fuel Assembly Storage," and TS 4.3.1, "Criticality," currently contain references to Joseph Oat spent fuel pool storage racks that have been physically removed from the spent fuel pool. Braidwood Station, Units 1 and 2, were issued Amendment 105 that approved the installation of new Boral high-density spent fuel storage racks (i.e., Holtec storage racks) on March 1, 2000 (i.e., Reference 6). This amendment was in response to Commonwealth Edison Company's application dated March 23, 1999 (i.e., Reference 7), as supplemented on October 21 and December 15, 1999 (i.e., References 8 and 9) respectively. This amendment supported the removal of all 23 of the then existing spent fuel storage racks (i.e., Joseph Oat spent fuel storage racks) and the replacement with 24 new (i.e., Holtec International) spent fuel storage racks.

During the installation of the new Holtec spent fuel storage racks, both Holtec and the then existing Joseph Oat spent fuel storage racks were in the spent fuel pool at the same time. The approved changes to TS 3.7.15, TS 3.7.16 and TS 4.3.1 (i.e., Reference 6) addressed the requirements for both the new Holtec storage racks, during and after installation, and the then existing Joseph Oat storage racks, during the Holtec rack installation.

The replacement of Joseph Oat spent fuel storage racks with Holtec spent fuel storage racks at Braidwood Station was completed in December 2001. Braidwood Station currently has only Holtec spent fuel storage racks in the spent fuel pool and therefore all references to Joseph Oat spent fuel storage racks are no longer needed since they are physically removed from the spent fuel pool.

The proposed revisions to TS 3.7.15, TS 3.7.16 and TS 4.3.1 to remove all references to Joseph Oat spent fuel pool storage racks is editorial only and will delete wording that is no longer required. The revisions will result in various editorial and formatting change requirements and as a result of the proposed revision to TS 3.7.16, Surveillance Requirement (SR) 3.7.16.3, and existing Figures 3.7.16-1, 3.7.16-2, and 3.7.16-3 will be deleted in their entirety.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Overview

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an amendment to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, respectively. The proposed amendment would revise Technical Specification (TS) 4.2.1, "Fuel Assemblies," to allow up to eight AREVA NP Inc. (AREVA), formerly Framatome, modified Advanced Mark-BW fuel assemblies (i.e., Advanced Mark-BW(A)fuel assemblies) containing M5 alloy to be placed in nonlimiting (i.e., for $F_{\Delta H}$, F_Q, and fuel assembly average power at hot full power normal operating conditions) Braidwood Station Unit 1 core regions (i.e., locations) for evaluation during Cycles 14, 15, and 16, and Safety Limit (SL) 2.1.1, "Reactor Core SLs," to incorporate the peak fuel centerline temperature equations associated with the AREVA NP fuel in SL 2.1.1.3. The proposed amendment also revises the existing Amendment 122 Additional Condition in the Operating License, Appendix C, "Additional Conditions," to address operation during Cycles 14, 15, and 16 with up to eight AREVA NP Advanced Mark-BW(A) fuel assemblies containing fuel pellets incorporating homogeneous poisons. The license for Braidwood Station Unit 2 is affected only due to the fact that Unit 1 and Unit 2 use common TS.

The proposed amendment will permit Braidwood Station Unit 1 to load up to eight AREVA NP Advanced Mark-BW(A) fuel assemblies in the reactor core for operation in Cycles 14, 15, and 16 for evaluation. The purpose of this evaluation program is to allow EGC to gain an understanding of the behavior of mixed fuel cores prior to a possible transition from Westinghouse fuel to AREVA fuel. The AREVA Advanced Mark-BW(A) fuel assemblies use an advanced zirconium-based M5 alloy for the fuel assembly structural tubing, fuel rod cladding, and grids. In addition, the Advanced Mark-BW fuel pellets may contain homogeneous poisons (i.e., gadolinia).

In addition to the above changes, EGC is requesting changes to TS 3.7.15, "Spent Fuel Pool Boron Concentration," TS 3.7.16, "Spent Fuel Assembly Storage," and TS 4.3.1, "Criticality." These changes are administrative and remove all references to Joseph Oat spent fuel pool storage racks that have been physically removed from the spent fuel pool. Braidwood Station, Units 1 and 2, were issued Amendment 105 that approved the installation of new Boral high-density spent fuel storage racks (i.e., Holtec storage racks) on March 1, 2000 (i.e., Reference 6). The replacement of Joseph Oat spent fuel storage racks with Holtec spent fuel storage racks at Braidwood Station was completed in December 2001.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

1. **The proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The AREVA Advanced Mark-BW(A) fuel is similar in design to the Westinghouse fuel that will be co-resident in the core. The Advanced Mark-BW(A) fuel assemblies are also similar in design to the Advanced Mark-BW assemblies using M5 alloy material for the cladding, structural tubing, and grids generically approved for use in Westinghouse 3- and 4-loop designed pressurized water reactors with 17 x 17 fuel rod arrays. The AREVA Advanced Mark-BW(A) fuel assemblies will be placed in nonlimiting regions (i.e., locations) of the core. The Cycle 14, 15, and 16 reload designs will meet all applicable design criteria. EGC will use the NRC-approved standard reload design models and methods to demonstrate that all applicable design criteria will be met. Evaluations will be performed as part of the cycle specific reload safety analysis for the operation of the AREVA Advanced Mark-BW(A) fuel to confirm that the acceptance criteria of the existing safety analyses continue to be met Operation of the AREVA Advanced Mark-BW(A) fuel will not significantly increase the predicted radiological consequences of accidents postulated in the Updated Final Safety Analysis Report.

The proposed change regarding removal of all references in TS to the Joseph Oat spent fuel racks is administrative and deletes unnecessary wording relating to equipment that is physically removed from the Braidwood Station spent fuel pool and therefore does not alter the design, configuration, operation, or function of any plant system, structure or component. As a result, the administrative change does not affect the outcome of any previously evaluated accidents.

Based on the above discussion, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The AREVA Advanced Mark-BW(A) fuel is similar in design to the Westinghouse fuel that will be co-resident in the core. The Advanced Mark-BW(A) fuel assemblies are also similar in design to the Advanced Mark-BW assemblies using M5 alloy material for the cladding, structural tubing, and grids generically approved for use in Westinghouse 3- and 4-loop designed pressurized water reactors with 17 x 17 fuel rod arrays. The Braidwood Station Unit 1 cores in which the fuel operates will be designed to meet all applicable design criteria and ensure that all pertinent licensing basis criteria are met. Demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. The reload core designs for the cycles in which the AREVA Advanced Mark-BW(A) fuel will operate will demonstrate that the use of up to eight AREVA Advanced Mark-BW(A) fuel assemblies in nonlimiting core regions (i.e., locations) is acceptable. The relevant design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of AREVA Advanced Mark-BW(A) fuel does not involve any alteration to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors.

The proposed change regarding removal of all references in TS to the Joseph Oat spent fuel racks is administrative and deletes unnecessary wording relating to equipment that is physically removed from the Braidwood Station spent fuel pool and therefore does not alter the design, configuration, operation, or function of any plant system, structure or component. As a result, the administrative change does not create any new or different kind of accident.

Based on this evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed TS change does not involve a significant reduction in a margin of safety.

Operation of Braidwood Station Unit 1 with up to eight AREVA Advanced Mark-BW(A) fuel assemblies in nonlimiting core regions (i.e., locations) does not change the performance requirements on any system or component such that any design criteria will be exceeded. The normal limits on core operation defined in the Braidwood Station TS will remain applicable for the use of up to eight AREVA Advanced Mark-BW(A) fuel assemblies during Cycles 14, 15, and 16. The reload core designs for the cycles in which the AREVA Advanced Mark-BW(A) fuel will operate will specifically evaluate any pertinent differences, including both mechanical design differences and the past irradiation history, between the AREVA Advanced Mark-BW(A) fuel product, and the Westinghouse fuel product that will be co-resident in the core. The use of up to eight AREVA Advanced Mark-BW(A) fuel assemblies will be specifically evaluated during the reload design process using reload design models and methods as approved by the NRC.

The proposed change regarding removal of all references in TS to the Joseph Oat spent fuel racks is administrative and deletes unnecessary wording relating to equipment that is physically removed from the Braidwood Station spent fuel pool and therefore does not alter the design, configuration, operation, or function of any plant system, structure or component. As a result, the administrative change does not affect the ability of any operable structure, system, or component to perform its designated safety function.

Based on this evaluation, the proposed change does not involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant hazards consideration under the criteria set forth in 10 CFR 50.92(c).

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires nuclear power reactors fueled with uranium oxide pellets within cylindrical Zircaloy or ZIRLO™ cladding to be provided with an emergency core cooling system with certain performance requirements. Although the AREVA Advanced Mark-BW(A) fuel assemblies incorporate cladding material other than those defined in 10 CFR 50.46 (i.e., Zircaloy and ZIRLO™), the criteria of this section will continue to be satisfied for the Braidwood Station Unit 1 core. Since 10 CFR 50.46 does not specifically address M5 alloy, an exemption to 10 CFR 50.46 has been requested.

10 CFR Part 50, Appendix K, "ECCS Evaluation Models," ensures that cladding oxidation and hydrogen generation are appropriately limited during a LOCA and conservatively accounted for in the ECCS model. This regulation sets forth requirements for plants that use either Zircaloy or

ZIRLO™ fuel cladding. Specifically, Paragraph I.A.5 of 10 CFR Part 50, Appendix K, requires that the Baker-Just equation be used in the ECCS evaluation model to determine the rate of energy release, hydrogen generation, and cladding oxidation. When M5 alloy is used as fuel rod cladding and structural material, the Baker-Just correlation bounds post-LOCA scenarios, and ECCS evaluation model criteria will be met. Because the Baker-Just equation does not explicitly address M5 alloy, an exemption to 10 CFR Part 50, Appendix K has been requested.

