ENCLOSURE 5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED

TO AMENDMENT NO. 251 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-19,

AMENDMENT NO. 244 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-25,

AMENDMENT NO. 264 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-29,

AND AMENDMENT NO. 259 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-30

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NOS. 2 AND 3, AND

QUAD CITIES NUCLEAR POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-237, 50-249, 50-254, AND 50-265

Proprietary information pursuant to Title 10 of Code of Federal Regulations (10 CFR) Section 2.390 has been redacted from this document. Redacted information is identified by blank space enclosed within double brackets as shown here:[[ ]]

ADAMS Accession No. ML16221A061

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC or Commission) dated February 6, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15055A154), as supplemented by letters dated September 1, 2015, January 20, January 28, April 26, June 22, and September 28, 2016 (ADAMS Accession Nos. ML15251A381, ML16020A232, ML16028A303, ML16117A187, ML16174A374, and ML16272A376, respectively) Exelon Generation Company, LLC (EGC, the licensee) requested changes to the technical specifications (TSs) for the Dresden Nuclear Power Station (DNPS), Unit Nos. 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Unit Nos. 1 and 2.

The proposed license amendment requests NRC approval to change several DNPS and QCNPS technical specifications, as discussed below, in support of transitioning from the currently used Westinghouse SVEA-96 OPTIMA2 (OPTIMA2) nuclear fuel design to the AREVA ATRIUM 10XM fuel design. The licensee's submittal describes the applicability of the AREVA safety analysis methodologies for DNPS and QCNPS, the fuel design and thermal hydraulic analysis, the analyses for anticipated operational occurrences (AOOs), and design basis accidents using AREVA methods.

The licensee intends to perform phased[1] fuel transitions to AREVA ATRIUM 10XM fuel at DNPS and QCNPS. Scheduled refueling outages (RFO) are to occur as follows:

Station	Unit	Refueling Outage	Outage Schedule	Implementing Cycle
Dresden	3	D3R24	Fall 2016	25
Quad Cites	1	Q1R24	Spring 2017	25
Dresden	2	D2R25	Fall 2017	26
Quad Cities	2	Q2R24	Spring 2018	25

The licensee's submittal (Reference 1) requested approval to operate the ATRIUM 10XM fuel design at extended power uprate (EPU) conditions with the Maximum Extended Load Line Limit Analysis (MELLLA) operating domain. Further discussions regarding the EPU/MELLLA operating domains for DNPS and QCNPS can be found in the perspective EPU safety evaluations[2] previously approved by the NRC; and included as part of each facilities licensing basis. The licensee's submittal proposed multiple TS changes to support transitioning to ATRIUM 10XM fuel at DNPS and QCNPS, and to adopt the AREVA fuel design methodologies and safety analyses. The proposed TS changes include:[3]

(1) (TS 3.2.3) addition of new surveillance requirement (SR 3.2.3.2) to TS 3.2.3, "Linear Heat Generation Rate (LHGR)." Since the transient analyses take credit for conservatism in the scram speed performance, demonstrating scram speed distribution is consistent with that used in the transient analyses,

(2) (TS 3.3.4.1) revision to surveillance requirement (SR 3.3.4.1.4.b) Reactor Pressure Vessel (RPV) Steam Dome Pressure-High Allowable Value (AV). The AV for DNPS, Unit Nos. 2 and 3, is lowered to less than or equal to (≤) 1198 pounds per square inch gauge (psig) (previously ≤ 1241 psig). The AV for QCNPS, Unit Nos. 1 and 2, is lowered to ≤ 1195 psig (previously ≤ 1219 psig). Reduction of the Steam Dome Pressure-High AV increases the margin to the maximum RPV acceptance criteria for certain anticipated transient without scram (ATWS) transients,

(3) (TS 3.7.7) revision to the TS limiting condition for operation (LCO) associated with the Main Turbine Bypass System (LCO 3.7.7) to include requirements to use the minimum critical power ratio (MCPR) limits (LCO 3.2.2) and the linear heat generation rate (LHGR) limits (LCO 3.2.3) during plant operations greater than or equal to (≥) 25 percent of rated thermal power when the Main Turbine Bypass System is inoperable, and

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1 DNPS/QCNPS will operate using mixed cores comprised of Westinghouse SVEA-96 OPTIMA2 and AREVA ATRIUM 10XM fuel. During successive one-third core reload cycles, each core will transition towards the exclusive use of ATRIUM 10XM fuel.

2 DNPS, Units 2 and 3 (ADAMS Accession No. ML013620048); QCNPS, Units 1 and 2 (ADAMS Accession No. ML013620116)

3 Section 3.3 of this Safety Evaluation documents the staff's evaluation of these proposed TS changes.

(4) (TS 5.6.5) revision to TS 5.6.5, “Core Operating Limits Report (COLR),” to delete the TS 5.6.5.b reference to the Commonwealth Edison Topical Report (TR) NFSR-0091, "Benchmark of CASMO/MICROBURN boiling-water reactor (BWR) Nuclear Design Methods," as this methodology is no longer used to develop core operating limits for DNPS and QCNPS. The licensee proposes to include references to 18 NRC-approved AREVA methodologies to be used to develop future core operating limits for the DNPS and QCNPS cores reloaded with AREVA ATRIUM 10XM fuel. TS 5.6.5.b lists the approved analytical methods which may be used to determine input to the core operating limit report (COLR).

2.0 REGULATORY EVALUATION

The following NRC regulatory requirements are applicable to the NRC staff's review for this proposed license amendment:

In Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, the NRC established its regulatory requirements related to the content of TSs. 10 CFR 50.36(b) requires that each license authorizing the operation of a facility will include TSs and that the TSs will be derived from the safety analysis. 10 CFR 50.36(c) specifies the categories that are to be included in the TS including (1) Safety limits, limiting safety system settings, and limiting control settings; (2) Limiting conditions for operation (LCOs); (3) Surveillance requirements (SRs); (4) Design Features; and (5) Administrative controls.

- Per 10 CFR 50.36(c)(1)(i)(A) Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shutdown.

- Per 10 CFR 50.36(c)(1)(ii)(A) Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

- Per 10 CFR 50.36(c)(2)(i) LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

- Per 10 CFR 50.36(c)(3) SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

- Per 10 CFR 50.36(c)(4) Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety.

- Per 10 CFR 50.36(c)(5) Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

- Per 10 CFR 50.46(a)(1)(i) Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in section 50.46(b). ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.

- 10 CFR Part 50, Appendix K, sets forth the documentation requirements for each evaluation model, and establishes required and acceptable features of evaluation models for heat removal by the ECCS.

The NRC staff reviewed the licensee's submittal to evaluate the applicability of AREVA methodologies to DNPS and QCNPS, to confirm that the use of the methodologies is within the NRC-approved ranges of its applicability, and to verify that the results of the analyses are in compliance with the applicable requirements of the following Design Criteria specified in the DNPS and QCNPS UFSARs. [4]

The DNPS, Units 2 and 3, UFSAR (Section 3.1.2) describes compliance with the following criteria:

- Design Criterion 10 - Reactor Design - The reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- Design Criterion 12 - Suppression of Reactor Power Oscillations - The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

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[4] DNPS and QCNPS were constructed and licensed prior to the implementation of the General Design Criteria required in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The Design Criteria included within this Safety Evaluation are specified in the DNPS and QCNPS UFSAR's.

- Design Criterion 15 – Reactor Coolant System Design - The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- Design Criterion 35 - Emergency Core Cooling - A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

  Suitable redundancy in components and features and suitable interconnections, leak detection, isolation and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The QCNPS, Units 1 and 2, UFSAR (Section 3.1.2) describes compliance with the following criteria:

- Criterion 6 - Reactor Core Design - The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

- Criterion 7 - Suppression of Reactor Power Oscillations - The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

- Criterion 9 – Reactor Coolant Pressure Boundary - The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

- Criterion 35 - Emergency Core Cooling (Section 3.1.6) - At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

In addition to the above regulatory requirements, the following guidance documents were considered during this review:

- NUREG-0800, Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, Section 4.2, "Fuel System Design,"
- NUREG-0800, Section 4.4, "Thermal and Hydraulic Design,"
- NUREG-0800, Section 15.8, "Anticipated Transients Without Scram," and
- NUREG-1433, Volume 1, Revision 4.0, "Standard Technical Specifications [STS] General Electric BWR/4 Plants."

3.0 TECHNICAL EVALUATION

Both DNPS and QCNPS are General Electric/Type-3 light-water boiling-water reactor designs (BWR/3). The facilities share similar core operating parameters exhibited in the following table.

Operating Parameters	DNPS	QCNPS
EPU Licensed Power	2,957 MWt*	2,957 MWt
Rated Core Flow	98.0 Mlbm/hr**	98.0 Mlbm/hr
Rated Steam Flow	11.713 Mlbm/hr	11.713 Mlbm/hr
Normal Feedwater Temperature (° F)	355.6 °F	355.6 °F
Nominal Dome Pressure (rated power)	1,015 psia***	1,015 psia
Number of Fuel Assemblies	724	724

* (Mega-Watt thermal) ** (Millions pounds-mass per hour) *** (pounds per square-inch absolute)

The licensee's submittal states:

> "Because of the similarity between the two stations, the application of the AREVA methodology to a representative core design [1] is sufficient to support transition to AREVA ATRIUM 10XM fuel at both stations. Application of the AREVA methodology involves key analyses for Loss of Coolant Accident (LOCA), transient, and ATWS."

The licensee's submittal also states:

> "The DNPS and QCNPS Emergency Core Cooling System (ECCS) components, although slightly different in performance, [2] are similar. The LOCA analysis break spectrum is dependent on system response inputs. These inputs are measured by the timing of reactor trip, ECCS actuation, refilling of the lower plenum and maintaining two-phase cooling in the hot assembly. Furthermore, similarities between the two stations include near identical reactor vessel geometry and dimensions as well as recirculation system parameters, single failure, and ECCS availability. Analyses for QCNPS are performed to demonstrate applicability of AREVA methodology. Because of similarity between the two stations, this demonstration is also applicable to DNPS. A DNPS plant specific LOCA break spectrum analysis will be performed in accordance with the NRC-approved methodologies prior to fuel introduction at DNPS."

This Safety Evaluation reflects the staff's determination that the licensee's proposed use of AREVA methodology to use a representative core design (i.e., QCNPS Unit 2, Cycle 24) is an adequate basis for the licensee's analysis, and is therefore acceptable. Section 3.2.6 below provides additional information regarding the AREVA methodology to use a representative core design. References in this safety evaluation to the "representative core," or "QCNPS Unit 2, Cycle 24," or "AREVA methodology" are used interchangeably to depict the modeling used in the analysis to transition DNPS and QCNPS to ATRIUM 10XM fuel. This Safety Evaluation documents the staff's review of the licensee's use of the representative core design to analyze key conditions such as LOCA, Transients, and ATWS. Specific results and final staff conclusions for these conditions are documented below in Section 3.1.2 (LOCA), Section 3.1.6 (Transients), and Section 3.1.4.2 (ATWS).

3.1 Accident and Transient Analysis

3.1.1 Thermal Limits

The thermal limits include the MCPR for the safety and operating limits, the LHGR, and the maximum average planar linear heat generation rate (MAPLHGR). Limits are provided to support two-loop operation (TLO), single-loop operation (SLO), and several equipment out-of-

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1 QCNPS Unit 2 Cycle 24 is the representative core design for modeling and analysis evaluated within this Safety Evaluation.

2 DNPS Units 2 and 3 designed with Isolation Condensers, QCNPS Units 1 and 2 designed with Reactor Core Isolation Cooling (RCIC).

service scenarios. The MCPR limits protect the fuel cladding integrity in accordance with DNPS (Design Criterion 10) and QCNPS (Criterion 6), as specified in the DNPS and QCNPS UFSARs.

3.1.1.1 Safety Limit Minimum Critical Power Ratio (SLMCPR)

The core MCPR must remain above the SLMCPR during steady state operation and during AOOs. The SLMCPR includes a margin for uncertainties in plant operating parameters such as the power distribution, nuclear instrumentation, and the critical power correlation. The SLMCPR is determined using an NRC-approved, statistical process to roll-up the various uncertainties as described in ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," dated June 2011 (ADAMS Accession No. ML112590057) (Reference 5). As described in Attachment 12 to the licensee's application dated February 6, 2015 (ADAMS Accession No. ML15055A154), the determination of the MCPR limits for the QCNPS, Unit 2, Cycle 24, representative core design is based on the analyses of the limiting AOOs. The MCPR operating limits are established so that less than 0.1 percent of the fuel rods in the core are expected to experience boiling transition during an AOO initiated from rated or off-rated conditions and are based on the TSs TLO SLMCPR limits and SLO [3] SLMCPR limits.

3.1.1.1.1 Conclusion

The NRC staff has determined that the licensee has appropriately evaluated and applied NRC-approved methodologies to establish adequate SLMCPR margins accounting for uncertainties from plant operating parameters. Therefore, the staff concludes the licensee's SLMCPR analysis is acceptable.

3.1.1.2 Operating Limit Minimum Critical Power Ration (OLMCPR)

The OLMCPR applies an additional margin to the SLMCPR for AOOs. The OLMCPR is determined on a cycle-specific basis using the NRC-approved suite of AREVA BWR safety analysis methods. The application of the Siemens Power Corporation BWR (SPCB) critical power ratio (CPR) correlation to co-resident OPTIMA2 legacy fuel followed the NRC approved indirect process described in TR EMF-2245(P)(A), "Application of Siemens Power Corporation's Critical Power Correlations to Co-resident Fuel," dated August 2000 (Reference 6).

