Exelon Generation

4300 Winfield Road
Warrenville, IL 60555

630 657 2000 Office

PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390

10 CFR 50.90

RS-15-008

February 6, 2015

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

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Request for License Amendment Regarding Transition to AREVA Fuel

References:

1. Letter from Brenda L. Mozafari (U. S. NRC) to Exelon Generation Company, LLC, "Summary of August 23, 2013, Meeting with Exelon Generation Company, LLC Regarding AREVA XM Fuel Transition Request for Dresden and Quad Cities Nuclear Stations (TAC Nos. MF2422, MF2423, MF2424, and MF2425)," dated September 11, 2014 (ADAMS Accession No. ML14241A633)

2. Letter from Brenda L. Mozafari (U. S. NRC) to Exelon Generation Company, LLC, "Summary of May 19, 2014, Meeting with Exelon Generation Company, LLC Regarding AREVA XM Fuel Transition for Dresden and Quad Cities Nuclear Stations (TAC Nos. MF2422, MF2423, MF2424, and MF2425)," dated September 11, 2014 (ADAMS Accession No. ML14226B012)

Attachments 7 through 15 transmitted herewith contain Proprietary Information. When separated from attachments, this document is decontrolled.

3. Letter from Brenda L. Mozafari (U. S. NRC) to Exelon Generation Company, LLC, "Summary of August 19, 2014, Closed Meeting with Exelon Generation Company, LLC (Exelon) Regarding AREVA XM Fuel Transition for Dresden and Quad Cities Nuclear Stations (TAC Nos. MF2422, MF2423, MF2424, and MF2425)," dated September 22, 2014 (ADAMS Accession No. 14254A153)

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS) Units 2 and 3, and Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2. The proposed change supports the transition from Westinghouse SVEA-96 Optima2 (Optima2) fuel to AREVA ATRIUM 10XM fuel at DNPS and QCNPS. Specifically, EGC proposes to revise Technical Specification (TS) 5.6.5, "Core Operating Limits Report (COLR)," paragraph b, to delete no longer used methodologies and to add the AREVA analysis methodologies to the list of approved methods to be used in determining the core operating limits in the COLR. Also, in support of the planned transition to AREVA ATRIUM 10XM fuel, EGC proposes to revise DNPS and QCNPS TS 3.2.3, "Linear Heat Generation Rate (LHGR)," and TS 3.7.7, "The Main Turbine Bypass System."

In addition to the proposed changes supporting the fuel transition, EGC is proposing a change to DNPS and QCNPS TS Surveillance Requirement (SR) 3.3.4.1.4, Allowable Value for the anticipated transient without scram recirculation pump trip (ATWS–RPT) on high reactor pressure vessel (RPV) steam dome pressure, in order to increase the margin to the maximum RPV pressure acceptance criteria on certain ATWS transients. EGC proposes to modify the Allowable Value for ATWS-RPT on high RPV steam dome pressure to ≤ 1198 psig for DNPS and ≤ 1195 psig for QCNPS.

This request is subdivided as follows.

- Attachment 1 provides an evaluation supporting the proposed change.

- Attachments 2 and 3 contain the marked-up Technical Specifications (TS) pages for DNPS and QCNPS, respectively, with the proposed changes indicated.

- Attachments 4 and 5 provide the marked-up TS Bases pages for DNPS and QCNPS, respectively, with the proposed changes indicated. These attachments are provided for information only.

- In support of the proposed TS changes, certain technical information related to a representative transition core design (i.e., a mixed core loading of Optima2 fuel and ATRIUM 10XM fuel) and licensing analyses, as well as information related to the AREVA analysis methodologies, has been provided in Attachments 6 through 25 of this submittal.

Attachments 7, 8, 9, 10, 11, 12, 13, 14, and 15 to this letter contain information that AREVA considers to be proprietary in nature and subsequently, pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4), it is requested that such information be withheld from public disclosure. AREVA, as the owner of the proprietary information, has executed an affidavit provided in Attachment 16, which identifies that the enclosed proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has previously been withheld from public disclosure. AREVA requests that the enclosed proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390. Attachments 17, 18, 19, 20, 21, 22, 23, 24, and 25 contain the redacted versions of the proprietary enclosures with the proprietary material removed, which are suitable for public disclosure.

EGC participated in several meetings with the NRC Staff regarding the planned transition from Westinghouse Optima2 fuel to the new AREVA ATRIUM 10XM fuel design at DNPS and Quad Cities Nuclear Power Station. During these meetings, EGC discussed our plan to submit this amendment request supporting the transition to AREVA ATRIUM 10XM fuel. The attached evaluation addresses all the issues discussed at these meetings as summarized in References 1, 2, and 3.

The proposed change has been reviewed by the Plant Operations Review Committee and approved by the Nuclear Safety Review Board for the respective facilities in accordance with the requirements of the EGC Quality Assurance Program.

As described above, EGC plans to transition fuel vendors for DNPS and QCNPS beginning with the DNPS Unit 3 refueling outage in November 2016. ATRIUM 10XM will then be inserted in QCNPS Unit 1 in March 2017, DNPS Unit 2 in November 2017, and QCNPS Unit 2 in March 2018. Therefore, EGC requests approval of the proposed changes by September 30, 2016, since the core operating limits using the new analytical methods added to TS Section 5.6.5.b will become effective upon startup following the DNPS Unit 3 refueling outage.

The cycle specific reload analyses for the affected units may result in the need for additional TS changes to support the specific cycle designs, such as a change to the safety limit minimum critical power ratio. These changes, if any, will be submitted to the NRC in a separate license amendment request.

In accordance with 10 CFR 50.91(b), EGC is notifying the State of Illinois of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Should you have any questions related to this letter, please contact Mr. Timothy A Byam at (630) 657-2818.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 6th day of February 2015.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachments:

Attachment 1:	Evaluation of Proposed Change
Attachment 2:	Markup of Proposed Technical Specifications Pages for DNPS
Attachment 3:	Markup of Proposed Technical Specifications Pages for QCNPS
Attachment 4:	Markup of Proposed Technical Specifications Bases Pages for DNPS (For Information Only)
Attachment 5:	Markup of Proposed Technical Specifications Bases Pages for QCNPS (For Information Only)
Attachment 6:	Boiling Water Reactor Licensing Methodology Compendium
Attachment 7:	Applicability of AREVA Methodology (Proprietary)
Attachment 8:	Mechanical Design Report (Proprietary)
Attachment 9:	Thermal-Hydraulic Design Report (Proprietary)
Attachment 10:	Fuel Rod Thermal-Mechanical Design Report (Proprietary)
Attachment 11:	Fuel Cycle Design Report (Proprietary)
Attachment 12:	Reload Safety Analysis Report (Proprietary)
Attachment 13:	LOCA Break Spectrum Analysis Report (Proprietary)
Attachment 14:	LOCA-ECCS Analysis MAPLHGR Limit Report (Proprietary)
Attachment 15:	Nuclear Fuel Design Report (Proprietary)
Attachment 16:	AREVA Affidavits
Attachment 17:	Applicability of AREVA Methodology (Non-Proprietary)
Attachment 18:	Mechanical Design Report (Non-Proprietary)
Attachment 19:	Thermal-Hydraulic Design Report (Non-Proprietary)
Attachment 20:	Fuel Rod Thermal-Mechanical Design Report (Non-Proprietary)
Attachment 21:	Fuel Cycle Design Report (Non-Proprietary)
Attachment 22:	Reload Safety Analysis Report (Non-Proprietary)
Attachment 23:	LOCA Break Spectrum Analysis Report (Non-Proprietary)
Attachment 24:	LOCA-ECCS Analysis MAPLHGR Limit Report (Non-Proprietary)
Attachment 25:	Nuclear Fuel Design Report (Non-Proprietary)

cc: Regional Administrator- NRC Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

**Attachments 7 through 15 transmitted herewith contain Proprietary Information.
When separated from attachments, this document is decontrolled.**

ATTACHMENT 1
EVALUATION OF PROPOSED CHANGE

Subject: Request for License Amendment Regarding Transition to AREVA Fuel

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS) Units 2 and 3, and Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2. The proposed change supports the transition from Westinghouse SVEA-96 Optima2 (Optima2) fuel to AREVA ATRIUM 10XM fuel at DNPS and QCNPS. Specifically, EGC proposes to revise DNPS and QCNPS Technical Specification (TS) 5.6.5, "Core Operating Limits Report (COLR)," paragraph b, to delete no longer used methodologies and to add the AREVA analysis methodologies to the list of approved methods to be used in determining the core operating limits in the COLR. Also, in support of the planned transition to AREVA ATRIUM 10XM fuel, EGC proposes to revise DNPS and QCNPS TS 3.2.3, "Linear Heat Generation Rate (LHGR)," and TS 3.7.7, "The Main Turbine Bypass System."

In addition to the proposed changes supporting the fuel transition, EGC is proposing a change to DNPS and QCNPS TS Surveillance Requirement (SR) 3.3.4.1.4, Allowable Value for the anticipated transient without scram recirculation pump trip (ATWS–RPT) on high reactor pressure vessel (RPV) steam dome pressure, in order to increase the margin to the maximum RPV pressure acceptance criteria on certain ATWS transients. EGC proposes to modify the Allowable Value for ATWS-RPT on high RPV steam dome pressure to ≤ 1198 psig for DNPS and ≤ 1195 psig for QCNPS.

2.0 DETAILED DESCRIPTION

2.1 Proposed Fuel Transition Changes

EGC intends to begin using the ATRIUM 10XM fuel design in DNPS Unit 3 Cycle 25 with the first load of ATRIUM 10XM targeted for insertion into the core in the November 2016 refueling outage. ATRIUM 10XM will then be inserted in QCNPS Unit 1 in March 2017, DNPS Unit 2 in November 2017, and QCNPS Unit 2 in March 2018. The ATRIUM 10XM product is a proven fuel design approved for use at two other Boiling Water Reactors (BWRs) in the United States. At this time, EGC is requesting approval for the transition to AREVA fuel at Extended Power Uprate (EPU) conditions. EGC has participated in pre-application meetings with the NRC Staff on August 27, 2013, May 19, 2014, and August 19, 2014 as documented in References 1, 2, and 3, respectively. During these pre-application meetings the plans for the EGC fuel transition were discussed including the methodologies to be used, the proposed TS changes, technical topics associated with the transition, and schedule. This evaluation addresses all the issues discussed at these meetings as summarized in References 1, 2, and 3.

In order to extend the use of this fuel design to DNPS Units 2 and 3 and QCNPS Units 1 and 2 and to adopt the AREVA fuel design and safety analyses, changes to the DNPS and QCNPS TS are required. The proposed change to TS 5.6.5.b will delete reference to Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," as this methodology is no longer used to develop core operating limits for DNPS and QCNPS. In addition, EGC proposes to insert the titles of eighteen NRC-approved AREVA methodologies that will be used to develop core operating limits for the DNPS and QCNPS cores loaded with AREVA ATRIUM 10XM fuel. TS 5.6.5.b addresses the analytical methods which may be used to determine input to the COLR. The proposed amendments will add the relevant AREVA Topical Reports to the list of analytical methods specified in TS 5.6.5.b that have been reviewed and approved by the NRC for determining core operating limits. The AREVA Topical reports to be added to DNPS and QCNPS TS 5.6.5.b include the following:

- XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.
- ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels, May 1995.
- EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998.
- BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.
- 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, March 1983.
- XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.
- 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, January 1987.
- 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, October 1999.
- EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.
- EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," AREVA NP, September 2009.
- ANP-10298PA Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," AREVA, March 2014.
- ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.
- XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.
- ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.
- EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001.

- EMF-2292(P)(A) Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000.
- ANF-1358(P)(A) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005.
- EMF-CC-074(P)(A) Volume 4 Revision 0, "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000.

In support of the transition to AREVA ATRIUM 10XM fuel, EGC also proposes to revise the DNPS and QCNPS TS 3.2.3, "Linear Heat Generation Rate (LHGR)," by adding a new surveillance requirement. Because the transient analyses take credit for conservatism in the scram speed performance, the new proposed surveillance requirement will demonstrate that the specific speed distribution is consistent with that used in the transient analyses. The new DNPS and QCNPS surveillance requirement will read as follows.

"SR	3.2.3.2	Determine the LHGR limits.	Once within 72 hours after each completion of SR 3.1.4.1 <u>AND</u> Once within 72 hours after each completion of SR 3.1.4.2 <u>AND</u> Once within 72 hours after each completion of SR 3.1.4.4"

In addition, EGC also proposes a revision to the DNPS and QCNPS TS 3.7.7, "The Main Turbine Bypass System," Limiting Condition for Operation (LCO) in support of the transition to AREVA ATRIUM 10XM fuel. Specifically, the proposed revision will add LHGR limits to the existing Minimum Critical Power Ratio (MCPR) limits as an alternative in the event the Main Turbine Bypass System is inoperable. The revised DNPS and QCNPS LCO will read as follows.

"LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

The following limits are made applicable:

a. LCO 3.2.2, "Minimum Critical Power Ratio (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, and

b. LCO 3.2.3, "Linear Heat Generation Rate (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR."

Mark-ups of the above proposed TS changes are provided in Attachments 2 and 3 for DNPS and QCNPS, respectively. In addition, mark-ups of the associated TS Bases pages are provided in Attachments 4 and 5 for DNPS and QCNPS, respectively. The Bases mark-ups are provided for information only, and do not require NRC approval.

2.2 Proposed Change to TS 3.3.4.1, "Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation"

The proposed change would require a revision to the Allowable Value (AV) defined in the DNPS and QCNPS TS 3.3.4.1, "Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation," SR 3.3.4.1.4.b, "Reactor Vessel Steam Dome Pressure-High," channel calibration. Specifically, the proposed change revises the ATWS-RPT Reactor Vessel – High AV to ≤ 1198 psig for DNPS and ≤ 1195 psig for QCNPS. The reduction in the ATWS-RPT AV is requested to reduce the peak reactor vessel pressure following an ATWS event during the short term phase of the event (i.e., within the first 30 seconds).

Currently the DNPS Units 2 and 3 TS AV for the ATWS-RPT on high reactor vessel steam dome pressure is ≤ 1241 psig. The proposed change revises DNPS TS SR 3.3.4.1.4 to read as follows.

"SR 3.3.4.1.4 Perform CHANNEL CALIBRATION. The Allowable Values shall be:

a. Reactor Vessel Water Level-Low Low: ≥ -54.15 inches with time delay set to ≥ 8.3 seconds and ≤ 9.7 seconds; and

b. Reactor Vessel Steam Dome Pressure-High: ≤ 1198 psig"

Currently the QCNPS Units 1 and 2 TS AV for the ATWS-RPT on high reactor vessel steam dome pressure is ≤ 1219 psig. The proposed change revises QCNPS TS SR 3.3.4.1.4 to read as follows.

"SR 3.3.4.1.4 Perform CHANNEL CALIBRATION. The Allowable Values shall be:

a. Reactor Vessel Water Level-Low Low: ≥ -56.3 inches with time delay set to ≥ 7.2 seconds and ≤ 10.8 seconds; and

b. Reactor Vessel Steam Dome Pressure-High: ≤ 1195 psig"

Attachments 2 and 3 provide TS marked-up pages indicating the above proposed changes for DNPS and QCNPS, respectively. In addition, in support of this proposed TS change, mark-ups of the associated TS Bases pages are provided in Attachments 4 and 5 for DNPS and QCNPS, respectively. The Bases mark-ups are provided for information only, and do not require NRC approval.

3.0 TECHNICAL EVALUATION

3.1 Proposed Fuel Transition Changes

3.1.1 Design Description - ATRIUM 10XM Fuel

The fuel design to be introduced into DNPS Units 2 and 3 and QCNPS Units 1 and 2 in 2016 through 2018 is the AREVA ATRIUM 10XM product. This design utilizes a 10x10 array of fuel rods, with 79 full length fuel rods and 12 partial length fuel rods. The partial length fuel rods are approximately one half the length of the full length fuel rods. The use of partial length rods improves fuel utilization in the high void upper region of the bundle and also enhances stability, pressure drop performance, and cold shutdown margin. The ATRIUM 10XM design does not utilize tie rods as the structural tie between the upper and lower tie plates. Instead, the design uses a central water channel having a mechanical connection to the two tie plates. The central water channel carries the mechanical loads during fuel handling. It displaces a 3x3 array of fuel rods within the bundle and serves to improve fuel economy by improving internal neutron moderation. The lower ends of the fuel rods rest on top of the lower tie plate with their lower ends laterally restrained by a spacer grid located just above the lower tie plate. No expansion springs are required on each fuel rod because a single, large reaction spring is used on the central water channel to hold the upper tie plate in the latched position. The ATRIUM 10XM design uses fuel rod spacers to provide lateral support for the fuel rods and to enhance thermal-hydraulic performance. The ATRIUM 10XM design to be employed utilizes a debris resistant lower tie plate to limit introduction of foreign material into the assembly from below.

Further description of the ATRIUM 10XM design is provided in Attachment 8.

3.1.2 Similarity between DNPS and QCNPS

The DNPS Units 2 and 3 and QCNPS Units 1 and 2 are similar in many respects. Both stations are General Electric BWR/3 designs that have the same operating conditions as indicated below in Table 1.

Table 1

Comparison of DNPS and QCNPS Operating Parameters

Operating Parameter	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	355.6 ˚F	355.6 ˚F
Nominal Dome Pressure (rated power)	1,015 psia	1,015 psia
Core Size - Fuel Assemblies	724	724

Because of the similarity between the two stations, the application of the AREVA methodology to a representative core design 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.

LOCA Analysis
The DNPS and QCNPS Emergency Core Cooling System (ECCS) components, although slightly different in performance, 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.

Transient Analysis
Application of the AREVA methodology for transient analysis has been demonstrated for a representative mixed core loading of Optima2 fuel and ATRIUM 10XM fuel at QCNPS. However, a cycle-specific transient analysis will be performed by AREVA to establish the

core thermal limits required for operating with the actual ATRIUM 10XM fuel loading at DNPS and QCNPS. 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. Relative to parameters important to transient analysis, similarities between DNPS and QCNPS include nearly identical scram times and setpoints, nearly identical steam line valve characteristics, similar equipment out of service (EOOS) scenarios, and a similar sequence of events for the various transient events such that the same AREVA methodology will be used for both stations.

ATWS Analysis

Analyses for the MSIV closure and pressure regulator failure – open to maximum demand event are performed to evaluate the effect on the peak vessel pressure for a representative mixed core loading of Optima2 fuel and ATRIUM 10XM fuel at QCNPS. For this analysis, a critical parameter is initiation of the ATWS-RPT causing a recirculation pump trip and adding negative reactivity. The ATWS-RPT is designed to lessen the effects of the event.

The analysis supports a change to the TS ATWS-RPT Reactor Vessel Steam Dome Pressure-High allowable value 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. 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. Relative to parameters important to ATWS analysis, similarities between the two stations include nearly identical steam line valve characteristics and recirculation system parameters, and identical ATWS-RPT AL.

3.1.3 Justification for Proposed TS Changes

TS 5.6.5, "Core Operating Limits Report (COLR)"

The AREVA analytical methods and topical reports to be added to DNPS and QCNPS TS 5.6.5.b are those utilized to establish the core operating limits identified in the COLR. Additionally, Reference 4 is being added to the TS as the basis for acceptance of the ATRIUM 10XM fuel design. Each analytical methodology being added to TS 5.6.5.b has been previously reviewed and approved by the NRC.

In accordance with TS 5.6.5.b, the core operating limits must be determined using methods previously reviewed and approved by the NRC. TS 5.6.5.b provides a listing of the specific NRC reviewed and approved methodologies to be used in determining the core operating limits. The methodologies being added have all been submitted to the

NRC and have been approved. The following table (i.e., Table 2) lists the topical reports being added to TS Section 5.6.5.b, along with the basis for adding it to TS 5.6.5.b. 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 applicable power distribution limit TS for each methodology is also identified in the following table (i.e., Table 2).

Table 2

BWR Approved Topical Reports for
QCNPS and DNPS Technical Specification
COLR References

Report	Applicable TS LCO	COLR Justification
XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.	3.2.1 3.2.2	Provides an analytical capability to predict BWR fuel thermal and mechanical conditions to establish initial conditions for non-LOCA and LOCA analyses.
ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.	3.2.1 3.2.2 3.2.3	Establishes a set of design criteria which assures that BWR fuel will perform satisfactorily throughout its lifetime.
EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998.	3.2.1 3.2.2	Extends the exposure limits of the RODEX2A code, which is a version of RODEX2 that includes a fission gas release model specific to BWR fuel designs.
BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.	3.2.3	Provides an analytical capability to predict fuel thermal and mechanical conditions. It is used to support the acceptability of the fuel LHGR limits.
XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors –	3.2.1 3.2.2 3.2.3	Development of BWR core analysis methodology which comprises codes for fuel neutronic parameters and

Report	Applicable TS LCO	COLR Justification
Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983.		assembly burnup calculations, reactor core simulation, diffusion theory calculations, and producing input for nuclear plant transient analysis. Subsequently, approved codes or methodologies have superseded portions of this report.
XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.	3.2.1 3.2.2 3.2.3	Summarizes the types of BWR licensing analyses performed, identifies the methodologies used.
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, January 1987.	3.2.2	Provides overall methodology for determining a MCPR operating limit.
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, October 1999.	3.2.1 3.2.2 3.2.3	Describes the reactor core simulator code MICROBURN-B2 and the lattice physics code CASMO-4.
EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.	3.2.2	Presents approaches to develop parameters necessary to model co-resident (non-AREVA) fuel with an approved AREVA critical power correlation.
EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," AREVA NP, September 2009.	3.2.2	Presents the AREVA critical power correlation that will be used for the Optima2 fuel design.
ANP-10298PA Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," AREVA,	3.2.2	Presents and justifies the critical power correlation applicable for the ATRIUM

Report	Applicable TS LCO	COLR Justification
March 2014.		10XM fuel design.
ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.	3.2.2	Presents a methodology for determining the safety limit minimum critical power ratio (SLMCPR).
XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.	3.2.2	Provides a capability to perform transient thermal-hydraulic and thermal margin analyses.
ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.	3.2.2	Provides a computer program for analyzing BWR system transients.
EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001	3.2.1	Describes an evaluation model for analysis of postulated LOCAs in jet pump BWRs. The methodology complies with 10 CFR 50.46 and 10 CFR 50 Appendix K criteria.
EMF-2292(P)(A) Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000.	3.2.1	Provides justification to use Appendix K spray heat transfer coefficients in LOCA analyses for 10x10 fuel designs.
ANF-1358(P)(A) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005	3.2.2	Presents a generic methodology that may be used to evaluate the loss of feedwater heating event.
EMF-CC-074(P)(A) Volume 4 Revision 0, "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000	3.3.1.3	Describes methodology for stability analysis with input from the MICROBURN-B2 reactor core simulator.