6.0 ENVIRONMENTAL CONSIDERATION

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

7.0 REFERENCES

1. Framatome ANP BAW-10239(P)-A, "Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report," dated July 2004

2. Letter from M. Chawla (NRC) to J. L. Skolds (Exelon Generation Company, LLC), "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 – Issuance of Amendments," dated April 19, 2002

3. Letter from S. A. Richards (NRC) to T. A. Coleman (Framatome Cogema Fuels), "Revised Safety Evaluation (SE) for Topical Report BAW-10227P: 'Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel,'" dated February 4, 2000

4. Letter from S. Monarque (NRC) to D. A. Christian (Virginia Electric and Power Company), "North Anna Power Station, Units 1 and 2, Issuance of Exemption from the Requirements of 10 CFR 50.44, 10 CFR 50.46, and 10 CFR Part 50, Appendix K, to Allow the use of the M5 Alloy for Fuel Cladding Material," dated September 23, 2003

5. Letter from H. N. Berkow (NRC) to J. F. Mallay (Framatome ANP), " Safety Evaluation of Framatome ANP Topical Report BAW-10186P-A, Revision 1, Supplement 1, 'Extended Burnup Evaluation,'" dated June 18, 2003

6. Letter from G. F. Dick (NRC) to O.D. Kingsley (Commonwealth Edison Company), "Byron and Braidwood – Issuance of Amendments on Spent Fuel Storage Racks," dated March 1, 2000

7. Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for an Amendment to Technical Specifications to Support Installation of New Spent Fuel Pool Storage Racks at Byron and Braidwood Stations," dated March 23, 1999

8. Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Response to Request for Additional and Clarifying Information Regarding Holtec International Report, HI-982083, 'Licensing Report for Spent Fuel Rack Installation at Byron and Braidwood Nuclear Stations,'" dated October 21, 1999

9. Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Editorial Correction to Technical Specification Amendment Request to Support Installation of New Spent Fuel Pool Storage Racks at Byron and Braidwood Stations," dated December 15, 1999

10. Letter from H. N. Berkow (NRC) to J. F. Mallay (Framatome ANP), "Final Safety Evaluation for Framatome ANP Topical Report BAW-10239(P), Revision 0, 'Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report'," dated July 1, 2004

11. Framatome ANP BAW-10227P-A, Revision 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," dated June 2003

12. Framatome ANP BAW-10162P-A, "TACO3 – Fuel Pin Thermal Analysis Computer Code," dated October 1989

13. Framatome ANP BAW-10184P-A, "GDTACO – Urania Gadolinia Fuel Pin Thermal Analysis Code," February 1995

14. Framatome ANP BAW-10183P-A, "Fuel Rod Gas Pressure Criterion (FRGPC)," dated July 1995

15. Framatome ANP BAW-10244P-A, "Mark-BW CHF Correlations Applied with XCOBRA-IIIC," dated October 2004

16. Topical Report XN-NF-75-21(P)(A), Revision 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Operation," dated January 1986

17. Letter from R. F. Kuntz (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2 – Issuance of Amendments Re: Alternative Source Term," dated September 8, 2006

18. Holtec Report HI-982094 Rev. 2, "Criticality Evaluation for the Byron/Braidwood Rack Installation Project for ComEd," dated July 12, 2000

19. AREVA NP Letter 12-9002633-01, "FANP Analysis for Braidwood LUA Project," dated November 9, 2005

20. AREVA NP Letter 51-5070238-002, "Braidwood LTA Design Parameters," dated May 9, 2006

21. Westinghouse Letter NF-CB-05-159 Revision 2, "Revision 2 ** Strategy for AREVA Lead Use Assembly Program," dated November 15, 2005

22. BRW-05-0084-M/BYR06-017, "Byron Units 1 and 2 and Braidwood Units 1 and 2 GSI-191 Downstream Effects – Vessel Blockage and Fuel Evaluation," Revision 0, dated January 31, 2006

23. Framatome ANP BAW-10243(P)(A), "Statistical Fuel Assembly Hold Down Methodology," dated September 2005

ATTACHMENT 2

BRAIDWOOD STATION
UNITS 1 and 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

License Amendment Request and Exemption Requests to Allow Use of
AREVA NP Inc. Advanced Mark-BW(A) Fuel Lead Assemblies

Markup of Operating License and Technical Specifications Page Changes

Operating License Additional Conditions
Page 2

Technical Specifications Pages
2.0-1
3.7.15-1
3.7.15-2
3.7.16-1 to 3.7.16-7
4.0-1 to 4.0-3

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-72

The licensee shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
113	The licensee shall implement modifications as discussed in Section 5.11.9 of the Safety Evaluation to maintain the stability of the Braidwood transmission grid including a reduction in the existing local breaker backup time settings.	Prior to implementation of full power up-rate conditions
113	The licensee shall submit to the NRC a confirmatory analysis using a model acceptable to the NRC justifying the value of 8.5 hours for the time of switchover to hot leg injection following a loss-of-coolant accident (Safety Evaluation Section 3.1.3); or recalculate the switchover time using the currently accepted methodology.	Submit by June 1, 2002
113	The licensee shall make the instrumentation changes as described in Section 4.15.2 of the Safety Evaluation.	Prior to implementation of full power up-rate conditions
122	The safety limit equation specified in TS 2.1.1.3 regarding fuel centerline melt temperature (i.e., less than 5080 °F, decreasing by 58 °F per 10,000 MWD/MTU burnup as described in WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995) is valid for uranium oxide fuel without the presence of poisons mixed homogeneously into the fuel pellets. If fuel pellets incorporating homogeneous poisons are used, the topical report documenting the fuel centerline melt temperature basis must be reviewed and approved by the NRC and referenced in this license condition. TS 2.1.1.3 must be modified to also include the fuel centerline melt temperature limit for the fuel with homogeneous poison.	With implementation of the amendment

During operation in Cycles 14, 15, and 16, up to eight (8) AREVA NP Advanced Mark-BW(A) fuel assemblies containing fuel pellets incorporating homogeneous poisons may be placed in nonlimiting Unit 1 core regions (i.e., locations). The design basis for the AREVA NP fuel rod centerline melt follows that given in BAW-10162P-A, "TACO3 – Fuel Pin Thermal Analysis Computer Code," October 1989, and BAW-10184P-A, "GDTACO – Urania Gadolinia Fuel Pin Thermal Analysis Code," February 1995.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell and ≥ 1.25 for the WRB-2 DNB correlation for a typical cell.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ~~< 5080°F, decreasing by 58°F per 10,000 MWD/MTU burnup.~~

as follows:

a. < 5080°F decreasing by 58°F per 10,000 MWD/MTU burnup for Westinghouse fuel,

b. < 5173°F decreasing by 65°F per 10,000 MWD/MTU burnup for AREVA NP fuel (Unit 1 only), and

c. < 5189°F decreasing by 65°F per 10,000 MWD/MTU burnup for AREVA NP fuel containing Gadolinia (Unit 1 only).

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Boron Concentration

LCO 3.7.15 The spent fuel pool boron concentration shall be, ~~as applicable:~~

~~a. ≥ 300 ppm for Holtec spent fuel pool storage racks; and~~

~~b. ≥ 2000 ppm for Joseph Oat spent fuel pool storage racks.~~

APPLICABILITY: Whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

----------NOTE----------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	A.1	Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND		
	A.2	Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel pool boron concentration is within limit.	7 days

(NO TEXT CHANGES TO THIS PAGE - INCLUDED FOR PAGINATION ONLY)

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 Each spent fuel assembly stored in the spent fuel pool shall, as applicable:

a. Region 1 of Joseph Oat spent fuel pool storage racks

Have an initial nominal enrichment of ≤ 4.7 weight percent U-235 or satisfy a minimum number of Integral Fuel Burnable Absorbers (IFBAs) for higher initial enrichments up to 5.0 weight percent U-235 to permit storage in any cell location.

b. Region 2 of Joseph Oat spent fuel pool storage racks

Have a combination of initial enrichment, burnup, and decay time within the Acceptable Burnup Domain of Figure 3.7.16-1, 3.7.16-2, or 3.7.16-3, as applicable for that storage configuration.

c. Interface Requirements for Joseph Oat spent fuel pool storage racks

Comply with the Interface Requirements within and between adjacent racks.

a. ~~d.~~ Region 1 of Holtec spent fuel pool storage racks

Have an initial nominal enrichment of ≤ 5.0 weight percent U-235 to permit storage in any cell location.

b. ~~e.~~ Region 2 of Holtec spent fuel pool storage racks

Have a combination of initial enrichment and burnup within the Acceptable Burnup Domain of Figure 3.7.16-~~4~~ 1.