The licensee provided the results of its AOO analyses for the QCNPS, Unit 2, Cycle 24, representative core design to demonstrate adequate core design to support the AREVA fuel transition (see Section 3.1.5.1). The power-dependent minimum critical power ratio ($MCPR_p$) and flow-dependent minimum critical power ratio ($MCPR_f$) limits presented in Tables 8.2 through 8.16 of ANP-3361NP , Revision 0, "Quad Cities Unit 2 Cycle 24 Representative Cycle Design Reload Safety Analysis" (ADAMS Accession No. ML15043A487), dated December 2014 (Reference 7), are operating limit MCPR values. The margin to the OLMCPR is determined using the limiting or highest MCPR limit from the applicable $MCPR_p$ or $MCPR_f$ limits for the given

[3] SLO restricted to < 50 % of rated core thermal power and < 51% of rated core flow.

power/flow state point. If there is a need to input $MCPR_p$ limits in the core monitoring system for power levels above 100 percent of rated, the rated power $MCPR_p$ limit can be used as it would be bounding. Stability analyses (see Section 3.1.5.2) provides a range of oscillation power range monitor (OPRM) setpoints as a function of the OLMCPR, in order to ensure that there is adequate protection from thermal-hydraulic instability at the chosen OLMCPR.

3.1.1.2.1 Conclusion

The NRC staff has completed its review of the licensee's analyses of the limiting AOOs on a cycle-specific basis to determine the OLMCPR, and because the THERMEX results indicated reasonable agreement with the results of the prior analyses, the staff concluded that the licensee's application of the THERMEX methodology to be acceptable. The staff determined the results indicated adequate thermal margins for the analyzed SLMCPR and supported OLMCPR values. Based on these determinations, the NRC staff concludes the licensee's analysis for OLMCPR is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.1.3 Linear Heat Generation Rate (LHGR)

Table 8.17 of ANP-3361NP lists the steady-state LHGR limits for ATRIUM 10XM fuel as a function of peak pellet exposure. The LHGR limits for OPTIMA2 fuel are lattice dependent and are specified in the plant specific COLR. An example is presented in the licensee's letter, "Core Operating Limits Report for Quad Cities Unit 2 Cycle 23," (SVP-14-034) (ADAMS Accession No. ML14126A600), dated May 2, 2014 (Reference 8). The flow-dependent linear heat generation rate factor ($LHGRFAC_f$) multipliers and power-dependent linear heat generation rate factor ($LHGRFAC_p$) multipliers are applied directly to the LHGR limits to protect against fuel melting and overstraining of the cladding during an AOO.

The ATRIUM 10XM $LHGRFAC_p$ multipliers are determined using the RODEX4 thermal-mechanical methodology discussed NRC approved topical report BAW-10247PA, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors", AREVA NP, dated February 2008 (Reference 9). A process consistent with the Westinghouse thermal-mechanical methodology was used to determine $LHGRFAC_p$ multipliers from the transient analyses for OPTIMA2 fuel. Exposure-dependent $LHGRFAC_p$ multipliers were established to support base case and equipment out of operation from beginning-of-cycle to near end-of-cycle (core average exposure of 34,702 MWd/MTU[4]) and from near end-of-cycle to end-of-cycle licensing basis (core average exposure of 36,774 MWd/MTU) for Nominal Scram Speed, Intermediate Scram Speed, and TS Scram Speed insertion times. The ATRIUM 10XM and OPTIMA2 $LHGRFAC_p$ multipliers are presented in Tables 8.18 through 8.24 of ANP-3361NP. The LHGR multipliers for the later exposure range can be used earlier in the cycle as they are the same or more conservative.

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[4] Core average exposure parameter applicable to the Quad Cities Unit 2 Cycle 24 representative core.

$LHGRFAC_f$ multipliers are established to provide protection against fuel centerline melt and overstraining of the cladding during a postulated slow flow excursion. For ATRIUM 10XM and OPTIMA2 fuel, the $LHGRFAC_f$ multipliers are presented Tables 8.25 and 8.26 of ANP-3361NP, respectively.

The $LHGRFAC_p$ and $LHGRFAC_f$ multipliers presented in Tables 8.18 through 8.26 of ANP-3361NP are applied to the ATRIUM 10XM and OPTIMA2 LHGR limits for this evaluation. Actual $LHGRFAC_p$ values will be determined on a plant-specific plant/cycle basis. In all conditions, the margin to the LHGR limits is determined by applying the lowest multiplier from the applicable $LHGRFAC_p$ and $LHGRFAC_f$ multipliers for the power/flow state point of interest to the steady-state LHGR limit. If there is a need to input $LHGRFAC_p$ multipliers in the core monitoring system for power levels above 100 percent of rated, the rated power $LHGRFAC_p$ multiplier can be used.

3.1.1.3.1 Conclusion

Based on the above considerations, the NRC staff has determined the licensee's LHGR limits analysis for the representative core demonstrate conformance to 10 CFR 50.46(b) acceptance criteria for a specific cycle design. Therefore, the NRC staff concludes the licensee's LHGR analysis is acceptable to support the proposed ATRIUM 10XM fuel transition at DNPS and QCNPS.

3.1.1.4 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

A maximum average planar linear heat generation rate (MAPLHGR) is applied to ensure that the fuel does not operate in a condition that would cause it to exceed the bounds of the ECCS evaluation. While the ECCS evaluation itself is performed for DNPS and QCNPS using a representative core design at beginning of life conditions, as documented in ANP-3328NP, "Quad Cities Units 1 and 2 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel," dated December 2014 (Reference 10), cycle-specific MAPLHGR analyses are performed using the initial fluid conditions from the ECCS evaluation, but with cycle-specific core Neutronics parameters. The MAPLHGR analysis assures that the core design conforms to the 10 CFR 50.46(b) acceptance criteria. The licensee included in the submittal of report ANP-3356NP, "Quad Cities Units 1 and 2 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel" (ADAMS Accession No. ML15043A493), dated December 2014 (Reference 11), for NRC staff review.

The MAPLHGR limits analysis is an element of the cycle-specific reload safety analysis. As such, the NRC staff bases its findings and conclusions with respect to ECCS evaluation on the LOCA break spectrum analysis. The MAPLHGR limits analysis captures the cycle-to-cycle variation in the predicted peak cladding temperature (PCT) and oxidation results.

3.1.1.4.1 Conclusion

Based on the above considerations, the NRC staff has determined the licensee's MAPLHGR limits analysis for the representative core demonstrate conformance to 10 CFR 50.46(b) acceptance criteria for a specific cycle design. Therefore, the NRC staff concludes the licensee's MAPLHGR analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.2 Postulated Design Basis Accidents (LOCA)

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the recirculation lines at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection system (RPS) and ECCS are provided to mitigate these accidents. The NRC staff's review covered: (1) the licensee's determination of break locations and break sizes, (2) postulated initial conditions, (3) the sequence of events, (4) the analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients, (5) calculations of peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling, (6) functional and operational characteristics of the reactor protection and ECCS systems, and (7) operator actions.

Chapter 6 of ANP-3361NP discusses postulated accidents which the NRC staff reviewed. The ECCS evaluation was reviewed in additional detail because it shows little margin to regulatory acceptance criteria as discussed below in Section 3.1.2.1. For the remainder of the design basis events, the NRC staff performed a limited-scope review to ensure that the licensee's use of AREVA fuel and analytic methods remains consistent with the DNPS and QCNPS licensing basis, and that the consequences of the analyzed events remain acceptable.

3.1.2.1 Emergency Core Cooling System (ECCS) Evaluation

The licensee's submittal proposes to implement the NRC-approved TR EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model (ADAMS Accession No. ML003772936), dated October 2000 (Reference 12). This report describes a revised evaluation model for the analysis of postulated LOCAs in jet pump BWRs using a methodology which complies with 10 CFR 50.46 and 10 CFR 50, Appendix K, criteria. The NRC staff reviewed the ECCS evaluation discussed in ANP-3328NP. Since the DNPS and QCNPS ECCS evaluation previously indicated less than 100 °F margin to the 2200 °F limit for predicted PCT as specified in the regulatory requirements in 10 CFR 50.46(b)(1), "Peak cladding temperature," and the AREVA evaluation results for ATRIUM 10XM fuel continue to indicate little margin to the acceptance criterion, the NRC staff reviewed the ECCS evaluation results in detail. The NRC staff evaluated the licensee's ECCS evaluation by:

- Comparing AREVA's results to the prior DNPS and QCNPS EPU results,
- Reviewing the general performance of the break spectrum, and
- Evaluating the phenomena associated with the limiting transient.

The LOCA calculations described in ANP-3328NP were performed in conformance with the acceptance criteria in 10 CFR 50.46 and 10 CFR 50, Appendix K, requirements. The break spectrum analyses were performed for a core composed entirely of ATRIUM 10XM fuel at beginning-of-life (BOL) conditions. Calculations assumed an initial core power of 102 percent of 2957 MWt (licensed thermal power of DNPS and QCNPS) or 3016.14 MWt. The limiting assembly in the core was assumed to be at a MAPLHGR limit of 11.7 kW/ft.

3.1.2.1.1 Conclusion

Based on the above considerations, the NRC staff has determined the licensee's analysis of a postulated jet pump LOCA and ECCS evaluations demonstrate conformance to 10 CFR 50.46(b) and 10 CFR 50, Appendix K criteria. The staff have determined the licensee has acceptably implemented the EXEM-BWR/2000 ECCS evaluation model (Reference 12), and has demonstrated that ATRIUM 10XM fuel can be used with adequate margins to the ECCS acceptance criteria set forth in 10 CFR 50.46(b). Based on the staffs evaluation that the licensees analysis supports the ATRIUM 10XM fuel design will maintain adequate margins from the PCT limits, as specified in the 10 CFR 50.46(b)(1), the NRC staff concludes the licensee's evaluations as discussed above are acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.2.2 Recirculation Line Break LOCA Analysis

Section 6.0 of ANP-3328NP, states:

> "The largest diameter recirculation system pipes are the suction line between the reactor vessel and the recirculation pump and the discharge line between the recirculation pump and the riser manifold ring. LOCA analyses are performed for breaks in both of these locations with consideration for both DEG [double-ended guillotine] and split break geometries. The break sizes considered included DEG breaks with discharge coefficients from 1.0 to 0.4 and split breaks with areas ranging between the full pipe area and 0.05 [foot-squared] ft^2. The single failures considered in the recirculation line break analyses are SF-LPCI [single failure-Low Pressure Coolant Injection], SF-DGEN [single failure-Diesel Generator], SF-HPCI [single failure-High Pressure Coolant Injection], SF-LSL [single failure-Loop Selection Logic], and SF-ADS [single failure-Automatic Depressurization System]."

The acceptance criteria contained in 10 CFR 50.46(b) requires that the fuel cladding remain in a geometry amenable to cooling, and that adequate cooling be available for the long-term removal of decay heat generated by the core. Provided that ECCS injection maintain an adequate water level to cover 2/3 core height, a stable quench can be maintained, and the top third of the core is adequately cooled with core spray.

3.1.2.2.1 Conclusion

The NRC staff determined the licensee's recirculation line break analysis is bounded by the disposition of the DNPS and QCNPS system designs to ensure the core remains 2/3 covered on a recirculation line break. The staff determined the licensee's proposed fuel transition and use of analytic models does not impact core geometry in respects to 10 CFR 50.46(b)(4) which continues to be met. Based the staff's determination that the licensee's analysis meets regulatory acceptance criteria, the NRC staff concludes the licensee's recirculation line break analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.2.3 Limiting Break Analysis Results

Section 6.1 of ANP-3328NP (ATRIUM 10XM fuel design), states:

> The analyses demonstrate that the limiting (highest PCT) recirculation line break is the 0.13 ft^2 split break in the pump discharge piping with an SF-HPCI failure and a top-peaked axial power shape when operating at 102 percent rated core power, and the PCT is 2127 °F.

The licensing basis PCT for OPTIMA2 legacy fuel is 2150 °F. The maximum local oxidation was less than 17 percent. The acceptance criteria established in 10 CFR 50.46(b) require PCT to be less than 2200 °F and maximum local oxidation to be less than 17 percent. On this basis, the NRC staff concluded that the license's analysis for a limiting line break is acceptable.

The licensee's ATRIUM 10XM analysis also included SLO with a 0.80 multiplier applied to the two-loop MAPLHGR limit resulting in an SLO MAPLHGR limit of 9.36 kW/ft. for ATRIUM 10XM fuel. The analyses were performed at BOL fuel conditions. The limiting SLO LOCA is the 0.1 ft^2 split pump discharge line break with SF-HPCI and a top-peaked axial power shape. The PCT for this case is 2047 °F.

The results provided by the licensee ensure that ECCS performance has been calculated for as required by 10 CFR 50.46(a)(1)(i). The licensee's results showed that the limiting LOCA remained bounded by 10 CFR 50.46(b) acceptance criteria. Based on these considerations, the NRC staff determined that the ECCS evaluation, its results, and the proposed implementation of the EXEM-BWR ECCS evaluation model, are acceptable and the proposed fuel design change is also acceptable.