EGC and AREVA have reviewed the associated NRC safety evaluations for these topical reports previously approved by the NRC. The use of the topical reports for DNPS and QCNPS reload core designs will be limited to the extent specified in the topical reports and under the conditions and limitations delineated in the NRC safety evaluations. The safety evaluation limitations/conditions and the measures taken to ensure compliance for each approved topical report are listed in Attachments 6 and 7. In addition, the applicability of the referenced AREVA methodologies to DNPS and QCNPS is addressed in Attachment 7. This attachment also discusses the changes and modifications to the approved methodologies to address the technical issues identified since the methodologies were approved. These changes were discussed in the pre-submittal meetings held with NRC (Reference 1, 2 and 3).

EGC previously benchmarked certain methodologies and performed licensing analyses in-house. To support this effort, EGC (Commonwealth Edison at the time) prepared and submitted for NRC review Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods." The current licensing analyses no longer utilize CASMO/MICROBURN. The methodology now used is CASMO4/MICROBURN-B2. Therefore, EGC proposes deleting the NFSR-0091 report from DNPS and QCNPS TS 5.6.5.b as this methodology is no longer used to develop core operating limits.

<u>TS 3.2.3, "Linear Heat Generation Rate (LHGR)"</u>

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations and anticipated operating conditions identified in the DNPS and QCNPS Updated Final Safety Analysis Report (UFSAR) Chapters 4 and 15.

The analytical methods and assumptions used in evaluating the fuel system design and establish LHGR limits are presented in Attachment 10. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20 and 50. A mechanism that could cause fuel damage during normal operations and operational transients and that is considered in fuel evaluations is a rupture of the fuel rod cladding caused by strain from the relative expansion of the $UO2$ pellet.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur. Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient excursions above the operating limit while still remaining within the fuel design limits, plus an allowance for densification power spiking.

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 used in the transient analyses. The reload safety analysis documented in Attachment 12 explicitly evaluates the transient and accident response of the mixed core. Therefore, a new surveillance requirement has been proposed to 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 an intermediate value 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 the measured scram speeds.

<u>TS 3.7.7, "The Main Turbine Bypass System"</u>

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 33.3% and 33.5% of the Nuclear Steam Supply System rated steam flow at QCNPS and DNPS, respectively. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of a nine valve manifold connected to the main steam lines between the main steam isolation valves and the main turbine stop valves. Each of the nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine Electro–Hydraulic Control System, as discussed in the DNPS and QCNPS UFSAR, Section 7.7.4. The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves sequentially. When the bypass valves open, the steam flows from the main steam equalizing header to the bypass manifold through the bypass valve, to its bypass line, where an orifice further reduces the steam pressure before the steam enters the condenser.

The Main Turbine Bypass System is assumed to function during the turbine trip, turbine generator load rejection and feedwater controller failure transients, as described in the DNPS and QCNPS UFSAR Sections 15.2.3.2, 15.2.2.2 and 15.1.2. The Main Turbine Bypass System is also assumed to function during other AOOs in which system pressurization may occur. Opening the bypass valves during the pressurization events mitigates the increase in reactor vessel pressure which affects the MCPR and nodal power excursion during the event. An inoperable Main Turbine Bypass System may result in MCPR and LHGR penalties.

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 as documented in Attachment 12. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. 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.

3.1.5 Fuel Design Changes

The ATRIUM 10XM design was developed using the thermal-mechanical design bases and limits outlined in Reference 4. Compliance with Reference 4 ensures the fuel design meets the fuel system damage, fuel failure, and fuel coolability criteria identified in NUREG-0800, "Standard Review Plan," (SRP) Section 4.2, "Fuel System Design." The NRC reviewed and approved in Reference 5 the use of Reference 4 for making changes and improvements to fuel designs; specifically stating such changes and improvements do not require specific NRC review and approval, provided the criteria are satisfied. The ATRIUM 10XM design fully complies with the criteria of Reference 4, and therefore meets all of the required fuel licensing criteria in the SRP.

The impact of the ATRIUM 10XM design on the DNPS and QCNPS UFSAR transient and accident analyses will be evaluated on a cycle-specific reload analysis basis. Limiting transients from UFSAR Chapter 15 categories of pressure increase events, vessel water temperature decrease events, control rod withdrawal error events, core flow increase events, and increase in vessel inventory events are evaluated each cycle.

For single loop operation (SLO), the maximum achievable power state point on the MELLLA upper boundary is assumed to be 70.2% licensed uprated power (i.e., 2076 MWth) at 55.1% flow (i.e., 54 Mlbm/hr) for Extended Power Uprate (References 17, 19 and 20). Since the SLO pump seizure ΔCPR results determined by AREVA are limiting for single loop operation and are used to establish the SLO MCPR operating limits, SLO is only supported for power levels ≤ 50% of rated and core flows ≤ 50 Mlbm/hr (i.e., 51% of rated). Therefore, AREVA performs the SLO pump seizure analysis at 50% rated power and a core flow of 50 Mlbm/hr. The input statepoints used by AREVA are controlled by DNPS and QCNPS procedures. Limiting analysis results for a representative transition cycle are presented in Attachment 12, Section 5.3.10.

Introduction of the ATRIUM 10XM fuel design will not adversely impact UFSAR accident analyses. AREVA evaluates the control rod drop accident (DNPS and QCNPS UFSAR Section 15.4.10) on a cycle-specific basis. Attachment 12 includes a cycle-specific evaluation of the control rod drop accident for a representative transition cycle. The evaluation shows the number of rods calculated to fail in this event remains well below the value of 850 assumed in the UFSAR radiological evaluation of this event. The doses from the control rod drop accident remain within limits required by 10 CFR 50.67, "Accident Source Term," and Regulatory Guide 1.183 (Reference 6).

Loss of Coolant Accident (LOCA) analyses (DNPS and QCNPS UFSAR Section 15.6.5) composed of a LOCA break spectrum analysis and a follow-on maximum average planar linear heat generation rate (MAPLHGR) analysis were performed for QCNPS at extended power uprated (EPU) conditions. These analyses are presented in Attachments 13 and 14. A break spectrum/ECCS single failure study, which is similar to the studies shown in the QCNPS and DNPS UFSAR by the other fuel suppliers, is performed using the AREVA LOCA Evaluation Model to determine the limiting break size and break location, in addition to the limiting ECCS single failure. Consistent with the current licensing basis, the ECCS pump flows are used as inputs to the LOCA analysis, without adjustment for uncertainty. The purpose of the break spectrum analysis is to determine the characteristics of the limiting break; the break that results in the highest cladding temperature during a postulated LOCA. To determine the limiting conditions, the following LOCA parameters are considered in the break spectrum analysis:

- Break location
- Break type (double-ended guillotine or split)
- Break size
- Limiting ECCS single failure
- Axial power shape (top- or mid-peaked)

The limiting break characteristics and conditions determined in the break spectrum analysis are used in the follow-on analyses to establish the exposure-dependent MAPLHGR limits for ATRIUM 10XM. Attachment 14 documents the results of the ATRIUM 10XM MAPLHGR analysis for QCNPS. MAPLHGR limits are established for normal (i.e., two-loop) operation and for operation when one of the recirculation loops is out-of-service (i.e., single-loop) as well as other modes of operation allowed by the COLR. The report also documents the licensing basis peak cladding temperature (PCT) and corresponding local cladding oxidation from the metal-water reaction (MWR) evaluation for ATRIUM 10XM fuel at QCNPS Units 1 and 2. The analyses demonstrate that the MAPLHGR limits ensure that the LOCA-ECCS criteria in 10 CFR 50.46 are satisfied for operation at or below these limits.

The break spectrum and MAPLHGR analyses documented in Attachments 13 and 14 were performed with the LOCA Evaluation Model developed by AREVA and approved by the NRC. Modifications to the approved methodology have been made to more accurately model advanced fuel designs and address regulatory concerns with the approved methodology. The modifications are described in Attachment 7, Appendix E.

The introduction of ATRIUM 10XM fuel will not challenge the PCT, cladding oxidation, or hydrogen generation limits specified in 10 CFR 50.46, paragraph (b). Satisfaction of the remaining two 10 CFR 50.46 acceptance criteria (i.e., coolable geometry and long-term core cooling) is demonstrated in Attachment 13.

The Fuel Handling Accident (FHA) for ATRIUM 10XM was evaluated and it was determined that the number of failed rods and the dose consequences of the accident are bounded by the current FHA analysis design basis results (see Attachment 12, Section 6.3).

The main steam line break accident (DNPS and QCNPS UFSAR section 15.6.4) is not affected by a change in fuel design. As stated in the UFSAR, no fuel failures are expected to occur as a result of this accident. The radionuclide inventory released from the primary coolant system is present in the coolant prior to the event. Therefore, the fuel design change does not alter the consequences of a main steam line break accident.

As noted above, the application of the AREVA methodologies to a representative transition core design of ATRIUM 10XM and Optima2 fuel (i.e., QCNPS Units 1 and 2) is sufficient to demonstrate the applicability of the methodologies for AREVA ATRIUM 10XM fuel at both stations (i.e. DNPS and QCNPS). The methods are applicable for a full core of AREVA ATRIUM 10XM fuel as well as transition cores containing AREVA ATRIUM 10XM fuel and co-resident legacy fuel at both stations. A cycle specific analysis will be completed prior to loading ATRIUM 10XM fuel in each of the four units beginning with Dresden Unit 3 in the fall of 2016.

3.1.6 Additional Supporting Technical Evaluations

The scope of the technical analyses provided in support of the DNPS and QCNPS transition to ATRIUM 10XM fuel is consistent with the technical analyses provided with the precedent submittals (see Section 4.2 below). The DNPS and QCNPS fuel transition employed the AREVA methods referenced in the proposed TS 5.6.5.b change (see Attachments 2 and 3) to evaluate both the ATRIUM 10XM fuel and the co-resident Westinghouse Optima2 fuel. The AREVA methodologies are applied in accordance with NRC approval for performance of design and licensing analyses for mixed cores.

Thermal-Hydraulic Analysis

The thermal-hydraulic characteristics of the Optima2 fuel design were explicitly accounted for and a detailed thermal-hydraulic analysis of the mixed core was performed as documented in Attachment 9.

For each operating cycle, an MCPR Safety Limit is established to ensure at least 99.9 percent of the fuel rods in the reactor core avoid boiling transition during normal operation and AOOs. The methodology described in Reference 18 as modified in accordance with the discussion presented in Attachment 7, Section 5, is applied on a cycle-specific basis to ensure an appropriate MCPR safety limit is used. The derivation of MCPR operating limits presented in the cycle-specific COLR, using the NRC-accepted methods described in the topical reports specified in Technical Specification 5.6.5.b, ensures the MCPR Safety Limit is not violated during all modes of plant operation as allowed by the COLR and anticipated operational occurrences.

The application of AREVA critical power correlations to co-resident fuel is governed by Reference 7. All AREVA critical heat flux (CHF) and CPR correlations are approved by the NRC staff to be applicable over specified ranges of assembly operating conditions. The NRC staff also approved specific corrective actions when the computed conditions fall outside of the approved range to assure that conservative calculations are obtained.

For Dresden and Quad Cities operating conditions, some analyses can predict assembly conditions to be outside the approved range of specified conditions for the CHF correlations. Consequently, the AREVA licensing methods are programmed to determine whether the computed assembly conditions fall outside of the approved range of applicability for the CHF correlation and impose approved corrective actions as appropriate to conservatively assess the critical power margin for the assembly. The CPR correlation used for the ATRIUM 10XM fuel is the ACE/ATRIUM 10XM critical power correlation and the corrective actions for when the computed conditions fall outside the approved range are provided in Reference 8. The application of the AREVA SPCB critical power correlation to the co-resident Optima2 fuel uses the methodology described in Reference 7. Details of the application are documented in Attachment 7 Section 4.

<u>Channel Bow and Control Blade Interference</u>

EGC has evaluated both the ATRIUM 10XM and legacy Westinghouse Optima2 fuel for impact from channel bow and deflection. Specifically, the evaluation was intended to determine if channel bow accounted for the possibility that a fluence gradient could induce variations in the sub-channel geometry.

When affected by channel bow the cross section geometry of the Westinghouse SVEA-channel is not significantly distorted. The net effect is a translation of the entire geometry in the lateral directions. Thus, the distances between sub-bundles are not significantly changed. Furthermore, since the fuel rods within each sub-bundle are held in place by individual sub-bundle spacers the relative distances between fuel rods in the same sub-bundle are not significantly impacted either. In this respect, the behavior is not different to how bundles without the water-cross behave. Since the impact of channel bow on relative distances between fuel rods is so small, the effect is not considered in the Westinghouse methodology.

The safety limit MCPR (SLMCPR) is determined such that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition during normal operation or an AOO. The SLMCPR for all fuel in the Quad Cities Unit 2 Cycle 24 representative core design was determined using the methodology described in Reference 18. The SLMCPR analysis explicitly includes the effects of channel bow and assumes that no fuel channels are used for more than one fuel bundle lifetime.

For the first cycle following the fuel transition, EGC intends to maintain the existing design and monitoring recommendations for Westinghouse fuel, as the first cycle AREVA fuel will not impact channel/control blade interference concerns. For future mixed core cycles, the design and monitoring will incorporate a channel exposure based model for both the AREVA and Westinghouse fuel to provide rechanneling and operating recommendations. The model will incorporate operating experience with AREVA and Westinghouse channels.

<u>Thermal Conductivity Degradation (TCD)</u>

Results of the AREVA fuel rod thermal-mechanical analyses are presented in Attachment 10 to demonstrate that the applicable design criteria are satisfied. The analysis results are evaluated according to the generic fuel rod thermal and mechanical design criteria contained in ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 4) along with design criteria provided in the RODEX4 fuel rod thermal-mechanical topical report (Reference 9). As described in Attachment 7, Appendix F, 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 impact and treatment of fuel conductivity degradation for licensing safety analyses supporting operation at DNPS and QCNPS are discussed in Attachment 7, Appendix F.

With the transition to AREVA ATRIUM 10XM fuel Westinghouse will continue to hold the LOCA analysis of record for the Westinghouse Optima2 fuel loaded in the core with inputs from the steady-state fuel rod performance analyses. Westinghouse provided a response to the NRC request regarding evaluation of TCD using Westinghouse methods in a letter dated February 17, 2012 (Reference 10). The supporting Westinghouse LOCA analyses for DNPS and QCNPS use the methods as discussed in Reference 10 and therefore, TCD has been accounted for in these analyses.

<u>Pellet-Clad Interaction</u>

Attachment 10, Section 3.1, describes the design of the ATRIUM 10XM fuel rod including the cladding design. The results of the fuel rod design analyses, including pellet-clad interaction (PCI), are provided in Attachment 10, Section 3.2.

EGC will be using the POWERPLEX-XD core monitoring system with the AREVA fuel conditioning guidance and the option to utilize XEDOR. The AREVA methods for fuel conditioning were incorporated into the applicable EGC procedures following the LaSalle County Station transition to ATRIUM 10 fuel. EGC will continue to apply the same PCI fuel conditioning guidance from Westinghouse that is in place today for the legacy Optima2 fuel. POWERPLEX has the capability to provide fuel-type specific monitoring criteria, and EGC is currently operating with Optima2 fuel monitored by POWERPLEX at QCNPS. In summary, EGC will continue to use the existing codes and Westinghouse criteria for Optima2 fuel and EGC will apply the AREVA criteria to AREVA fuel.

<u>Revised Dose Analysis</u>

EGC and AREVA have evaluated the impact of the fuel transition on the dose analyses of record using the approved Alternative Source Term (AST) methodology. The radiological source term for ATRIUM 10XM fuel loaded into DNPS and QCNPS has been calculated with the use of SCALE 4.4a, SAS2H/ORIGEN-S. It was concluded that the change in fuel type resulted in only slight variations in the radiological source term for the reactor core and a corresponding slight difference in the overall dose consequences compared to analysis of record. On the basis that the accident dose as a result of the transition to ATRIUM 10XM is bounded by the analysis of record, the AST analyses will not be revised and the results will not be submitted for NRC review. At no location did the calculated dose using ATRIUM 10XM fuel exceed the previously-reported dose based on the analysis of record. Dose consequences for the LOCA, FHA, and CRDA with an ATRIUM 10XM fuel source term remained well below the regulatory limit of 10 CFR 50.67 and Regulatory Guide 1.183. The Main Steam Line Break was not reviewed in detail because it is unaffected by the change in fuel design.

Thermal-Hydraulic Stability

The stability performance of a core is strongly dependent on the core power, core flow, and power distribution in the core. Therefore, core stability is evaluated on a cycle-specific basis and addressed in the reload licensing report. Associated with these calculations is confirmation of existing power/flow range exclusion regions or redefinition of the regions, as necessary. The cycle-specific stability evaluations are performed consistent with the BWR Owner's Group Option III solution. Attachment 9 discusses an evaluation of the relative stability performance of ATRIUM 10XM against the performance of a previously approved AREVA fuel design. The evaluation concluded that ATRIUM 10XM has decay ratios equivalent to or better than other approved AREVA designs. Attachment 12 provides a representative reload safety analysis and documents the results of a representative stability evaluation for a mixed core loading of Optima2 fuel and ATRIUM 10XM fuel. As part of this cycle specific analysis, the Backup Stability Protection (BSP) curves are evaluated to determine endpoints that meet decay ratio criteria for the BSP Base Minimal Region I (scram region) and Base Minimal Region II (controlled entry region). The details of this evaluation are provided in Attachment 12, Section 4.3.

Oxidation, Hydriding, and Crud Buildup

As noted in Attachment 6, Section 2.2.5, the AREVA fuel design basis for cladding corrosion and crud buildup is to prevent 1) significant degradation of the cladding strength, and 2) unacceptable temperature increases. Cladding corrosion reduces cladding wall thickness and results in less cladding load carrying capacity. At normal light water reactor operating conditions, this mechanism is not limiting except under unusual conditions where high cladding temperatures greatly accelerate the corrosion rate. Because of the thermal resistance of corrosion and crud layers, formation of these products on the cladding results in an elevation of temperature within the fuel as well as the cladding.

Attachment 10, Section 3.2.7 provides an overview of the analysis of ATRIUM 10XM fuel for the potential of oxidation and hydriding at DNPS and QCNPS. Cladding external oxidation is calculated using RODEX4. The corrosion model includes an enhancement factor that is derived from poolside measurement data to obtain a fit of the expected oxide thickness.

In the event abnormal crud is observed for a plant, a specific analysis is required to address the higher crud level. An abnormal level of crud is defined by a formation that increases the calculated fuel average temperature by 25°C above the design basis calculation. The corrosion model in RODEX4 also takes into consideration the effect of the higher thermal resistance from the crud on the corrosion rate. A higher corrosion rate is therefore included as part of the abnormal crud evaluation. A similar specific analysis is required if a plant experiences higher corrosion instead of crud. The current measurements of crud at QCNPS Unit 2 indicate normal low crud levels. However, in order to address the potential impact of changing water chemistry conditions, the input parameters have been conservatively selected in order to generate corrosion in excess of the current operating experience at QCNPS. Attachment 10, Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

Compliance with Safety Evaluation Restrictions and Limitations

Attachment 6 provides a compendium of AREVA methodologies and design criteria, which are described in topical reports that the NRC has found acceptable for referencing in boiling water reactor (BWR) licensing applications. This compendium provides a concise, organized source for BWR topical reports. It presents information about the application of each topical report, the associated safety evaluation report (SER) and its conclusions and restrictions for each topical report, the relationships among the topical reports, and, for certain methodologies, descriptions of their unique characteristics or applications. Compliance with the SER restrictions is assured by implementing them within the engineering guidelines or by incorporating them into the computer codes.

In addition, Attachment 7 provides reviews of the AREVA approved licensing methodologies to demonstrate that they are applicable to licensing and operation of the DNPS Units 2 and 3 and QCNPS Units 1 and 2 with the introduction of ATRIUM 10XM fuel.

3.2 Proposed Change to TS 3.3.4.1, "Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation"

The ATWS-RPT System initiates an RPT, adding negative reactivity, following events in which a Reactor Protection System (RPS) automatic shutdown (i.e., scram) should occur, but does not. The ATWS-RPT is designed to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity to the reactor core through an increase in steam voiding in the core area as core flow decreases. When either the Reactor Vessel Water Level—Low Low or Reactor Vessel Steam Dome Pressure—High setpoint is reached, power is secured to the associated reactor recirculation pump motor.

The ATWS-RPT is assumed to mitigate certain pressurization transients as specified in fuel cycle specific reload analysis. The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur.

Excessively high RPV pressure could damage the reactor coolant pressure boundary (RCPB). An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and reactor thermal power, which could potentially result in fuel failure and RCPB overpressurization. The Reactor Vessel Steam Dome Pressure—High function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety valves, limits the peak RPV pressure to less than the American Society of Mechanical Engineers (ASME) Section III Code Service Level C limits (i.e., 1500 psig). The Reactor Vessel Steam Dome Pressure—High AL is chosen to provide an adequate margin to the ASME Section III Code Service Level C allowable reactor coolant system (RCS) pressure.

The ASME overpressurization event is analyzed to demonstrate that the peak vessel pressure is less than 110% of the design pressure (i.e., 1375 psig) and the peak dome pressure is less than the Technical Specifications dome pressure safety limit (i.e., 1345 psig). The trip of the reactor recirculation pumps has two main effects on the ASME overpressurization event. It causes a reduction in core flow, which increases the core void generation, thus introducing negative reactivity and decreasing the reactor power. At the same time, the increase in core void causes a water level swell which results in a smaller gas volume in the upper region of the vessel and a higher peak dome pressure. The impact from using the proposed Reactor Vessel Steam Dome Pressure – High AL is evaluated.

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 AL of 1200 psig, determined the TS AV associated with the proposed ATWS-RPT Reactor Vessel Steam Dome Pressure—High function to be ≤ 1198 psig for DNPS and ≤ 1195 psig for QCNPS.

Revision 6 of the EGC setpoint methodology standard is equivalent to previous revisions of the standard which have been reviewed by the NRC as discussed below:

In a letter dated March 30, 2001 (Reference 11), the NRC issued license amendments and the associated Safety Evaluation (SE) for DNPS (ADAMS Accession Number ML011130121) implementation of Improved TSs. This SE documents the NRC's review of Revisions 1 (ADAMS Accession No. ML003698624) and 2 (ADAMS Accession No. ML003721342) of NES-EIC-20.04. In addition, as stated in the NRC's SE, EGC provided Revision 3 of NES-EIC-20.04 to the NRC in a letter dated November 30, 2000, (ADAMS Accession No. ML003776648). The March 30, 2001, NRC SE concluded that, "The staff also finds that the instrument setpoint methodology used by the licensee to determine the allowable values is acceptable." The same reviews and acceptance of EGC Setpoint Methodology were documented for QCNPS in Reference 12.