APPLICABILITY: Whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

----------NOTE----------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly into a location which restores compliance.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.16.1 ----------NOTE---------- Item a is only applicable for storage of fuel assemblies in Region 1 Holtec spent fuel pool storage racks. Item b is only applicable for storage of fuel assemblies in Region 1 Joseph Oat spent fuel pool storage racks. ---------- Verify by administrative means the ~~following requirements are met:~~ a. ~~Initial~~ initial nominal enrichment of the fuel assembly is ≤ 5.0 weight percent U-235. ~~AND~~	Prior to storing the fuel assembly in Region 1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.7.16.1 (continued) b. Initial nominal enrichment of the fuel assembly is ≤ 4.7 weight percent U-235 with less than the minimum number of IFBAs or ≤ 5.0 weight percent U-235 with the minimum number of IFBAs.	
SR 3.7.16.2 ----------NOTE---------- Figures 3.7.16-1, 3.7.16-2, and 3.7.16-3 are only applicable for storage of fuel assemblies in Region 2 Joseph Oat spent fuel pool storage racks. Figure 3.7.16-4 is only applicable for storage of fuel assemblies in Region 2 Holtec spent fuel pool storage racks. ---------- Verify by administrative means the combination of initial enrichment, burnup, and decay time, as applicable, of the fuel assembly is within the Acceptable Burnup Domain of Figure 3.7.16-1, ~~3.7.16-2, 3.7.16-3, or 3.7.16-4.~~	Prior to storing the fuel assembly in Region 2
SR 3.7.16.3 ----------NOTE---------- Only applicable for storage of fuel assemblies in Joseph Oat spent fuel pool storage racks. ---------- Verify by administrative means the interface requirements within and between adjacent racks are met.	Prior to storing the fuel assembly in the spent fuel pool

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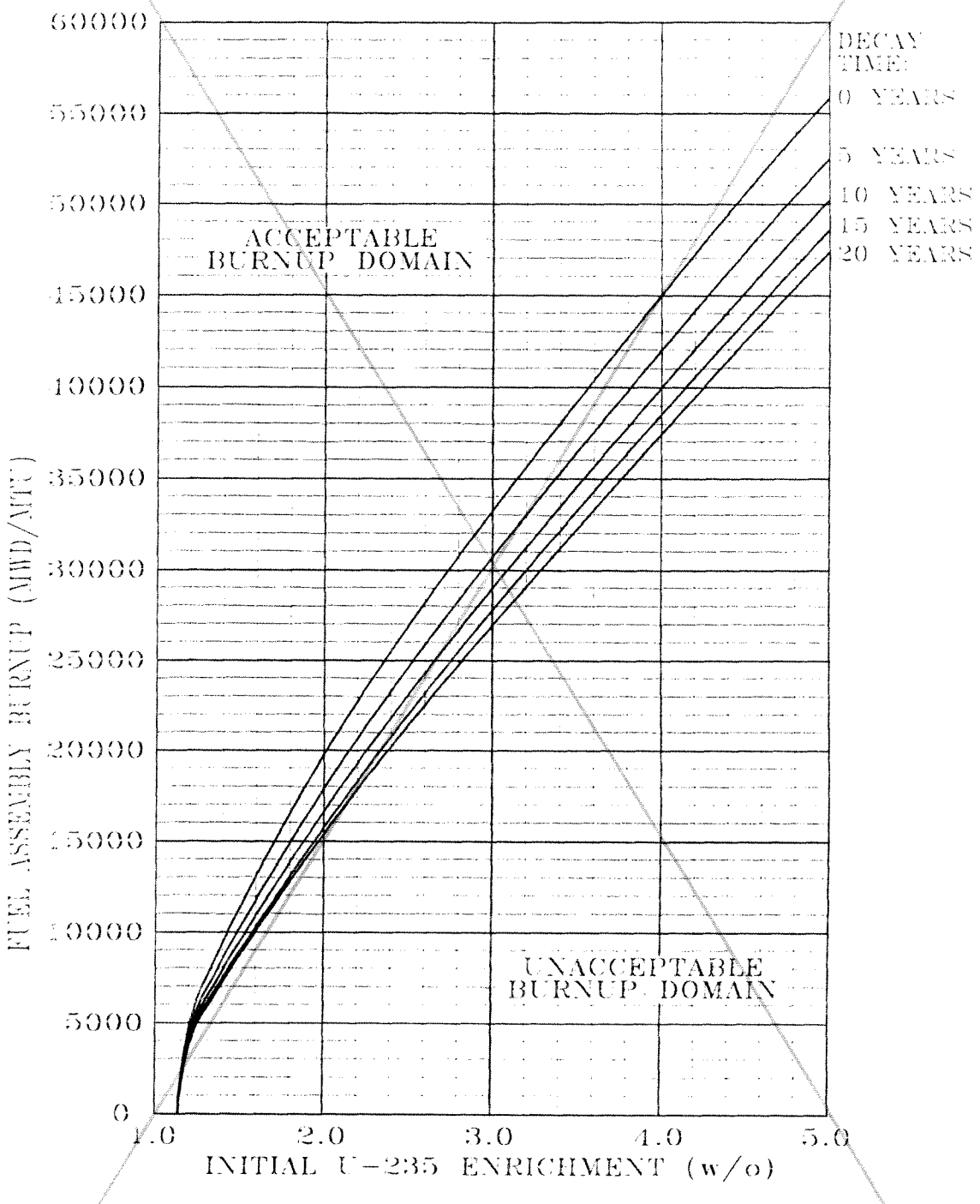

Figure 3.7.16-1 (page 1 of 1)
Region 2 All Cell Configuration Burnup Credit Requirements
(Joseph Oat Spent Fuel Pool Storage Racks)

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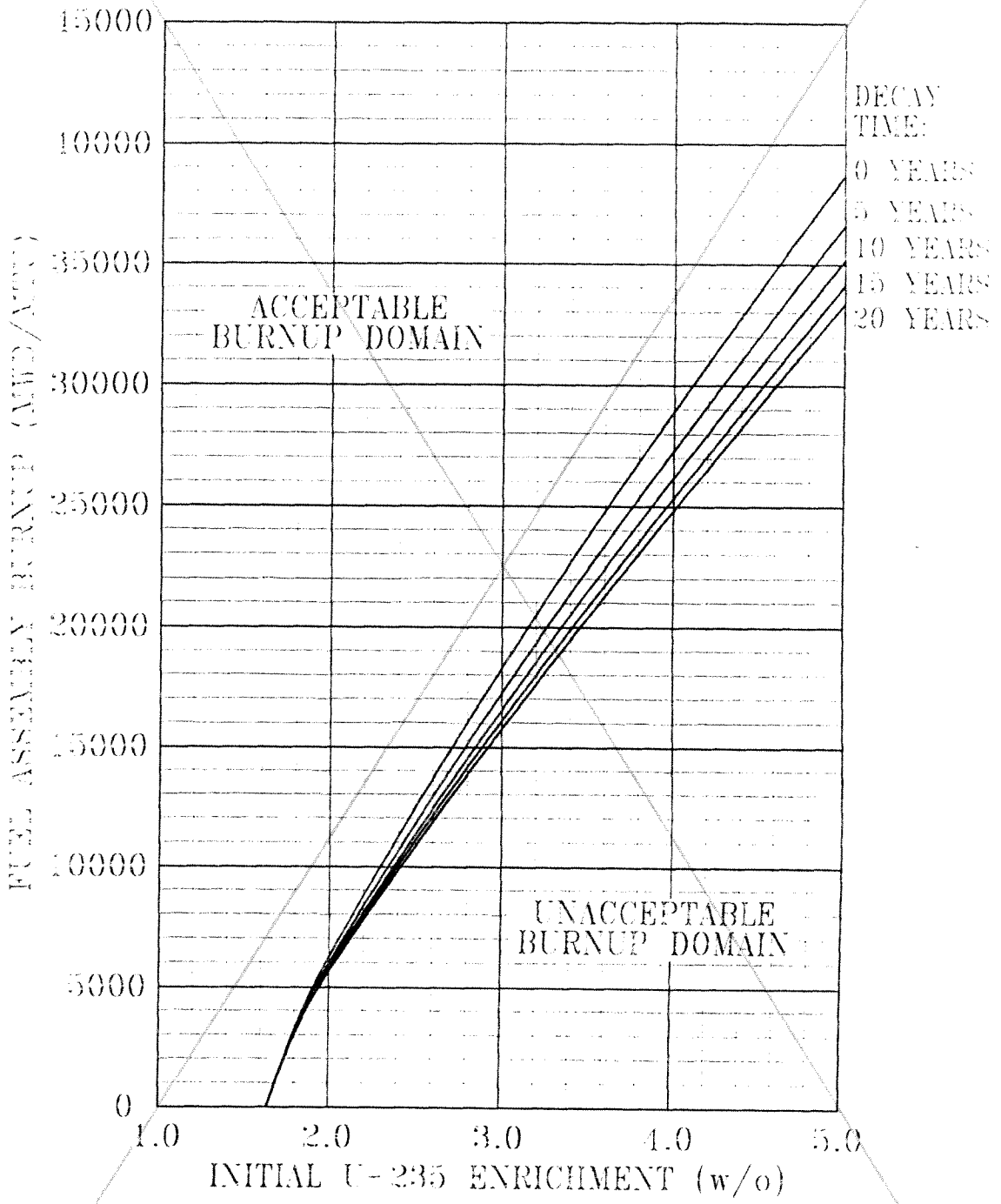

Figure 3.7.16-2 (page 1 of 1)
Region 2 3-out-of-4 Checkerboard Configuration Burnup Credit Requirements
(Joseph Oat Spent Fuel Pool Storage Racks)