The licensee proposes to implement an NRC-approved evaluation model, in conformance with 10 CFR 50, Appendix K. In turn, this implementation would also conform to 10 CFR 50.46(a)(1)(ii), which states:

> "Alternately, an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K, ECCS Evaluation Models."

3.1.2.3.1 Conclusion

The NRC staffs review of the proposed ECCS evaluation results included a comparison to the prior analyses of record. The staff determined that the predicted PCTs were reasonably consistent, and that differences in the break spectrum and limiting results were acceptable, given the differences between the two evaluation models. Therefore, the staff concludes the licensee's limiting break analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.3 Postulated Design Basis Accidents (DBAs) (Non-LOCA)

The licensee described the effect that the fuel and safety analysis methods transition would have on the plant's predicted performance for the remaining, non-LOCA postulated accidents within the DNPS and QCNPS licensing basis, including; control rod drop accident, fuel and equipment handling accident, and fuel loading error.

3.1.3.1 Control Rod Drop Accident (CRDA)

The NRC staff evaluated the consequences of a CRDA in the area of reactor physics. The staff's review covered the occurrences that lead to the accident, safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, the analytical model used for analyses, and the results of the analyses.

Chapter 6 of ANP-3361NP, states:

> Quad Cities Unit 2 uses an analyzed rod sequence with a bank position withdrawal sequence (BPWS) rod group definition to limit high worth control rod movements. A CRDA evaluation was performed for A sequence startups consistent with the withdrawal sequence specified by Exelon...

The NRC-approved AREVA generic CRDA methodology is described in topical report XN-NF-80-19(P)(A), Volume 1, and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, Exxon Nuclear Company," dated March 1983 (Reference 13). The NRC staff concludes the licensee's calculations demonstrate that the methodology is applicable to fuel modeled using an NRC-approved CASMO4/MICROBURN-B2 code system and, therefore, is acceptable. The NRC staff applied two acceptance criteria to the results of the CRDA: (1) the maximum deposited fuel enthalpy should be less than 280 calories per gram (cal/g) to assure core coolability, and (2) a fuel damage threshold of 170 cal/g is applied for the purposes of determining the number of rods with cladding failure for the radiological consequences. For the first scenario, the licensee confirmed that the maximum deposited fuel enthalpy was 190.6 cal/g and the maximum number of rods exceeding 170 cal/g was 546 rods which is well below the more limiting value of 850 rods as specified in the DNPS and QCNPS Updated Final Safety Analysis Report (UFSAR), Chapter 15, "Accident and Transient Analysis," Section 15.4.10.5.4.1 "Control Rod Drop Accident (CRDA)," which applies for all fuel designs assumed in the DNPS or QCNPS UFSARs. For the second scenario, since the number of failed rods is less than the number of failed rods assumed in the UFSAR, the radiological consequences of fuel rod cladding damage remains bounded by the existing UFSAR analysis. Since the licensee applied NRC-approved analytical

methods and determined that core coolability and cladding failure criteria remain satisfied, the NRC staff concluded that the proposed fuel transition is acceptable with respect to the CRDA.

3.1.3.1.1 Conclusion

The NRC staff reviewed the licensee's analysis of non-LOCA postulated DBAs for a CRDA and has determined that the licensee correctly applied NRC-approved analytical methods to determine that core coolability and cladding failure criteria remain satisfied. Based on this determination, the NRC staff concludes the licensee's CRDA analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.3.2 Fuel and Equipment Handling Accident (FEHA)

Section 6.3 of ANP-3361NP discusses the AREVA fuel handling accident analysis for the ATRIUM 10XM fuel design which determined that a postulated maximum of 162 fuel rods fail. Since this number of failed fuel rods is bounded by plant licensing basis calculations for the representative core reload analysis, the current fuel handling accident Alternate Source Term (AST) analysis remains applicable for the introduction of ATRIUM 10XM fuel at DNPS and QCNPS.

3.1.3.2.1 Conclusion

The NRC staff reviewed the licensee's analysis of non-LOCA postulated DBAs for a FEHA and determined the licensee's AST analysis remains bounded by previous plant licensing bases calculations. Based on this evaluation, the NRC staff concludes the licensee's FEHA analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.3.3 Fuel Loading Error (FLE)

Within the DNPS and QCNPS licensing basis, the fuel loading error is characterized as an infrequent event. There are two types of fuel loading errors possible in a BWR: (1) the mislocation of a fuel assembly in a core position prescribed to be loaded with another fuel assembly, and (2) the misorientation of a fuel assembly with respect to the control blade. The NRC acceptance criteria are that the offsite dose consequences due to the event shall not exceed a small fraction of the 10 CFR 50.67, "Accident source term," limits.

As stated in the ANP-3361NP, the licensee performed both cycle-specific fuel assembly mislocation and misorientation error analysis for the representative core design covering both OPTIMA2 and ATRIUM 10XM fuel designs. The analyses included the evaluation of the impact of a mislocated or a misoriented fuel assembly against potential fuel rod failure mechanisms due to increased LHGR and reduced critical power ratio (CPR), or a 90 degree or 180 degree misorientation (i.e., during depleted cycle or no operator interaction), respectively. In both fuel loading error scenarios, the 10 CFR 50.67 offsite dose criteria was conservatively satisfied.
3.1.3.3.1 Conclusion

The NRC staff has determined the licensee's analysis of non-LOCA postulated DBAs for fuel loading errors satisfy the proposed fuel and analytical methods for the representative core design. Therefore, the NRC staff concludes the licensee's FLE analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.4 Special Analyses

Section 7.0 of ANP-3361NP discusses the American Society of Mechanical Engineers (ASME) overpressurization analysis and the ATWS event which includes an evaluation of the standby liquid control system (SLCS).

3.1.4.1 ASME Overpressurization Protection

A reactor overpressure condition could result from a load rejection or similar event in the steam and power conversion system, a spurious main steam isolation valve closure, or a malfunction in the control systems causing feedwater supply or recirculation flow to exceed steam demand.

The overpressure protection system and the RPS mitigate the adverse effects of such events. The NRC staff evaluated the effect of the fuel and methods transition on the ASME overpressure protection, as discussed in Section 7.1 of ANP-3361NP. The NRC staff verified that the licensee performed analyses using the AREVA plant transient simulator code COTRANSA2. COTRANSA2 is a BWR system transient analysis code with models representing the reactor core, reactor vessel, steam lines, recirculation loops, and control systems. It is used to evaluate key reactor system parameters during core-wide BWR transient events. These parameters, such as power, flow, pressure, and temperature, are provided as boundary conditions to the hot channel analyses for ΔCPR determination. The code has been generically approved by the NRC to analyze system responses to fast transients in BWRs. The licensee used this code for transient analyses for 102 percent reactor power and both 95.3 percent and 108 percent core flow at the highest representative core design exposure where rated power operation can be attained. The following events were analyzed:

- Main steam isolation valve (MSIV) closure,
- Turbine control valve (TCV) closure,
- Turbine stop valve (TSV) closure, and
- Feedwater control failure (FWCF) event with turbine bypass valve out of service (TBVOOS).

The limiting over pressurization event was the FWCF with TBVOOS. The base FWCF AOO event is discussed in Section 5.1.3 of ANP-3361NP. The base FWCF event credits direct scram on TSV position and TBV available for pressure relief, whereas the FWCF over-pressurization event assumes scram on high neutron flux and no TBV pressure relief capacity. Sensitivity analyses showed that crediting the ATWS-RPT resulted in higher peak vessel and peak dome pressures. The following modeling assumptions were made in the analysis:

- The most critical active component (direct scram on valve position) was assumed to fail. However, scram on high neutron flux and high dome pressure is available,
- The plant configuration analyzed assumed that one of the lowest setpoint safety or safety/relief valves is inoperable. No credit was taken for relief valve operation,
- The turbine bypass valves are assumed out of service,
- TSs scram speed insertion times were used,
- A nominal ATWS-RPT set point of was used, and,
- The initial dome pressure was set at the maximum allowed by the TSs, 1019.7 psia (1005 psig).

Results of the limiting over pressurization analyses presented in ANP-3361NP show the response of various reactor plant parameters during the limiting FWCF with TBVOOS event as it applies to the representative core. The maximum pressure of 1362 psig occurs in the lower plenum. The maximum dome pressure for the same event is 1342 psig. These peak pressure results have been adjusted to address NRC concerns associated with the void-quality correlation and Doppler effects. The effects of exposure-dependent thermal conductivity degradation were included in the analysis.

The results demonstrate that the maximum vessel pressure limit of 1375 psig (110 percent of design limit) and dome pressure limit of 1345 psig are not exceeded.

3.1.4.1.1 Conclusion

The NRC staff evaluated the licensee's analysis discussed above and determined the licensee properly applied NRC-approved COTRANSA2 computer code to analyze the ASME overpressure protection for the representative core design. The licensee included additional conservative adders to the final pressure result, thereby accounting for non-conservative models that had been identified subsequent to the code's approval (increased pressure margin). The staff determined since the licensee's methodology used NRC-approved code and added margin, the integrity of the DNPS and QCNPS reactor coolant system boundaries will be adequately protected when using the ATRIUM 10XM fuel design. Therefore, the NRC staff concludes the licensee's ASME overpressure protection analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.4.2 Anticipated Transient without Scram (ATWS)

The NRC staff evaluated the effect of the fuel and methods transition from an ATWS event as discussed in Section 7.2 of ANP-3361NP. The NRC staff verified the licensee's analysis of the ATWS overpressurization at 100 percent reactor power at 95.3 percent and 108 percent flow (the maximum extended operating domain) is acceptable. The MSIV closure and pressure regulator failure open (PRFO) events were also evaluated. In each event, the RPS was assumed to fail, and the plant shutdown was accomplished through SLCS actuation. A more immediate power reduction occurs due to an automatic recirculation pump trip. Table 7.2 of ANP-3361NP exhibits the ATWS overpressurization parameters during the limiting PRFO event, the event which resulted in the maximum vessel pressure. The maximum lower plenum pressure was 1489 psig and the maximum dome pressure was 1473 psig. The results demonstrate that the ATWS maximum vessel pressure limit of 1500 psig acceptance criterion was not exceeded. In the event that the control rod scram function becomes incapable of rendering the core in a shutdown state, the SLCS is required to be capable of bringing the reactor from full power to a cold shutdown condition at any time in the core life. The licensee has performed an analysis that demonstrates that the SLCS meets the required shutdown capability for the representative core design.

3.1.4.2.1 Conclusion

The NRC staff determined the licensee's ATWS analysis properly applied acceptable ATWS analytic methods to support the proposed fuel transition. In a letter dated January 28, 2016 (Reference 3), the licensee stated that the analysis used the NRC-approved COTRANSA2 plant simulator code. Since the licensee explicitly addressed the limiting ATWS events by analyzing them using the approved code, and since the results of the analysis were less than the 1500 psi acceptance criterion, the NRC staff determined that the licensee's disposition for ATWS mitigation is acceptable. The licensee demonstrated that it will continue to meet the requirements of 10 CFR 50.62, "Requirements for reduction of risk from ATWS events for light-water-cooled nuclear power plants," and the analysis acceptance criteria following implementation of the proposed fuel transition. Therefore, the NRC staff concludes the licensee's ATWS mitigation analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.5 Anticipated Operational Occurrences (AOOs) and Stability Analysis

3.1.5.1 Anticipated Operational Occurrences (AOOs)

Section 5.0 of ANP-3361NP discusses plant responses to the limiting AOOs analyzed for each reload cycle. To support the proposed fuel and safety analysis methods transition, the licensee provided the results of its reload transient analysis which covers the projected operating conditions within the licensed power-to-flow map, equipment out of service options, and SCRAM speed options (i.e., Nominal Scram Speed, Intermediate Scram Speed, and TS Scram Speed insertion times). For the initial application of AREVA fuel and methodology for DNPS and QCNPS, the reload analysis consisted of simulation of transient events to cover the rated and off-rated operating conditions. The results were used to determine the OLMCPR limits for ATRIUM 10XM and co-resident OPTIMA2 fuel using the QCNPS, Unit 2, Cycle 24, representative core design.

The thermal limits are determined following the NRC-approved TR XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, Exxon Nuclear Company, dated January 1987 (Reference 14). The methodology employs several NRC-approved codes, including; system transient simulation code (COTRANSA2), thermal-hydraulic codes used for steady-state and transient analysis (XCOBRA and XCOBRA-T), and a Neutronics code (CASMO-4/MICROBURN-B2) (Reference 46). The licensee's methodology along with the use of these computer codes, is specifically approved by the NRC for BWR transient analysis and thermal limits assessment. As such, the NRC staff determined that they are acceptable for application to DNPS and QCNPS in its MELLLA operating domain.

In letter dated January 20, 2016 (Reference 2), the licensee explained two errors identified (post-submittal) in the computer modeling code MICROBURN-B2 used in the licensee's fuel transition analysis. The MICROBURN-B2 errors apply to both a potential hydraulic non-convergence condition and a void quality correlation. The potentially affected analyses includes the following:

- Stability Option III Delta Over Initial MCPR Versus Oscillation Magnitude (DIVOM) calculations,
- MICROBURN-B2/STAIF calculations for Backup Stability Protection (BSP),

- OPRM Setpoint Analyses,

- Flow Run-up Analyses including both flow-dependent $LHGRFAC_f$ and flow-dependent operating limits ($MCPR_f$), and

- Transients, infrequent events, or design basis accidents where MICROBURN-B2 is used to set the initial core conditions for low flows.