Subsequent to the March 30, 2001, license amendments and associated SEs, EGC issued Revisions 4, 5, and 6 to NES-EIC-20.04. The scope of the change in Revision 4 was limited. For example, Revision 4 corrected a typographical error to a table in Appendix J, "Guideline for the Analysis and Use of As-Found/As-Left Data," added clarification to Appendix J Sections 2.1.1 and 2.1.2.2, and incorporated the changing of the company name from "ComEd" to "Exelon." Revision 5 captures Rosemount guidance for addressing static head effects on differential pressure transmitters, updates references, and corrects minor typographical errors. Previously, EGC calculations would reference Rosemount documents for guidance on addressing static head effects. EGC provided NES-EIC-20.04, Revision 5, to the NRC as Enclosure 1 in a letter from J. L. Hansen, "Additional Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation," dated September 15, 2010 (ADAMS Accession No. ML102590347). In Reference 13 the NRC documented its review of NES-EIC-20.04, Revision 5 as follows: "The NRC staff evaluated NES-EIC-20.04, Revision 5, and the calculations provided for each instrument trip channel using RG 1.105 and RIS 2006-17. Based upon this evaluation, the NRC staff finds that the methodology, as applied to the DNPS SDC Isolation Calculated Setpoints, Expanded Tolerances, Setting Tolerances and AVs is acceptable for the Shutdown Cooling System Isolation, Reactor Vessel Pressure-High function." Finally, EGC issued NES-EIC-20.04, Revision 6 to incorporate the requirements for the adoption of Technical Specifications Task Force (TSTF) Traveler TSTF-493, "Clarify Application of Setpoint Methodology for LSSS

Functions." The methodology that was used for this proposed change to DNPS and QCNPS TS SR 3.3.4.1.1.b (i.e., NES-EIC-20.04 Revision 6) is the same methodology reviewed by the NRC in 2001 and 2010 (i.e., Revisions 3 and 5 of NES-EIC-20.04).

The proposed change to TS SR 3.3.4.1.4.b follows the guidance of TSTF-493. EGC has elected to use Option A when implementing TSTF-493 for DNPS Units 2 and 3 and QCNPS Units 1 and 2 for SR 3.3.4.1.4.b. TSTF-493 footnotes are not required for ATWS-RPT Instrumentation surveillance requirements when Option A is used. Attachment 4 provides the markup of the DNPS TS Bases for SR 3.3.4.1.4.b and Attachment 5 provides the markup of the QCNPS TS Bases for SR 3.3.4.1.4.b that implement the guidance of TSTF-493.

The potentially limiting analyses for the ASME overpressurization and ATWS events were performed using a representative transition cycle (Attachment 12, Sections 7.1 and 7.2). It is sufficient to limit the technical evaluation to these events since they are identified as potentially being limiting and are also the events that are reanalyzed each fuel cycle for DNPS and QCNPS. The ATWS-RPT AV change will not make another event more limiting.

In summary, an ATWS-RPT on the revised high reactor steam dome pressure AV (i.e., ≤ 1198 psig for DNPS and ≤ 1195 psig for QCNPS) ensures that the peak RPV pressure during an ATWS transient is maintained less than 1500 psig and the peak pressure during an ASME overpressure transient is maintained less than applicable limits. Additionally, operating in the proposed manner will not result in any adverse ancillary effects on DNPS and QCNPS AOOs, analyzed accidents, special events, or other operational considerations.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

4.1.1 <u>Proposed Fuel Transition Changes</u>

The following regulatory requirements relate to fuel related design and licensing criteria specified in the SRP Section 4.2.

The following 10 CFR 50, Appendix A, "General Design Criteria," are applicable to this amendment request. While DNPS and QCNPS are not formally committed to the GDC due to the vintage of the stations, an evaluation was performed addressing the DNPS and QCNPS conformance with the GDC. This evaluation is documented in the DNPS and QCNPS UFSAR Section 3.1, "Conformance with NRC General Design Criteria." This evaluation concluded that DNPS and QCNPS fully satisfy the intent of the (then draft) GDC.

General Design Criterion (GDC) 10, "Reactor design," states that 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. To ensure compliance with GDC 10, EGC performs plant-specific critical power limit analyses using NRC-approved methodologies. The MCPR Safety Limit ensures that sufficient conservatism exists in the operating limit MCPR such that, in the event of an anticipated operational occurrence, there is a reasonable expectation that at least 99.9 percent of the fuel rods in the core will avoid boiling transition for the power distribution within the core including all uncertainties. The analysis documented above verifies that DNPS and QCNPS will continue to satisfy the intent of this GDC following the transition to ATRIUM 10XM fuel.

GDC 27, "Combined reactivity control system capability," states that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. The plant design contains two independent and different principle reactivity control systems. Control of reactivity is operationally provided by a combination of movable control rods, burnable neutron absorbers contained in the fuel, and reactor coolant recirculation system flow. The reactivity control system is designed such that, under conditions of normal operation, sufficient reactivity compensation is always available to make the reactor adequately subcritical from its most reactive condition. The analysis documented above verifies that DNPS and QCNPS will continue to satisfy the intent of this GDC following the transition to ATRIUM 10XM fuel.

GDC 35, "Emergency core cooling," states that 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. The emergency core cooling systems (ECCS) are designed so that at least two different ECCS of different phenomena are provided to prevent clad melt over the entire spectrum of postulated design basis reactor primary system breaks. Such capability is available concurrently with the loss of all offsite ac power. The ECCS individual systems themselves are designed to various levels of component redundancy such that no single active component failure in addition to the accident will negate the required emergency core cooling capability. The analysis documented above verifies that DNPS and QCNPS will continue to satisfy the intent of this GDC following the transition to ATRIUM 10XM fuel.

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires that each BWR must be provided with an emergency core cooling system (ECCS) that is designed so that it's calculated cooling performance following a postulated LOCA conforms to the criteria defined in 10 CFR 50.46(b). As shown by the QCNPS analysis documented above, application of the AREVA LOCA methodology will ensure that DNPS and QCNPS will continue to conform to the requirements of the specified regulations.

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in a licensee's TS. 10 CFR 50.36, paragraph (c)(5) states that TS will include administrative controls that address 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. The COLR is required as part of the reporting requirements specified in the DNPS and QCNPS TS administrative controls. In addition, it is required that the analytical methods used to determine the core

operating limits be approved and described in the administrative controls section of the TS. The proposed change ensures that these requirements are met.

4.1.2 ATWS-RPT Technical Specification Change

The ATWS-RPT is assumed to mitigate certain pressurization transients as specified in the cycle specific reload analysis. The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur. Based on its contribution to the reduction of overall plant risk, however, the instrumentation meets Criterion 4 of 10 CFR 50.36(c)(2)(ii). Operation in the proposed manner will continue to aid in preserving fuel cladding integrity; therefore, the change will have no impact on compliance with this requirement.

The ATWS Rule, 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," provides requirements for reduction of risk from ATWS events. 10 CFR 50.62(c)(5) requires that each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS, and requires that this equipment be designed to perform its function in a reliable manner. The protection and monitoring functions of the ATWS-RPT instrumentation have been designed to ensure safe operation of the reactor by lessening the effects of an ATWS event. The ATWS-RPT instrumentation initiates an RPT to insert negative reactivity into the reactor whenever reactor water level and/or reactor vessel steam dome pressure exceed their specified limits. RPT actuation aids in preserving the integrity of the fuel cladding following events in which a RPS scram should occur, but does not. Operation in the proposed manner ensures that an RPT will occur under ATWS conditions, and that the requirements of 10 CFR 50.62(c)(5) are met.

4.2 Precedent

The NRC has previously approved the use of the AREVA methodologies in support of the transition to ATRIUM 10XM fuel as documented in the following approved amendments.

Brunswick Steam Electric Plant, Units 1 and 2, Safety Evaluation, dated April 8, 2011 (Reference 14). This amendment approved the use of ACE/ATRIUM 10XM Critical Power Correlation, Revision 0 (March 2010) and supported the plant's transition to ATRIUM 10XM fuel design and associated core design methodologies. The final Safety Evaluation for this amendment included a license condition to ensure the limits generated with the NRC-approved methods appropriately bounded the effects of the K-factor calculation. The issues associated with this license condition have been resolved with the NRC review and approval of ANP-10298PA, "ACE/ATRIUM 10XM Critical Power Correlation," Revision 1 (Reference 8). This precedent is directly applicable to this DNPS and QCNPS fuel transition LAR in that it provides precedent for the ATRIUM 10XM fuel design and the associated critical power correlation requested herein.

Browns Ferry Nuclear Plant (BFN), Units 1, 2 and 3 Safety Evaluation, dated July 31, 2014 (Reference 15). This amendment approved the addition of three AREVA analysis methodologies to the list of approved methods to be used in determining core operating limits in the Core Operating Limits Report. In addition, the amendments implement a change to the Safety Limit Minimum Critical Power Ratio value for BFN Unit 2. The changes support a planned transition to AREVA ATRIUM 10XM (XM) fuel design. This precedent is also directly applicable to the DNPS and QCNPS fuel transition in that it provides precedent for the ATRIUM 10XM fuel design and the associated methodologies including the critical power correlation included in this amendment request.

A similar amendment request is currently under review by the NRC. Northern States Power Company submitted an amendment request for Monticello Nuclear Generating Plant on July 15, 2013 (Reference 16). The proposed amendment involves a fuel transition from the Global Nuclear Fuels (GNF) GE14 fuel design to the ATRIUM 10XM fuel design. Specifically, the amendment request proposes to revise TS 5.6.3 to add the AREVA analysis methodologies to the list of approved methods to be used in determining the core operating limits in the Core Operating Limits Report (COLR). While the Monticello amendment request is still under review and has not yet been approved, EGC has used this precedence in the development of this DNPS and QCNPS amendment request.

4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (EGC) will be transitioning to AREVA ATRIUM 10XM fuel at Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. AREVA, in conjunction with EGC, will be designing future core reloads for DNPS and QCNPS beginning with DNPS Unit 3 Cycle 25, which is scheduled to begin in November 2016. Technical Specifications (TS) 3.2.3, "Linear Heat Generation Rate (LHGR)," TS 3.7.7, "The Main Turbine Bypass System," and TS Section 5.6.5, "Core Operating Limits Report (COLR)," require revision to support this transition. Specifically, the proposed change: (1) to take credit for conservatism in the scram speed performance, TS Surveillance Requirement (SR) 3.2.3.2 will be revised to demonstrate that the specific speed distribution is consistent with that used in the transient analyses; (2) the proposed revision to TS 3.7.7 will add the LHGR limits to the Minimum Critical Power Ratio (MCPR) limits as an alternative in the event the Main Turbine Bypass System is inoperable; and (3) revises TS Section 5.6.5.b to remove no longer used methodologies and to add analytical methods that support design of core reloads utilizing AREVA ATRIUM 10XM fuel.

In addition to the proposed changes supporting the fuel transition, EGC is proposing a change to DNPS and QCNPS TS SR 3.3.4.1.4, Allowable Value for the anticipated transient without scram recirculation pump trip (ATWS–RPT) on high reactor pressure vessel (RPV) steam dome pressure, in order to increase the margin to the maximum RPV pressures on certain ATWS transients.

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

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

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

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

EGC has evaluated the proposed change to the TS for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

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

   Response: No

   The proposed change has no effect on any accident initiator or precursor previously evaluated and does not change the manner in which the core is operated. The type of fuel is not a precursor to any accident. The new methodologies for determining core operating limits have been validated to ensure that the output accurately models predicted core behavior, and use of the methodologies will be within the ranges previously approved. The new methodologies being referenced have all been submitted to the NRC, and have been approved.

   The proposed changes to the TS associated with LHGR and the Main Turbine Bypass System, support the new analyses performed as part of the transition to ATRIUM 10XM fuel. These changes do not require modification to the plant and do not impact any initiators of an accident previously analyzed. Implementation of these changes will ensure that the basis for the accident and transient analyses are maintained throughout the operating cycle.

   The proposed change to the ATWS-RPT high RPV steam dome pressure does not require modification to the facility beyond the conservative reduction of the allowable value (AV). The proposed change will be implemented through revision of the associated surveillance test procedures, where the revised AV will replace the existing value.

   Calculation of the AV to plant-specific parameters provides additional confidence that protective instrumentation that passes the surveillance testing criteria will perform its design function without exceeding the associated limit.

   The revised AV for the ATWS-RPT is not considered an initiator to any previously analyzed accident and therefore, cannot increase the probability of any previously evaluated accident. Implementation of the revised AV will ensure that the instrumentation will perform its required function to meet the accident analysis assumptions. The proposed AV will ensure that the fuel is adequately cooled and overpressurization of the nuclear steam supply system is prevented following an accident or transient. The proposed change does not increase the probability of any accident previously evaluated.

   There is no change in the consequences of an accident previously evaluated. The proposed change in the administratively controlled analytical methods does not affect the ability to successfully respond to previously evaluated accidents and does not affect radiological assumptions used in the evaluations. The source term from ATRIUM 10XM fuel will be bounded by the source term assumed in the accident analyses. Since the proposed change ensures the same level of protection as assumed in the accident analyses, the conclusions of the accident scenarios remain valid. As a result, no changes to radiological release parameters are involved. There is no effect on the type or amount of radiation released, and there is no effect on predicted offsite doses in the event of an accident.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed change does not affect the performance of any DNPS or QCNPS structure, system, or component credited with mitigating any accident previously evaluated. The use of new analytical methods, which have been reviewed and approved by the NRC, for the design of a core reload will not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not introduce any new modes of system operation or failure mechanisms. The proposed TS changes ensure operation in compliance with the accident and transient analyses.

   The proposed change to the ATWS-RPT AV does not involve any physical changes to the ATWS-RPT system or associated components beyond the reduction in the ATWS-RPT AV for high reactor vessel steam dome pressure, or the manner in which the ATWS-RPT system functions. The proposed change will not alter the manner in which equipment operation is initiated nor will the functional demands on credited equipment be changed. The change in methods governing normal plant operation is consistent with the current ATWS analysis assumptions specified in the DNPS and QCNPS Updated Final Safety Analysis Report (UFSAR)

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

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

   Response: No

   The proposed change to TS 3.2.3 provides assurance the operating parameters are consistent with the inputs to the transient analyses which take credit for conservatisms in scram speed performance. The proposed change does not alter the acceptance criteria for control rod scram times. The proposed revision to TS 3.7.7 allows the flexibility to take credit for LHGR limits defined in the COLR based on the analyses supporting the transition to ATRIUM 10XM fuel. The proposed change to TS Section 5.6.5.b adds new analytical methods for design and analysis of core reloads to the list of methods currently used to determine the core operating limits. The NRC has previously approved the analytical methods being added.

The proposed change also lowers the ATWS-RPT AV for RPT on high reactor steam dome pressure. There is no decrease in the margin of safety, since the maximum reactor vessel pressure for a postulated ATWS event and ASME overpressure event is maintained below the acceptance criteria. The proposed change will be implemented through revisions to the associated surveillance test procedures where the revised AV replaces the existing AV. Since the availability of the ATWS-RPT system will be maintained and since the system design is unaffected, the proposed change ensures the instrumentation is capable of performing its intended function.

Since the setpoint at which the ATWS-RPT is activated is not a safety limit, the proposed change does not modify any safety limits at which protective actions are initiated, and does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety.

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

Based upon the above, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

5.0 ENVIRONMENTAL CONSIDERATION

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

6.0 REFERENCES

1. Letter from Brenda L. Mozafari (U. S. NRC) to Exelon Generation Company, LLC, "Summary of August 23, 2013, Meeting with Exelon Generation Company, LLC Regarding AREVA XM Fuel Transition Request for Dresden and Quad Cities Nuclear Stations (TAC Nos. MF2422, MF2423, MF2424, and MF2425)," dated September 11, 2014 (ADAMS Accession No. ML14241A633)

2. Letter from Brenda L. Mozafari (U. S. NRC) to Exelon Generation Company, LLC, "Summary of May 19, 2014, Meeting with Exelon Generation Company, LLC Regarding AREVA XM Fuel Transition for Dresden and Quad Cities Nuclear Stations (TAC Nos.

MF2422, MF2423, MF2424, and MF2425)," dated September 11, 2014 (ADAMS Accession No. ML14226B012)

3. Letter from Brenda L. Mozafari (U. S. NRC) to Exelon Generation Company, LLC, "Summary of August 19, 2014, Closed Meeting with Exelon Generation Company, LLC (Exelon) Regarding AREVA XM Fuel Transition for Dresden and Quad Cities Nuclear Stations (TAC Nos. MF2422, MF2423, MF2424, and MF2425)," dated September 22, 2014 (ADAMS Accession No. 14254A153)

4. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Designs," Advanced Nuclear Fuels Corporation, dated May 1995

5. Letter from R.C. Jones (NRC) to R. Copeland (Siemens Power Corporation), "Acceptance for Referencing of Topical Report ANF-89-98(P), Revision 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," (TAC No. M81070)," dated April 20, 1995

6. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, dated July 2000

7. EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000

8. ANP-10298PA Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," AREVA, March 2014

9. BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP Inc., February 2008

10. Letter from Westinghouse to U. S. NRC, "Title," LTR-NRC-12-18, dated February 17, 2012

11. Letter from Stewart N. Bailey (U. S. NRC) to Oliver D. Kingsley (Exelon Generation Company, LLC), Issuance of Amendments (TAC Nos. MA8382 and MA8383), dated March 30, 2001 (ADAMS Accession No. ML011130121)

12. Letter from Stewart N. Bailey (U. S. NRC) to Oliver D. Kingsley (Exelon Generation Company, LLC), "Issuance of Amendments (TAC Nos. MA8378 and MA8379)," dated March 30, 2001 (ADAMS Accession No. ML011130309)

13. Letter from Eva A. Brown (U. S. NRC) to Michael J. Pacilio (Exelon Generation company, LLC), "Dresden Nuclear Power Station, Units 2 and 3 – Issuance of Amendments Regarding Changes to Shutdown Cooling System Isolation Instrumentation (TAC Nos. ME3354 and ME3355)," dated February 7, 2011 (ADAMS Accession No. ML110210619)

14. Letter from Farideh E. Saba (U. S. NRC) to Michael J. Annacone (Carolina Power & Light Company), "Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of

Amendments Regarding Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5 (TAC Nos. ME3856 and ME3857)," dated April 8, 2011 (ADAMS Accession No. ML111010234)

15. Letter from Farideh E. Saba (U. S. NRC) to Joseph W. Shea (Tennessee Valley Authority), "Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Issuance of Amendments Regarding Technical Specification (TS) Change TS-478 Addition of Analytical Methodologies to TS 5.6.5 and Revision of TS 2.1.1.2 for Unit 2 (TAC Nos. MF0877, MF0878 and MF0879)," dated July 31, 2014 (ADAMS Accession No. ML14113A286)

16. Letter from Mark A. Schimmel (Northern States Power Company – Minnesota) to U. S. NRC, "License Amendment Request for Transition to AREVA ATRIUM 10XM Fuel and AREVA Safety Analysis Methodology," dated July 15, 2013 (ADAMS Accession No. ML13200A187)

17. Letter from R. M. Krich (Commonwealth Edison Company) to U. S. Nuclear Regulatory Commission, "Request for License Amendment for Power Uprate Operation," dated December 27, 2000 (ADAMS Accession No. ML010080047)

18. ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP Inc., June 2011

19. Letter from Lawrence W. Rossbach (U. S. NRC) to Oliver D. Kingsley (Exelon Nuclear), "Dresden Nuclear Power Station, Units 2 and 3 – Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0844 and MB0845)," dated December 21, 2001 (ADAMS Accession No. ML013510595)

20. Letter from Stewart N. Bailey (U. S. NRC) to Oliver D. Kingsley (Exelon Nuclear), "Quad Cities Nuclear power Station, Units 1 and 2 – Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0842 and MB0843)," dated December 21, 2001 (ADAMS Accession No. ML013530380)

ATTACHMENT 2

Markup of Proposed Technical Specifications Pages for DNPS

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP AND In accordance with the Surveillance Frequency Control Program
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

SURVEILLANCE	FREQUENCY
	72 hours after each completion of SR 3.1.4.4

SURVEILLANCE REQUIREMENTS

---NOTE---

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.2	Calibrate the trip units.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.3	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.4	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level—Low Low: ≥ -54.15 inches with time delay set to ≥ 8.3 seconds and ≤ 9.7 seconds; and b. Reactor Vessel Steam Dome Pressure-High: ≤ ~~1241~~ 1198 psig.	In accordance with the Surveillance Frequency Control Program

(continued)

3.7 PLANT SYSTEMS

3.7.7 The Main Turbine Bypass System

LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, ~~are made applicable.~~ and

~~a.~~b. **LCO 3.2.3, "Linear Heat Generation Rate (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.**

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.7.1 Verify one complete cycle of each main turbine bypass valve.	In accordance with the Surveillance Frequency Control Program

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. The LHGR for Specification 3.2.3.

4. Control Rod Block Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.

5. The OPRM setpoints for the trip function for SR 3.3.1.3.3

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. ~~Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."~~

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

3. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.

4. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel."

5. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2."

6. WCAP-15682-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application."

7. WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel."

8. WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

9. WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENPD-287."

10. CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors."

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

12. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels, May 1995.

13. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998.

14. BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.

15. 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, March 1983.

16. XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.

17. 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, January 1987.

18. 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, October 1999.

19. EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.

20. EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," AREVA NP, September 2009.

21. ANP-10298PA Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," AREVA, March 2014.

22. ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

23. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.
24. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.
25. EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001.
26. EMF-2292(P)(A) Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000.
27. ANF-1358(P)(A) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005.
28. EMF-CC-074(P)(A) Volume 4 Revision 0, "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000.

The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

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

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

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

ATTACHMENT 3

Markup of Proposed Technical Specifications Pages for QCNPS

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP AND In accordance with the Surveillance Frequency Control Program
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

SURVEILLANCE	FREQUENCY
	72 hours after each completion of SR 3.1.4.4

SURVEILLANCE REQUIREMENTS

-----NOTE-----
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.2	Calibrate the trip units.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.3	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.4	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level—Low Low: ≥ -56.3 inches with time delay set to ≥ 7.2 seconds and ≤ 10.8 seconds; and b. Reactor Vessel Steam Dome Pressure-High: ≤ ~~1219~~ 1195 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.5	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.7 The Main Turbine Bypass System

LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

The following limits are made applicable:

a. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, ~~are made applicable.~~and

~~a.~~b. LCO 3.2.3, "Linear Heat Generation Rate (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.7.1 Verify one complete cycle of each main turbine bypass valve.	In accordance with the Surveillance Frequency Control Program

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. The LHGR for Specification 3.2.3.