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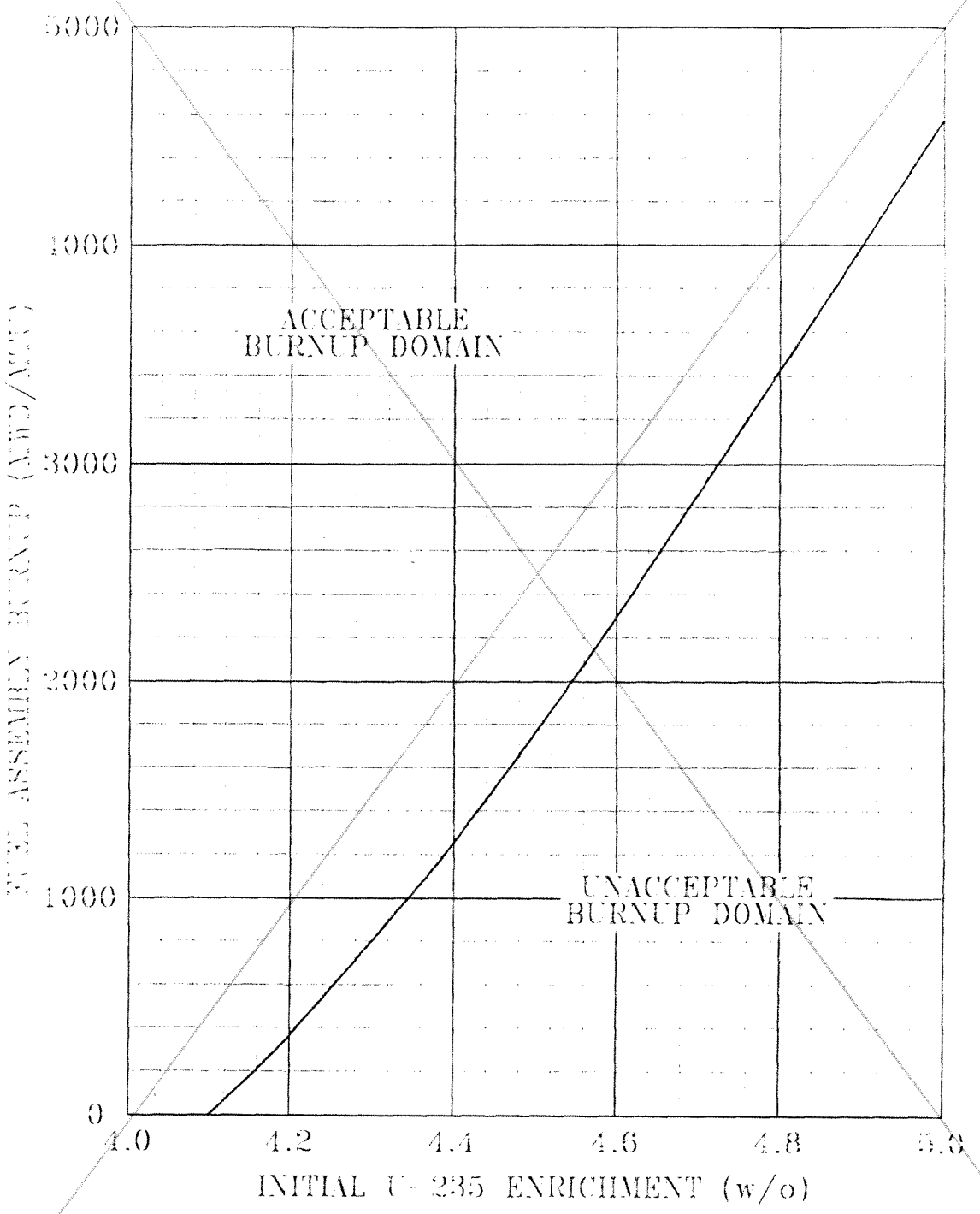
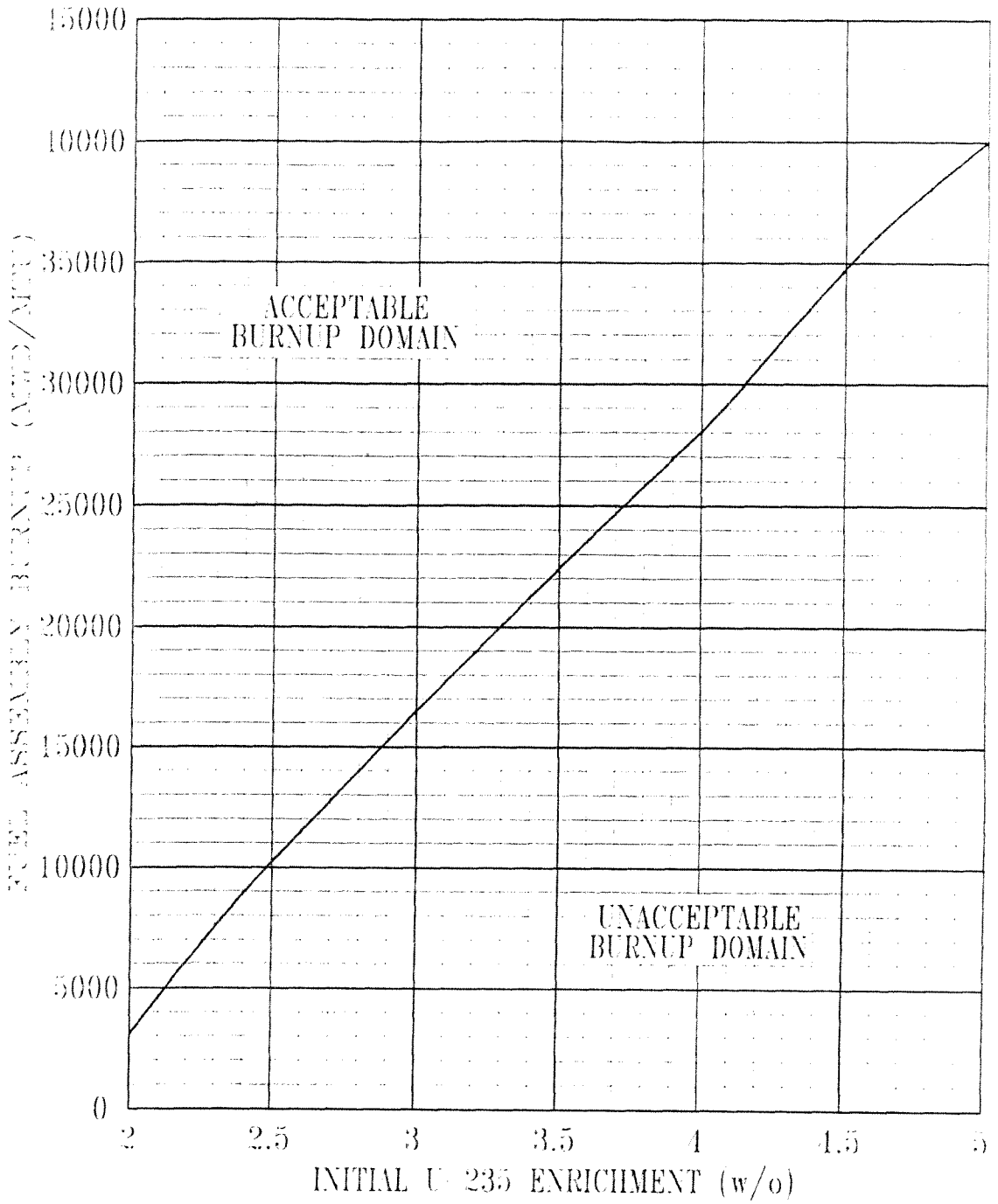


Figure 3.7.16-3 (page 1 of 1)
Region 2 2-out-of-4 Checkerboard Configuration Burnup Credit Requirements
(Joseph Oat Spent Fuel Pool Storage Racks)



3.7.16-1

Figure ~~3.7.16-4~~ (page 1 of 1)
Region 2 Fuel Assembly Burnup Requirements
(Holtec Spent Fuel Pool Storage Racks)

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site Location

The site is located in Reed Township, approximately 20 mi (32 km) south-southwest of the city of Joliet in northern Illinois.

4.1.2 Exclusion Area Boundary (EAB)

The EAB shall not be less than 1591 ft (485 meters) from the outer containment wall.

4.1.3 Low Population Zone (LPZ)

The LPZ shall be a 1.125 mi (1811 meter) radius measured from the midpoint between the two reactors.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods or vacancies for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver indium cadmium, hafnium, or a mixture of both types.

DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

The spent fuel storage racks are designed and shall be maintained, as applicable, with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. For Joseph Oat spent fuel pool storage racks, $k_{eff} < 1.0$ if fully flooded with unborated water which includes an allowance for uncertainties as described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology";

c. For Joseph Oat spent fuel pool storage racks, $k_{eff} \leq 0.95$ if fully flooded with water borated to 550 ppm, which includes an allowance for uncertainties as described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology";

d. For Joseph Oat spent fuel pool storage racks, a nominal 10.32 inch north-south and 10.42 inch east-west center to center distance between fuel assemblies placed in Region 1 racks; and

e. For Joseph Oat spent fuel pool storage racks, a nominal 9.03 inch center to center distance between fuel assemblies placed in Region 2 racks.

b. ~~f.~~ For Holtec spent fuel pool storage racks, $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Holtec International Report HI-982094, "Criticality Analysis for Byron/Braidwood Rack Installation Project," Project No. 80944, 1998;

c. ~~g.~~ For Holtec spent fuel pool storage racks, a nominal 10.888 inch north-south and 10.574 inch east-west center to center distance between fuel assemblies placed in Region 1 racks; and

d. ~~h.~~ For Holtec spent fuel pool storage racks, a nominal 8.97 inch center to center distance between fuel assemblies placed in Region 2 racks.