The licensee's evaluation of these errors determined that there were non-limiting analyses adversely affected by the error corrections. In contrast, the limiting analysis improved as a result of the error corrections. Additionally, the licensee has determined that only the ANP-3361NP, "Reload Safety Analysis Report," was moderately impacted by these errors. The corrections to the MICROBURN-B2 code will be factored into cycle-specific licensing calculations for the first reload of ATRIUM 10XM fuel the DNPS and QCNPS facilities. Based on the cycle-specific safety analysis, the OLMCPR is established as discussed above in Section 3.1.1.2. The TS requires that MCPR and LHGR limits only need to be monitored at power levels $\geq$ 25 percent of rated thermal power.

All pressurization transients assumed that the single most beneficial relief, safety or safety/relief valve was out of service. For DNPS and QCNPS, this was identified as a Target Rock safety valve, a valve with the highest flow capacity resulting in the greatest pressure relief. The licensee analyzed several AOO events to determine the OLMCPR, which included: load rejection no bypass, turbine trip no bypass, feedwater controller failure to maximum demand, an inadvertent actuation of the high pressure coolant injection (HPCI) system, loss of stator cooling, loss of feedwater heating, and control rod withdrawal error. The licensee determined that there were no new potentially limiting events identified as a result of the introduction of the ATRIUM 10XM fuel design. The system response to the various transients and accidents will have no significant impacts due to the change in fuel design from OPTIMA2 to ATRIUM 10XM at DNPS and QCNPS.

3.1.5.1.1 Conclusion

The staff evaluated the MICROBURN-B2 code modeling errors, identified by the licensee, and determined the modeling discrepancies were minor and only occurred at relatively low flow conditions bounded by operating limits provided in the proposed amendment. The staff is satisfied the licensee adequately dispositioned the errors which will be accounted for in forthcoming cycle-specific calculations and that the licensee also documented discrepancies in the AREVA correction action program. Also, the staff determined the results indicated adequate thermal margins for the analyzed SLMCPR and supported OLMCPR values. The NRC staff has determined that the licensee properly analyzed the limiting AOOs on a cycle-specific basis to determine the OLMCPR, and because the THERMEX results indicated reasonable agreement with the results of the prior analyses, the staff concluded that the licensee's application of the THERMEX methodology to be acceptable. Since the ATRIUM 10XM and OPTIMA2 (co-resident fuel) indicated similar thermal margin performance, the NRC staff concludes the licensee's AOO analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.1.5.2 Stability

Previous staff concerns associated with operating experience regarding uncontrolled power oscillations in BWRs[5] has been addressed by the licensee. Section 4.3 of ANP-3361(NP)(A) discusses the licensee's implementation[6] of the Boiling Water Reactor Owners Group (BWROG) Long Term Stability Solution (LTS) Option III (Oscillation Power Range Monitor-OPRM) licensing methodology. This is reinforced as referenced in Section 4.3.2.3, "Stability," of the DNPS and QCNPS UFSARs. This methodology relies on an OPRM to detect stability decay ratios and trip the reactor if destabilizing power oscillations are detected. The licensee evaluated two postulated conditions: (1) steady-state operation at 45 percent core flow, and (2) a transient associated with a two recirculation pump trip from the full-power operation state point.

The licensee used the RAMONA5-FA computer code in accordance with an NRC-approved methodology described in TR BAW-10255PA, Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," dated May 2008 (Reference 15), to calculate the relative change in CPR as a function of the calculated hot channel oscillation magnitude. A stability-based OLMCPR is calculated using the most limiting of: (1) the RAMONA5-FA-calculated change in relative ΔCPR for a given oscillation magnitude, or (2) a generic value calculated in accordance with the General Electric Option III methodology. The licensee's calculations determined that the generic value was limiting for the QCNPS, Unit 2, Cycle 24, representative core design. Both DNPS and QCNPS implement BSP when the OPRM system is inoperable.

The NRC-approved topical report EMF-CC-074(P)(A), Volume 4, Revision 0, "BWR Stability Analysis: Assessment of STAIF [Stability Analysis in the Frequency Domain] with Input from MICROBURN-B2," Siemens Power Corporation, dated August 2000 (Reference 16), includes specific requirements for operator action as well as restrictions on operation in certain regions of the power/flow map. The STAIF methodology uses a frequency domain code that provides best-estimate calculations to determine exclusion regions for BWR stability. The regions are based on criteria related to decay ratio, or the measure of growth (or decay) of power oscillations. The decay ratio is calculated based on the neutronic feedback and the thermal-hydraulic conditions at any given region on the power-to-flow map. The results of the STAIF analysis are used to define where immediate scram is required (Region I) and a less severe scenario where manual intervention is required (Region II) to exit the power-to-flow conditions in that region. These regions occur at the high-power, low-flow extent of the MELLLA operating domain.

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5 NRC Bulletin 88-07, "Supplement 1: Power Oscillations in Boiling Water Reactors" (ADAMS Accession No. ML031220139)

6 Applies to the Quad Cities Unit 2 Cycle 24 representative core design

3.1.5.2.1 Conclusion

The NRC staff has determined the licensee's properly utilized the AREVA suite of stability analysis methods in a manner consistent with NRC approval. The staff recognized the licensee properly verified (in their analysis) that a range of OPRM setpoints were available that supported a variety of assumed OLMCPRs, such that the stability solution would provide acceptable protection at an OLMCPR that is supported by the AOO analyses. The NRC staff previously reviewed the licensee's stability solution within the MELLLA operating domain during the review of the DNPS and QCNPS EPU and found the results to be acceptable. Since the licensee will operate the ATRIUM 10XM fuel within the same MELLLA operating domain, the previous NRC EPU findings related to operator actions and general LTS Option III methodology, remain applicable for the proposed DNPS and QCNPS fuel transition when operating with the ATRIUM 10 XM fuel design. The licensee's implementation of the LTS Option III licensing methodology by addition of a detection and suppression system in their design basis, mitigates the effects of power oscillations in the core, as discussed above. This satisfies the requirements to meet Design Criterion 12, "Suppression of Reactor Power Oscillations," as specified in the DNPS and QCNPS UFSARs. Therefore, based on these considerations, the NRC staff concludes the licensee's Stability analysis is acceptable to support the proposed transition of fuel at DNPS and QCNPS.

3.1.6 Conclusions Regarding Accident and Transient Analysis

Based on the considerations discussed in the preceding sections the staff's review supported the following conclusions:

- The overpressure and ATWS analyses show that DNPS and QCNPS can use ATRIUM 10XM with adequate overpressure protection to protect the integrity of the reactor coolant pressure boundary,

- The licensee has acceptably implemented COTRANSA2 for analysis of the overpressure events,

- The licensee has acceptably implemented the EXEM-BWR/2000 ECCS evaluation model, and has demonstrated that ATRIUM 10XM fuel can be used with adequate margins to the ECCS acceptance criteria set forth in 10 CFR 50.46(b),

- Remaining design basis accidents are unaffected by the proposed fuel transition, since the licensee's analyses indicated that radiological consequences would remain bounded within those previously established in the DNPS and QCNPS licensing basis,

- The licensee's use of the THERMEX methodology, along with its current constituent computer codes, shows that the ATRIUM 10XM fuel can perform with similar analytic margins to co-resident OPTIMA2 fuel, and with similar analytic margins as those previously demonstrated in the current vendor's safety analysis,

- The limiting thermal margin events at DNPS and QCNPS will continue to be analyzed on a cycle-specific basis to determine the OLMCPR,

- The licensee will continue to use the GE LTS Option III stability solution, and has demonstrated using RAMONA5-FA that the generic thermal margin protection setpoints remain adequate with ATRIUM 10XM fuel, but this fact will be conformed on a cycle-specific basis, and more conservative plant-specific setpoints will be adopted if necessary,

- BSP setpoints have been assessed and will be confirmed using NRC-approved STAIF codes. However, the manual operator actions required to provide adequate BSP remain unchanged and unaffected by the change in fuel design.

- The licensee has adequately exhibited in their analysis for ATRIUM 10XM that fuel design limits are not exceeded during any conditions of normal operation, including the effects of AOOs. This satisfies the requirements to meet the reactor design criteria at DNPS (Design Criterion 10) and QCNPS (Criterion 6), and

- The licensee's implementation of methodologies factored into the use of ATRIUM 10XM fuel by addition of detection and suppression systems in their design basis mitigates the effects of power oscillations in the core. This satisfies the requirements to meet the "Suppression of Reactor Power Oscillations." DNPS (Design Criterion 12) and QCNPS (Criterion 7).

In summary, the NRC staff has determined that the proposed transition to AREVA fuel and safety analysis methods at DNPS and QCNPS is acceptable.

3.2 ATRIUM 10XM Fuel Design, Mixed Core, and Nuclear Performance

3.2.1 ATRIUM 10XM Fuel Rod Mechanical Evaluation

ANP-3305NP, Revision 1, "Mechanical Design Report for Quad Cities and Dresden ATRIUM 10XM Fuel Assemblies," dated August 2015 (Reference 17), provides an overview of the mechanical design of the ATRIUM 10XM fuel design to be used in DNPS, Units 2 and 3, and QCNPS, Units 1 and 2. The fuel design is comprised of a 10x10 array of fuel rods with a square internal water channel that displaces a 3x3 array of rods, with **[[**
]] The active length of the PLFR is **[[**
]] The use of the PLFRs is expected to improve the fuel utilization in the high void upper region of the bundle, enhance the shutdown margin, improve stability, and pressure drop performance. ATRIUM 10XM components included in this mechanical evaluation include the following:

3.2.1.1 Fuel Assembly

The ATRIUM 10XM fuel assembly consists of a lower tie plate (LTP) and an upper tie plate (UTP), 91 fuel rods, **[[** **]]** spacer grids, a central water channel with **[[** **]]**, and miscellaneous assembly hardware. **[[** **]]**

3.2.1.2 Spacer Grid

The spacer grids are made of a **[[** **]]** version of the ULTRAFLOW design that consists of **[[** **]]**

3.2.1.3 Water Channel

[[**]]**

3.2.1.4 Lower and Upper Tie Plates (LTP and UTP)

The LTP and UTP are made of **[[** **]]**

3.2.1.5 Fuel Rods

The fuel rods of ATRIUM XM fuel design are made with **[[**

]]

3.2.1.6 Fuel Channel

The fuel channel is a square duct with rounded corners and is open at both ends and encloses the sides of each fuel assembly. Its main purpose is to provide a flow boundary between the active coolant flow and the core bypass flow. [[

]]

Table 2-1 and Table 2-3 of ANP-3305NP lists the values and descriptions for fuel assemblies and components, and fuel channels and fasteners, respectively.

3.2.2 Fuel Design Evaluation

This fuel design evaluation contains only fuel structural analyses where the fuel rod evaluation is documented in ANP-3324NP, Revision 1, "ATRIUM 10XM Fuel Rod Thermal-Mechanical Design for Quad Cities, Unit 2, Cycle 24, Representative Fuel Cycle Design," dated August 2015 (Reference 18), and is discussed in Section 3.2.6 of this safety evaluation. Section 3.0 of ANP-3305NP summarizes the mechanical methodology and the objectives of the fuel design and states: (1) the fuel assembly shall not fail as a result of normal operation and AOOs, (2) fuel assembly damage shall never prevent control rod insertion when required, (3) the number of fuel rod failures shall be conservatively estimated for postulated accidents, (4) fuel coolability shall always maintained, (5) the mechanical design of the fuel assemblies shall be compatible with co-resident fuel and the reactor core internals, and (6) fuel assemblies shall be designed to withstand the loads from handling and shipping. The first four objectives are addressed in Section 4.2 of the SRP, the latter two are to assure the structural integrity of the fuel and the compatibility with the existing reload fuel (co-resident fuel).

3.2.2.1 Stress, Strain, and Loading Limits on Assembly Components

The ASME Boiler and Pressure Vessel Code (B&PV), Section III, Division 1, ASME was used as guidance in establishing acceptable stress, deformation, and load limits for standard fuel assembly components. These limits are applied to the mechanical design and evaluation of the UTP, LTP, spacer grids, springs, and load chain components, as necessary. The fuel assembly structural component criteria under faulted conditions are based on Appendix F of the ASME

B&PV Code, Section III, with some criteria derived from component tests. Outside of faulted conditions, most structural components are under the most limiting loading conditions during fuel handling activities. Although normal operation and AOO loads are often not limiting for structural components, **[[**

]] The fuel assembly

]] Table 3-2 of ANP-3305NP, provides information regarding the description, criteria, and results for the ATRIUM 10XM fuel channel during several conditions, including stresses related to, pressure differential, fatigue, oxidation and hydriding, long-term deformation, load limits, channel bending, and gusset strength.

In a response to additional information requested by the NRC staff, EGC/AREVA stated that to evaluate stresses under normal operating conditions a **[[**

]] Comparison of the maximum normal operation

]] for Quad Cities and Dresden analyses against the limit has ensured adequate margin.