4. Control Rod Block Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.

5. The OPRM setpoints for the trip function for SR 3.3.1.3.3.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

2. ~~Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."~~

3. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.

4. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel."

5. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2."

6. WCAP-15682-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application."

7. WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1."

9. WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to CENPD-287."

10. CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors."

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

12. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels, May 1995.

13. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998.

14. BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.

15. 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, March 1983.

16. XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.

17. 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, January 1987.

18. 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, October 1999.

19. EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.

20. EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," AREVA NP, September 2009.

21. ANP-10298PA Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," AREVA, March 2014.

22. ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.

(continued)

5.6 Reporting Requirements

5.6.5 ~~CORE OPERATING LIMITS REPORT (COLR)~~ (continued)

23. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.

24. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.

25. EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001.

26. EMF-2292(P)(A) Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000.

27. ANF-1358(P)(A) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005.

28. EMF-CC-074(P)(A) Volume 4 Revision 0, "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000.

The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

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

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

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

ATTACHMENT 4

Markup of Proposed Technical Specifications Bases Pages for DNPS
(For Information Only)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

The AREVA ACE/ATRIUM 10XM correlation is valid for critical power calculations at pressures > 300 psia and at bundle mass fluxes > 0.1 x 10^6 lb/hr-ft^2 (Reference 3). The AREVA SPCB correlation is valid for critical power calculations at pressures > 600 psia and bundle mass fluxes > 0.16 x 10^6 lb/hr-ft^2 (Reference 4). ~~The use of the Siemens Power Corporation correlation (ANFB) is valid for critical power calculations at pressures > 600 psia and bundle mass fluxes > 0.1 x 10^6 lb/hr-ft^2 (Refs. 2 and 3). The use of the General Electric (GE) Critical Power correlation (GEXL) is valid for critical power calculations at pressures > 785 psig and core flows > 10% (Ref. 4).~~ The use of the Westinghouse (WEC) Critical Power correlation (D4.1.1) is valid for critical power calculations at pressures > 362 psia and bundle mass fluxes > 0.23 x 10^6 lb/hr-ft^2 (Ref. 7). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/hr (approximately a mass velocity of 0.25 x 10^6 lb/hr-ft^2), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale critical power test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative. Although the ~~ANFB correlation is~~ACE/ATRIUM 10XM and SPCB correlations are valid at reactor steam dome pressures > 600 psia, and the Westinghouse correlation is valid at reactor steam dome pressures > 362 psia, application of the fuel cladding integrity SL at reactor steam dome pressure < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.2.2.1.

2. ~~ANF-524(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors, (as specified in Technical Specification 5.6.5).~~ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," (as specified in Technical Specification 5.6.5).

3. ~~ANF-1125(P)(A) and Supplements 1 and 2, ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation, (as specified in Technical Specification 5.6.5).~~ANP-10298PA Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," (as specified in Technical Specification 5.6.5).

4. ~~NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR) (as specified in Technical Specification 5.6.5)~~EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," (as specified in Technical Specification 5.6.5).

5. ~~ANF-1125(P)(A), Supplement 1, Appendix E, ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, Siemens Power Corporation, (as specified in Technical Specification 5.6.5).~~EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," (as specified in Technical Specification 5.6.5).

6. 10 CFR 50.67.

7. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2" (as specified in Technical Specification 5.6.5).

8. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (as specified in Technical Specification 5.6.5).

BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

assure isolation capability). These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated. This (ensuring components are OPERABLE) may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial operating condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 43). For the SDM

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. The control rods can be electrically disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note, which allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

D.1 and D.2

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ 10% RTP, the analyzed rod position sequence analysis (Refs. 6, ~~7, and~~ 8, and 9) requires inserted control rods not in compliance with the analyzed rod position sequence to be separated by at least two OPERABLE control rods in all directions, including the diagonal (i.e., all other control rods in a five-by-five array centered on the inoperable control rod are OPERABLE). Therefore, if two or more inoperable control rods are not in compliance with the analyzed rod position sequence and not separated by at least two OPERABLE control rods in all directions, action must be taken to restore compliance with the analyzed rod position sequence or restore the control rods to OPERABLE status. Condition D is modified by a Note indicating that the Condition is not applicable when > 10% RTP, since the analyzed rod position sequence is not required to be

(continued)

BASES

REFERENCES (continued)

5. UFSAR, Section 4.6.3.4.2.1.

6. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.

7. ~~NFSR-0091, Commonwealth Edison Topical Report, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, (as specified in Technical Specification 5.6.5).~~Not used

8. CENPD-284-P-A, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification."

9. 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," (as specified in Technical Specification 5.6.5).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1, 2, and 3.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, 3, 4, ~~5,~~ and 14. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 6), the fuel design limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Ref. 7). Generic evaluations (Refs. 8 and 9) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 10)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

and the calculated offsite doses will be well within the required limits (Ref. 11). Cycle specific CRDA analyses are performed that assume eight inoperable control rods with at least two cell separation and confirm fuel energy deposition is less than 280 cal/gm.

Control rod patterns analyzed in the cycle specific analyses follow predetermined sequencing rules (analyzed rod position sequence). The analyzed rod position sequence is applicable from the condition of all control rods fully inserted to 10% RTP (Ref~~s~~. ~~5 and~~ 14). The control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Cycle specific analyses ensure that the 280 cal/gm fuel design limit will not be violated during a CRDA under worst case scenarios. The cycle specific analyses (Refs. 1, 2, 3, 4, ~~5,~~ and 14) also evaluate the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods. Specific analysis may also be performed for atypical operating conditions (e.g., fuel leaker suppression).

When performing a shutdown of the plant, an optional rod position sequence (Ref. 13) may be used provided that all withdrawn control rods have been confirmed to be coupled. The rods may be inserted without the need to stop at intermediate positions since the possibility of a CRDA is eliminated by the confirmation that withdrawn control rods are coupled. When using the optional (Ref. 13) control rod sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved control rod insertion process.

In order to use the Reference 13 shutdown process, an extra check is required in order to consider a control rod to be "confirmed" to be coupled. This extra check ensures that no single operator error can result in an incorrect coupling check. For purposes of this shutdown process, the method for confirming that control rods are coupled varies depending on the position of the control rod in the core.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Details on this coupling confirmation requirement are provided in Reference 13. If the requirements for use of the control rod insertion process contained in Reference 13 are followed, the plant is considered in compliance with the rod position sequence as required by LCO 3.1.6. Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the analyzed rod position sequence. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the analyzed rod position sequence.
APPLICABILITY	In MODES 1 and 2, when THERMAL POWER is ≤ 10% RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > 10% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA (Refs. 4~~, 5,~~ and 14). In MODES 3 and 4, the reactor is shutdown and the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied, therefore, a CRDA is not postulated to occur. In MODE 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.
ACTIONS	A.1 and A.2 With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours.

(continued)

BASES

REFERENCES (continued)

5. ~~NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).~~Not used.

6. NUREG-0979, Section 4.2.1.3.2, April 1983.

7. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

8. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

9. NEDO-10527, "Rod Drop Accident Analysis for Large BWRs," (including Supplements 1 and 2), March 1972.

10. ASME, Boiler and Pressure Vessel Code.

11. 10 CFR 50.67.

12. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

13. NEDO-33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," July 2004.

14. CENPD-284-P-A, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification."

BASES

ACTIONS

B.1 (continued)

the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and periodically thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. ~~EMF-94-217(NP), "Boiling Water Reactor Licensing Methodology Summary," Revision 1, November 1995.~~EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model" (as specified in Technical Specification 5.6.5).
2. UFSAR, Chapter 4.
3. UFSAR, Chapter 6.
4. UFSAR, Chapter 15.
5. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (as specified in Technical Specification 5.6.5).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, ~~5, 6, and~~ 7, and 8. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow state ($MCPR_f$) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in UFSAR, Chapter 15 (Ref. 4).

Flow-dependent MPCR limits, MCPR(F), ensure that the Safety Limit MCPR (SLMCPR) is not violated during recirculation flow events. The design basis flow increase event is a slow-flow power increase event which is not terminated by scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. Flow runout events are simulated along a constant xenon flow control line assuming a quasi stead-state plant heat balance. The MCPR(F) limit is specified as an absolute value and is based on the MCPR safety limit. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

~~For GE methodology, above the power at which the scram is bypassed, bounding power-dependent trend functions have been developed. These trend functions, K(P), are used as multipliers to the rated MCPR operating limits to obtain the power-dependent MCPR limits, MCPR(P). Below the power at which the scram is automatically bypassed, the MCPR(P) limits are actual absolute Operating Limit MCPR (OLMCPR) values. The power-dependent limits are established to protect the core from plant transients other than core flow increases, including pressurization and local control rod withdrawal events.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

For Westinghouse methodology, the MCPR(P) limits are actual absolute OLMCPR values. The flow dependent limits are established to protect the safety limit. Although MCPR(P) limits are calculated, multipliers derived from these limits can be provided to support the COLR.

For AREVA methodology, the MCPR(P) limits are actual absolute OLMCPR values. The flow dependent limits are established to protect the safety limit.

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the appropriate $MCPR_f$ or the rated condition MCPR limit.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a low recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2

the specific scram speed distribution is consistent with that used in the transient analyses.

~~For GE methodology, SR 3.2.2.2 determines the value of T, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4) and Option B (realistic scram times) analyses. This determination of the parameter T for GE methodology must be performed once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in the actual scram speed distribution expected during the fuel cycle.~~

The Westinghouse and 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 an intermediate value 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. Note that the Westinghouse and AREVA methodologies do not support interpolation of the operational limits between scram speeds.~~The Westinghouse reload safety analysis results are performed using Nominal Scram Speed (NSS) 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 these scram 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 an intermediate value between the NSS and the Technical Specification Scram Speed (TSSS) control rod insertion times. In the event ISS times are exceeded, the operational limits for the TSSS are applied. Note that the Westinghouse methodology does not support interpolation of the operational limits between scram speeds.~~

REFERENCES

1. NUREG-0562, June 1979.
2. UFSAR, Chapter 4.
3. UFSAR, Chapter 6.
4. UFSAR, Chapter 15.

BASES

REFERENCES (continued)

5. ~~NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).~~Not used.

6. ~~NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR) (as specified in Technical Specification 5.6.5).~~Not used.

7. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (as specified in Technical Specification 5.6.5).

8. XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," (as specified in Technical Specification 5.6.5).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, (i.e., steady state), anticipated operational occurrences (AOOs)~~, and to ensure that the peak cladding temperature during the postulated design basis loss of coolant accident does not exceed the limits specified in 10 CFR 50.46~~. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations and anticipated operating conditions identified in References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design and establish LHGR limits are presented in References 1, 2, and 3~~, 4, 5, and 6~~. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20 and 50. A mechanism that could cause fuel damage during normal operations and operational transients and that is considered in fuel evaluations is a rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. ~~7~~4).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	includes allowances for short term transient excursions above the operating limit while still remaining within the ~~AOO~~ fuel design limits, plus an allowance for densification power spiking. Flow-dependent LHGR limits were designed to assure adherence to all fuel thermal-mechanical design bases in the case of a slow recirculation flow runout event. From the bounding overpowers, the limits were derived such that during these events, the peak transient LHGR would not exceed the fuel mechanical limits. Power-dependent LHGR limits are used to assure adherence to the fuel thermal-mechanical design bases at reduced power conditions. Incipient centerline melting of the fuel and plastic strain of the cladding are considered. Appropriate limits are selected based on the plant-specific transient analysis. These limits are derived to assure that peak transient LHGR for any transient is not increased above the fuel design bases. The LHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The LHGR (steady state) limit to accomplish this objective is specified in the COLR.
APPLICABILITY	The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 25% RTP.

(continued)

BASES (continued)

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER TO < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and periodically thereafter. They are compared to the LHGR limits (steady state) in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.2.3.2

Westinghouse reload analyses do not provide scram dependent LHGR limits. However, for AREVA, 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 used in the transient analyses.

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 an intermediate value 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. Note that the AREVA methodologies do not support interpolation of the operational limits between scram speeds.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Chapter 4.
2. UFSAR, Chapter 15.
3. ~~XN-NF-80-19(P)(A), Advanced Nuclear Fuel Methodology for Boiling Water Reactors.~~BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors" (as specified in Technical Specification 5.6.5).
~~4. XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.~~
~~5. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (as specified in Technical Specification 5.6.5).~~
~~6. EMF-85-74(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.~~
~~7~~4. NUREG-0800, Section 4.2.II.A.2(g), Revision 2, July 1981.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints and allowable values derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 30% RTP and a non-peripheral control rod is selected. Below this power level, or if a peripheral control rod is selected, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3).

2. Rod Worth Minimizer

The RWM enforces the analyzed rod position sequence to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, 7, ~~8,~~ and 14. The analyzed rod position sequence requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the analyzed rod position sequence are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional control rod sequence (Ref. 13) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 13 control rod insertion sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved control rod insertion process.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.9

LCO 3.1.3 and LCO 3.1.6 may require individual control rods to be bypassed (taken out of service) in the RWM to allow insertion of an inoperable control rod or correction of a control rod pattern not in compliance with the analyzed rod position sequence. With the control rods bypassed (taken out of service) in the RWM, the RWM will provide insert and withdraw blocks for bypassed control rods that are fully inserted and a withdraw block for bypassed control rods that are not fully inserted. To ensure the proper bypassing and movement of these affected control rods, a second licensed operator (Reactor Operator or Senior Reactor Operator) or other task qualified member of the technical staff (e.g., shift technical advisor or reactor engineer) must verify the bypassing and position of these control rods. Compliance with this SR allows the RWM to be OPERABLE with these control rods bypassed.

REFERENCES

1. UFSAR, Section 7.6.1.5.3.

2. UFSAR, Section 7.7.2.

3. UFSAR, Section 15.4.2.3.

4. UFSAR, Section 15.4.10.

5. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1 Exxon Nuclear Methodology for Boiling Water Reactor-Neutronics Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).

6. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (as specified in Technical Specification 5.6.5).

7. Letter to T.A. Pickens (BWROG) from G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.

8. ~~NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).~~Not used.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1.4 (continued)

measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the pump breakers is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel(s) would be inoperable.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 7.8.
2. UFSAR, Section 15.8
3. GENE-770-06-1-A, "Bases for Changes To Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications," December 1992.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 2.a. Reactor Vessel Water Level—Low Low (continued)

The Reactor Vessel Water Level—Low Low Allowable Value is chosen to allow time for the low pressure core flooding systems to activate and provide adequate cooling.

Four channels of CS Reactor Vessel Water Level—Low Low Function are only required to be OPERABLE when the CS or DG(s) are required to be OPERABLE to ensure that no single instrument failure can preclude CS and DG initiation. Also, four channels of the LPCI Reactor Vessel Water Level—Low Low Function are only required to be OPERABLE when the LPCI System is required to be OPERABLE to ensure no single instrument failure can preclude LPCI initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS—Shutdown," for Applicability Bases for the low pressure ECCS subsystems; LCO 3.8.1, "AC Sources—Operating"; and LCO 3.8.2, "AC Sources—Shutdown," for Applicability Bases for the DGs.

1.b, 2.b. Drywell Pressure—High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS and associated DGs are initiated upon receipt of the Drywell Pressure—High Function in order to minimize the possibility of fuel damage. The Drywell Pressure—High Function, along with the Reactor Water Level—Low Low Function, is directly assumed in the ~~small break LOCA analysis (Ref. 4)~~analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure switches that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

The Drywell Pressure—High Function is required to be OPERABLE when the ECCS or DG is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS Drywell Pressure—High Function are required to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.b. Drywell Pressure-High (continued)

possibility of fuel damage. The Drywell Pressure-High Function, along with the Reactor Water Level-Low Low Function, is directly assumed in the analysis of the recirculation line break (Ref. 2)~~small break LOCA analysis (Ref. 4)~~. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure switches that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure-High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for the Applicability Bases for the HPCI System.

3.c. Reactor Vessel Water Level-High

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Reactor Vessel Water Level-High Function signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs). The Reactor Vessel Water Level-High Function is not assumed in the plant specific accident and transient analyses. It was retained since it is a potentially significant contributor to risk.

Reactor Vessel Water Level-High signals for HPCI are initiated from two differential pressure transmitters from the medium range water level measurement instrumentation. Both signals are required in order to close the HPCI injection valve. This ensures that no single instrument failure can preclude HPCI initiation. The Reactor Vessel Water Level-High Allowable Value is chosen to prevent flow from the HPCI System from overflowing into the MSLs.

Two channels of Reactor Vessel Water Level-High Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.4 and SR 3.3.5.1.5 (continued)

adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.5.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety function.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 5.2.
2. UFSAR, Section 6.3.
3. UFSAR, Chapter 15.
4. ~~EMF-97-025(P), Revision 1, "LOCA Break Spectrum Analysis for Dresden Units 2 and 3," May 30, 1997.~~Not used.
5. NEDC-30936-P-A, "BWR Owners' Group Technical Specification Improvement Analyses for ECCS Actuation Instrumentation, Part 1 and Part 2," December 1988.

BASES

BACKGROUND (continued)

to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., 55 to 100% of RTP) without having to move control rods and disturb desirable flux patterns.

Each recirculation loop is manually started from the control room. The adjustable speed drive provides regulation of individual recirculation loop drive flows. The flow in each loop is manually controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). ~~The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was~~The AREVA LOCA analyses assume mismatched flow in the recirculation loops. In the LOCA analyses for non-AREVA fuel, the analyses assume that both loops are operating at the same flow prior to the accident. The non-AREVA LOCA analyses were reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgement. ~~The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the UFSAR.~~Mismatched flow has an insignificant impact on the fuel thermal margin during the abnormal operational transients which are analyzed in Chapter 15 of the UFSAR (Reference 2).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) and LINEAR HEAT GENERATION RATE (LHGR) requirements are modified accordingly (Ref. 1). For AREVA fuel, there is no requirement to modify the LHGR limits to support operation with only one operating recirculation loop.

The transient analyses in Chapter 15 of the UFSAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MINIMUM CRITICAL POWER RATIO (MCPR) requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) and the Rod Block Monitor Allowable Values is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR, LHGR, and MCPR limits for single loop operation are specified in the COLR. The APRM Flow Biased Neutron Flux-High Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." The Rod Block Monitor-Upscale Allowable Value is in LCO 3.3.2.1, "Control Rod Block Instrumentation."

Recirculation loops operating satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Alternatively, with only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), LHGR limits (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), APRM Flow Biased Neutron Flux-High Allowable Value (LCO 3.3.1.1), and the Rod Block Monitor-Upscale Allowable Value (LCO 3.3.2.1), as applicable, must be

(continued)

BASES

LCO (continued)

applied to allow continued operation consistent with the assumptions of Reference 1. For AREVA fuel, there is no requirement to modify the LHGR limits to support operation with only one operating recirculation loop.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1 and A.2

With no recirculation loops in operation, the probability of thermal-hydraulic oscillations is greatly increased. Therefore, action must be taken as soon as practicable to reduce power to assure stability concerns are addressed and place the unit in at least MODE 2 within 8 hours and to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and transients and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1 and C.1

With both recirculation loops operating but the flows not matched, the flows must be matched within 2 hours. If matched flows are not restored, the recirculation loop with the lower flow must be declared "not in operation," as required by Required Action B.1. This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing

(continued)

BASES

BACKGROUND (continued)

through a discharge line to a point below the minimum water level in the suppression pool. The safety valves discharge directly to the drywell.

In addition to the safety valves and S/RV, each unit is designed with four relief valves which actuate in the relief mode to control RCS pressure during transient conditions to prevent the need for safety valve actuation (except S/RV) following such transients. The relief valves are also located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The relief valves are of the Electromatic type, which are opened by automatic or manual switch actuation of a solenoid. The switch energizes the solenoid to actuate a plunger, which contacts the pilot valve operating lever, thereby opening the pilot valve. When the pilot valve opens, pressure under the main valve disc is vented. This allows reactor pressure to overcome main valve spring pressure, which forces the main valve disc downward to open the main valve. Two of the five relief valves are the low set relief valves and all of the relief valves, including the S/RV, are Automatic Depressurization System (ADS) valves. The low set relief requirements are specified in LCO 3.6.1.6, "Low Set Relief Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS—Operating."

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient. The ~~MSIV closure without direct scram on valve position event, a turbine trip with a failure of the turbine bypass system and without direct scram on turbine stop valve position event, and a load reject with a failure of the turbine bypass system and without direct scram on turbine control valve fast closure event~~limiting pressurization transients are evaluated each reload (Ref. 1). For the purpose of the analyses, all nine safety valves are assumed to operate in the safety mode. The relief valves and the relief function of the S/RV are not credited to function during this event. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.7 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 33.5% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of a nine valve manifold connected to the main steam lines between the main steam isolation valves and the main turbine stop valves. Each of the nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine Electrohydraulic Control System, as discussed in the UFSAR, Section 7.7.4 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves sequentially. When the bypass valves open, the steam flows from the main steam equalizing header to the bypass manifold through the bypass valve, to its bypass line, where an orifice further reduces the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the turbine trip, turbine generator load rejection and feedwater controller failure transients, as discussed in the UFSAR, Sections 15.2.3.2, 15.2.2.2, and 15.1.2 (Refs. 2, 3, and 4, respectively). The Main Turbine Bypass System is also assumed to function during other AOOs in which system pressurization may occur. Opening the bypass valves during the pressurization events mitigates the increase in reactor vessel pressure, which affects the MCPR and nodal power excursion during the event. An inoperable Main Turbine Bypass System may result in ~~an~~ MCPR and LHGR ~~penalty~~penalties.

The Main Turbine Bypass System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

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 (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 ~~is~~ are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure or reactor dome pressure as selected by the operator. This response is within the assumptions of the applicable analyses (Refs. 2, 3, and 4).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during the turbine generator load rejection, turbine trip, and feedwater controller failure transients. As discussed in the Bases for LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (~~one~~ two or more bypass valves inoperable as specified in the COLR), and the MCPR and LHGR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR and LHGR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status and the MCPR and LHGR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the turbine generator load rejection, turbine trip, and feedwater controller failure transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.15 (continued)

allowed since the main purpose of the Surveillance can be met by performing the test on either unit. If the DG fails one of these Surveillances, the DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit.

SR 3.8.1.16

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 13 seconds. The 13 second time is derived from the requirements of the accident analysis for responding to a design basis ~~large break~~ LOCA (Ref. 13). In addition, the DG is required to maintain proper voltage and frequency limits after steady state is achieved. The time for the DG to reach the steady state voltage and frequency limits is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by three Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The requirement that the diesel has operated for at least 2 hours at approximately full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing. To minimize testing of the common DG, Note 3 allows a single test of the common DG (instead of two tests, one for each unit) to satisfy the requirements for both units. This is allowed since the main purpose of the Surveillance can be met by performing the test on either unit. If the DG fails one of these Surveillances, the DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit.