DESIGN FEATURES (continued)

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 410 ft, 0 inches. |

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 2984 fuel assemblies. |

(NO TEXT CHANGES TO THIS PAGE - INCLUDED FOR PAGINATION ONLY)

ATTACHMENT 3

BRAIDWOOD STATION
UNITS 1 and 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

License Amendment Request and Exemption Requests to Allow Use of AREVA NP Inc. Advanced Mark-BW(A) Fuel Lead Assemblies

Revised Operating License and Technical Specifications Pages

Operating License Additional Conditions
Page 2

Technical Specifications Pages
2.0-1
3.7.15-1
3.7.16-1 to 3.7.16-3
4.0-1 to 4.0-2

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-72

The licensee shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
113	The licensee shall implement modifications as discussed in Section 5.11.9 of the Safety Evaluation to maintain the stability of the Braidwood transmission grid including a reduction in the existing local breaker backup time settings.	Prior to implementation of full power uprate conditions
113	The licensee shall submit to the NRC a confirmatory analysis using a model acceptable to the NRC justifying the value of 8.5 hours for the time of switchover to hot leg injection following a loss-of-coolant accident (Safety Evaluation Section 3.1.3); or recalculate the switchover time using the currently accepted methodology.	Submit by June 1, 2002
113	The licensee shall make the instrumentation changes as described in Section 4.15.2 of the Safety Evaluation.	Prior to implementation of full power uprate conditions
	The safety limit equation specified in TS 2.1.1.3 regarding fuel centerline melt temperature (i.e., less than 5080 °F, decreasing by 58 °F per 10,000 MWD/MTU burnup as described in WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995) is valid for uranium oxide fuel without the presence of poisons mixed homogeneously into the fuel pellets. If fuel pellets incorporating homogeneous poisons are used, the topical report documenting the fuel centerline melt temperature basis must be reviewed and approved by the NRC and referenced in this license condition. TS 2.1.1.3 must be modified to also include the fuel centerline melt temperature limit for the fuel with homogeneous poison. During operation in Cycles 14, 15, and 16, up to eight (8) AREVA NP Advanced Mark-BW(A) fuel assemblies containing fuel pellets incorporating homogeneous poisons may be placed in nonlimiting Unit 1 core regions (i.e., locations). The design basis for the AREVA NP fuel rod centerline melt follows that given in BAW-10162P-A, "TACO3 – Fuel Pin Thermal Analysis Computer Code," October 1989, and BAW-10184P-A, "GDTACO – Urania Gadolinia Fuel Pin Thermal Analysis Code," February 1995.	With implementation of the amendment

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell and ≥ 1.25 for the WRB-2 DNB correlation for a typical cell.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained as follows:

a. < 5080°F decreasing by 58°F per 10,000 MWD/MTU burnup for Westinghouse fuel,

b. < 5173°F decreasing by 65°F per 10,000 MWD/MTU burnup for AREVA NP fuel (Unit 1 only), and

c. < 5189°F decreasing by 65°F per 10,000 MWD/MTU burnup for AREVA NP fuel containing Gadolinia (Unit 1 only).

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Boron Concentration

LCO 3.7.15 The spent fuel pool boron concentration shall be ≥ 300 ppm.

APPLICABILITY: Whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

----------NOTE----------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 Each spent fuel assembly stored in the spent fuel pool shall, as applicable:

a. Region 1 of Holtec spent fuel pool storage racks

Have an initial nominal enrichment of ≤ 5.0 weight percent U-235 to permit storage in any cell location.

b. Region 2 of Holtec spent fuel pool storage racks

Have a combination of initial enrichment and burnup within the Acceptable Burnup Domain of Figure 3.7.16-1.

APPLICABILITY: Whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

----------NOTE----------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly into a location which restores compliance.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means the initial nominal enrichment of the fuel assembly is ≤ 5.0 weight percent U-235.	Prior to storing the fuel assembly in Region 1
SR 3.7.16.2	Verify by administrative means the combination of initial enrichment, burnup, and decay time, as applicable, of the fuel assembly is within the Acceptable Burnup Domain of Figure 3.7.16-1.	Prior to storing the fuel assembly in Region 2

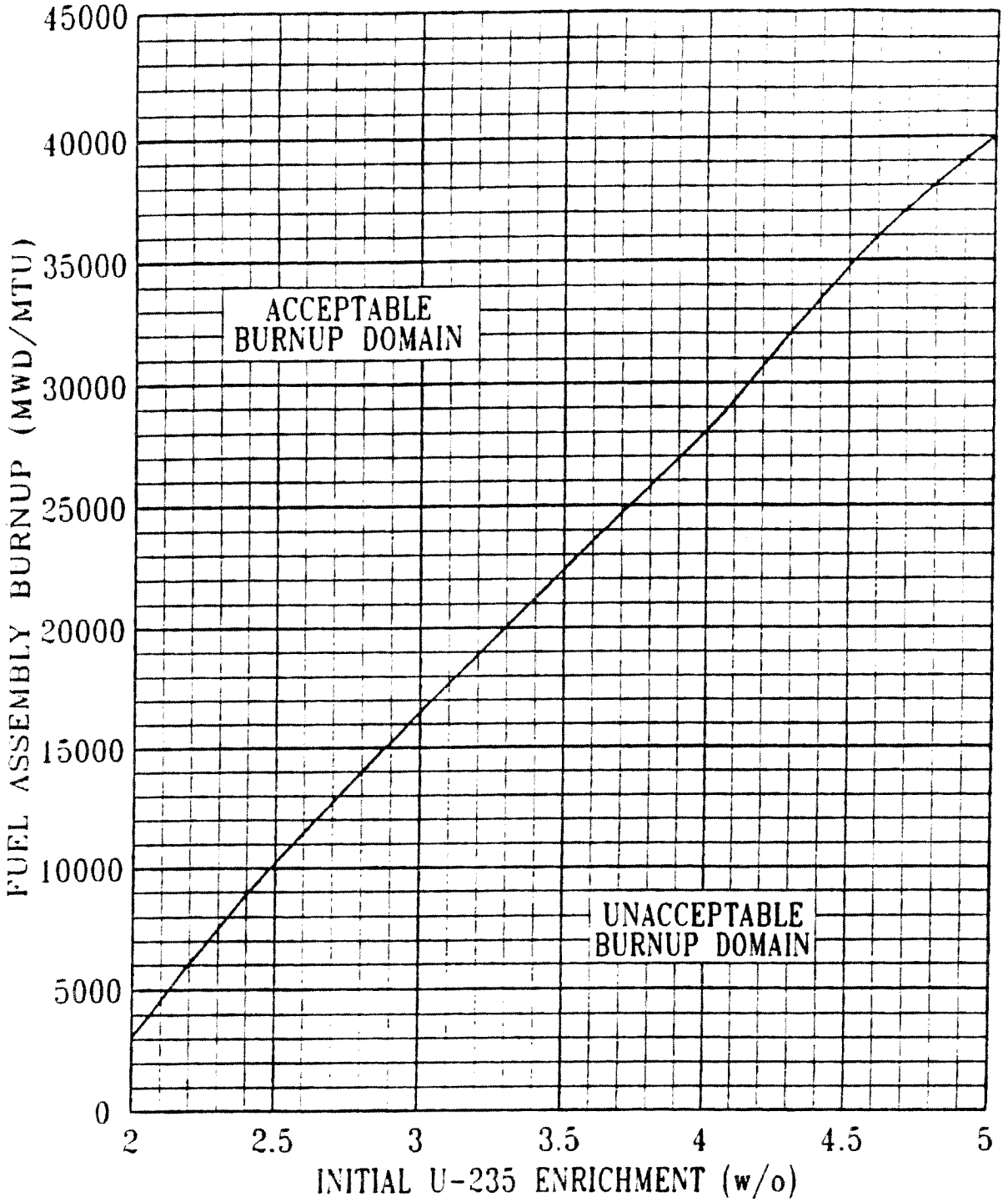


Figure 3.7.16-1 (page 1 of 1)
Region 2 Fuel Assembly Burnup Requirements
(Holtec Spent Fuel Pool Storage Racks)

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site Location

The site is located in Reed Township, approximately 20 mi (32 km) south-southwest of the city of Joliet in northern Illinois.

4.1.2 Exclusion Area Boundary (EAB)

The EAB shall not be less than 1591 ft (485 meters) from the outer containment wall.

4.1.3 Low Population Zone (LPZ)

The LPZ shall be a 1.125 mi (1811 meter) radius measured from the midpoint between the two reactors.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly, with exceptions as noted below, shall consist of a matrix of Zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods or vacancies for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

Up to 8 AREVA NP Advanced Mark-BW(A) fuel assemblies containing M5 alloy may be placed in nonlimiting Unit 1 core regions for evaluation during Cycles 14, 15, and 16.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver indium cadmium, hafnium, or a mixture of both types.

DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

The spent fuel storage racks are designed and shall be maintained, as applicable, with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. For Holtec spent fuel pool storage racks, $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Holtec International Report HI-982094, "Criticality Analysis for Byron/Braidwood Rack Installation Project," Project No. 80944, 1998;

c. For Holtec spent fuel pool storage racks, a nominal 10.888 inch north-south and 10.574 inch east-west center to center distance between fuel assemblies placed in Region 1 racks; and

d. For Holtec spent fuel pool storage racks, a nominal 8.97 inch center to center distance between fuel assemblies placed in Region 2 racks.

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 410 ft, 0 inches.

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 2984 fuel assemblies.

ATTACHMENT 4

BRAIDWOOD STATION
UNITS 1 and 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

License Amendment Request and Exemption Requests to Allow Use of
AREVA NP Inc. Advanced Mark-BW(A) Fuel Lead Assemblies

Revised Technical Specification Bases Pages

Technical Specifications Bases Pages

B 3.7.15-1 to B 3.7.15-8

B 3.7.16-1 to B 3.7.16-8

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

The spent fuel pool provides for storage of various Westinghouse Optimized Fuel Assembly (OFA) types of different initial fuel enrichments and exposure histories in two distinct regions. (For this discussion, the term OFA is intended to refer to the specific reduced fuel rodlet diameter, and includes all analyzed fuel types with this diameter, such as Vantage 5.) The spent fuel pool is provided with 24 Holtec spent fuel pool storage racks, which provide placement locations for a total of 2984 new or used fuel assemblies. Of the 24 Holtec spent fuel pool storage racks, four are designated "Region 1" with the remaining 20 racks designated as "Region 2." The analytical methodology used for the criticality analyses is in accordance with established NRC guidelines (Ref. 2).