Stresses under AOO and accident conditions were evaluated using the **[[**

]]

3.2.2.1.1 Conclusion

The NRC staff has completed its review of the mechanical design of the various components comprised within the ATRIUM 10XM fuel design. The evaluation and comparison of the results in respect to the analysis used by the licensee to evaluate fuel design stresses and load limits, demonstrates that the fuel assembly structural component criteria is satisfied. Therefore, the NRC staff concludes the methodologies and analysis performed by the licensee includes a sufficient design margin under normal operating and AOO conditions. Furthermore, the NRC staff concludes that the stress, strain, and loading limits of the ATRIUM 10XM fuel design is acceptable to support the licensee's proposed fuel transition at DNPS and QCNPS.

3.2.2.2 Fatigue and Fretting Wear

[[

]] Though there is no specific

]]

3.2.2.2.1 Conclusion

The NRC staff has completed its review of the licensee's analysis regarding fatigue and fretting wear. The staff finds that the licensee's evaluation demonstrates that the wear at the spacer spring/fuel rod interface at relaxed to end of life (EOL) conditions is not significant and provides a reasonable assurance that no significant fretting of the fuel rod will occur at the higher exposure levels. Therefore, the NRC staff concludes the licensee's evaluation regarding fatigue and fretting wear is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.2.2.2.2 Rod Bow

Differential expansion between the fuel rods and cage structure, and lateral thermal and flux gradients can lead to lateral creep bow of the rods in the spans between spacer grids. This lateral creep bow alters the pitch between the rods and may affect the peaking and local heat transfer. The criterion for fuel rod bowing is **[[**

]]

In a response to a request from NRC staff for additional information, the licensee stated that the current AREVA methodology for evaluating the impact of rod bow on thermal margins is composed of two steps: (1) **[[**

]], and (2) **[[**
]]

The licensee's response to a NRC staff request for information indicated that **[[**

]]

3.2.2.2.2.1 Conclusion

The NRC staff has determined that the licensee adequately evaluated AREVA's methodology to meet the fuel rod bow criterion in respects thermal margins. Therefore, the NRC staff concludes the licensee's rod bow analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.2.2.2.3 Axial Irradiation Growth

Section 3.3.6 of ANP-3305NP, discusses fuel assembly characteristics and the calculations considered for the ATRIUM 10XM fuel design. Components such as the fuel channel must maintain clearances and engagements throughout their design life. There are three specific growth calculations for the XM fuel design: (1) minimum fuel rod clearance between LTP and UTP, (2) minimum engagement of the fuel channel with the LTP seal spring, and (3) external interfaces. ANP-3305NP states: rod growth, assembly growth, and fuel channel growth are calculated using correlations derived from post-irradiation data. The evaluation of initial engagements and clearances accounts for the combination of fabrication tolerances on individual component dimensions. **[[**

]] Assembly growth is dictated by the water channel growth. The upper and lower **[[**
]], as appropriate, are used to obtain EOL growth values.

3.2.2.2.3.1 Conclusion

The NRC staff determined the licensee properly evaluated NRC-approved methodologies to validate that fuel assembly axial irradiation growth criteria are being met and appropriate calculations are being used, as discussed above. Therefore, the NRC staff concludes the licensee's axial irradiation growth analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.2.2.2.4 Assembly Liftoff

NRC-approved TR ANF-89-98PA, Revision 1, Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," (May 1995) (Reference 19), states the fuel assembly shall not levitate under normal operating, AOO, or faulted conditions. Under postulated accident conditions, the fuel shall not become disengaged from the fuel support. These criteria assure control blade insertion is not impaired. For normal operating conditions, the calculated net axial force acting on the assembly due to addition of the loads from gravity, hydraulic resistance from coolant flow, difference in fluid flow entrance and exit momentum, and buoyancy will be in the downward direction, indicating no assembly liftoff. The net force calculation is performed at maximum hot channel conditions because the greater two-phase flow losses produce a higher uplift force. For faulted conditions, the XM design was evaluated for full lift at **[[**

]] Mixed core conditions for assembly lift-off are considered on a cycle-specific basis, as determined by the plant and other fuel types. In response to previous requests from NRC staff for additional information, the licensee provided details on typical calculations demonstrating that the margins to fuel assembly lift-off under normal operating and faulted conditions are adequate. The licensee stated that liftoff calculations were performed for the ATRIUM 10XM fuel design using previously approved criteria for normal operation and AOOs. The submerged fuel assembly weight, including the channel, must be greater than the hydraulic loads. For accident conditions, the normal hydraulic loads plus accident loads shall not cause the assembly to become disengaged from the fuel support; ensuring that control blade insertion

is not impaired. The fuel assembly is **[[**

]] The results from these calculations indicate that the net force on the fuel assembly is downward and prevents the assembly from liftoff during normal operating conditions, AOOs, and accident conditions.

3.2.2.2.4.1 Conclusion

The NRC staff determined that the licensee properly evaluated approved methodologies and met NRC-approved fuel assembly liftoff criteria, as referenced above. Therefore, the NRC staff conclude the licensee's assembly liftoff analysis and use of approved methodologies is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.2.3 Structural Deformation Evaluation

Evaluations for structural deformation or stresses from postulated accidents are limited according to requirements contained in the ASME B&PV Code, Section III, Division 1, Appendix F, and SRP, NUREG-0800, Section 4.2, Appendix A. Dynamic characteristics of the fuel assembly, such as stiffness, natural frequencies, and damping values for the assemblies and dynamic characteristics of the spacer grids derived from the tests are used as inputs for analytical models of the fuel assembly and fuel channel. Fuel assemblies are tested with and without a fuel channel. In addition, the analytical models are compared to the test results to ensure an accurate characterization of the fuel. Testing and analyses for the ATRIUM 10XM design are similar to other BWR fuel designs that have the same channel configuration and weight. Evaluations performed for the fuel under accident loadings include mechanical fracturing of the fuel rod cladding, assembly structural integrity, and fuel assembly liftoff.
The ATRIUM 10XM fuel design was analyzed under a limiting top guide and core support time history supplied by Exelon for QCNPS, 1 and 2, and DNPS, Units 2 and 3. The time histories were input to the non-linear dynamic two-assembly model as described in NRC-approved TR EMF-93-177 (P)(A), Revision 1, dated August 2005 (Reference 20). Tables 3-1 and 3-2 of ANP-3305, Revision 1, list the minimum design margins for the assembly structural components and fuel channel.

3.2.3.1 Conclusion

The NRC staff has reviewed the details of the SRP and ASME requirements as well as the licensee's analyses and finds that the minimum design margins for the structural integrity of the fuel assemblies and associated components have been maintained during normal and accident operational conditions. Therefore, the NRC staff concludes the licensee's evaluation of the structural and fuel channel design of ATRIUM 10XM fuel is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.2.4 ATRIUM 10XM Fuel Rod Thermal-Mechanical Design Evaluation

This section presents the results of the NRC staff's review of fuel rod thermal-mechanical analyses for ATRIUM 10XM fuel. The analyses were performed using approved codes and methodologies as discussed in TR BAW-10247PA, Revision 0, and TR ANF-89-98PA, Revision 1, and Supplement 1. The fuel cladding external oxidation limit was reduced according when the RODEX4 code was first implemented. The RODEX4 fuel rod thermal-mechanical analysis code and methodology are used to analyze the fuel rod for fuel centerline temperature, cladding strain, rod internal pressure, cladding collapse, cladding fatigue, and external oxidation.

3.2.4.1 Fuel Rod Design

ATRIUM 10XM fuel rod design configuration is very similar to the past fuel designs of ATRIUM-9 and ATRIUM-10. **[[**

]]

Table 3-1 of ANP-3305NP, lists key fuel rod design parameters. **[[**

]] A summary of fuel rod thermal mechanical design criteria are summarized by the following:

3.2.4.2 Internal Hydriding

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and formation of hydride platelets. This is prevented by moisture control during fuel fabrication which reduces the potential for hydrogen absorption on the inside of the cladding.

3.2.4.3 Cladding Collapse

Creep collapse of the cladding and subsequent potential for fuel failure is avoided in the design by limiting the gap formation due to fuel densification subsequent to pellet-clad contact. Creep collapse of the clad is evaluated using NRC-approved RODEX4 methodology. The RODEX4 code uses a statistical method and gives best-estimate results for nominal inputs. The maximum gap formation is calculated such that the expected fraction of fuel rods below the maximum value is 99.9 percent with a 95 percent confidence level.

3.2.4.4 Overheating of Fuel Pellets

To avoid fuel failure from overheating of the fuel pellet, the centerline temperature of the fuel pellets must remain below the melting point during normal operation and AOOs. The melting point is adjusted for gadolinia content in the fuel. AREVA establishes a LHGR to protect against fuel centerline melting during steady-state operation and during AOOs. Fuel centerline temperature is evaluated using the RODEX4 code, as discussed in NRC-approved TR BAW-10247PA, Revision 0, for both normal operating conditions and AOOs. RODEX4 fuel model considers the fuel column divided in to axial and radial regions, gap region, cladding, gas plena and the fill gas and released fission gases. The operational conditions are controlled by the **[[**

]]

In response to the NRC staff's request for additional information, the licensee explained how radial depression of the thermal neutron flux is accounted for in defining the local volumetric heat generation rate. Specifically, the licensee states:

> "The radial depression of the thermal flux is one component of the radial power profile model of RODEX4. [[
>
>]] The volumetric thermal power at any location in the fuel rod is the product of the value of the radial power profile factor at that radius, the input linear power at the axial location and the volume of [[
>]]

Mechanical processes include [[

]]

Fuel rod power histories are generated based typically [[
]] In a response to NRC staff request for information, the licensee explained that neutronic fuel assembly grouping or types are identified by enrichment and gadolinia distribution within the fuel rods that comprise the assembly. For a given type, the mechanical fuel assembly designs are identical with respect to number of fuel rods; number, location and length of part-length fuel rods; plenum volumes for each fuel rod; spacer grid design, and water channel design. Two sets of histories are created, one for an equilibrium core design and one for the upcoming cycle under evaluation. Once the fuel has completed the cycle, the histories are updated to account for actual core follow data and the next cycle is reanalyzed taking credit for reduced operational uncertainties for the past cycle(s) of operation.

RODEX4 is a best estimate code and, therefore, the uncertainties are taken in to account by a statistical method. The overall reactor power uncertainty is the uncertainty associated with the core thermal energy balance and is supplied by the plant operator. The power distribution measurement uncertainty is the uncertainty associated with the LHGR estimates provided by

[[ ]] Operational flexibility uncertainty is the uncertainty attributable to the difference between [[ ]] The operational LHGR uncertainty is modified to [[ ]]

[[ ]] Model uncertainties that are included in the analysis are [[ ]]

3.2.4.5 Stress and Strain Limits

Cladding strain caused by transient-induced deformations of the cladding is calculated using the NRC-approved RODEX4 code and methodology. The calculated strain is reported to be less than 1 percent.

Cladding stresses are calculated using solid mechanics elasticity solutions and finite element methods. Stresses are calculated for the primary and secondary loadings. [[ ]] The stresses are found to be less than the design limits prescribed by ASME B&PV Code, Section III.

3.2.4.6 Fuel Densification and Swelling

Fuel densification and swelling are limited by the design criteria for fuel temperature, cladding strain, cladding collapse, and rod internal pressure criteria. The effect of these phenomena are included in the RODEX4 code.

3.2.4.7 Fatigue

[[ ]] A maximum value that encompasses 99.9 percent of the fuel rods with a 95-percent confidence is determined. The maximum cumulative usage factor for the cladding remains below the design criterion.

3.2.4.8 Oxidation, Hydriding, and Crud Buildup

The RODEX4 methodology for calculation of cladding external oxidation includes an enhancement factor that is derived from poolside measurement data to obtain a fit of the expected oxide thickness. An uncertainty on the model enhancement factor also is determined from the data. The RODEX4 analysis implicitly includes the thermal effect from normal levels of crud. Specific analyses are performed for higher than normal crud deposition. An abnormal level of crud is defined by a formation that increases the calculated fuel average temperature by 25 °C [degrees Centigrade (approx. 77 °F] above the design basis calculation. The corrosion model also takes into consideration the effect of the higher thermal resistance from the crud on the corrosion rate.

A safety evaluation report (SER) restriction imposed on RODEX4 required that the calculations account for an expected, design basis crud thickness and it may be based on plant-specific history. As part of information provide to the NRC upon approval of the RODEX4 topical report, it was stated that the existing corrosion model includes a design basis level of crud.

During the first reload application of RODEX4, the initial approved limit of corrosion was challenged by NRC staff due to a concern about the effect of spallation on the cladding integrity. To avoid the issue of spallation, the limit was reduced to **[[ ]]**. The **[[ ]]** limit was established from a review of historical liftoff measurement data on AREVA BWR fuel. This new limit was established, in part, as a means **[[ ]]**. The NRC staff accepted the new fuel rod oxide limit, and thereby finds **[[ ]]** continued acceptable fuel performance.

For the representative core design used to support the DNPS and QCNPS ATRIUM 10XM fuel transitions, the licensee reports the current measurements indicate normal low crud levels. In order to address potential change in coolant chemistry conditions, the input parameters have been conservatively selected. It has been shown that the values selected for this plant are the results from these conservative calculations. The NRC staff has reviewed the crud calculations submitted by the licensee and determined crud levels to be at acceptable levels as to not reduce fuel reliability at DNPS and QCNPS.

3.2.4.9 Rod Internal Pressure

Fuel rod internal pressure is calculated using the RODEX4 code and methodology. The maximum rod pressure is calculated under steady-state conditions and transients. Rod internal pressure is limited to **[[ ]]** above the rated system pressure.