(continued)

B 3.10 SPECIAL OPERATIONS

B 3.10.6 Control Rod Testing—Operating

BASES

BACKGROUND	The purpose of this Special Operations LCO is to permit control rod testing, while in MODES 1 and 2, by imposing certain administrative controls. Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), such that only the specified control rod sequences and relative positions required by LCO 3.1.6, "Rod Pattern Control," are allowed over the operating range from all control rods inserted to the low power setpoint (LPSP) of the RWM. The sequences effectively limit the potential amount and rate of reactivity increase that could occur during a control rod drop accident (CRDA). During these conditions, control rod testing is sometimes required that may result in control rod patterns not in compliance with the prescribed sequences of LCO 3.1.6. These tests include SDM demonstrations, control rod scram time testing, and control rod friction testing. This Special Operations LCO provides the necessary exemption to the requirements of LCO 3.1.6 and provides additional administrative controls to allow the deviations in such tests from the prescribed sequences in LCO 3.1.6.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, 3, and 4~~, and 5~~. CRDA analyses assume the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analyses. The RWM provides backup to operator control of the withdrawal sequences to ensure the initial conditions of the CRDA analyses are not violated. For special sequences developed for control rod testing, the initial control rod patterns assumed in the safety analysis of References 1, 2, 3, and 4~~, and 5~~ may not be preserved. Therefore special CRDA analyses are required to demonstrate that these special sequences will not result in unacceptable consequences, should a CRDA occur during the testing. These analyses, performed in accordance with an NRC approved methodology, are dependent on the specific test being performed.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.10.6.2 When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be met if SR 3.10.6.1 is satisfied.

REFERENCES

1. UFSAR, Section 15.4.10.
2. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1, Exxon Nuclear Methodology for Boiling Water Reactor Neutronics Methods for Design Analysis, (as specified in Technical Specification 5.6.5).
3. NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel, (as specified in Technical Specification 5.6.5).
4. Letter from T. Pickens (BWROG) to G.C. Lainas (NRC) "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.
5. ~~NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1, 2, 3, and 4~~, and 5~~ is applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1, 2, 3, and 4~~, and 5~~ may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, are required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1, 2, 3, and 4~~, and 5~~). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, the APRMs are also required to be OPERABLE (LCO 3.3.1.1, Functions 2.a and 2.d as though the reactor were in MODE 2. Because multiple control rods will be withdrawn and the reactor will potentially become critical, the approved control rod withdrawal sequence must be enforced by the RWM

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.7.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. Since the reactor is depressurized in MODE 5, there is insufficient reactor pressure to scram the control rods. Verification of charging water header pressure ensures that if a scram were required, capability for rapid control rod insertion would exist. The minimum pressure of 940 psig is well below the expected pressure of approximately 1500 psig while still ensuring sufficient pressure for rapid control rod insertion. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 15.4.10.
2. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1, Exxon Nuclear Methodology for Boiling Water Reactor Neutronics Methods for Design Analysis, (as specified in Technical Specification 5.6.5).
3. NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel, (as specified in Technical Specification 5.6.5).
4. Letter from T. Pickens (BWROG) to G.C. Lainas (NRC) "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.
5. ~~NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commomwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).~~

ATTACHMENT 5

Markup of Proposed Technical Specifications Bases Pages for QCNPS
(For Information Only)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

The AREVA ACE/ATRIUM 10XM correlation is valid for critical power calculations at pressures > 300 psia and at bundle mass fluxes > 0.1 x 10^6 lb/hr-ft^2 (Reference 3). The AREVA SPCB correlation is valid for critical power calculations at pressures > 600 psia and bundle mass fluxes > 0.16 x 10^6 lb/hr-ft^2 (Reference 4).~~The use of the Siemens Power Corporation correlation (ANFB) is valid for critical power calculations at pressures > 600 psia and bundle mass fluxes > 0.1 x 10^6 lb/hr-ft^2 (Refs. 2 and 3). The use of the General Electric (GE) Critical Power correlation (GEXL) is valid for critical power calculations at pressures > 785 psig and core flows > 10% (Ref. 4).~~ The use of the Westinghouse critical power correlation (D4.1.1) is valid for critical power calculations at pressures > 362 psia and bundle mass fluxes > 0.23 x 10^6 lb/hr-ft^2 (Ref. 8). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/hr (approximately a mass velocity of 0.25 x 10^6 lb/hr-ft^2), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale critical power test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative. ~~Although the ANFB correlation is~~ACE/ATRIUM 10XM and SPCB correlations are valid at reactor steam dome pressures > 600 psia, and the Westinghouse D4.1.1 correlation is valid at reactor steam dome pressures > 362 psia, application of the fuel cladding integrity SL at reactor steam dome pressure < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the fuel vendor's critical power correlation. References 2, 3, 4, 5, ~~6,~~ 8, and 9 describe the methodology used in determining the MCPR SL.

The fuel vendor's critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the fuel vendor's correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.2.1.

2. ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," (as specified in Technical Specification 5.6.5).~~ANF-524(P)(A), Revision 2, Supplement 1, Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, (as specified in Technical Specification 5.6.5).~~

3. ANP-10298PA Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," (as specified in Technical Specification 5.6.5).~~ANF-1125(P)(A) and Supplements 1 and 2, ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation, (as specified in Technical Specification 5.6.5).~~

4. EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," (as specified in Technical Specification 5.6.5).~~NEDE 24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR)" (as specified in Technical Specification 5.6.5).~~

5. EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," (as specified in Technical Specification 5.6.5).~~ANF-1125(P)(A), Supplement 1, Appendix E, ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, Siemens Power Corporation, (as specified in Technical Specification 5.6.5).~~

6. ~~EMF-1125(P)(A), Supplement 1, Appendix C, ANFB Critical Power Correlation Application for Coresident Fuel, Siemens Power Corporation, (as specified in Technical Specification 5.6.5).~~Not used.

7. 10 CFR 50.67.

8. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2" (as specified in Technical Specification 5.6.5).

9. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (as specified in Technical Specification 5.6.5).

BASES

ACTIONS (continued)

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. The control rods can be electrically disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note, which allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

D.1 and D.2

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ 10% RTP, the analyzed rod position sequence analysis (Refs. 6, ~~7~~8, and ~~8~~9) requires inserted control rods not in compliance with the analyzed rod position sequence to be separated by at least two OPERABLE control rods in all directions, including the diagonal (i.e., all other control rods in a five-by-five array centered on the inoperable control rod are OPERABLE). Therefore, if two or more inoperable control rods are not in compliance with the analyzed rod position sequence and not separated by at least two OPERABLE control rods in all directions, action must be taken to restore compliance with the analyzed rod position sequence or restore the control rods to OPERABLE status. Condition D is modified by a Note indicating that the Condition is not applicable when > 10% RTP, since the analyzed rod position sequence is not required to be

(continued)

BASES

REFERENCES (continued)

5. UFSAR, Section 4.6.3.4.2.1.

6. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.

7. ~~NFSR-0091, Commonwealth Edison Topical Report, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, (as specified in Technical Specification 5.6.5).~~Not used.

8. CENPD-284-P-A, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification."

~~8.~~9. 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," (as specified in Technical Specification 5.6.5).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1, 2, and 3.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, 3, 4, ~~5,~~ and 14. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 6), the fuel design limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Ref. 7). Generic evaluations (Refs. 8 and 9) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 10) and the calculated offsite doses will be well within the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

required limits (Ref. 11). Cycle specific CRDA analyses are performed that assume eight inoperable control rods with at least two cell separation and confirm fuel energy deposition is less than 280 cal/gm.

Control rod patterns analyzed in the cycle specific analyses follow predetermined sequencing rules (analyzed rod position sequence). The analyzed rod position sequence is applicable from the condition of all control rods fully inserted to 10% RTP (Ref~~s~~. ~~5 and~~ 14). The control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Cycle specific analyses ensure that the 280 cal/gm fuel design limit will not be violated during a CRDA under worst case scenarios. The cycle specific analyses (Refs. 1, 2, 3, 4, ~~5,~~ and 14) also evaluate the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods. Specific analysis may also be performed for atypical operating conditions (e.g., fuel leaker suppression).

When performing a shutdown of the plant, an optional rod position sequence (Ref. 13) may be used provided that all withdrawn control rods have been confirmed to be coupled. The rods may be inserted without the need to stop at intermediate positions since the possibility of a CRDA is eliminated by the confirmation that withdrawn control rods are coupled. When using the optional (Ref. 13) control rod sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved control rod insertion process.

In order to use the Reference 13 shutdown process, an extra check is required in order to consider a control rod to be "confirmed" to be coupled. This extra check ensures that no single operator error can result in an incorrect coupling check. For purposes of this shutdown process, the method for confirming that control rods are coupled varies depending on the position of the control rod in the core. Details on this coupling confirmation requirement are

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	provided in Reference 13. If the requirements for use of the control rod insertion process contained in Reference 13 are followed, the plant is considered in compliance with the rod position sequence as required by LCO 3.1.6. Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the analyzed rod position sequence. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the analyzed rod position sequence.
APPLICABILITY	In MODES 1 and 2, when THERMAL POWER is ≤ 10% RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > 10% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA (Refs. 4~~, 5,~~ and 14). In MODES 3 and 4, the reactor is shutdown and the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied, therefore, a CRDA is not postulated to occur. In MODE 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.
ACTIONS	<u>A.1 and A.2</u> With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

When nine or more OPERABLE control rods are not in compliance with the analyzed rod position sequence, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is periodically verified to be in compliance with the analyzed rod position sequence to ensure the assumptions of the CRDA analyses are met. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The RWM provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. UFSAR, Section 15.4.10.
2. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1 Exxon Nuclear Methodology for Boiling Water Reactor-Neutronics Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).
3. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (as specified in Technical Specification 5.6.5).
4. Letter from T.A. Pickens (BWROG) to G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.
5. ~~NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).~~Not used.

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (as specified in Technical Specification 5.6.5).
2. UFSAR, Chapter 4.
3. UFSAR, Chapter 6.
4. UFSAR, Chapter 15.
5. EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model" (as specified in Technical Specification 5.6.5). ~~EMF-94-217(NP), Revision 1, "Boiling Water Reactor Licensing Methodology Summary," November 1995.~~
6. UFSAR, Section 6.3.3.2.2.4.
7. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (as specified in Technical Specification 5.6.5).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, ~~6,~~ and 7. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow state

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

($MCPR_f$) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in UFSAR, Chapter 15 (Ref. 5).

Flow-dependent MPCR limits, MCPR(F), ensure that the Safety Limit MCPR (SLMCPR) is not violated during recirculation flow events. The design basis flow increase event is a slow-flow power increase event which is not terminated by scram, but which stablizes at a new core power corresponding to the maximum possible core flow. Flow runout events are simulated along a constant xenon flow control line assuming a quasi steady-state plant heat balance. The MCPR(F) limit is specified as an absolute value and protects the MCPR Safety Limit. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

~~For GE methodology above the power at which the scram is bypassed, bounding power-dependent trend functions have been developed. These trend functions, K(P), are used as multipliers to the rated MCPR operating limits to obtain the power-dependent MCPR limit, MCPR(P).~~

~~Below the power at which the scram is automatically bypassed, the MCPR(P) limits are actual absolute Operating Limit MCPR (OLMCPR) values. The power dependent limits are established to protect the core from plant transients other than core flow increases, including pressurization and local control rod withdrawal events.~~

For Westinghouse methodology, the MCPR(P) limits are actual absolute OLMCPR values. The flow dependent limits are established to protect the safety limit. Although MCPR(P) limits are calculated, multipliers derived from these limits can be provided to support the COLR.

For AREVA methodology, the MCPR(P) limits are actual absolute OLMCPR values. The flow dependent limits are established to protect the safety limit.

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the appropriate $MCPR_f$ or the rated condition MCPR limit.

(continued)

BASES

ACTIONS (continued)

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and periodically thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.2.2.2

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 used in the transient analyses.

~~For GE methodology, SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4) and Option B (realistic scram times) analyses. This determination of the parameter τ for GE methodology must be performed once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.2 (continued)

~~during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in the actual scram speed distribution expected during the fuel cycle.~~

The Westinghouse and 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 an intermediate value 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. Note that the Westinghouse and AREVA methodologies do not support interpolation of the operational limits between scram speeds.~~The Westinghouse reload safety analysis results are performed using Nominal Scram Speed (NSS) 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 these scram 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 an intermediate value between the NSS and the Technical Specifications Scram Speed (TSSS) control rod insertion times. In the event ISS times are exceeded, the operational limits for the TSSS are applied. Note that the Westinghouse methodology does not support interpolation of the operational limits between scram speeds.~~

REFERENCES

1. NUREG-0562, June 1979.

2. XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," (as specified in Technical Specification 5.6.5). ~~NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (as specified in Technical Specification 5.6.5).~~

3. UFSAR, Chapter 4.

4. UFSAR, Chapter 6.

5. UFSAR, Chapter 15.

6. ~~NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).~~Not used.

7. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (as specified in Technical Specification 5.6.5).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs) ~~and to ensure that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46~~. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations and anticipated operating conditions identified in References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design and establish LHGR limits are presented in References 1, 2, and 3~~, 4, 5, and 6~~. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20 and 50. A mechanism that could cause fuel damage during normal operations and operational transients and that is considered in fuel evaluations is a rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. ~~7~~4).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient excursions

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	above the operating limit while still remaining within the ~~AOO~~ fuel design limits, plus an allowance for densification power spiking. Flow-dependent LHGR limits were designed to assure adherence to all fuel thermal-mechanical design bases in the case of a slow recirculation flow runout event. From the bounding overpowers, the limits were derived such that during these events, the peak transient LHGR would not exceed the fuel mechanical limits. Power-dependent LHGR limits are used to assure adherence to the fuel thermal-mechanical design bases at reduced power conditions. Incipient centerline melting of the fuel and plastic strain of the cladding are considered. Appropriate limits are selected based on the plant-specific transient analysis. These limits are derived to assure that peak transient LHGR for any transient is not increased above the fuel design bases. The LHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.
APPLICABILITY	The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 25% RTP.

(continued)

BASES (continued)

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER TO < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and periodically thereafter. They are compared to the LHGR limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.2.3.2

For Westinghouse fuel, the reload analyses do not provide scram dependent LHGR limits. However, for AREVA, 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 used in the transient analyses.

(continued)

BASES (continued)

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 an intermediate value 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. Note that the AREVA methodologies do not support interpolation of the operational limits between scram speeds.

REFERENCES

1. UFSAR, Chapter 4.
2. UFSAR, Chapter 15.

(continued)

BASES (continued)

REFERENCES (continued)

3. BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors" (as specified in Technical Specification 5.6.5). ~~XN-NF-80-19(P)(A), Advanced Nuclear Fuel Methodology for Boiling Water Reactors.~~

~~4. XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.~~

~~5. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (as specified in Technical Specification 5.6.5).~~

~~6. EMF-85-74(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.~~

~~7~~4. NUREG-0800, Section 4.2.II.A.2(g), Revision 2, July 1981.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	1. Rod Block Monitor (continued)

<u>1. Rod Block Monitor</u> (continued)

from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 30% RTP and a non-peripheral control rod is selected. Below this power level, or if a peripheral control rod is selected, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3).

<u>2. Rod Worth Minimizer</u>

The RWM enforces the analyzed rod position sequence to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, 7, ~~8,~~ and 13. The analyzed rod position sequence requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the analyzed rod position sequence are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional control rod sequence (Ref. 12) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 12 control rod insertion sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved control rod insertion process.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.9

LCO 3.1.3 and LCO 3.1.6 may require individual control rods to be bypassed (taken out of service) in the RWM to allow insertion of an inoperable control rod or correction of a control rod pattern not in compliance with the analyzed rod position sequence. With the control rods bypassed (taken out of service) in the RWM, the RWM will provide insert and withdraw blocks for bypassed control rods that are fully inserted and a withdraw block for bypassed control rods that are not fully inserted. To ensure the proper bypassing and movement of these affected control rods, a second licensed operator (Reactor Operator or Senior Reactor Operator) or other task qualified member of the technical staff (e.g., shift technical advisor or reactor engineer) must verify the bypassing and position of these control rods. Compliance with this SR allows the RWM to be OPERABLE with these control rods bypassed.

REFERENCES

1. UFSAR, Section 7.6.1.5.3.
2. UFSAR, Section 7.7.2.
3. UFSAR, Section 15.4.2.3.
4. UFSAR, Section 15.4.10.
5. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1 Exxon Nuclear Methodology for Boiling Water Reactor-Neutronics Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).
6. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (as specified in Technical Specification 5.6.5).
7. Letter to T.A. Pickens (BWROG) from G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.
8. ~~NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).~~Not used.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1.4 (continued)

measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the pump breakers is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if an ASD feed breaker or ASD emergency stop circuit is incapable of operating, the associated instrument channel(s) would be inoperable.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 7.8.
2. UFSAR, Section 15.8
3. GENE-770-06-1-A, "Bases for Changes To Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications," December 1992.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.b. Drywell Pressure—High (continued)

possibility of fuel damage. The Drywell Pressure—High Function, along with the Reactor Water Level—Low Low Function, is directly assumed in the ~~small break LOCA analysis~~ analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure switches that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure—High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for the Applicability Bases for the HPCI System.

3.c. Reactor Vessel Water Level—High

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Reactor Vessel Water Level—High Function signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs). The Reactor Vessel Water Level—High Function is not assumed in the plant specific accident and transient analyses. It was retained since it is a potentially significant contributor to risk.

Reactor Vessel Water Level—High signals for HPCI are initiated from two differential pressure instruments from the narrow range water level measurement instrumentation. Both signals are required in order to close the HPCI injection valve. This ensures that no single instrument failure can preclude HPCI initiation. The Reactor Vessel Water Level—High Allowable Value is chosen to prevent flow from the HPCI System from overflowing into the MSLs.

Two channels of Reactor Vessel Water Level—High Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

(continued)

BASES

BACKGROUND (continued)

to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., 55 to 100% of RTP) without having to move control rods and disturb desirable flux patterns.

Each recirculation loop is manually started from the control room. The ASD provides regulation of individual recirculation loop drive flows. The flow in each loop is manually controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). ~~The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was~~The AREVA LOCA analyses assume mismatched flow in the recirculation loops. In the LOCA analyses for non-AREVA fuel, the analyses assume that both loops are operating at the same flow prior to the accident. The non-AREVA LOCA analyses were reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgement. ~~The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the UFSAR.~~Mismatched flow has an insignificant impact on the fuel thermal margin during abnormal operational transients which are analyzed in Chapter 15 of the UFSAR (Reference 2).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the Average Planar Linear Heat Generation Rate (APLHGR) and Linear Heat Generation Rate (LHGR) requirements are modified accordingly (Ref. 3). For AREVA fuel, there is no requirement to modify the LHGR limits to support operation with only one operating recirculation loop.

The transient analyses in Chapter 15 of the UFSAR have also been performed for single recirculation loop operation (Ref. 4) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) and the Rod Block Monitor Allowable Values is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR, LHGR, and Minimum Critical Power Ratio (MCPR) limits for single loop operation are specified in the COLR. The APRM Flow Biased Neutron Flux-High Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." The Rod Block Monitor-Upscale Allowable Value is in LCO 3.3.2.1, "Control Rod Block Instrumentation."

Recirculation loops operating satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Alternatively, with only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), LHGR limits (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), APRM Flow Biased Neutron Flux-High Allowable Value (LCO 3.3.1.1), and the Rod Block Monitor-Upscale Allowable Value (LCO 3.3.2.1), as applicable, must be applied to allow continued operation consistent with the assumptions of Reference 3. For AREVA fuel, there is no requirement to modify the LHGR limits to support operation with only one operating recirculation loop.

(continued)

BASES

BACKGROUND (continued)

In addition to the safety valves and S/RV, each unit is designed with four relief valves which actuate in the relief mode to control RCS pressure during transient conditions to prevent the need for safety valve actuation (except S/RV) following such transients. The relief valves are also located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. These valves are sized by assuming a turbine trip, a coincident scram and a failure of the turbine bypass system. Four of the relief valves are of the Electromatic type, which are opened by automatic or manual switch actuation of a solenoid. The switch energizes the solenoid to actuate a plunger, which contacts the pilot valve operating lever, thereby opening the pilot valve. When the pilot valve opens, pressure under the main valve disc is vented. This allows reactor pressure to overcome main valve spring pressure, which forces the main valve disc downward to open the main valve. Two of the five relief valves are the low set relief valves and all of the relief valves, including the S/RV, are Automatic Depressurization System (ADS) valves. The low set relief requirements are specified in LCO 3.6.1.6, "Low Set Relief Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS—Operating."

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient. The ~~MSIV closure without direct scram on valve position event, a turbine trip with a failure of the turbine bypass system and without direct scram on turbine stop valve position event, and a load reject with a failure of the turbine bypass system and without direct scram on turbine control valve fast closure event~~limiting pressurization transients are evaluated each reload (Ref. 1). For the purpose of the analyses, nine safety valves (including the S/RV) are assumed to operate in the safety mode. The relief valves are not credited to function during this event. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event.

For the turbine trip or generator load rejection with Main Turbine Bypass System failure described in References 2 and 3, respectively, the relief

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.7 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 33.3% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of a nine valve manifold connected to the main steam lines between the main steam isolation valves and the main turbine stop valves. Each of the nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine Electro-Hydraulic Control System, as discussed in the UFSAR, Section 7.7.4 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves sequentially. When the bypass valves open, the steam flows from the main steam equalizing header to the bypass manifold through the bypass valve, to its bypass line, where an orifice further reduces the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the turbine trip, turbine generator load rejection and feedwater controller failure transients, as discussed in the UFSAR, Sections 15.2.3.2, 15.2.2.2, and 15.1.2 (Refs. 2, 3, and 4, respectively). The Main Turbine Bypass System is also assumed to function during other AOOs in which system pressurization may occur. Opening the bypass valves during the pressurization events mitigates the increase in reactor vessel pressure, which affects the MCPR and nodal power excursion during the event. An inoperable Main Turbine Bypass System may result in ~~an~~ MCPR ~~penalty~~and LHGR penalties.