The storage of AREVA Advanced Mk-BW(A) fuel assemblies in the spent fuel pool storage racks was analyzed in Reference 6 and found to conform to the design basis described herein for Westinghouse OFA fuel. The requirements of LCO 3.7.15 and LCO 3.7.16 are applicable to the storage of both Westinghouse OFA and AREVA Advanced Mk-BW(A) fuel assemblies.

BASES

BACKGROUND (continued)

Region 1 racks contain 396 cells which are analyzed for storing Westinghouse OFAs in an "All Cells" arrangement (that is, the criticality analysis assumes that spent fuel assemblies reside in all available cell locations). The stored fuel assemblies may contain an initial nominal enrichment of ≤ 5.0 weight percent U-235 (with or without IFBAs installed) (Ref. 4).

Region 2 racks contain 2588 cells which are also analyzed for storing Westinghouse OFAs in an "All Cells" arrangement (that is, the criticality analysis assumes that spent fuel assemblies reside in all available cell locations). For the "All Cells" storage configuration, the stored fuel assemblies may contain an initial nominal enrichment of ≤ 5.0 weight percent U-235 with credit for burnup.

The water in the spent fuel pool normally contains soluble boron which results in large subcriticality margins under actual operating conditions.

APPLICABLE SAFETY ANALYSES

Methodologies in accordance with established NRC guidelines were used to develop the criticality analyses (Ref. 1) for the Holtec spent fuel pool storage racks. The fuel handling accident analyses are described in Reference 3.

The criticality analyses for the spent fuel assembly storage racks confirm that k_{eff} remains ≤ 0.95 for the Holtec spent fuel pool storage racks (including uncertainties and tolerances) at a 95% probability with a 95% confidence level (95/95 basis), based on the accident condition of the pool being flooded with unborated water. Thus, the design of both regions assumes the use of unborated water while maintaining stored fuel in a subcritical condition.

BASES

APPLICABLE SAFETY ANALYSES (continued)

However, the presence of soluble boron has been credited to provide adequate safety margin to maintain spent fuel assembly storage rack $k_{eff} \leq 0.95$ (also on a 95/95 basis) for all postulated accident scenarios involving dropped or misloaded fuel assemblies. Crediting the presence of soluble boron for mitigation of these scenarios is acceptable based on applying the "double contingency principle" which states that there is no requirement to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident (Refs. 4 and 5).

The accident analyses address the following five postulated scenarios:

1) fuel assembly drop on top of rack;
2) fuel assembly drop between rack modules;
3) fuel assembly drop between rack modules and spent fuel pool wall;
4) change in spent fuel pool water temperature; and
5) fuel assembly loaded contrary to placement restrictions.

Of these, only scenarios 2, 3, and 5 have the capacity to increase reactivity for the Holtec spent fuel pool storage racks.

Calculations were performed to determine the reactivity change caused by a change in spent fuel pool water temperature outside the normal range (50 - 160°F).

BASES

APPLICABLE SAFETY ANALYSES (continued)

Calculations were performed for the Holtec spent fuel pool storage racks, for a spent fuel pool temperature of 4°C (39°F) which is well below the lowest normal operating temperature (50°F). Because the temperature coefficient of reactivity in the spent fuel pool is negative, temperatures greater than 4°C will result in a decrease in reactivity.

In all cases, additional reactivity margin is available to the 0.95 k_{eff} limit to allow for temperature accidents.

For the fuel assembly misload accident, calculations were performed to show the largest reactivity increase caused by a Westinghouse 17X17 OFA fuel assembly misplaced into a Holtec Region 2 storage cell for which the restrictions on enrichment or burnup are not satisfied. The assembly misload accident can only occur during fuel handling operations in the spent fuel pool.

The AREVA Advanced Mk-BW(A) fuel assemblies were analyzed (Ref. 6) for storage in the Holtec racks. Calculations were performed using the same assumptions as in Reference 2. Calculation results for the Westinghouse OFA fuel were compared to that in Reference 2. Calculation results for the AREVA Advanced Mk-BW(A) fuel were compared to those for the Westinghouse OFA fuel and to the regulatory limit of $k_{eff} \leq 0.95$.

Reference 6 shows that a fresh AREVA Advanced Mk-BW(A) fuel assembly is less reactive than a fresh Westinghouse OFA assembly. Therefore, the AREVA Advanced Mk-BW(A) fuel may be used in the Holtec Region 1 spent fuel pool storage racks since the reactivity is bounded by that of the fresh Westinghouse OFA fuel assemblies. Reference 6 shows that a burned AREVA Advanced Mk-BW(A) fuel assembly is more reactive than a Westinghouse OFA assembly of the same burnup, but that placement of AREVA Advanced Mk-BW(A) fuel assemblies in Holtec Region 2 spent fuel pool storage racks meets the regulatory limit of $k_{eff} \leq 0.95$.

BASES

APPLICABLE SAFETY ANALYSES (continued)

For the above postulated accident conditions, the double contingency principle can be applied. Specifically, the presence of soluble boron in the spent fuel pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event. For the Holtec spent fuel pool storage racks, spent fuel pool soluble boron has been credited in the criticality safety analysis to offset the reactivity caused by postulated accident conditions. Because the Region 1 racks are designed for the storage of fresh fuel assemblies, a fuel assembly misload accident has no consequences from a criticality standpoint (i.e., the acceptance criteria for storage are satisfied by all assemblies in the spent fuel pool).

Based on the above discussion for the Holtec spent fuel pool storage racks, should a fuel assembly misload accident occur in the Region 2 storage cells, k_{eff} will be maintained ≤ 0.95 due to the presence of at least 300 ppm of soluble boron in the spent fuel pool water.

The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The spent fuel pool boron concentration is required to be ≥ 300 ppm for the Holtec spent fuel pool storage racks. The specified concentration of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in References 2 and 3. The dissolved boron concentration of 300 ppm bounds the minimum required concentration for accidents occurring during fuel assembly movement within the spent fuel pool for the Holtec spent fuel pool storage racks.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

The ACTIONS have been modified by a Note indicating that LCO 3.0.3 does not apply.

A.1 and A.2

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. Immediate actions are also taken to restore spent fuel pool boron concentration.

If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day frequency is appropriate based on operating experience and takes into consideration that no major replenishment of spent fuel pool water is expected to occur over such a short period of time.

BASES

REFERENCES

1. NRC Memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants," dated August 19, 1998.

2. Holtec International Report, HI-982094, "Criticality Analysis for the Byron/Braidwood Rack Installation Project," Project No. 80944, 1998.

3. UFSAR, Section 15.7.4.

4. Double contingency principle of ANSI N16.1 - 1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

5. ANSI/ANS 8.1 - 1983 "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

6. AREVA NP Report, 32-5069924-00, “Braidwood Fuel Rack Criticality Evaluation,” dated September 9, 2005.

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B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel pool provides for storage of various Westinghouse Optimized Fuel Assembly (OFA) types of different initial fuel enrichments and exposure histories in two distinct regions. (For this discussion, the term OFA is intended to refer to the specific reduced fuel rodlet diameter, and includes all analyzed fuel types with this diameter, such as Vantage 5.) The spent fuel pool is provided with 24 Holtec spent fuel pool storage racks, which provide placement locations for a total of 2984 new or used fuel assemblies. Of these 24 Holtec spent fuel pool storage racks, four are designated "Region 1" with the remaining 20 racks designated as "Region 2." The analytical methodology used for the criticality analyses is in accordance with established NRC guidelines (Ref. 2).

The storage of AREVA Advanced Mk-BW(A) fuel assemblies in the spent fuel pool storage racks was analyzed in Reference 6 and found to conform to the design basis described herein for Westinghouse OFA fuel. The requirements of LCO 3.7.15 and LCO 3.7.16 are applicable to the storage of both Westinghouse OFA and AREVA Advanced Mk-BW(A) fuel assemblies.

BASES

BACKGROUND (continued)

Region 1 racks contain 396 cells which are analyzed for storing Westinghouse OFAs in an "All Cells" arrangement (that is, the criticality analysis assumes that spent fuel assemblies reside in all available cell locations). The stored fuel assemblies may contain an initial nominal enrichment of ≤ 5.0 weight percent U-235 (with or without IFBAs installed) (Ref. 4).

Region 2 racks contain 2588 cells which are also analyzed for storing Westinghouse OFAs in an "All Cells" arrangement (that is, the criticality analysis assumes that spent fuel assemblies reside in all available cell locations). For the "All Cells" storage configuration, the stored fuel assemblies may contain an initial nominal enrichment of ≤ 5.0 weight percent U-235 with credit for burnup.

The water in the spent fuel pool normally contains soluble boron which results in large subcriticality margins under actual operating conditions.

APPLICABLE SAFETY ANALYSES

Methodologies in accordance with established NRC guidelines were used to develop the criticality analyses (Ref. 1) for the Holtec spent fuel pool storage racks. The fuel handling accident analyses are described in Reference 3.

The criticality analyses for the spent fuel assembly storage racks confirm that k_{eff} remains ≤ 0.95 for the Holtec spent fuel pool storage racks (including uncertainties and tolerances) at a 95% probability with a 95% confidence level (95/95 basis), based on the accident condition of the pool being flooded with unborated water. Thus, the design of both regions assumes the use of unborated water while maintaining stored fuel in a subcritical condition.