3.2.4.10 Conclusion

The staff finds that the licensee's application using NRC-approved code and methodologies for this analysis supports the proposed fuel transition. Furthermore, the staff finds that the fuel design criteria, as set forth by the applicable regulations and Section 4.2 of the SRP, have been satisfied by the licensee for safe operation of ATRIUM 10XM fuel at DNPS and QCNPS.

Therefore, the NRC staff concludes that the licensee's proposed fuel transition at DNPS and QCNPS is acceptable with respect to the thermal-mechanical design of ATRIUM 10XM fuel.

3.2.5 Thermal-Hydraulic Design of ATRIUM 10XM Fuel Assemblies

This section describes the results of the NRC staff's review regarding the representative core design for the thermal-hydraulic analyses to demonstrate the hydraulic compatibility of the ATRIUM 10XM fuel with the Westinghouse OPTIMA2 (co-resident) fuel design. This evaluation extends to the hydraulic characterization of the ATRIUM 10XM and OPTIMA2 fuel design for the proposed fuel transition. The OPTIMA2 fuel assembly description is as follows:

The OPTIMA2 fuel assembly description, fuel channel, fuel design criteria, fuel assembly components, methodology for evaluation of fuel rods, technical data, and operating experience are provided in NRC-approved TR WCAP-15492PA, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENP-387," Westinghouse, dated March 2006 (Reference 21) and Section 4 of the DNPS and QCNPS UFSARs. The OPTIMA2 fuel assembly consists of three basic components, the fuel bundle, the fuel channel, and the handle. The fuel bundle consists of 96 fuel rods, arranged in four 5x5 sub-bundles. The sub-bundles are separated by a cruciform internal structure (water cross) in the channel. The water cross has a square central canal and smaller water channels in each of the four wings for non-boiling water during operation. The sub-bundles are inserted in the channel from the top and are **[[**

]] In each sub-bundle, **[[**

]]. Each sub-bundle is assembled as a separate unit with its own top and bottom tie plates and is held together by two tie rods.

Thermal-hydraulic analyses are performed to verify that design criteria are satisfied and to establish that the thermal operating limits are within acceptable margins of safety during normal reactor operation and AOOs. NRC approved design criteria for BWR fuels are discussed in TR ANF-89-98(P)(A), Revision 1, Supplement 1.

Though the analysis is performed on a generic basis, many of the analyses supporting these thermal-hydraulic operating limits are performed on a plant-specific and cycle-specific basis due to differences in reactor and cycle operating features. Applicable thermal-hydraulic design criteria of the fuel includes: analyses, and results for hydraulic compatibility, thermal margin performance, fuel centerline temperature, rod bow, bypass flow, stability, LOCA analysis, CRDA analysis, ASME overpressurization analysis, and seismic/LOCA liftoff. The sections below summarize the results from selected design criteria and analyses results.

3.2.5.1 Thermal-Hydraulic Characterization

Tables 3.2 (pg. 3-9) and 3.3 (pg. 3-10) ANP-3287NP, Revision 1, "Quad Cities Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM 10XM Fuel Assemblies," dated November 2014 (Reference 22), lists the geometric parameters and component loss coefficients for ATRIUM 10XM and OPTIMA2 fuel designs. The loss coefficients include test data modifications **[[**

]] Tests are performed to obtain the bare rod friction, ULTRAFLOW™ spacer loss, and LTP and UTP losses for the ATRIUM 10XM fuel design. **[[**

]]

Thermal-hydraulic characterization for the ATRIUM 10XM fuel design was performed using the XCOBRA code. For the OPTIMA2 fuel design, thermal-hydraulic characterization included **[[**

]]

[[

]]

3.2.5.1.1 Conclusion

Based on the NRC staff review of the licensee's application of approved methodologies, test data, approved codes, and vendor modeling, as discussed above, the staff determined that the introduction of ATRIUM 10XM fuel design does not significantly affect the hydraulic characterization for loss coefficients and pressure drops with mixed cores at DNPS and QCNPS. Therefore, the NRC staff concludes the licensee's thermal-hydraulic compatibility analysis is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.2.5.2 Thermal-Hydraulic Compatibility

The thermal-hydraulic analyses were performed in accordance with the AREVA thermal-hydraulic methodology for BWRs. The NRC-approved XCOBRA code predicts steady-state thermal-hydraulic performance of the fuel assemblies of BWR cores at various operating conditions and power distributions. Thermal-hydraulic criteria for hydraulic compatibility is that the hydraulic flow resistance of the reload fuel assemblies shall be sufficiently similar to the existing fuel in the reactor such that there is no significant impact on total core flow distribution among the assemblies in the core. Section 3.2 of ANP-3287NP, states:

> The hydraulic compatibility analysis is based on [[
>
>]]

In letter dated January 28, 2016 (Reference 3), the licensee responded to an NRC staff question describing the process of [[

.]]

Tables 3.7 and 3.8 of ANP-3287NP present the QCNPS analysis results for the following core loadings: full core OPTIMA2, full core ATRIUM 10XM, and mixed cores with ATRIUM 10XM and OPTIMA2 fuel designs. The three axial power shapes, as illustrated in Figure 3.1, were analyzed at [[

]]

Table 3.4 of ANP-3287NP summarizes the input conditions for the analyses for the two of the state points considered in the analyses: 100 power power/100 percent flow and 55 percent power/38.5 percent flow. Additionally, Table 3.4 defines the core loading for the transition core configurations including full core OPTIMA2, first transition core ([[ ]]), second transition core ([[ ]]), and full core ATRIUM 10XM. TR ANP-3287NP further provides a summary of thermal-hydraulic results using the first transition core configuration for assembly flow, exit quality, exit void fraction, CPR, assembly bypass fraction, and active flow for ATRIUM 10XM and OPTIMA2 fuel designs for both assembly radial peaking factors of [[ ]]

The thermal compatibility analysis has been performed using ANP-10298(P)(A), Revision 1, "ACE [AREVA Critical Power Evaluator]/ATRIUM 10XM Correlation," dated March 2014 (Reference 23) for the ATRIUM XM fuel. EMF-2209(P)(A) Revision 3, "SPCB Critical Power

Correlation," dated September 2009 (Reference 24), and EMF-2245(P)(A), Revision 0 (Reference 6), for the OPTIMA2 fuel design. The hydraulic compatibility analysis results show that the assembly flow rates and core bypass are within the criteria previously approved by NRC and are acceptable for all core configurations and for rated conditions. Also for off-rated conditions the difference in flow to the maximum power ATRIUM 10XM fuel assembly and the flow to the OPTIMA2 assembly is within the criteria established by AREVA and licensee for hydraulic compatibility between the two fuel designs.

3.2.5.2.1 Conclusion

The NRC staff reviewed the licensee's analysis of fuel assembly pressure drops and hydraulic flow characteristics, and finds that the licensee's proposed transition from a fully loaded core of OPTIMA2 fuel to a fully loaded core of ATRIUM 10XM fuel, to be hydraulically compatible. Therefore, the staff concludes the licensee's proposed fuel transition at DNPS and QCNPS is acceptable with respect to the fuels thermal-hydraulic compatibly.

3.2.5.3 Thermal Margin Performance

Thermal margin analyses were performed using the thermal hydraulic methodology and the XCOBRA code. The calculation of fuel assembly CPR (thermal margin performance) is established by means of an empirical correlation based on results of boiling transition test programs. An evaluation of the margin to the thermal limits for BWRs is performed using the AREVA's CPR methodology. CPR values for ATRIUM 10XM fuel are calculated using the AREVA Critical Power Evaluator (ACE)/ATRIUM 10XM critical power correlation and the CPR values for the OPTIMA2 fuel are calculated with the SPCB critical power correlation. **[[**

]] The acceptability of the use of SPCB correlation for computing CPR values for OPTIMA2 fuel design is justified (Reference 6).

For DNPS and QCNPS operating conditions, some analyses results in assembly conditions to be **[[**

]]

Though the SPCB correlation was found to be adequate to model OPTIMA2 fuel, **[[**

]]. The following table lists the SPCB overall statistics for the SPCB correlation application to the OPTIMA2 fuel design.

[[

]]

3.2.5.3.1 Conclusion

The NRC staff reviewed the licensee's process and calculations for the application of SPCB correlation to the OPTIMA2 fuel design. The staff finds that the licensee is using an approved methodology and the results indicate that the proposed use of ATRIUM 10XM fuel design will not result in thermal margin problems for the co-resident fuel (OPTIMA2) and is therefore acceptable for use in support of the proposed fuel transition at DNPS and QCNPS.

3.2.6 QCNPS, Unit 2, Cycle 24, Representative Fuel Cycle Design

To support the licensee's proposed request to transition fuel at DNPS and QCNPS, AREVA performed a representative fuel cycle design and fuel management calculations based on expected Cycle 24 operation of the QCNPS Unit 2 core. The analysis has been performed using the NRC-approved AREVA neutronic methodology. The lattice depletion code, CASMO-4 is used to generate cross sections and local power peaking factors. MICROBURN-B2, a three dimensional core simulator code, is used to model the core. Thermal margin calculations for this report are performed using the pin power construction model in MICROBURN-B2. The ACE critical power correlation was used for the ATRIUM 10XM fuel design while for the co-resident OPTIMA2 fuel assemblies used the SPCB critical power correlation using appropriate additive constants consistent with the NRC-approved methodology. As reflected in the licensee's response to NRC staff questions, the features of the MICROBURN-B2 code used in the analysis include:

- MICROBURN-B2 (version 2),
- Explicit control blade modeling,
- Control blade B-10 depletion,
- Explicit neutronic treatment of the spacer grids,
- Explicit thermal-hydraulic modeling of the water rod flow, and
- Explicit modeling of the plenum/spring region above the PLFRs.

The version 2 of MICROBURN-B2 is the same methodology as defined in TR EMF-2158(P)(A), EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," Siemens Power Corporation, dated October 1999 (Reference 25) with a few enhanced constituent models which have been demonstrated to meet the same requirements.

Explicit control (rod) blade modeling indicates that the specific material composition and mechanical design of various control blades present in the DNPS and QCNPS operating cycles are modeled in CASMO-4 and provided to MICROBURN-B2 to be applied for appropriate conditions. The absorber material in modern control blades includes B-10 and Hf (Hafnium) nuclides. These blades may also contain varying absorber and non-absorber composition along horizontal control blade wing. The cross-section representation in MICROBURN-B2 has been expanded to include multiple controlled states defined by the specific control blade present in the core adjacent to the fuel. The explicit modeling also includes an explicit depletion of B-10 neutron absorber while inserted adjacent to fuel assemblies. This accounts for reduction in control blade strength for sub-critical reactor shutdown and transient reactor scram.

Explicit neutronic treatment of spacer grids is an enhanced model recognizing that **[[**

]]

The licensee provided a detailed description of explicit neutronic treatment to the NRC staff in response to the staffs request for additional information, the licensee stated:

[[

]]

3.2.6.1 Conclusion

The NRC staff determined, as discussed above, that the licensee has adequately applied NRC-approved neutronic methodologies in the assessment of using the QCNPS, Unit 2, Cycle 24, Representative Fuel Cycle Design as a representative model for the proposed fuel transition. Therefore, the NRC staff concludes the licensee's use of the representative core modeling technique is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.2.6.2 Cross-Section Representation

CASMO-4 performs a multi-group spectrum calculation using a detailed heterogeneous description of the fuel lattice components. Fuel rods, absorber rods, water rods/channels and structural components are modeled explicitly. Depletion calculations are performed using predictor-corrector algorithm in each fuel or absorber rod. The two-dimensional transport solution based on method of characteristics provides pin power and exposure distributions, homogeneous multi-group microscopic cross-sections as well as macroscopic cross-sections. Discontinuity factors are determined from the solution.

MICROBURN-B2 performs microscopic fuel depletion on a nodal basis. The neutron diffusion equation is solved with a full two energy group method. This nodal method uses flux discontinuity factors for different regions and a multilevel iteration technique for efficiency. The model uses [[ ]] methods for accurate representation of in-reactor configuration.

In response to the staffs request for additional information the licensee explained the concept of [[ ]] as follows:

> "Although a BWR bundle is of relatively small size (about 15 cm width), a significant burnup and spectral history difference is developed across a radial plane of a bundle loaded in a heterogeneous environment of an actual core design and operation, especially on the core periphery or subject to a long period of controlled depletion. When a control blade is inserted near the wide-wide corner of a BWR fuel bundle, it produces a large flux and spectrum tilt across the

bundle which in turn causes a burnup and spectral history gradient. Spectral history is a history parameter for BWR where the bundle experiences evolution of varying neutron energy spectrum during their lifetime. Spectral history is used to determine the dependency of microscopic cross sections and macroscopic cross background cross-sections on the neutron spectrum history. Microscopic cross-section and background cross-section interpolation is performed using a quadratic Lagrangian method with the three instantaneous void state-points. Cross-section interpolation in exposure space is performed using a piecewise linear interpolation method. **[[**

]]

A full three-dimensional pin power reconstruction method is utilized. Traversing in-core probe (TIP) (neutron and gamma) and LPRM [Local Power Range Monitor] response models are included to compare calculated and measured instrument responses. Modern steady state thermal-hydraulics models define the flow distribution among the assemblies. Models for the calculation of CPR, LHGR, and MAPLHGR are included in the model for direct comparisons to the operating limits. Nodal macroscopic cross sections by summing the contribution of the various nuclides as a function of coolant density, nodal spectral history, nodal exposure and control fraction (Reference 26). The functional representations of microscopic and macroscopic cross-sections are obtained from three void depletion calculations with CASMO-4. At any exposure point, a quadratic fit of the three CASMO-4 data points is used to represent the continuous cross-section over instantaneous variation of void or water density. Cross-section changes due to spectral changes during depletion have been included. Also, cross-section changes due to self-shielding that occurs with isotopic concentration change have been accounted for using void history and exposure. Sophisticated interpolation methods (quadratic) have been employed to generate curves representing the behavior of the cross-sections as a function of the historical void fraction during the plant operation. The processed cross-sections for all isotopes in MICROBURN-B2 were compared to the cross-sections from CASMO-4 calculations with continuous operation at all possible void fractions. AREVA reports that the results show very good agreement for the entire exposure range of plant operation.