The Main Turbine Bypass System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

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 (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 ~~is~~ are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analyses (Refs. 2, 3, and 4).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during the turbine generator load rejection, turbine trip, and feedwater controller failure transients. As discussed in the Bases for LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

Also, between ≥ 25% RTP and the Main Turbine's Power/Load Unbalance relay trip setpoint, the main generator load reject trip must be OPERABLE, which requires the Unit Auxiliary Transformer (UAT) to be connected to at least one 4 kV bus.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (~~one~~ two or more bypass valves inoperable as specified in the COLR), and the MCPR and LHGR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR and LHGR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status and the MCPR and LHGR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the turbine generator load rejection, turbine trip, and feedwater controller failure transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.15 (continued)

purpose of the Surveillance can be met by performing the test on either unit. If the DG fails one of these Surveillances, the DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit.

SR 3.8.1.16

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 13 seconds. The 13 second time is derived from the requirements of the accident analysis for responding to a design basis ~~large break~~ LOCA (Ref. 15). In addition, the DG is required to maintain proper voltage and frequency limits after steady state is achieved. The time for the DG to reach the steady state voltage and frequency limits is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by three Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The requirement that the diesel has operated for at least 2 hours at approximately full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing. To minimize testing of the common DG, Note 3 allows a single test of the common DG (instead of two tests, one for each unit) to satisfy the requirements for both units. This is allowed since the main purpose of

(continued)

B 3.10 SPECIAL OPERATIONS

B 3.10.6 Control Rod Testing—Operating

BASES

BACKGROUND	The purpose of this Special Operations LCO is to permit control rod testing, while in MODES 1 and 2, by imposing certain administrative controls. Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), such that only the specified control rod sequences and relative positions required by LCO 3.1.6, "Rod Pattern Control," are allowed over the operating range from all control rods inserted to the low power setpoint (LPSP) of the RWM. The sequences effectively limit the potential amount and rate of reactivity increase that could occur during a control rod drop accident (CRDA). During these conditions, control rod testing is sometimes required that may result in control rod patterns not in compliance with the prescribed sequences of LCO 3.1.6. These tests include SDM demonstrations, control rod scram time testing, and control rod friction testing. This Special Operations LCO provides the necessary exemption to the requirements of LCO 3.1.6 and provides additional administrative controls to allow the deviations in such tests from the prescribed sequences in LCO 3.1.6.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, 3, and 4~~, and 5~~. CRDA analyses assume the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analyses. The RWM provides backup to operator control of the withdrawal sequences to ensure the initial conditions of the CRDA analyses are not violated. For special sequences developed for control rod testing, the initial control rod patterns assumed in the safety analysis of References 1, 2, 3, and 4~~, and 5~~ may not be preserved. Therefore special CRDA analyses are required to demonstrate that these special sequences will not result in unacceptable consequences, should a CRDA occur during the testing. These analyses, performed in accordance with an NRC approved methodology, are dependent on the specific test being performed.

(continued)

BASES

APPLICABILITY (continued)

Special Operations LCO are necessary to perform special tests that are not in conformance with the prescribed sequences of LCO 3.1.6.

While in MODES 3 and 4, control rod withdrawal is only allowed if performed in accordance with Special Operations LCO 3.10.2, "Single Control Rod Withdrawal—Hot Shutdown," or Special Operations LCO 3.10.3, "Single Control Rod Withdrawal—Cold Shutdown," which provide adequate controls to ensure that the assumptions of the safety analysis of References 1, 2, 3, and 4~~, and 5~~ are satisfied. During these Special Operations and while in MODE 5, the one-rod-out interlock (LCO 3.9.2, "Refuel Position One-Rod-Out Interlock,") and scram functions (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and LCO 3.9.5, "Control Rod OPERABILITY—Refueling"), or the added administrative controls prescribed in the applicable Special Operations LCOs, provide mitigation of potential reactivity excursions.

ACTIONS

A.1

With the requirements of the LCO not met (e.g., the control rod pattern is not in compliance with the special test sequence, the sequence is improperly loaded in the RWM) the testing is required to be immediately suspended. Upon suspension of the special test, the provisions of LCO 3.1.6 are no longer excepted, and appropriate actions are to be taken to restore the control rod sequence to the prescribed sequence of LCO 3.1.6, or to shut down the reactor, if required by LCO 3.1.6.

SURVEILLANCE REQUIREMENTS

SR 3.10.6.1

With the special test sequence not programmed into the RWM, a second licensed operator (Reactor Operator or Senior Reactor Operator) or other task qualified member of the technical staff (e.g., shift technical advisor or reactor engineer) is required to verify conformance with the approved sequence for the test. This verification must be performed during control rod movement to prevent deviations from the specified sequence. A Note is added to indicate that this Surveillance does not need to be met if SR 3.10.6.2 is satisfied.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.6.2

When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be met if SR 3.10.6.1 is satisfied.

REFERENCES

1. UFSAR, Section 15.4.10.
2. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1, Exxon Nuclear Methodology for Boiling Water Reactor Neutronics Methods for Design Analysis, (as specified in Technical Specification 5.6.5).
3. NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel, (as specified in Technical Specification 5.6.5).
4. Letter from T. Pickens (BWROG) to G.C. Lainas (NRC) "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.
~~5. NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1, 2, 3, and 4~~, and 5~~ are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1, 2, 3, and 4~~, and 5~~ may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, are required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1, 2, 3, and 4~~, and 5~~). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, the APRMs are also required to be OPERABLE (LCO 3.3.1.1, Functions 2.a and 2.d as though the reactor were in MODE 2. Because multiple control rods will be withdrawn and the reactor will potentially become critical, the approved control rod withdrawal sequence must be enforced by the RWM

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.7.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. Since the reactor is depressurized in MODE 5, there is insufficient reactor pressure to scram the control rods. Verification of charging water header pressure ensures that if a scram were required, capability for rapid control rod insertion would exist. The minimum pressure of 940 psig is well below the expected pressure of approximately 1500 psig while still ensuring sufficient pressure for rapid control rod insertion. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 15.4.10.
2. XN-NF-80-19(P)(A), Volume 1, Supplement 2, Section 7.1, Exxon Nuclear Methodology for Boiling Water Reactor Neutronics Methods for Design Analysis, (as specified in Technical Specification 5.6.5).
3. NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel, (as specified in Technical Specification 5.6.5).
4. Letter from T. Pickens (BWROG) to G.C. Lainas, NRC, "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.
5. ~~NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).~~

ATTACHMENT 6

Boiling Water Reactor Licensing Methodology Compendium

AREVA

ANP-2637
Revision 6

Boiling Water Reactor Licensing Methodology Compendium

October 2014

AREVA Inc.

ANP-2637
Revision 6

Boiling Water Reactor
Licensing Methodology Compendium

Nature of Changes

Item	Paragraph or Page(s)	Description and Justification
Revision 6 changes		
1.	Section 2.2.12 Page 2-12	Revised last sentence of the first paragraph to read: "When the criteria are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose calculations required by 10 CFR 100 (Reference 4) or 10 CFR 50.67 (Reference 5) and Regulatory Guide 1.183 (Reference 6) as appropriate."
2.	Section 5.2.2 Page 5-10	Revised last sentence of the second paragraph to read: "If the minimum CPR during the event remains above the safety limit MCPR, the dose calculation is not needed since operation at or above the safety limit MCPR meets the requirements of less than a small fraction of the 10 CFR 100 or 10 CFR 50.67 and Regulatory Guide 1.183 dose limits as appropriate."
3.	Section 5.2.2 Page 5-10	Revised second sentence of the third paragraph to read: "For an infrequent event, the dose calculation result must remain below a small fraction (10%) of the 10 CFR 100 or 10 CFR 50.67 and Regulatory Guide 1.183 limits as appropriate. For a limiting fault/design basis accident, the dose calculation result must not exceed 10 CFR 100 or 10 CFR 50.67 and Regulatory Guide 1.183 limits as appropriate."
4.	Section 6.0 Page 6-1	Removed revision 4.2 from the SCALE reference since other version of SCALE may be used.
5.	Section 7.0 Page 7-1	Added References 5 and 6.
6.	Section 7.0 Page 7-2	Removed reference 19. References 4 through 41 were renumbered.

Item	Paragraph or Page(s)	Description and Justification
Revision 5 changes		
1.	Throughout	Minor editorial changes for consistency and accuracy.
2.	Page 1-9	Table 1-2 updated report numbers in References 4-6 and 4-7.
3.	Page 3-14	Modified description of SE implementation to reflect the removal of the 10% penalty.
4.	Pages 4-14 and 4-15	Items 4-6 and 4-7 updated.
5.	Pages 5-25 and 5-26	Updated Implementation of SER Restrictions and Observations for NRC-accepted topical reports Reference 5-8.
6.	Page 5-27	Updated Observations for NRC-accepted topical reports Reference 5-9.
7.	Page 7-4	Added Reference 41.

Changes are further identified with (|)
in the right-hand margin.

Contents

List of Tables

Nomenclature

Acronym	Definition
ANF	Advanced Nuclear Fuels
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
BOC	Beginning-of-Cycle
BWR	Boiling Water Reactors
CHF	Critical Heat Flux
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRDA	Control Rod Drop Accident
DIVOM	Delta CPR over Initial Versus Oscillation Magnitude
ECCS	Emergency Core Cooling System
EFW	Expanded Flow Window
ENC	Exxon Nuclear Company
EO-III	Enhanced Option III
EOC	End-of-Cycle
EOL	End-of-Life
EPU	Extended Power Uprate
FCTF	Fuel Cooling Test Facility
FDL	Fuel Design Limit
FSAR	Final Safety Analysis Report
FWHOOS	Feedwater Heater Out of Service
GDC	General Design Criteria
HPCI	High Pressure Coolant Injection
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate
$LHGRFAC_f$	Flow Dependent LHGR Multiplier
$LHGRFAC_p$	Power Dependent LHGR Multiplier
LOCA	Loss-of-Coolant Accident

Nomenclature *(continued)*

Acronym	Definition
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
$MCPR_f$	Flow-Dependent Minimum Critical Power Ratio
$MCPR_p$	Power-Dependent Minimum Critical Power Ratio
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
MEOD	Maximum Extended Operating Domain
MSIV	Main Steam Isolation Valve
MWR	Metal-Water Reaction
NRC	Nuclear Regulatory Commission, U.S.
OLMCPR	Operating Limit MCPR
OPRM	Oscillation Power Range Monitor
PA	Postulated Accident
PAPT	Protection Against the Power Transient
PBDA	Period Based Detection Algorithm
PCT	Peak Cladding Temperature
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RIA	Reactivity Initiated Accident
RPS	Recirculation Pump Seizure
SAFDL	Specified Acceptable Fuel Design Limit
SER	Safety Evaluation Report
SLMCPR	Safety Limit Minimum Critical Power Ratio
SPT	Stability Protection Trip
SRP	Standard Review Plan
TER	Technical Evaluation Report
TR	Topical Report
TS	Technical Specification

Abstract

This report is a compendium of AREVA methodologies and design criteria that are described in topical reports that the NRC has found acceptable for referencing in boiling water reactor (BWR) licensing applications. This compendium provides a concise, organized source for NRC-approved BWR topical reports.

The methodologies and topical reports addressed in this report are designed to give BWR licensees using AREVA fuel the methodologies needed to conform to their original licensing bases and to meet cycle-specific parameter limits that have been established using NRC-approved methodologies. These methodologies may also be used to predict changes to limits consistent with all applicable limits of the plant safety analysis that are addressed in the FSAR.

1.0 INTRODUCTION

This report is a compendium of AREVA Inc.* (AREVA) methodologies and design criteria, which are described in topical reports that the NRC has found acceptable for referencing in boiling water reactor (BWR) licensing applications. This compendium provides a concise, organized source for BWR topical reports. It presents information about the application of each topical report, the associated safety evaluation report (SER) and its conclusions and restrictions for each topical report, the relationships among the topical reports, and, for certain methodologies, descriptions of their unique characteristics or applications. Compliance with the SER restrictions is assured by implementing them within the engineering guidelines or by incorporating them into the computer codes.

The methods and topical reports addressed herein are designed to give BWR licensees using AREVA fuel the methodologies needed to conform to their original licensing bases and to meet "…cycle-specific parameter limits that have been established using an NRC-approved methodology…," as stated in Generic Letter 88-16. These methodologies may also be used to predict "…changes [to limits]…consistent with all applicable limits of the plant safety analysis that are addressed in the [updated] final safety analysis report ([U]FSAR)." Additionally, these methodologies are used to demonstrate that AREVA fuel is compatible with co-resident fuel.

The organization of this report parallels the major sections of the Standard Review Plan (SRP) (Reference 1) that apply to reload fuel, specifically, 4.2 Fuel System Design, 4.3 Nuclear Design, 4.4 Thermal and Hydraulic Design of Chapter 4 Reactor, and all appropriate sub-chapters of Chapter 15 Accident Analysis. Table 1-1 includes a list of all the SRP numbers addressed by AREVA BWR methodologies. Table 1-2 provides a list of topical reports that are used by AREVA to support operation of BWRs. Table 1-2 also provides an index to topical reports that may be used to establish operating limits

* AREVA Inc. is an AREVA Company

reported in the core operating limits reports (COLR) and that may be referenced in the *technical specifications.*

There are two styles for citations of references used herein. References to an approved methodology addressed within Section 2.0, 3.0, 4.0, and 5.0 are cited as “Reference section number-number (see Table 1-2 for a list of References).” Other supporting references found in Section 7.0 are cited by the reference number.

Table 1-1 SRP No. Addressed by AREVA Methodologies

SRP No.	Chapter 4 Reactor
4.2	Fuel System Design
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design
SRP No.	**Chapter 15 Accident Analysis**
15.1.1 – 15.1.3	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow (AOO)
15.2.1 – 15.2.5	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed) (AOO)
15.2.7	Loss of Normal Feedwater Flow (AOO)
15.3.1 – 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions (AOO)
15.3.3 – 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break (Postulated Accident (PA))
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power (AOO)
15.4.4 – 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate (AOO)
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (PA)
15.4.9	Spectrum of Rod Drop Accidents (BWR) (PA)
15.4.9a	Radiological Consequences or Rod Drop Accident (BWR) (PA)
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory (AOO)

Table 1-1 SRP No. Addressed by AREVA Methodologies *(Continued)*

SRP No.	**Chapter 15 Accident Analysis** *(Continued)*
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve (AOO)
15.6.5	Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary (PA)
15.7.4	Radiological Consequences of Fuel Handling Accidents (PA)
15.8	Anticipated Transients without Scram

Table 1-2 Reference Index

Reference No.	Methodology	Page No.(s)	Typically Referenced in Core Operating Limits Report
2-1	XN-NF-79-56(P)(A) Revision 1 and Supplement 1, "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," Exxon Nuclear Company, November 1981.	2-22	
2-2	XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, October 1983. (Base document not approved.)	2-23	
2-3	XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.	2-24	yes
2-4	XN-NF-81-51(P)(A), "LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, May 1986.	2-25	
2-5	XN-NF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Exxon Nuclear Company, August 1986.	2-26	
2-6	XN-NF-85-67(P)(A) Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1986.	2-27	yes
2-7	XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986.	2-29	

Table 1-2 Reference Index *(Continued)*

Reference No.	Methodology	Page No.(s)	Typically Referenced in Core Operating Limits Report
2-8	XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986.	2-30	
2-9	XN-NF-82-06(P)(A) Supplement 1 Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Supplement 1, "Extended Burnup Qualification of ENC 9x9 BWR Fuel," Advanced Nuclear Fuels Corporation, May 1988.	2-31	
2-10	ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.	2-32	yes
2-11	ANF-90-82(P)(A) Revision 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995.	2-34	
2-12	EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998.	2-35	yes
2-13	EMF-93-177(P)(A) Revision 1, "Mechanical Design for BWR Fuel Channels," Framatome ANP, August 2005.	2-37	

Table 1-2 Reference Index *(Continued)*

Reference No.	Methodology	Page No.(s)	Typically Referenced in Core Operating Limits Report
2-14	BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.	2-39	yes
3-1	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, March 1983.	3-6	yes
3-2	XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.	3-8	yes
3-3	EMF-CC-074(P)(A) Volume 1, "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2 "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report," Siemens Power Corporation, July 1994.	3-9	
3-4	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, October 1999.	3-10	yes

Table 1-2 Reference Index *(Continued)*

Reference No.	Methodology	Page No.(s)	Typically Referenced in Core Operating Limits Report
3-5	EMF-CC-074(P)(A) Volume 4, Revision 0, "BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000.	3-12	yes
3-6	BAW-10255PA Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," AREVA NP, May 2008	3-14	
3-7	ANP-10262PA Revision 0, "Enhanced Option III Long Term Stability Solution," AREVA NP, May 2008	3-15	Yes, EO-III Applications only
4-1	XN-NF-79-59(P)(A), "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, November 1983.	4-7	
4-2	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, January 1987.	4-8	yes
4-3	ANF-524(P)(A) Revision 2 and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation, November 1990.	4-10	yes

Table 1-2 Reference Index *(Continued)*

Reference No.	Methodology	Page No.(s)	Typically Referenced in Core Operating Limits Report
4-4	EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.	4-11	yes
4-5	EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," AREVA NP, September 2009.	4-12	yes
4-6	ANP-10249P-A Revision 2 "ACE/ATRIUM-10 Critical Power Correlation," AREVA, March 2014	4-14	yes
4-7	ANP-10298P-A Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," AREVA, March 2014.	4-15	yes
4-8	ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.	4-16	yes
5-1	XN-CC-33(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual," Exxon Nuclear Company, November 1975.	5-15	
5-2	XN-NF-80-19(P)(A) Volumes 2, 2A, 2B and 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, September 1982.	5-17	

Table 1-2 Reference Index *(Continued)*

Reference No.	Methodology	Page No.(s)	Typically Referenced in Core Operating Limits Report
5-3	XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, November 1982.	5-18	
5-4	XN-NF-825(P)(A), "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$," Exxon Nuclear Company, May 1986.	5-19	
5-5	XN-NF-825(P)(A) Supplement 2, "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operations within the Extended Operating Domain," Exxon Nuclear Company, October 1986.	5-20	yes, for BWR/6
5-6	XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.	5-21	yes
5-7	ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.	5-23	yes
5-8	ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," Advanced Nuclear Fuels Corporation, January 1993.	5-25	

Table 1-2 Reference Index *(Continued)*

Reference No.	Methodology	Page No.(s)	Typically Referenced in Core Operating Limits Report
5-9	ANF-91-048(P)(A) Supplements 1 and 2, "BWR Jet Pump Model Revision for RELAX," Siemens Power Corporation, October 1997.	5-27	
5-10	EMF-2292(P)(A) Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000.	5-28	yes
5-11	EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001.	5-29	yes
5-12	ANF-1358(P)(A) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005.	5-30	yes

2.0 FUEL SYSTEM DESIGN

AREVA builds fuel assemblies to several specific design criteria to ensure that:

- The fuel assembly shall not fail as a result of normal operation and anticipated operational occurrences (AOOs). The fuel assembly dimensions shall be designed to remain within operational tolerances and the functional capabilities of the fuel shall be established to either meet or exceed those assumed in the safety analysis.
- Fuel assembly damage shall never prevent control rod insertion when it is required.
- The number of fuel rod failures shall be conservatively estimated for postulated accidents.
- Fuel coolability shall always be maintained.
- The mechanical design of fuel assemblies shall be compatible with co-resident fuel and the reactor core internals.
- Fuel assemblies shall be designed to withstand the loads from in-plant handling and shipping.

The first four objectives are those cited in Section I. of 4.2 Fuel System Design of the SRP. The last two objectives were established by AREVA to ensure structural integrity of the fuel and the compatibility of the fuel with existing reload fuel. All six of these objectives, which are found in Reference 2-10, are satisfied by AREVA design criteria approved by the NRC, which include:

- Preparing controlled documentation of the fuel system description and fuel assembly design drawings.
- Performing analyses with NRC-approved and accepted models and methods for AREVA fuels.

- Testing significant new design features with prototype testing and/or lead test assemblies prior to full reload implementation.

- Continued irradiation surveillance programs including post irradiation examinations to confirm fuel assembly performance.

- Using AREVA's approved QA procedures, QC inspection program, and design control requirements identified in FMM Revision 0 (Reference 2).

2.1 *Regulatory Requirements*

SRP Section 4.2 Fuel System Design, establishes criteria to provide assurance that the fuel system is not damaged as a result of normal operation or anticipated operational occurrences, that fuel system damage is never so severe that control rod insertion is prevented when it is required, that the number of fuel rod failures is not underestimated for postulated accidents, and that coolability is always maintained. These design criteria are necessary to meet the requirements of General Design Criteria (Reference 3) (GDC) 10, 27, and 35; 10 CFR Part 100, (Reference 4) and 10 CFR Part 50 (Reference 7) (50.46 and Appendix K).

2.2 *Fuel System Design Analyses*

The design criteria used for fuel system design analyses should not be exceeded during normal operation and AOOs. These criteria, described below, address the physical aspects of fuel assemblies and the behavior of the fuel and cladding.

2.2.1 Stress

Design Criteria

The design criteria for evaluating the structural integrity of the fuel assemblies are:

- Fuel assembly handling - The assembly must withstand dynamic axial loads based on the fuel assembly weight multiplied by a load factor.

- For all applied loads for normal operation and anticipated operational occurrences - The fuel assembly component structural design criteria are established for the two primary material categories: austenitic stainless steels (tie plates) and Zircaloy (tie rods, grids, spacer capture rod tubes, channels). The stress categories and strength theory for austenitic stainless steel presented in the ASME Boiler and Pressure Vessel Code, Section III (Reference 8) are used as a general guide.
- Steady-state stress design limits are given in Table 3-1 of Reference 2-10. Stress nomenclature is per the ASME Boiler and Pressure Vessel Code, Section III.
- Loads during postulated accidents - Deflection or failure of components shall not interfere with reactor shutdown or emergency cooling of the fuel rods.

Bases

In keeping with the GDC 10 specified acceptable fuel design limits (SAFDLs), the fuel damage design criteria for cladding stress assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Conservative stress limits are derived from the ASME Boiler and Pressure Vessel Code, Section III, Article III-2000 (Reference 8), and the specified 0.2% offset yield strength and ultimate strength for Zircaloy.

The structural integrity of the fuel assemblies is assured by setting design limits on stresses, deformations, and loadings due to various handling, operational, and accident loads. These limits are applied to the design and evaluation of upper and lower tie plates, grid spacers, tie rods, spacer capture rod, water rods, water channels, fuel channels, fuel assembly cage, and springs where applicable. The allowable component stress limits are based on the ASME Boiler and Pressure Vessel Code, Section III, with some criteria derived from component tests. Cladding stress categories include the primary membrane and bending stresses, and the secondary stresses. The loadings considered are fluid pressure, internal gas pressure, thermal gradients, restrained mechanical bow, flow induced vibration, and spacer contact. Table 3.1 of Reference 2-10 gives the ASME stress level criteria.