BASES

APPLICABLE SAFETY ANALYSES (continued)

However, the presence of soluble boron has been credited to provide adequate safety margin to maintain spent fuel assembly storage rack $k_{eff} \leq 0.95$ (also on a 95/95 basis) for all postulated accident scenarios involving dropped or misloaded fuel assemblies. Crediting the presence of soluble boron for mitigation of these scenarios is acceptable based on applying the "double contingency principle" which states that there is no requirement to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident (Refs. 4 and 5).

The accident analyses address the following five postulated scenarios:

1) fuel assembly drop on top of rack;
2) fuel assembly drop between rack modules;
3) fuel assembly drop between rack modules and spent fuel pool wall;
4) change in spent fuel pool water temperature; and
5) fuel assembly loaded contrary to placement restrictions.

Of these, only scenarios 2, 3, and 5 have the capacity to increase reactivity for the Holtec spent fuel pool storage racks.

Calculations were performed to determine the reactivity change caused by a change in spent fuel pool water temperature outside the normal range (50 - 160°F).

BASES

APPLICABLE SAFETY ANALYSES (continued)

Calculations were performed for the Holtec spent fuel pool storage racks, for a spent fuel pool temperature of 4°C (39°F) which is well below the lowest normal operating temperature (50°F). Because the temperature coefficient of reactivity in the spent fuel pool is negative, temperatures greater than 4°C will result in a decrease in reactivity.

In all cases, additional reactivity margin is available to the 0.95 k_{eff} limit to allow for temperature accidents.

For the fuel assembly misload accident, calculations were performed to show the largest reactivity increase caused by a Westinghouse 17X17 OFA fuel assembly misplaced into a Holtec Region 2 storage cell for which the restrictions on enrichment or burnup are not satisfied. The assembly misload accident can only occur during fuel handling operations in the spent fuel pool.

The AREVA Advanced Mk-BW(A) fuel assemblies were analyzed (Ref. 6) for storage in the Holtec racks. Calculations were performed using the same assumptions as in Reference 2. Calculation results for the Westinghouse OFA fuel were compared to that in Reference 2. Calculation results for the AREVA Advanced Mk-BW(A) fuel were compared to those for the Westinghouse OFA fuel and to the regulatory limit of $k_{eff} \leq 0.95$.

Reference 6 shows that a fresh AREVA Advanced Mk-BW(A) fuel assembly is less reactive than a fresh Westinghouse OFA assembly. Therefore, the AREVA Advanced Mk-BW(A) fuel may be used in the Holtec Region 1 spent fuel pool storage racks since the reactivity is bounded by that of the fresh Westinghouse OFA fuel assemblies. Reference 6 shows that a burned AREVA Advanced Mk-BW(A) fuel assembly is more reactive than a Westinghouse OFA assembly of the same burnup, but that placement of AREVA Advanced Mk-BW(A) fuel assemblies in Holtec Region 2 spent fuel pool storage racks meets the regulatory limit of $k_{eff} \leq 0.95$.

BASES

APPLICABLE SAFETY ANALYSES (continued)

For the above postulated accident conditions, the double contingency principle can be applied. Specifically, the presence of soluble boron in the spent fuel pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event. For the Holtec spent fuel pool storage racks, spent fuel pool soluble boron has been credited in the criticality safety analysis to offset the reactivity caused by postulated accident conditions. Because the Region 1 racks are designed for the storage of fresh fuel assemblies, a fuel assembly misload accident has no consequences from a criticality standpoint (i.e., the acceptance criteria for storage are satisfied by all assemblies in the spent fuel pool).

Based on the above discussion for the Holtec spent fuel pool storage racks, should a fuel assembly misload accident occur in the Region 2 storage cells, k_{eff} will be maintained ≤ 0.95 due to the presence of at least 300 ppm of soluble boron in the spent fuel pool water.

The configuration of fuel assemblies in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool in accordance with the requirements in the accompanying LCO ensure that the k_{eff} of the spent fuel pool will always remain ≤ 0.95 assuming the pool is flooded with unborated water for the Holtec spent fuel pool storage racks.

For the Holtec spent fuel pool storage racks, in LCO Figure 3.7.16-1, the Acceptable Burnup Domain lies on, above, and to the left of the line.

The use of linear interpolation between minimum burnups in Figure 3.7.16-1 is acceptable.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

The ACTIONS have been modified by a Note indicating that LCO 3.0.3 does not apply.

A.1

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with the requirements of the LCO, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance.

If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

Item a and item b are performed, as applicable, is performed prior to storing the fuel assembly in the intended spent fuel pool storage location. The frequency is appropriate because compliance with the SR ensures that the relationship between the fuel assembly and its storage location will meet the requirements of the LCO and preserve the assumptions of the analyses.

This SR verifies by administrative means that the initial nominal enrichment of the fuel assembly is met to ensure that the assumptions of the safety analyses are preserved.

SR 3.7.16.2

SR 3.7.16.2 is performed prior to storing the fuel assembly in the intended spent fuel pool storage location. The frequency is appropriate because compliance with the SR ensures that the relationship between the fuel assembly and its storage location will meet the requirements of the LCO and preserve the assumptions of the analyses.

This SR verifies by administrative means that the combination of initial enrichment, burnup, and decay time, as applicable, of the fuel assembly is within the Acceptable Burnup Domain of Figure 3.7.16-1 for the intended storage configuration to ensure that the assumptions of the safety analyses are preserved.

BASES

REFERENCES

1. NRC Memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants," dated August 19, 1998.

2. Holtec International Report, HI-982094, "Criticality Analysis for the Byron/Braidwood Rack Installation Project," Project No. 80944, 1998.

3. UFSAR, Section 15.7.4.

4. Double contingency principle of ANSI N16.1 - 1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

5. ANSI/ANS 8.1 - 1983 "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

6. AREVA NP Report, 32-5069924-00, “Braidwood Fuel Rack Criticality Evaluation,” dated September 9, 2005.

ATTACHMENT 5

BRAIDWOOD STATION
UNITS 1 and 2

Docket Nos. STN 50-456 and STN 50-457
License Nos. NPF-72 and NPF-77

License Amendment Request and Exemption Requests to Allow Use of
AREVA NP Inc. Advanced Mark-BW(A) Fuel Lead Assemblies

Justification for Exemption from 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models"

ATTACHMENT 5
Justification for Exemption

Specific Exemption Request

In accordance with 10 CFR 50.12, "Specific exemptions," Exelon Generation Company, LLC (EGC) is requesting temporary exemptions from the requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models." Specifically, the requested exemptions will allow Braidwood Station Unit 1 to place up to eight AREVA NP Inc. (AREVA), formerly Framatome, modified Advanced Mark-BW fuel assemblies (Advanced Mark-BW(A) fuel assemblies) containing M5 alloy clad fuel rods in nonlimiting (i.e., for $F_{\Delta H}$, F_Q, and fuel assembly average power at hot full power normal operating conditions) core regions (i.e., locations) for evaluation during Cycles 14, 15, and 16. The temporary exemptions are requested for the period of time when these fuel assemblies reside in the core.

Basis For Exemption Request

The Braidwood Station Unit 1 core consists of 193 fuel assemblies. The core may consist of any combination of Westinghouse VANTAGE 5 and VANTAGE+ fuel assemblies arranged in a checkered low-leakage pattern. Each fuel assembly consists of 264 fuel rods arranged in a 17 x 17 array. The VANTAGE+ fuel assembly design includes the following features: ZIRLO™ clad fuel rods, ZIRLO™ thimble and instrumentation tubes, and variable pitch plenum spring. The VANTAGE 5 design has added features, known as PERFORMANCE+ design features, which are: ZIRLO™ intermediate grids and flow mixer grids, an oxide protective coating at the lower end of the fuel rod cladding, and a protective bottom grid.

EGC intends to place up to eight AREVA Advanced Mark-BW(A) fuel assemblies in nonlimiting regions (i.e., locations) of the core for evaluation during Cycles 14, 15, and 16. The Advanced Mark-BW(A) fuel assemblies are similar in design to the Advanced Mark-BW assemblies using M5 alloy material for the cladding, structural tubing, and grids generically approved for use in Westinghouse 3- and 4-loop designed pressurized water reactors with 17 x 17 fuel rod arrays (i.e., Reference 2). The Advanced Mark-BW(A) fuel assemblies incorporate the following minor modifications relative to Advanced Mark-BW assemblies: removable upper end fitting with quarter-turn quick-disconnect feature, M5 MONOBLOC™ guide tubes, welded structure, application of different spacer grid types, and a FUELGUARD™ lower end fitting. The purpose of this evaluation program is to allow EGC to gain an understanding of the behavior of mixed fuel cores prior to a possible transition from Westinghouse fuel to AREVA fuel.

<u>10 CFR 50.12(a) Requirements</u>

The requested exemptions to the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, to allow the use of fuel assemblies constructed with M5 alloy meet the requirements of 10 CFR 50.12 as discussed below.

10 CFR 50.12 states that the NRC may grant an exemption from requirements contained in 10 CFR 50 provided that the following is satisfied.

ATTACHMENT 5
Justification for Exemption

1. The requested exemption is authorized by law

   No law exists which precludes the activities covered by these exemption requests. Transition to an alternate, but equivalent fuel product is not precluded by law. The AREVA Advanced Mark-BW(A) fuel assemblies to be irradiated at Braidwood Station Unit 1 contain a cladding material that does not conform to the cladding material designations defined in 10 CFR 50.46 (i.e., Zircaloy or ZIRLO™). The Baker-Just equation, set forth in 10 CFR Part 50, Appendix K, paragraph I.A.5, by its terms applies only to cladding made of Zircaloy, and is used to calculate the rate of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction. The criteria of these sections will, however, continue to be satisfied for the operation of Braidwood Station Unit 1 cores containing up to eight AREVA Advanced Mark-BW(A) fuel assemblies in nonlimiting regions (i.e., locations) of the core.