[[

]]

MICOBURN-B2 uses water density rather than void fraction in order to account for pressure changes as well as subcooled density changes. This transformation does not change the basic behavior as water density is proportional to void fraction. Also, MICROBURN-B2 uses spectral history rather than void history in order to account for other spectral influences due to actual core conditions (fuel loading, control rod inventory, leakage, etc.). Doppler feedback is modeled by accumulating Doppler broadening microscopic cross sections of each nuclide using branch calculations performed with CASMO-4 at various exposures and void fractions for each void history depletion.

In the letter dated January 28, 2016, the licensee responded to an NRC staff question stating that MICOBURN-B2 methodology models a wide range of thermal-hydraulic conditions including EPU and extended power/flow operating map conditions. The steady-state thermal-hydraulic methodology implemented in MICROBURN-B2 is an expanded version of the AREVA thermal-hydraulic design and transient analysis methodology XCOBRA. This expansion enables MICOBURN-B2 code to analyze thermal-hydraulic network consisting of several hundreds of heat sources, active coolant channels, water channels and bypass channels in the core without homogenizing them into smaller number of channels. This expansion enables to calculate the nodal specific heat deposition from neutron and gamma from their transport calculation by CASMO-4 and reconstructed by MICOBURN-B2 code to each component of the network instead of the conventional constant heat deposition fractions. The MICOBURN-B2 methodology contains two key constituent correlations in addition to the ASME steam table: (1) the void-quality correlation and (2) the channel component flow friction correlation. The range of thermal-hydraulic parameters underlying the measurement covers anticipated normal operation transient conditions. The MICOBURN-B2 methodology continues to be applied for a large variety of conditions including plants already operating with EPU. The maximum void fraction remains below 0.90 under steady state conditions throughout the power/flow operating domain.

AREVA has reported that recent analysis has shown that there was some concern about the void-quality correlation under low flow conditions when **[[ ]]** was implemented. This issue has been addressed and demonstrated that the limiting values reported in this LAR submittal either remain valid or are conservative. In the licensing analyses this **[[ ]]** is being eliminated and this issue no longer has any impact on the results.

3.2.6.2.1 Conclusion

The NRC staff has determined that based on the above evaluation, the licensee has properly analyzed and applied the MICROBURN-B2 code and XCOBRA methodologies to support the proposed fuel transition. Therefore, the NRC staff concludes the licensee's analysis for cross-section representation and supporting calculations are acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.2.7 Mixed Core Methodology

The DNPS and QCNPS will operate with a mix of both OPTIMA2 (Westinghouse) and ATRIUM 10XM (AREVA) fuel designs during the transition or refueling cycles. For each core design, analyses are performed to confirm that all design and licensing criteria are satisfied. Thermal-hydraulic characteristics are determined for each fuel type that will be present in the core. The thermal hydraulic characteristics used in core design, safety analysis, and core monitoring are developed on a consistent basis for both AREVA fuel and Westinghouse fuel to minimize variability due to methods. For core design and nuclear safety analyses, each fuel assembly is explicitly modeled in MICROBURN-B2 using cross-section data from CASMO-4 and geometric data appropriate for the fuel design. Fuel assembly thermal-mechanical limits for both ATRIUM

10XM and OPTIMA2 fuel are verified and monitored for each mixed core designed by AREVA. The thermal-mechanical limits established by the co-resident fuel vendor continue to be applicable for mixed (transition) cores. AREVA has performed design and licensing analyses to demonstrate that the core design meets steady-state limits and that transient limits are not exceeded during AOOs.

The critical power ratio (CPR) is evaluated for each fuel type in the core using calculated local fluid conditions and an appropriate critical power correlation. The CPR correlation used for the ATRIUM 10XM fuel is the ACE/ATRIUM 10XM critical power correlation (Reference 23). The SPCB critical power correlation is used for monitoring OPTIMA2 fuel present in transition cycles of operation at DNPS and QCNPS. Analyses performed to determine the SLMCPR explicitly address mixed core effects.

Each fuel type present in the core is explicitly modeled using appropriate geometric data, thermal-hydraulic characteristics, and power distribution information from CASMO-4 and MICROBURN-B2 analyses.

An OLMCPR is established for each fuel type in the core. Critical power performance is evaluated using local fluid conditions and fuel type specific CPR correlation coefficients. The transient CPR response is used to establish an OLMCPR for each fuel type. MAPLHGR operating limits are established and monitored for each fuel type in the core to ensure that 10 CFR 50.46 acceptance criteria are met during a postulated LOCA. MAPLHGR limits are established using each fuel vendor's LOCA methodology. For ATRIUM 10XM fuel the RELAX code is used to determine the overall system response during a postulated LOCA and provides boundary conditions for a RELAX hot channel model.

3.2.7.1 Conclusion

The NRC staff has reviewed the licensee's proposed mixed core methodology to support the fuel transition to AREVA ATRIUM 10 fuel at DNPS and QCNPS. The NRC staff have determined that the design and licensing of the mixed core at QCNPS and DNPS have been performed using NRC-approved methodologies for thermal-hydraulic and transients and accident analyses. The NRC finds that the licensee has explicitly considered each fuel type in the mixed core configuration. The licensee has established limits for each fuel type; operation within these limits is verified by the monitoring system during operation. Therefore, the NRC staff concludes that the licensee's proposed fuel transition at DNPS and QCNPS is acceptable with respect to the thermal-hydraulic and accident analysis of a mixed fuel core.

3.2.8 Void-Quality Correlations

This section describes the two void-quality correlations that are used in the nuclear and safety analyses used by AREVA for DNPS and QCNPS. This section also describes the void quality uncertainties and biasing of the correlation.

The AREVA analysis methods and the correlations used by the methods are applicable for modern fuel designs in both pre-EPU and EPU conditions. Though void-quality correlation uncertainty is not a direct input to either OLMCPR or SLMCPR methodologies, the impact of void-correlation uncertainty is inherently incorporated in both methodologies. The SLMCPR methodology explicitly considers important uncertainties in the Monte Carlo calculation performed to determine the number of rods in boiling transition. The bundle power uncertainty in the SLMCPR methodology is determined through comparison of calculated to measured core power distributions.

The transient analysis methodology is a deterministic, bounding approach that contains conservatism to offset uncertainties in individual phenomena. Conservatism is incorporated in the methodology in two ways: (1) computer code models are developed to produce conservative results on an integral basis relative to benchmark tests, and (2) important input parameters are biased in a conservative direction in licensing calculations. The transient analysis methodology results in predicted power increases that are bounding relative to benchmark tests. In addition, for licensing calculations a 110 percent multiplier is applied to the calculated integral power to provide additional conservatism to offset uncertainties in the transient analyses methodology. Therefore, uncertainty in the void-quality correlation is inherently incorporated in the transient analysis methodology.

3.2.8.1 Conclusion

By letter dated January 28, 2016, the licensee addressed a staff request for additional information associated with void-quality correlation uncertainty. Based on the licensee's response the staff determined that the impact of void-quality correlation uncertainty is incorporated in NRC-approved analytical methods which were used by the licensee to determine the OLMCPR. Therefore, the NRC staff concludes that the licensee has acceptably addressed void-quality correlations to support the proposed fuel transition at DNPS and QCNPS.

3.2.9 Fuel Thermal Conductivity Degradation

This section summarizes the impact and treatment of fuel thermal conductivity degradation (TCD) for licensing safety analyses that supports the licensee's proposed fuel transition for DNPS and QCNPS.

By letter dated October 8, 2016, the NRC issued Information Notice (IN) 2009-23 (ADAMS Accession No. ML091550527), to inform industry of concerns regarding the use of historical reactor fuel thermal conductivity models in the safety analysis of operating reactor plants. IN 2009-23 discusses how historical fuel thermal mechanical codes may over predict fuel rod thermal conductivity at higher burn-ups based on new experimental data. This new experimental data showed significant degradation of fuel pellet thermal conductivity with exposure.

At the time of approval of RODEX2 and RODEX2A, the TCD with fuel exposure was not well characterized by irradiation tests or post-irradiation specific-effects tests at high burnups. The fuel performance codes developed at that time did not accurately account for this phenomenon. Analyses performed with RODEX2/2A are impacted by the lack of an accurate thermal conductivity degradation model. Similarly, conductivity models in the transient codes COTRANSA2 and XCOBRA-T do not account for thermal conductivity degradation. NRC-approved fuel performance methodology code RODEX4 is a best-estimate, state-of-the-art fuel code that fully accounts for burnup degradation of fuel thermal conductivity. RODEX4, therefore, can be used to quantify the impact of burnup-dependent fuel thermal conductivity degradation and its effect on key analysis parameters. Thermal-mechanical licensing safety analyses for DNPS and QCNPS are performed with RODEX4 and, therefore, explicitly account for thermal conductivity degradation.

The issues identified in IN 2009-23 were entered into the AREVA corrective action program in 2009. A summary of the investigation was provided to the NRC in a white paper by AREVA (ADAMS Accession No. ML092010160). The white paper presented results of an extensive evaluation for BWRs, the assessments consisted primarily of ATRIUM-10 fuel. Following an NRC review (Reference 27) the staff requested additional information (Reference 28). AREVA provided responses (Reference 29); items relevant to these previous exchanges between the NRC staff and the licensee are also discussed in the following subsections.

The computer codes COTRANSA2 and XCOBRA-T are used in AOO analyses. Both codes use uranium dioxide (UO_2) thermal conductivity correlations that do not address TCD. In addition, the core average gap conductance used in the COTRANSA2 system calculations and the hot channel gap conductance used in XCOBRA-T. Hot channel calculations are obtained from RODEX2 calculations. In general, the sensitivity to conductivity and gap conductance for AOO analyses is in the opposite direction for the core and hot channel, i.e., putting more energy into the coolant (higher thermal conductivity/higher gap conductance) is non-conservative for the system calculation but conservative for the hot channel calculation. The competing effects between the core and hot channel calculation minimize the overall impact of thermal conductivity degradation.

Based on the inherent conservatisms associated with the transient analysis codes and the small impact of thermal conductivity degradation with exposure for the AOO analysis, it is concluded that MCPR and LHGR operating limits based on the AOO methodology are not impacted.

LOCA analyses are performed using the EXEM BWR-2000 methodology and include the use of the RODEX2, RELAX, and HUXY computer codes. In addition to the initial stored energy, the

RODEX2 code is used to calculate fuel mechanical parameters for use in the HUXY computer code that potentially impact the clad ballooning and rupture models. Assessments of the potential impact of exposure-dependent degradation of UO2 thermal conductivity on the fuel mechanical parameters were made using the RODEX4 computer code. **[[**

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Following NRC approval of RODEX2, more Halden tests were performed with fuel centerline temperature monitoring. As with the RODEX4 submittal, **[[**

]]

3.2.9.1 Conclusion

The NRC staff finds that the licensee has acceptably implemented the thermal conductivity degradation with fuel burnup in the codes that performed fuel performance analyses and transients and accidents analyses. Therefore, the staff concludes that the licensee's analysis regarding the nuclear and safety aspects of fuel thermal conductivity degradation is acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.3 Technical Specifications

The following safety evaluation is a review of changes to the DNPS and QCNPS technical specifications. The NRC staff reviewed the proposed changes by comparing the licensee proposed TSs and SRs to the regulatory criteria using the guidance documents specified in the Regulatory Evaluation section above to determine the acceptability to meet the requirements of 10 CFR 50.36.

3.3.1 TS 3.2.3, "Linear Heat Generation Rate (LHGR)"

The licensee proposes to add new SR 3.2.3.2 to the DNPS and QCNPS TS 3.2.3; worded as follows:

	Surveillance	Frequency
SR 3.2.3.2	Determine the LHGR limits	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

This SR is not part of NUREG-1433, Standard Technical Specifications, GE plants, Revision 4. However; to support the licensee's proposed SR 3.2.3.2 addition to the TS, the licensee explains the following reason for the new SR requirement:

> Because the transient analyses take credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that use transient analysis. Therefore, a new surveillance requirement has been proposed determine the LHGR limits within 72 hours of determining control rod scram times, consistent with the requirements for the minimum critical power ratio.
>
> The AREVA reload safety analyses are performed to support three sets of scram speed control rod insertion times. These scram times are based on a conservative interpretation of as-found scram time measurements. In the event

> that plant surveillance data shows nominal scram speed (NSS) control rod insertion times are exceeded, the thermal margin limits are modified to the values corresponding to the intermediate scram speed (ISS) control rod insertion times. The ISS times have been chosen to provide and intermediate valve between the NSS and the Technical Specifications Scram Speed (TSSS) control rod insertion times. In the event the ISS times are exceeded, the operational limits for the TSSS are applied. The new surveillance will verify the correct LHGR limits are being applied based on measured scram speeds.