The stress calculations use conventional elasticity theory equations. A general purpose finite element stress analysis code such as ANSYS (Reference 9) may be used to calculate the spacer spring contact stresses. The fuel assembly structural component stresses under faulted conditions are evaluated using primarily the criteria outlined in Appendix F of the ASME Boiler and Pressure Vessel Code, Section III.

The AREVA analysis methods for calculating fuel rod cladding and assembly steady-state stresses are discussed and approved in References 2-6 and 2-9. The methods for calculating fuel channel stresses are discussed and approved in Reference 2-13.

2.2.2 **Strain**

Design Criteria

The design criteria for fuel rod cladding strain is that the transient-induced deformations must be less than 1% uniform. With the RODEX2A methodology, the strain limit is reduced at higher exposures to account for lower ductility.

Bases

The design criteria for cladding strain are intended to preclude excessive cladding deformation and failure from normal operations and AOOs.

With the RODEX2A methodology, AREVA uses the NRC-approved methods given in References 2-5 and 2-12 to calculate steady-state cladding strain during normal operation. Transient cladding strain is calculated as described in Supplement 1 of Reference 2-3.

With the RODEX4 methodology, the cladding transient strain is calculated as described in Reference 2-14. The evaluation of the transient criteria also includes fuel centerline melt. Cladding transient strain and fuel melt are evaluated for fast transients and for slow transients at off-rated conditions using the methodology given in Reference 2-14 for fast transients.

2.2.3 **Strain Fatigue**

Design Criteria

The design criteria for strain fatigue limits the cumulative fatigue usage factor based on a defined design fatigue life.

Bases

Cycle loading associated with relatively large changes in power can cause cumulative damage, which may eventually lead to fatigue failure. Therefore, AREVA requires that the cladding not exceed the fatigue usage design life as reduced by a proprietary factor. The fatigue usage factor is the number of expected cycles divided by the number of allowed cycles. The total cladding usage factor is the sum of the individual usage factors for each duty cycle.

The AREVA RODEX2 methodology for determining fuel assembly strain fatigue is based on Supplement 1 of Reference 2-3 and the O'Donnell and Langer fatigue design curves (Reference 10). The fatigue curves have been adjusted to incorporate the recommended safety factor of two on stress amplitude or 20 on number of cycles, whichever is more conservative. The RODEX2 and RAMPEX codes are used to provide fuel rod stress conditions for AREVA fatigue analysis.

With RODEX4, the fatigue usage design limit along with the fatigue design curve are the same as used with RODEX2. However, the cyclic stresses are calculated using the RODEX4 code as described in Reference 2-14. Rather than analyze a set of design duty cycle cases, RODEX4 uses a realistic approach and makes use of the power changes contained in the operating power histories. The calculated cumulative usage factor is scaled up according to a maximum expected quantity of duty cycles.

Fuel channel fatigue is evaluated with finite element calculations to evaluate channel stresses due to pressure variations in the channel as a function of bundle power and flow

(Reference 2-13). The same O'Donnell and Langer fatigue design curve is used as for the fuel rod evaluations.

2.2.4 Fretting Wear

Design Criteria

The design criteria for fretting wear requires that fuel rod failure due to fretting shall not occur.

Bases

AREVA controls fretting wear by use of design features, such as a spacer spring dimple system, which assure that fuel rods are positively supported by the grid spacers throughout the expected irradiation period. Spacer grid spring systems are designed such that the minimum rod contact forces throughout the design life are greater than the maximum fuel rod flow vibration forces. AREVA performs fretting tests to verify consistent fretting performance for new spacer designs. Examination of a large number of irradiated BWR rods, fuel assemblies, and channels has substantiated the absence of fretting in AREVA designs.

2.2.5 Oxidation and Crud Buildup

Design Criteria

There is no specific limit for oxide thickness or crud buildup. The effects of oxidation and crud buildup are considered in the fuel rod thermal and internal gas pressure analyses.

Bases

The AREVA fuel design basis for cladding corrosion and crud buildup is to prevent 1) significant degradation of the cladding strength, and 2) unacceptable temperature increases. Cladding corrosion reduces cladding wall thickness and results in less cladding load carrying capacity. At normal light water reactor operating conditions, this mechanism is not limiting except under unusual conditions where high cladding temperatures greatly accelerate the corrosion rate. Because of the thermal resistance of corrosion and crud layers, formation of these products on the cladding results in an elevation of temperature within the fuel as well as the cladding.

In the case of the RODEX2A methodology, there is no specific limit for oxidation. However, the BWR fuel performance code RODEX2A (Reference 2-12) includes the oxide buildup in the fuel performance predictions. That is, the crud and oxidation models are a part of the approved models and therefore impact the temperature calculation. AREVA includes an enhancement in the RODEX2A calculations for the corrosion analysis and fuel temperature analysis. This enhancement is a factor that is input to the code. This factor increases the amount of oxidation predicted by the corrosion model. The factor is selected, based on the particular design power history, to provide an end-of-life (EOL) oxidation thickness that is equivalent to the maximum peak oxidation observed for AREVA BWR fuel.

As part of the RODEX2A methodology, AREVA data show that even at higher exposures and residence times, cladding oxide thickness is relatively low. Mechanical properties of the cladding are not significantly affected by thin oxide or crud layers. For the thermal analyses, the effect of oxidation is included. There is sufficient conservatism in the gas pressure analysis to account for the effect of cladding oxidation without the use of an additional enhancement factor. For steady-state strain, transient strain, and cyclic stress, the effect of wall thinning is insignificant since cladding deformation is strain dependent. That is, the change in cladding diameter during a power change is primarily determined by the change in the pellet diameter since pellet-cladding contact occurs at higher exposures. For the cladding EOL stress analysis, the wall thickness is reduced consistent with the peak oxide thickness.

With RODEX4, the approved licensing limit is a proprietary value based on operating experience. A regulatory commitment was made to the NRC during the first application of RODEX4 to limit the maximum oxidation to a lower value. The commitment arises from the concern for the effect of non-uniformities in the cladding, such as spallation and localized hydride formations, on the 1% strain ductility limit. The cladding oxidation is calculated using RODEX4 as described in Reference 2-14. There is a SER restriction that requires an additional account for the thermal effect of crud in the event the methodology is applied to a plant that has higher crud levels of a given significance.

2.2.6 Rod Bowing

Design Criteria

The AREVA design criteria for rod bowing is that lateral displacement of the fuel rods shall not be of sufficient magnitude to degrade thermal margins.

Bases

Differential expansion between the fuel rods, 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 rods and may affect the peaking and local heat transfer. Rather than placing design limits on the amount of bowing that is permitted, the effects of bowing are included in the cladding overheating analysis by limiting fuel rod powers when bowing exceeds a predetermined amount. AREVA uses an approved methodology (Reference 2-9) to determine a rod-to-rod clearance closure limit below which a penalty is addressed on the minimum critical power ratio (MCPR) and above which no reduction in MCPR is necessary. The methodology is based on empirical data (Reference 2-2) to calculate minimum EOL rod-to-rod spacing. The potential effect of this rod bow on thermal margin is negligible. Rod bow at extended burnup does not affect thermal margins due to the lower powers achieved at high exposure.

2.2.7 Axial Growth

Design Criteria

AREVA requires that the fuel assembly be compatible with the channel throughout the fuel assembly lifetime. In addition, AREVA requires that clearances and engagements in the fuel assembly structure be maintained throughout the lifetime of the fuel.

Bases

AREVA evaluates fuel channel-fuel assembly differential growth to assure that the fuel channel to lower tie plate engagement is maintained to the design burnup. Another condition for BWR fuel assemblies is to maintain engagement between the fuel rod end cap shank and the assembly tie plates to prevent fuel rod disengagement from the tie plates. The change in BWR fuel rod-to-tie plate engagement (and possible disengagement) is due to the differential growth rate between the fuel rods and the tie rods for 9x9 fuel designs. For the 10x10 fuel, where the water channel connects the bottom and top tie plates, the goal is to ensure adequate clearance for growth of the fuel rods.

The analysis method (Reference 2-9) for evaluating rod-to-tie plate engagement is based on a statistical upper bound of measured differential rod-to-tie plate growth data (Reference 2-12) for 9x9 and 10x10 designs. The correlation predicts differential growth that bounds the differential growth data with a given statistical tolerance. This analysis uses fabrication tolerances in order to maintain conservatism in the calculated initial engagements and clearances.

2.2.8 Rod Internal Pressure

Design Criteria

AREVA limits maximum fuel rod internal pressure relative to system pressure. With the use of RODEX2A, there is an additional AREVA requirement when fuel rod pressure exceeds system pressure; the pellet-clad gap has to remain closed if it is already closed or that it should not tend to open for steady-state or increasing power operations.

Bases

Rod internal pressure is limited to prevent unstable thermal behavior and to maintain the integrity of the cladding. Outward circumferential creep which may cause an increase in pellet-to-cladding gap must be prevented since it would lead to higher fuel temperature and higher fission gas release. The maximum internal pressure is also limited to protect against embrittlement of the cladding caused by hydride reorientation during cooldown and depressurization conditions. A proprietary limit above system pressure has been justified by AREVA in Reference 2-7.

With RODEX2A, the gap requirement is checked as part of each calculation while with RODEX4, the rod pressure limit was established to protect the gap from re-opening at higher pressures.

2.2.9 **Fuel Assembly Liftoff**

Design Criteria

AREVA requires that the assembly not levitate from hydraulic or accident loads.

Bases

Levitation of a fuel assembly could result in the assembly becoming disengaged from the fuel support and interfering with control rod movement. For normal operation, including AOOs, the submerged fuel assembly weight, including the channel, must be greater than the hydraulic loads. The criterion is applicable to both cold and hot conditions and uses the technical specification limits on total core flow. For accident conditions, the normal hydraulic loads plus additional accident loads shall not cause the assembly to become disengaged from the fuel support. This assures that control blade insertion is not impaired.

2.2.10 **Fuel Assembly Handling**

Design Criteria

The assembly design must withstand all normal axial loads from shipping and fuel handling operations without permanent deformation.

Bases

AREVA uses either a stress analysis or testing to demonstrate compliance. The analysis or test uses an axial load factor on the static fuel assembly weight. At this load, the fuel assembly structural components must not show any yielding. Because of design features, for example grooved end caps, failure from axial loads will occur at the tie rod end caps rather than in the cladding or tie plates.

The rod plenum spring also has design criteria associated with handling requirements. The spring must maintain a force against the stack weight to prevent column movement during handling. The component drawing specifies the fabricated cold spring force.

2.2.11 **Miscellaneous Component Criteria**

2.2.11.1 **Compression Spring Forces**

Design Criteria

The compression spring(s) must support the weight of the upper tie plate and the channel throughout the design life of the fuel. Therefore, there is a requirement on the minimum compression spring force. There is also a maximum spring force limit requirement that the force be less than the calculated fuel rod buckling load in the case of the 9x9 designs.

Bases

The compression springs aid in seating the fuel rods against the lower tie plate while allowing for non-uniform growth and expansion of the same. The compression springs also exert an upward load to maintain the upper tie plate against the latching mechanism. The design criterion for the minimum force ensures the upper tie plate is fully latched throughout the lifetime of the fuel. A maximum force limit for the compression spring ensures fuel rods are not inadvertently damaged during tie plate removal and installation.

The maximum force requirements do not apply to the ATRIUM™-10* design as there is only one large spring on the water channel.

2.2.11.2 Lower Tie Plate Seal Spring

Design Criteria

The seal accommodates the channel deformation to limit the leak rate of coolant between the lower tie plate and channel wall.

Bases

The lower tie plate seal spring limits the leak rate of coolant between the lower tie plate and the channel wall. The seal shall have adequate corrosion resistance and be able to withstand the operating stresses without yielding. The design also considers the differential axial growth between the channel and the fuel assembly. Flow testing of prototypic components verifies the leakage rate and fretting resistance. A stress analysis provides the seal stresses.

2.2.12 Fuel Rod Failure

The fuel rod failure design criteria and bases cover normal operation conditions, AOOs, and postulated accidents. When the fuel rod failure criteria are applied in normal operation and AOOs, they are used as limits (SAFDLs) since fuel failure under those conditions must not occur according to GDC 10 (Reference 3). When the criteria are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose calculations required by 10 CFR 100 (Reference 4) or 10 CFR 50.67 (Reference 5) and Regulatory Guide 1.183 (Reference 6) as appropriate.

* ATRIUM is a trademark of AREVA Inc.

2.2.12.1 **Internal Hydriding**

Design Criteria

AREVA limits internal hydriding by imposing a fabrication limit for total hydrogen in the fuel pellets.

Bases

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and formation of hydride platelets. Hydriding, as a cladding failure mechanism, is precluded by controlling the level of moisture and other hydrogenous impurities during fuel pellet fabrication. The hydrogen concentration criteria are met by maintaining moisture control during fuel fabrication (Reference 2-7).

2.2.12.2 **Cladding Collapse**

Design Criteria

Creep collapse of the cladding is avoided in the AREVA fuel system design by eliminating the formation of significant axial gaps in the pellet column.

Bases

If axial gaps in the fuel pellet column were to occur due to handling, shipping, or fuel densification, the cladding would have the potential of collapsing into the gap. Because of the large local strains that would result from the collapse, the cladding is assumed to fail. Creep collapse of the cladding and the subsequent potential for fuel failure is avoided in the AREVA fuel system design by eliminating the formation of significant axial gaps. The evaluation must show that the pellet column is compact at a specified burnup. The internal plenum spring provides an axial load on the fuel stack that is sufficient to assist in the closure of any gaps caused by handling, shipping, and densification. Evaluation of cladding creep stability in the unsupported condition is performed considering the compressive load on the cladding due to the difference between primary system pressure and the fuel rod internal pressure. AREVA fuel is designed to minimize the potential for the formation of axial gaps in the fuel and to minimize clad creepdown that would prevent the closure of axial gaps or allow creep collapse.

The RODEX2A code (Reference 2-12) is used to provide initial in-reactor fuel rod conditions to the COLAPX (Reference 11) method described in Reference 2-7 which is used to predict creep collapse. COLAPX calculates ovality changes and creep deformation of the cladding as a function of time. The goal of the analysis is to demonstrate that the initial fuel densification occurs prior to the time of pellet-clad gap closure as based on the initial, minimum pellet-to-cladding gap.

With RODEX4, the cladding collapse analysis is performed in a similar manner except the maximum axial gap formation is calculated and compared to an allowable axial gap that protects against creep collapse. The ovalization calculation of the cladding is integrated into the RODEX4 code and the analysis is performed as part of the statistical methodology described in Reference 2-14.

2.2.12.3 **Overheating of Cladding**

Design Criteria

The design basis to preclude fuel rod cladding overheating is 99.9% of the fuel rods shall not experience transition boiling.

Bases

It has been traditional practice to assume that fuel failures will occur if the thermal margin criterion is violated. Thermal margin is stated in terms of the minimum value of the critical power ratio (CPR) for the most limiting fuel assembly in the core. Prevention of potential fuel failure from overheating of the cladding is accomplished by minimizing the probability of exceeding thermal margin limits on limiting fuel rods during normal operation and anticipated operational occurrences. Compliance with this criterion as part of the reload thermal hydraulics analysis is discussed in Section 4.2 of this report.

2.2.12.4 **Overheating of Fuel Pellets**

<u>Design Criteria</u>

Fuel failure from overheating of the fuel pellets is not allowed. The centerline temperature of the fuel pellets must remain below melting during normal operation and AOOs.

<u>Bases</u>

Steady-state and transient design linear heat generation rate (LHGR) limits are established for each fuel system to protect against centerline melting. Operation within these LHGR limits prevents centerline melting during normal operation and anticipated operational occurrences throughout the design lifetime of the fuel.

In the case of the RODEX2A application, a correlation is used for the fuel melting point that accounts for the effect of burnup and gadolinia content. This fuel melting limit has been reviewed and approved (Reference 2-7) with respect to the extended burnup of fuel and gadolinia bearing fuel.

With the RODEX2A methodology, AREVA uses the RODEX2A computer code (Reference 2-12) to calculate the maximum possible fuel centerline temperature for normal operations and for AOOs. Conservative LHGR power histories are used to perform the centerline temperature calculations. For AOOs and accidents, AREVA also uses the RODEX2A code to calculate maximum possible fuel centerline temperatures at LHGRs that are higher than the steady-state LHGR history used for normal operation.

With RODEX4, the maximum fuel centerline temperature is calculated using operational power histories that include AOO transients. Adjustments are provided for the effect of gadolinia content on the fuel melt temperature. In addition, the RODEX4 code explicitly includes the effect of thermal conductivity degradation in modeling the fuel.

The methodology to assess the maximum fuel centerline temperatures for fast AOOs is described in Reference 2-14. This methodology evaluates the fuel centerline temperature on the limiting nodal axial region of a fuel rod to demonstrate that the transient design criteria are satisfied with 95% probability and a 95% confidence level. The same methodology also is used to evaluate slow transients at off-rated conditions to determine power dependent operational limits that assure satisfaction of the transient criteria.

2.2.12.5 **Pellet/Cladding Interaction**

Design Criteria

The Standard Review Plan (Reference 1) does not contain an explicit criterion for pellet/cladding interaction. However, it does present two related criteria. The first is that transient-induced deformations must be less than 1% uniform cladding strain. The second is that fuel melting cannot occur.

Bases

The cladding strain requirement is addressed in Section 2.2.2. The centerline temperature requirement is addressed in Section 2.2.12.4.

2.2.12.6 **Cladding Rupture**

Design Criteria

10 CFR 50 Appendix K (Reference 7) requires that cladding rupture must not be underestimated when analyzing a loss-of-coolant accident.

Bases

Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure conditions during a loss-of-coolant accident (LOCA). Since there are no specific design criteria in the Standard Review Plan (Reference 1) associated with cladding rupture, AREVA has established a rupture temperature correlation to be used during the LOCA emergency core cooling system (ECCS) analysis.

The effects of cladding rupture are an integral part of the AREVA ECCS evaluation model. The cladding ballooning and rupture models used are those presented in

NUREG-0630 (Reference 12) for cladding rupture evaluation. These models are described in XN-NF-82-07(P)(A) Revision 1 (see Reference 5-3).

2.2.12.7 **Fuel Rod Mechanical Fracture**

Design Criteria

AREVA limits the combined stresses from postulated accidents to the stresses given in the ASME Code, Section III, Appendix F (Reference 8) for faulted conditions.

Bases

A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force, such as a hydraulic load or a load derived from core plate motion induced by a seismic or LOCA event. The design bases and criteria for mechanical fracturing of AREVA BWR reload fuel are presented in Reference 2-4, which describes AREVA's LOCA-seismic structural response analysis. The design basis is that the channeled fuel assemblies must withstand external loads due to earthquake and postulated pipe breaks without fracturing the fuel rod cladding. The stresses due to postulated accidents in combination with normal steady-state fuel rod stresses should not exceed the stress limits given in Reference 2-4. The allowable stresses are derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix F, for faulted conditions.

The mechanical fracture analysis is done as part of the plant specific seismic-LOCA loading analysis. Consideration can be given to the fuel assembly dynamic properties in determining the need for reanalysis when the fuel design is changed. AREVA verifies the assembly characteristics for new designs to ascertain that these characteristics (assembly weight and vibration mode) are similar to the co-resident fuel.

2.2.12.8 **Fuel Densification and Swelling**

Design Criteria

Fuel densification and swelling are limited by the design criteria specified for fuel temperature, cladding strain, cladding collapse, and internal pressure criteria.

Bases

AREVA uses the NRC reviewed and accepted densification and swelling models in the fuel performance code, RODEX2A (Reference 2-12) and RODEX2 (Reference 2-3). Or, in the application of RODEX4, the approved densification and swelling models are described in Reference 2-14.

2.2.13 **BWR Fuel Coolability**

For accidents in which severe fuel damage might occur, core coolability and the capability to insert control blades are essential. Normal operation or anticipated operational occurrences must remain within the thermal margin criteria. Chapter 4.2 of the Standard Review Plan (Reference 1) provides several specific areas important to the coolability and the capability of control blade insertion. The sections below discuss these areas.

2.2.13.1 **Fragmentation of Embrittled Cladding**

Design Criteria

ECCS evaluations meet the 10 CFR 50.46 (Reference 7) limits of 2200°F peak cladding temperature, local and core-wide oxidation, and long term coolability.

Bases

The requirements on cladding embrittlement relate to the LOCA requirements of 10 CFR 50.46. The principal cause of cladding embrittlement is the high cladding temperatures that result in severe cladding oxidation.

The models to compute the temperatures and oxidation are those prescribed by 10 CFR 50 Appendix K (Reference 7) (see Reference 5-1). LOCA analyses are performed on a plant specific basis.

2.2.13.2 **Violent Expulsion of Fuel**

Design Criteria

AREVA limits the radially-averaged enthalpy deposition at the hottest axial location to 280 cal/gm for severe reactivity initiated accidents.

Bases

In a severe reactivity initiated accident (RIA), large and rapid deposition of energy in the fuel could result in melting, fragmentation, and dispersal of the fuel. The AREVA methodology complies with the fission product source term guideline in Regulatory Guide 1.77 (Reference 13) and the Standard Review Plan (Reference 1) that restricts the radially-averaged energy deposition.

The limiting RIA for AREVA fuel in a BWR is the control rod drop accident (CRDA). AREVA calculates the maximum radially averaged enthalpy for the CRDA for each reload core in order to assure that the maximum calculated enthalpy is below the 280 cal/gm limit. The control rod drop calculation methodology approved by the NRC is described in Reference 3-1. The parameterized AREVA control rod drop methodology determines maximum deposited enthalpy as a function of dropped rod worth, effective delayed neutron fraction, Doppler coefficient, and four-bundle local peaking factor.

The CRDA analysis is not part of the normal fuel assembly mechanical analysis but is part of the cycle specific safety analysis performed for each BWR.

2.2.13.3 **Cladding Ballooning**

Design Criteria

There are no specific design limits associated with cladding ballooning, other than a requirement in 10 CFR 50 Appendix K (Reference 7) that the degree of swelling not be underestimated.

Bases

Zircaloy cladding will balloon (swell) under certain combinations of temperature, heat rate, and stress during a LOCA. Cladding ballooning can result in flow blockage; therefore, the LOCA analysis must consider the cladding ballooning impacts on the flow.

The effects of cladding ballooning are an integral part of the AREVA ECCS evaluation model. The cladding ballooning and rupture models used are those presented in NUREG-0630 (Reference 12) for cladding rupture evaluation. These models are described in XN-NF-82-07(P)(A) Revision 1 (see Reference 5-3).

The RODEX2 fuel performance code (Reference 2-3) is used to provide burnup dependent input to the LOCA analysis, e.g., stored energy and rod pressures, that are a function of the initial steady-state operation of the fuel. This initial steady-state fuel condition is also important to cladding ballooning.