2. The requested exemption does not present an undue risk to the public health and safety

   The AREVA Advanced Mark-BW(A) fuel has been evaluated for use in reloads containing up to eight AREVA Advanced Mark-BW(A) fuel assemblies in nonlimiting regions (i.e., locations) of the Braidwood Station Unit 1 core to confirm that operation with the AREVA Advanced Mark-BW(A) fuel does not significantly increase the probability of occurrence or the consequences of an accident at Braidwood Station Unit 1 and will not create the possibility for a new or different type of accident that could pose a risk to public health and safety. In addition, appropriate mixed-core safety analyses will be performed as part of the reload analysis to demonstrate that AREVA Advanced Mark-BW(A) fuel does not present an undue risk to the public health and safety. EGC employs NRC approved methods for the reload design process for the Braidwood Station Unit 1.

3. The requested exemption is consistent with the common defense and security

   The Advanced Mark-BW(A) fuel contains low enrichment uranium, similar to the reload fuel assemblies currently used at Braidwood Station Unit 1. The special nuclear material in this fuel product will continue to be handled and controlled in accordance with approved procedures. Use of Advanced Mark-BW(A) fuel will not affect the operation of Braidwood Station Unit 1 or endanger the common defense and security.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.46 and 10 CFR Part 50, Appendix K

   In accordance with 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of paragraph (a)(2)(ii), "Application of the regulation is not necessary to achieve the underlying purpose of the rule."

   10 CFR 50.12(a)(2)(ii)

   The underlying purpose of 10 CFR 50.46 is to ensure that nuclear power facilities have adequately demonstrated the cooling performance of their Emergency Core Cooling System (ECCS). The effectiveness of the ECCS at Braidwood Station Unit 1 will not be affected by the use of up to eight Advanced Mark-BW(A) fuel assemblies. 10 CFR 50.46 applies to fuel with Zircaloy and ZIRLO™ cladding. Although the AREVA Advanced Mark-BW(A) fuel assemblies incorporate a M5 cladding material that is not addressed by 10 CFR 50.46, the

criteria of 10 CFR 50.46 (e. g., peak cladding temperature limit of 2200°F) will continue to be satisfied for the Braidwood Station Unit 1 core. Normal reload safety analyses will confirm that the Braidwood Station Unit 1 safety analyses acceptance criteria continue to be met with the use of up to eight Advanced Mark-BW(A) fuel assemblies. Consequently, the use of the M5 alloy cladding in Advanced Mark-BW(A) fuel will not have a detrimental impact on the performance of the Braidwood Station Unit 1 core under Loss-of-Coolant Accident (LOCA) conditions.

The underlying purpose of 10 CFR Part 50, Appendix K, is to ensure that cladding oxidation and hydrogen generation are appropriately limited during a LOCA and conservatively accounted for in the ECCS model. This regulation sets forth requirements for plants that use either Zircaloy or ZIRLO™ fuel cladding; the AREVA Advanced Mark-BW(A) fuel assemblies use M5 fuel cladding, which is not addressed by 10 CFR Part 50, Appendix K. Specifically, Paragraph I.A.5 of 10 CFR Part 50, Appendix K, requires that the Baker-Just equation be used in the ECCS evaluation model to determine the rate of energy release, hydrogen generation, and cladding oxidation. This equation conservatively bounds all post-LOCA scenarios. In References 1 and 3, the NRC concluded that the Baker-Just correlation is conservative for determining high temperature M5 alloy oxidation for LOCA analysis, and that the correlation is acceptable for LOCA ECCS analysis up to the currently approved burn-up levels. Therefore, when M5 alloy is used as fuel rod cladding and structural material, the Baker-Just correlation bounds post-LOCA scenarios, and ECCS evaluation model criteria will be met. Accordingly, application of the rule requirements to use Zircaloy or ZIRLO™ is not necessary to achieve the underlying purpose of 10 CFR Part 50, Appendix K.

Therefore, the underlying purpose of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, will continue to be satisfied for operation in Cycles 14, 15, and 16 with up to eight AREVA Advanced Mark-BW(A) fuel assemblies placed in nonlimiting regions (i.e., locations) of the core. Issuance of an exemption from these regulations will not compromise the safe operation of Braidwood Station Unit 1.

Environmental Assessment

In accordance with 10 CFR 51.30, “Environmental assessment,” and 10 CFR 51.32, “Finding of no significant impact,” the following information is provided in support of an environmental assessment and finding of no significant impact for the proposed action.

The proposed action would grant exemptions from requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," to allow the use of up to eight AREVA Advanced Mark-BW(A) fuel assemblies containing M5 alloy cladding in nonlimiting regions (i.e., locations) of the core for operation during Cycles 14, 15, and 16.

The requested exemption is needed because Exelon Generation Company, LLC (EGC) intends to place up to eight AREVA Advanced Mark-BW(A) fuel assemblies in nonlimiting regions (i.e., locations) of the core for evaluation during Cycles 14, 15, and 16. The purpose of this evaluation program is to allow EGC to gain an understanding of the behavior of mixed fuel cores prior to a possible transition from Westinghouse fuel to AREVA fuel.

The principal alternative to the proposed action would be to deny the requested exemption and require adherence to the current 10 CFR 50.46 and 10 CFR Part 50, Appendix K,

requirements. Denial of the exemption requests would result in no change in environmental impacts. The environmental impacts of the proposed action and the alternative action are similar. Based on the assessment above, the proposed action will not have a significant effect on the quality of the human environment.

Regarding alternative use of resources, granting the requested exemptions will not involve the use of resources not previously considered in the Final Environmental Statements for Braidwood Station (i.e., NUREG-1026, "Final Environmental Statement related to the operation of Braidwood Station, Units 1 and 2," dated June 1984).

The proposed action (i.e., granting the exemption request) will not significantly increase the probability or consequences of accidents, no changes are being made in the types or quantities of any radiological effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

The proposed action does not affect non-radiological plant effluents and has no other environmental impact. Therefore, there are no significant non-radiological impacts associated with the proposed action.

The environmental impacts of the proposed action and the alternative action are similar. Based on the assessment presented above, the proposed action will not have a significant effect on the quality of the human environment.

Conclusion

Title 10 CFR 50.46, and 10 CFR Part 50, Appendix K, only apply to the use of fuel rods clad with Zircaloy or ZIRLO™. These regulations do not apply to the proposed use of M5 alloy. In order to support EGC evaluations to gain experience with mixed-fuel cores in preparation for a possible transition to full core loads consisting of AREVA Advanced Mark-BW(A) fuel assemblies, an exemption from the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, is requested. Evaluations demonstrate that the intent of the regulations continue to be met because aspects of safety, including mechanical, neutronic, thermal hydraulic, transient, and LOCA accident analyses results for cores containing up to eight AREVA Advanced Mark-BW(A) fuel assemblies in nonlimiting core regions (i.e., locations) will be bounded by the current Braidwood Station Unit 1 safety analyses.

EGC respectfully requests that the NRC approve this exemption request by August 15, 2007, in order to support Braidwood Station Unit 1 Cycle 14 operations scheduled to begin in November 2007.

References

1. Letter from S. A. Richards (NRC) to T. A. Coleman (Framatome Cogema Fuels), "Revised Safety Evaluation (SE) for Topical Report BAW-10227P: 'Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel'," dated February 4, 2000

2. Framatome ANP BAW-10239(P)-A, "Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report," Revision 0, dated July 2004

3. Letter from H. N. Berkow (NRC) to J. F. Mallay (Framatome ANP), " Safety Evaluation of Framatome ANP Topical Report BAW-10186P-A, Revision 1, Supplement 1, 'Extended Burnup Evaluation,'" dated June 18, 2003

ATTACHMENT 6

BRAIDWOOD STATION
UNITS 1 and 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

License Amendment Request and Exemption Requests to Allow Use of
AREVA NP Inc. Advanced Mark-BW(A) Fuel Lead Assemblies

List of Regulatory Commitments

ATTACHMENT 6
List of Regulatory Commitments

The following table identifies commitments made in this document. (Any other actions discussed in this submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.)

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	Programmatic (Yes/No)
To provide assurance that the current bounding evaluations performed for Braidwood Station Unit 1 reloads will remain valid the AREVA fuel assemblies being used for Braidwood Station Unit 1 Cycles 14, 15, and 16 will be placed in nonlimiting core regions (i.e., for $F_{\Delta H}$, F_Q, and fuel assembly average power at hot full power normal operating conditions) and the nuclear design of the Braidwood Station Unit 1 Cycles 14, 15, and 16 cores performed by Westinghouse will ensure sufficient margin between the lead Westinghouse fuel assembly and the Advanced Mark-BW(A) assemblies for $F_{\Delta H}$, F_Q, and for fuel assembly average power; these margins will be a minimum of 5%. The reload safety evaluation (RSE) will ensure that: • The applicable reload analysis acceptance criteria continue to be met. • The AREVA fuel assemblies are not placed in locations containing rod cluster control assemblies. • The AREVA fuel assemblies do not have an adverse impact on the co-resident Westinghouse fuel. The Westinghouse fuel will be shown to meet its mechanical and thermal-hydraulic limits as described in the Braidwood Station UFSAR. • Confirmatory analyses demonstrate that the AREVA fuel assemblies satisfy the operating and safety limits established by the current Westinghouse Analysis of Record (AOR).	Prior to operation in Cycles 14, 15, and 16	Yes	No
The AREVA fuel assemblies will meet AREVA's own mechanical and thermal-hydraulic limits per Topical Report BAW-10239(P)(A) and other approved methodologies as discussed in this submittal.	Prior to operation in Cycles 14, 15, and 16	Yes	No