The licensee's submittal reflects the following from Section 5.1 of ANP-3361NP, Revision 0, "Quad Cities, Unit 2, Cycle 24, Representative Cycle Design Reload Safety Analysis," which states in part:

> [T]he results of the system pressurization transients are sensitive to the scram speed used in the calculations. To take advantage of average scram speeds faster than those associated with the Technical Specifications requirements, scram speed-dependent $MCPR_p$ limits are provided. The nominal scram speed (NSS), intermediate scram speed (ISS), and the Technical Specifications scram speed (TSSS) insertion times are presented in Table 5.2. The NSS and ISS $MCPR_P$ limits can only be applied if the 5 percent, 20 percent, 50 percent, and 90 percent, of the scram speed test results meet the NSS and ISS insertion times. System transient analyses were performed to establish $MCPR_p$ limits for NSS, ISS, and TSSS insertion times.

3.3.1.1 Conclusion

The NRC staff finds that the proposed SR is an appropriate activity to assure the necessary quality of components that facility operation will be within safety limits and the LCO will be met. Therefore, the staff concludes the proposed SR 3.2.3.2 meets 10 CFR 50.36(c)(3) for both DNPS and QCNPS and is, therefore, acceptable to support the proposed fuel transition.

3.3.2 TS 3.3.4.1, "Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT) Instrumentation"

The licensee proposes a change to the allowable value (AV) for Reactor Vessel Steam Dome Pressure High in SR 3.3.4.1.4 for each site. The licensee's submittal states:

> "The DNPS and QCNPS ATWS-RPT Reactor Vessel Steam Dome Pressure-High setpoint analyses were performed according to the EGC Engineering Setpoint Methodology Standard NES-EIC-20.04, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy," Revision 6. The calculation, based on the revised analytical limit (AL)[7] of 1200 psig, determined the TS AV

[7] The analytical limit is a calculated variable established to ensure that a safety limit is not exceeded.

associated with the proposed ATWS-RPT Reactor Vessel Steam Dome Pressure-High function to be 1198 psig for DNPS and 1195 psig for QCNPS. The analysis supports a change to the TS ATWS-RPT Reactor Vessel Steam Dome Pressure-High allowable value (AV) from ≤ 1241 psig to ≤ 1198 psig for DNPS and from ≤ 1219 psig to ≤ 1195 psig for QCNPS. These new AVs are based on an associated new AL of 1200 psig. Reducing the ATWS-RPT reactor vessel steam dome pressure allowable value will lower the peak vessel pressure during an ATWS event.

The reduction in the ATWS-RPT Reactor Vessel Steam Dome Pressure-High allowable value is being proposed to reduce the peak vessel pressure following an ATWS event during the short term phase of the event (i.e., within the first 30 seconds). For the existing ATWS-RPT AL, the calculated peak vessel pressure for the short term ATWS is often very close to the ATWS peak pressure limit of 1500 psig. This analysis is performed on a cycle-specific basis for all four units of DNPS and QCNPS. Reducing the allowable value as proposed would provide additional margin by reducing the ATWS peak vessel pressure. The analysis was performed using the representative core design to demonstrate applicability of AREVA methodology. Due to the similarities between DNPS and QCNPS designs, this demonstration is also applicable to DNPS.

The Reactor Vessel Steam Dome Pressure – High function initiates an RPT [Recirculation Pump Trip] for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation."

3.3.2.1 Conclusion

Based on the licensee's analysis of the ATWS-RPT and its association with the RPV Steam Dome Pressure-High setpoint, the NRC staff finds that the licensee's approach is conservative and increases the margin to safety from peak vessel pressure during ATWS conditions. The staff concludes that changes to SR 3.3.4.1.4 to lower the Reactor Pressure Vessel Steam Dome Pressure – High setpoint will continue to meet 10 CFR 50.36(c)(1) and is therefore acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.3.3 TS 3.7.7, "The Main Turbine Bypass System"

In the licensee's evaluation of proposed change to LCO 3.7.7 for DNPS and QCNPS. The licensee states the following:

"The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable (defined as two or more bypass valves inoperable as specified in the COLR),

> modifications to the MCPR limits (LCO 3.2.2, "Minimum Critical Power Ratio (MCPR)" and LHGR limits (LCO 3.2.3, "Linear Heat Generation Rate (LHGR)" may be applied to allow this LCO to be met. The MCPR and LHGR limits for the inoperable Main Turbine Bypass System are specified in the COLR and evaluated on a cycle specific basis. The modification of the MCPR and LHGR limits in response to an inoperable Main Turbine Bypass System is within the assumptions of the applicable analyses."

As indicated in the DNPS and QCNPS UFSARs (Chapter 15), a main turbine trip without having the Main Turbine Bypass System (MTBS) operable, is analyzed as a limiting transient. This event factors into calculations for the thermal limits to protect fuel cladding integrity. A key safety function of the Main Turbine Bypass System is to limit peak reactor pressure during abnormal plant events thereby keeping within analyzed MCPR and LHGR limits. Applying cycle-specific MCPR and LHGR limits to LCO 3.7.7 ensures the LCO can be satisfied during events when the Main Turbine Bypass System is inoperable. The proposed revision increases the margin to safety by adding flexibility to take credit for LHGR limits as defined in the cycle-specific COLR based on the analyses supporting the transition to ATRIUM 10XM fuel.

In the DNPS/QCNPS UFSAR it states:

> "The DNPS/QCNPS UFSARs (chapter 15) state that the fuel-specific operating limit MCPR is determined for each reload core based on bounding events for the cycle. The operating limit MCPR is established to preclude violation of the fuel cladding integrity safety limit."

> "The maximum drop in CPR (delta-CPR) calculated is adequate for protection of all fuel types against boiling transition. Since a typical rated conditions operating limit MCPR is 1.46 (typical value for OLMCPR, the cycle specific OLMCPR can be found in the Core Operating Limits Report or applicable cycle specific reload documents), the MCPR will remain above the Technical Specification Safety Limit and the fuel cladding integrity safety limit is not violated. The MCPR will remain above the Technical Specification Safety Limit and the fuel cladding integrity safety limit is not violated."

3.3.3.1 Conclusion

The NRC staff has evaluated the licensee's proposed TS change to modify LCO 3.7.7 to include LHGR and MCPR limits as an alternative means to meet the LCO. Since the MCPR and LHGR limits are established assuming no operation of the MTBS the occurrence of an AOO while meeting these limits would not violate fuel integrity acceptance criteria. Since the licensee's analysis included the use of NRC-approved ANP-3361NP to calculate the MCPR and LHGR limits the NRC staff concludes the proposed change to TS LCO 3.7.7 meets 10 CFR 50.36(c)(2), and is therefore acceptable to support the proposed fuel transition at DNPS and QCNPS.

3.3.4 TS 5.6.5, "Core Operating Limits Report"

Both DNPS and QCNPS TS 5.6.5 COLR analytical methods currently include in TS 5.6.5 (b)(1), "Commonwealth Edison Company Topical Report NFSR-0091," and "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods." The licensee's submittal states that they previously had used this methodology for in-house analysis, however, in the future, this methodology will no longer be used. The licensee is proposing to introduce TR EMF-2158(P)(A) Revision 0, (Reference 25) to the methodologies listed in the DNPS and QCNPS TS (i.e., TS 5.6.5.b item no. 18), as a replacement. Additionally, the licensee proposes to add those AREVA analytical methods and topical reports used to establish the core operating limits specified in the COLR to the DNPS and QCNPS TS 5.6.5.b. The NRC staff reviewed the proposed additions as specified in Table 2 of Attachment 1, pages 9, 10 and 11, of the licensee's proposed amendment request (LAR). The additions are also marked on pages 5.6-4 and 5.6-5 of Attachment 2 of the LAR for DNPS and pages 5.6-4 and 5.6-5 of Attachment 3 of the LAR for QCNPS. The licensee states that each analytical methodology being added to TS 5.6.5.b have been approved by the NRC and that in accordance with TS 5.6.5.b core operating limits must be determined using approved methods previously reviewed and approved by the NRC. The licensee states the following regarding Table 2 of Attachment 1, pages 9, 10 and 11, of the LAR:

> "Each of the listed methodologies supports one or more analyses used to establish or support the core operating limits defined in TS 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," TS 3.2.2, "Minimum Critical Power Ratio," TS 3.2.3, "Linear Heat Generation Rate," and TS 3.3.1.3, "Oscillation Power Range Monitor (OPRM) Instrumentation."

The use of NRC-approved topical reports for DNPS and QCNPS reload core methodologies, as listed in the DNPS and QCNPS TS 5.6.5.b, will be limited to the extent specified under the TRs conditions and limitations as delineated in the staff safety evaluation for each approved topical report.

3.3.4.1 Conclusion

The NRC staff finds that the licensees reasoning and justification for removing no longer used methodologies, (1) the removal of code no longer in use has no impact or adverse effect and is not needed by other TS requirements, and (2) the proposed codes to be added to TS 5.6.5.b are NRC-approved and will be used in accordance with those approvals. Therefore, the staff has determined that replacement with used methodologies to be acceptable and approves the deletions/additions to TS 5.6.5.b, as proposed.

The staff has concluded that TS 5.6.5 will continue to meet 10 CFR 50.36(c)(5) in support of the proposed fuel transition at DNPS and QCNPS.

3.3.5 Fuel Transition Implementation

During unit refueling intervals each core will be partially loaded with AREVA ATRIUM 10XM fuel as equal amounts of Westinghouse Optima2 fuel are unloaded, at the rate of an approximate one-third core load/unload, per RFO. As evaluated within this Safety Evaluation, each core will operate using a mixed core (transition core) until successive refueling outages have transitioned the cores fuel exclusively to AREVA ATRIUM 10XM. Due to the logistical and scheduling requirements associated with alternating RFOs at DNPS and QCNPS (sites with common unit TSs), the licensee supplemented the submittal by letter dated September 28, 2016 (ADAMS Accession No. ML16272A376) with interim TS pages. The specific TS changes, as evaluated within this Safety Evaluation, become effective upon unit startup (MODE 2) following the initial reload cycle of ATRIUM 10XM fuel, for each unit.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (*Federal Register Notice*; 80 FR 67800, dated November 3, 2015) and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: M. Razzaque, NRR/SRXB
M. Panicker, NRR/SNPB
P. Snyder, NRR/STSB

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49. N. Zuber and J. A. Findlay, "Average Volumetric Concentration in Two-Phase Flow Systems," J. Heat Transfer, dated 1965.

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to include requirements to use the minimum critical power ratio (MCPR) limits (limiting condition for operation (LCO) 3.2.2) and the LHGR limits (LCO 3.2.3) during plant operations when at $\geq$ 25percent of rated thermal power and the main turbine bypass system is inoperable. Lastly, TS 3.3.4.1, "Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation," will be revised. The associated Allowable Value (AV) in SR 3.3.4.1.4.b, "Reactor Vessel Steam Dome Pressure-High," will be lowered to increase the margin to the maximum reactor pressure vessel (RPV) acceptance criteria for certain anticipated transient without scram (ATWS) transients. For DNPS, Unit Nos. 2 and 3, the AV will be lowered to less than or equal to 1198 pounds per square inch gauge (psig) (originally $\leq$ 1241 psig). For QCNPS, Unit Nos. 1 and 2, the AV will be lowered to less than or equal to 1195 psig (originally $\leq$ 1219 psig).

The NRC staff has determined that its safety evaluation (SE) for the subject amendments contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 2.390. Accordingly, the NRC staff has prepared a redacted, publicly available, non-proprietary version of the SE. Both versions of the SE are enclosed.

The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice. If you have any questions concerning this licensing action, please contact me at 301-415-1129 or by e-mail at Russell.Haskell@nrc.gov.

Sincerely,

/RA/

Russell S. Haskell II, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249, 50-254, and 50-265

Enclosures:
1. Amendment No. 251 to DPR-19
2. Amendment No. 244 to DPR-25
3. Amendment No. 264 to DPR-29
4. Amendment No. 259 to DPR-30
5. Safety Evaluation (Non-Proprietary)
6. Safety Evaluation (Proprietary)

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ADAMS Accession Nos.: PKG ML16218A498 ML16218A485 Amend **ML16218A502** (prop SE) **ML16221A061** (Non prop SE)*by email

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NAME	RHaskell	SRohrer	EOesterle*	RLukes*	AKlein*	BHarris	GEMiller	RHaskell
DATE	8/6/2016	8/6/2016	10/17/2016	10/17/2016	10/13/2016	10/11/2016	10/20/2016	10/20/2016

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Letter to Bryan C. Hanson from Russell S. Haskell II dated:

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNIT NOS. 2 AND 3, AND QUAD CITIES NUCLEAR POWER STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-19, DPR-25, DPR-29, AND DPR-30, TO REVISE TECHNICAL SPECIFICATIONS TO SUPPORT TRANSITIONING TO AREVA NUCLEAR FUEL (CAC NOS. MF5736, MF5737, MF5738, AND MF5739)

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