2.2.13.4 **Fuel Assembly Structural Damage from External Forces**

Design Criteria

The AREVA design criteria for fuel assembly structural damage from external forces are discussed in Sections 2.2.1, 2.2.9, and 2.2.12.7.

Bases

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. The Standard Review Plan (Reference 1) states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control blade insertion when required during these accidents. The AREVA design basis is that the fuel assembly will maintain a geometry that is capable of being cooled under the worst case accident and that system damage is never so severe as to prevent control blade insertion. AREVA ensures these design bases are met by placing ASME design limits on the stresses that the fuel channel and critical fuel assembly components can experience. These limits have been approved for AREVA fuel assemblies in References 2-4 and 2-13.

2.3 *NRC-Accepted Topical Report References*

The NRC has approved the following licensing topical reports that describe the methods and assumptions used by AREVA to demonstrate the adequacy of its BWR fuel system design. These reports address mechanical design criteria and required mechanical and thermal conditions. The purpose of each topical report and the restrictions that have been placed on the methods presented are described in the following sections.

2-1: XN-NF-79-56(P)(A) Revision 1 and Supplement 1, "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," Exxon Nuclear Company, November 1981.

- Purpose: Justify gadolinia fuel properties for up to 5 wt% gadolinia loading in uranium dioxide fuel.

- SER Restrictions:

  1. The concentration of gadolinia is limited to 5 wt%.

  2. The report is acceptable based on a commitment to acquire more data for gadolinia bearing rods.

- Implementation of SER Restrictions:

  1. This SER restriction is no longer applicable. The limit on gadolinia concentration was increased to 8 wt% in Reference 2-8 with the application of RODEX2 and RODEX2A.

  2. The additional data was gathered and was provided to the NRC in Reference 16.

- Observations:

  1. The limitation on the concentration of gadolinia was raised to 8 wt% by the topical report XN-NF-85-92(P)(A). Additional data was gathered on gadolinia from Prairie Island, Tihange, and other reactors.

  2. The topical report is no longer referenced in conjunction with the RODEX4 methodology. The gadolinia concentration limit increased to 10 wt% in Reference 2-14 and an updated thermal conductivity correlation is used.

2-2: XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, October 1983. (Base document not approved.)

- Purpose: Develop an empirical method for determining fuel rod bow.
- SER Restrictions: The technical evaluation of the methodology was limited to the fuel designs, exposures, and conditions stated in the topical report and, in part, on assumptions made in formulating the methodology. It was recommended that Exxon continue fuel surveillance to ensure confidence in the assumptions and bases.
- Implementation of SER Restrictions: The application of the rod bow model to higher burnup and other fuel designs was approved in Reference 2-9.
- Observations: AREVA has continued to gather data from fuel surveillance and CPR experiments.

2-3: XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.

- Purpose: Provide an analytical capability to predict BWR and PWR fuel thermal and mechanical conditions for normal core operation and to establish initial conditions for power ramping, non-LOCA and LOCA analyses.

- SER Restrictions:

  1. The NRC concluded that the RODEX2 fission gas release model was acceptable to burnups up to 60 MWd/KgU. This implies a burnup limit of 60 MWd/KgU (nodal basis).

  2. The creep correlation accepted by the NRC is the one with the designation MTYPE = 0.

- Implementation of SER Restrictions:

  1. This restriction no longer applies. The exposure limits for BWR fuel were increased to 54 MWd/kgU for an assembly and to 62 MWd/kgU for a rod in Reference 2-12. These exposure limits are reflected in engineering guidelines.

  2. This restriction is implemented in the engineering guidelines and through computer code controls (defaults, override warning messages).

- Observations: The computer code that is used to perform analyses is now called RODEX2-2A. The NRC approved models, RODEX2 or RODEX2A, are chosen by input. A single code is maintained in order to assure that the NRC approved models are implemented correctly. RODEX2 is the fuel performance code that provides input to BWR LOCA and transient thermal-hydraulic methodologies.

  RODEX2 and RODEX2A may be used to model fuel with up to 8% gadolinia loading (See Reference 2-8).

2-4: XN-NF-81-51(P)(A), "LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, May 1986.

- Purpose: Develop a methodology for performing LOCA-Seismic structural analyses of BWR jet pump fuel assemblies.

- SER Restrictions: The allowable stress values reported for BWR jet pump fuel channel and assembly components are acceptable and licensees referencing the topical report for other non-GE manufactured channels are required to show that the calculated allowable stresses for seismic and LOCA loading conditions are bounded by those in the topical report.

- Implementation of SER Restrictions: This restriction is no longer applicable. The requirements for fuel channels are now described in Reference 2-13.

- Observations: The analyses reported were for an 8x8 fuel assembly. The channeled fuel assembly seismic analysis was performed using the response spectrum method of dynamic analysis in the NASTRAN finite element program (Reference 15). Current analyses make use of the KWUSTOSS dynamic analysis code for fuel channels (with fuel assembly) as described in Reference 2-13. The LOCA seismic criteria are specified in Reference 2-10.

2-5: XN-NF-85-74(P)(A) Revision 0, "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model" Exxon Nuclear Company, August 1986.

- Purpose: The purpose of this topical report was to obtain NRC approval of a modification of the RODEX2 (Reference 2-3) fission gas release model for application to BWRs. This code version was named RODEX2A.

- SER Conclusions / Restrictions:

  1. The code RODEX2A is acceptable for mechanical analyses but RODEX2 must continue to be used for LOCA and transient analysis input generation.

  2. The RODEX2A calculation of fuel rod pressure must be performed to a minimum burnup of 50 MWd/kgU using the approved power history.

- Implementation of SER Restrictions:

  1. This SER restriction is implemented in engineering guidelines.

  2. The code RODEX2A was approved to a rod average burnup of 62 MWd/kgU in Reference 2-12. The analyzed burnup for all current designs is greater than 58 MWd/kgU.

- Observations: The RODEX2A code was approved to a maximum rod average burnup of 62 MWd/kgU in Reference 2-12.

2-6: XN-NF-85-67(P)(A) Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1986.

- Purpose: Demonstrate that mechanical design criteria are not violated when fuel is operated at the LHGR limits for both 8x8 fuel and 9x9 fuel with maximum assembly discharge exposures of 35,000 MWd/MTU and 40,000 MWd/MTU, respectively.

- SER Restrictions:

  1. LHGR limit curves (Figures 3.1, 3.2, and 3.3) are to be used for the fuel described.

  2. Discharge exposure is limited to previously approved 30,000 MWd/MTU batch average exposure pending approval of Reference 2-9.

  3. Additional rod bow data are required for burnup extensions beyond 30,000 MWd/MTU for 8x8 fuel and 23,000 MWd/MTU for 9x9 fuel.

- Implementation of SER Restrictions:

  1. This restriction no longer applies since the 8x8 and 9x9 fuel addressed by this report are no longer being supplied.

  2. and 3. These restrictions no longer apply. The exposure limits for BWR fuel were increased to 54 MWd/kgU for an assembly and to 62 MWd/kgU for a rod in Reference 2-12. These exposure limits are reflected in engineering guidelines.

- Observations:

  1. Although Reference 2-6 only discusses applications to 8x8 and 9x9 fuel types, the report includes a description of the process used to develop linear heat generation rates for fuel designs. Subsequent to the approval of this topical report, AREVA developed and the NRC approved the use of generic design

criteria for new fuel designs (Reference 2-10). Reference 2-12 describes the use of the same LHGR methodology for application to the ATRIUM-9 and ATRIUM-10 designs.

2. This topical report is not needed in the case where the RODEX4 methodology is used.

2-7: XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986.

- Purpose: Provide the design bases, analyses and test results in support of the qualification of BWR fuel (8x8 and 9x9) for burnup extension to 35,000 MWd/MTU assembly batch exposure. (Note: This topical report also addressed burnup extension to 45,000 MWd/MTU for PWR fuel.)

- SER Restrictions: If fuel at extended burnup levels experiences a plant depressurization accident, the licensee must address possible cladding hydride reorientation prior to further irradiation of the fuel.

- Implementation of SER Restrictions: This and other issues would be addressed in response to a request from a licensee to justify continued operation of BWR fuel following an accident.

- Observations: Reference 2-10 references this topical report as the approved method for setting a fuel pressure limit above system pressure and a criterion which requires that a radial fuel-cladding gap be maintained during constant and increasing power operation under normal reactor operating conditions.

2-8: XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986.

- Purpose: Justify gadolinia fuel properties for up to 8 wt % gadolinia loading in uranium dioxide fuel to be used in BWR fuel designs. The topical report applies with the application of the RODEX2 and RODEX2A methodology.

- SER Restrictions: Based on a commitment to confirm the fission gas release model with in-reactor data, the gadolinia fuel properties are acceptable for licensing applications up to 8 wt% gadolinia concentration.

- Implementation of SER Restrictions: The SER restriction on 8 wt% gadolinia is implemented in engineering guidelines.

- Observations:

  1. In-reactor fission gas release test results (Reference 16) were provided to the NRC. The thermal conductivity model supersedes the previously approved model (Reference 2-1).

  2. With the use of RODEX4, the gadolinia concentration limit is increased to 10 wt%. In addition, an updated thermal conductivity model is provided. This topical report does not need to be referenced with the application of RODEX4.

- Clarifications: NRC concurrence with a clarification related to the topical report was requested in Reference 34. The NRC concurrence with the clarification was provided in Reference 35. The clarification was with respect to the use of one conductivity equation for UO_2-only fuel and a separate gadolinia-bearing fuel conductivity equation for all gadolinia concentrations greater than zero wt%.

2-9: XN-NF-82-06(P)(A) Supplement 1 Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Supplement 1, "Extended Burnup Qualification of ENC 9x9 BWR Fuel," Advanced Nuclear Fuels Corporation, May 1988.

- Purpose: Provide the design bases, analyses, and test results in support of the qualification of BWR fuel (9x9) for burnup extension to 40,000 MWd/MTU peak assembly exposure and to obtain approval of the rod bow method for extended burnup.

- SER Restrictions: The LHGR limit curves (Figures 3.1, 3.2, and 3.3) in XN-NF-85-67(P)(A) Revision 1 continue to be applicable as bounding LHGR limits.

- Implementation of SER Restrictions: This restriction no longer applies. LHGR limit curves can be established as allowed in Reference 2-10.

- Observations: The rod bow model approved in XN-75-32(P)(A) was approved for application to 9x9 fuel for assembly exposures to 40,000 MWd/MTU. The extended burnup data used to confirm the rod bow model indicated that rod bow at extended burnup does not affect thermal margins due to the lower rod powers at high exposure. The use of the same rod bow model up to 54,000 MWd/MTU for the ATRIUM-9 and ATRIUM-10 designs is described in Reference 2-12.

- Clarifications: NRC concurrence with a clarification related to the topical report was requested in References 28 and 29. The NRC concurrence with this clarification was provided in Reference 30. The clarification is that Reference 2-10 removes the need for a specific LHGR limit curve for BWR fuel designs and allows for LHGR limits to be established in accordance with the approved mechanical design criteria.

2-10 : ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

- Purpose: Establish a set of design criteria which assures that BWR fuel will perform satisfactorily throughout its lifetime.

- SER Restrictions:

  1. Peak pellet burnup shall not be increased beyond 60,000 MWd/MTU unless axial growth and fretting wear data have been collected from lead test assemblies of the modified design.

  2. Exposure beyond 60,000 MWd/MTU peak pellet must be approved by the NRC.

  3. Approval does not extend to the development of additive constants for ANFB to co-resident fuel.

  4. For each application of the mechanical design criteria, AREVA must document the design evaluation and provide a summary of the evaluation for the NRC.

- Implementation of SER Restrictions:

  The revised SER restrictions on burnup are implemented in engineering guidelines.

  1. The NRC approved higher burnup values as presented in Reference 2-12.

  2. The exposure limit was extended to a rod-average burnup of 62 GWd/MTU by the approval of Reference 2-12.

  3. The ANFB correlation is no longer used.

4. It was clarified in References 28 and 29 that this requirement applies to generic evaluations that are independent of plant specific evaluations. The NRC concurred with this in Reference 30.

- Observations:

1. The application of the processes and criteria described in this topical report do not require prior NRC approval.

2. With the use of RODEX4, some fuel rod design criteria are updated. The RODEX4 LTR (Reference 2-14) references this topical report for the remaining fuel rod criteria.

3. The mechanical design of the fuel channel is performed using the criteria and methods described and approved in Reference 2-13.

4. The design methodology for the reconstitution of a BWR fuel assembly complies with Reference 2-11.

2-11 : ANF-90-82(P)(A) Revision 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995.

- Purpose: Develop a methodology to justify reinsertion of irradiated fuel assemblies, which have been reconstituted with replacement rods, into a reactor core. Replacement rods can be fuel rods containing natural uranium pellets, water rods, and inert rods containing Zircaloy or stainless steel inserts.

- SER Restrictions: The reconstitution methodology is acceptable for reload licensing applications with the following conditions:

  1. BWR reconstituted assemblies are limited to 9 rods per assembly.

  2. The seismic LOCA analysis will be reassessed if the reconstructed weight drops below a proprietary value.

- Implementation of SER Restrictions: The SER restrictions are implemented in engineering guidelines.

- Observations: The reconstitution methodology is applicable to all fuel designs.

  The SER restrictions on the number of replacement rods apply only to inert rods containing Zircaloy or stainless steel inserts.

  NOTE: Extension of this Methodology for use with the SPCB or ACE CHF correlations would require Licensees to perform a 10 CFR 50.59 evaluation/justification or a plant specific methodology approval within a License Amendment Request.

2-12 : EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998.

- Purpose: Extend the exposure limits of the RODEX2A (Reference 2-5) code, which is a version of RODEX2 that includes a fission gas release model specific to BWR fuel designs.

- SER Restrictions: RODEX2A is acceptable for steady-state licensing applications to 62,000 MWd/MTU rod-average burnup and the fuel rod growth, fuel assembly growth, and fuel channel growth models and analytical methods are acceptable for ATRIUM-9 and -10 fuel designs up to 54,000 MWd/MTU assembly-average burnup.

- Implementation of SER Restrictions: The SER restrictions on burnup are implemented in engineering guidelines.

- Observations: The RODEX2A code, which is used for BWR fuel design applications, is a derivative of AREVA's base fuel performance code RODEX2.

  In the approved topical report, the NRC acknowledges the following observations as correct:

  1. Steady-state analyses of maximum wall thinning from oxidation for end of life conditions will be performed.
  2. The growth correlations reviewed are applicable to all AREVA 9x9 fuel designs.
  3. Transient strain is to be calculated with the version of RODEX referenced in XN-NF-81-58(P)(A) Revision 2 Supplement 1 (Reference 2-3). Strain is limited to 1.0% and the limit is reduced at high exposures.
  4. Steady-state strain is to be calculated with RODEX2A and is limited to 1%.
  5. RODEX2A is to be used to calculate fuel temperatures for fuel melt analyses.

6. RODEX2 shall be used as the base fuel performance code to interface with the AREVA LOCA and transient thermal-hydraulic methodologies. The RODEX2 code was also approved for BWR analyses to 62 GWd/MTU rod average burnup.

With the use of RODEX4, some of the fuel rod criteria present in this topical report were updated. For example, the transient strain limit and the corrosion limit were changed. However, the RODEX4 topical report references this topical report for other fuel rod-related design criteria such as those for cladding steady-state stresses.

- Clarifications: NRC concurrence with clarifications related to this topical report was requested in References 38 and 39. The NRC concurrence with these clarifications was provided in Reference 40. The clarification was associated with applying the exposure limits to only the full length fuel rods and not the part length fuel rods.

2-13 : EMF-93-177(P)(A) Revision 1, "Mechanical Design for BWR Fuel Channels," Framatome ANP, August 2005.

- Purpose: Demonstrate that analytical methods are adequate to perform evaluations which ensure that fuel channels perform as designed for normal operations and during anticipated operational occurrences and that for postulated accident loadings channel damage does not prevent control blade insertion and assembly coolability is maintained.

- SER Restrictions: Subject to certain conditions, the analyses conducted by AREVA are acceptable for licensing applications.

  1. The fuel channel TR (Technical Report) methods and criteria may be applied to fuel channel designs similar to the configuration of a square box with radiused corners open at the top and bottom ends. The wall thickness shall fall within the range of current designs. The channels shall be fabricated from either Zircaloy-2 or Zircaloy-4. AREVA will not use Zircaloy material for channels which has less strength than specified in the TR, and if the strength of material is greater than that in the TR, AREVA will not take credit for the additional strength without staff review.

  2. Updates to channel bulge and bow data are permitted without review by the NRC staff; however, AREVA shall resubmit the channel bulge and bow data statistics if the two-sigma upper and lower bounds change by more than one standard deviation

  3. This TR is approved using the ABAQUS or ANSYS codes in the deformation analysis. The use of other codes in the deformation analysis, i.e., NASTRAN, is beyond the current approval.

  The following restrictions are carried over from EMF-93-177(P)(A) Revision 0; for specific plant applications the following conditions are to be met:

4. The allowable differential pressure loads and accident loads should bound those of the specific plant.

5. Lattice dimensions should be compatible to those used in the analyses reported such that the minimum clearances with control blades continue to be acceptable.

6. Maximum equivalent exposure and residence time should not exceed the values used in the analyses.

- Implementation of SER Restrictions: The SER restrictions are implemented in engineering guidelines.

- Observations: The methodology approved is appropriate for exposures and minor dimensional changes beyond those evaluated and reported in the topical. Use of the methodology to extended exposure must be validated against the original design criteria.

The Reference 27 letter was provided to the NRC to inform them that Revision 0 of the topical report had been used to confirm the fuel channel design met the design criteria at an approved assembly exposure for which results had not been previously provided. No NRC response was requested.

2-14 : BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.

- Purpose: Provide an analytical capability to predict BWR fuel thermal and mechanical conditions for normal core operation and to establish initial conditions for power ramping, and non-LOCA. At this time RODEX4 is not approved for use with LOCA analyses.

- SER Restrictions:

  1. Due to limitations within the FGR model, the analytical fuel pellet grain size shall not exceed 20 microns 3-D when the as-manufactured fuel pellet grain size could exceed 20 microns 3-D.

  2. RODEX4 shall not be used to model fuel above incipient fuel melting temperatures.

  3. The hydrogen pickup model within RODEX4 is not approved for use.

  4. Due to the empirical nature of the RODEX4 calibration and validation process, the specific values of the equation constants and tuning parameters derived in TR BAW-10247(P), Revision 0 (as updated by RAI responses) become inherently part of the approved models. Thus, these values may not be updated without necessitating further NRC review.

  5. RODEX4 has no crud deposition model. Due to the potential impact of crud formation on heat transfer, fuel temperature, and related calculations, RODEX4 calculations must account for a design basis crud thickness. The level of deposited crud on the fuel rod surface should be based upon an upper bound of expected crud and may be based on plant-specific history. Specific analyses would be required if an abnormal crud or corrosion layer (beyond the design basis) is observed at any given plant. For the purpose of this evaluation, an abnormal

crud/corrosion layer is defined by a formation that increases the calculated fuel average temperature by more than 25°C beyond the design basis calculation.

- Implementation of SER Restrictions:

1. The RODEX4 methodology is automated using the AUTOFROST and AUTOPOSTFROST codes. In RODEX4 fuel rod thermal-mechanical analyses, AUTOFROST is used to prepare RODEX4 inputs and set up the RODEX4 runs. AUTOFROST has internal automatic checks to ensure the proper grain size is input that meets SER requirements. (Note: Both RODEX4 and the manufacturing specifications define the grain size using the MLS (mean line intercept) method instead of a 3-D measurement. The equivalent MLS grain size is on the order of 12.8 µm.)

2. Fuel melting is a fuel rod thermal-mechanical design criterion. No fuel design is allowed that would exceed fuel melting temperature. Thus RODEX4 will not be applied above fuel melting.

3. AUTOPOSTFROST, the post-processor of AUTOFROST/RODEX4 runs, writes hydrogen pickup calculation results to output with the qualifier “for information only”.

4. The model parameters documented in BAW-10247PA have been programmed into AUTOFROST as default values. By default, the automation will override user inputs of model parameters with the approved values.

5. Because the crud deposition is plant specific and operation dependent, and thus so are the RODEX4 calculations/applications. This requirement was implemented in a guideline by requiring that liftoff measurement data and/or visual examinations be examined to characterize plant crud conditions. If crud conditions (total liftoff) indicate higher than normal crud levels, then a heat transfer coefficient must be estimated and input to account for the added thermal resistance as required by the SER.

- Observations: RODEX4 is approved for modeling BWR fuel rods with the following conditions:

    a. Peak rod average burnup limit of 62 GWd/MTU (full length rod).

    b. Solid UO_2 fuel pellet with a maximum gadolinia content of 10.0 weight percent.

    c. CWSR Zr-2 fuel clad material.

  A regulatory commitment was made to the NRC during the first application of RODEX4 that restricts the maximum external oxidation to a proprietary limit. Until further data are available, it is expected that applications of RODEX4 for other plants will require the same limitation until data to support a higher limit is obtained. The limit is included in the fuel rod analysis engineering guideline.

  RODEX4 is not yet approved for use with LOCA analyses.

3.0 NUCLEAR DESIGN

Nuclear design analyses are used for nuclear fuel assembly design and core design. The core design analysis demonstrates operating margins for minimum critical power ratio (MCPR), maximum average planar linear heat generation rate (MAPLHGR), and linear heat generation rate (LHGR). Two LHGR limits are established for each fuel design. One is a steady-state operating fuel design limit (FDL), and the other is the protection against the power transient (PAPT) limit.

An exposure dependent LHGR limit is established for each fuel assembly design as part of the mechanical design analysis. The LHGR limit is consistent with the power histories established to perform the mechanical analyses. Hence, operation of the fuel assembly within the steady-state LHGR limit ensures that the power history assumptions remain valid.

3.1 *Regulatory Requirements*

SRP Section 4.3 Nuclear Design discusses GDC 10-13, 20, and 25-28 that pertain to nuclear design. Many of the GDCs relate to mechanical properties of the fuel assembly that are satisfied by meeting appropriate thermal and reactivity margin limits while the fuel resides in the reactor core. AREVA standard design practice is to define these limits and demonstrate that the fuel maintains appropriate margin to these limits by calculating the expected margins in simulated projections of the cycle prior to the fuel being loaded in the reactor core. In addition, by demonstrating that appropriate licensing criteria are met when certain postulated accidents are modeled to occur during the cycle in which the fuel is loaded, the safety aspects of the fuel are assured.

Of the GDCs mentioned in 4.3 Nuclear Design, only GDC 11 is principally related to the neutronic response of the fuel. GDC 11 requires that “in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.”

3.2 *Nuclear Design Analyses*

The nuclear design analyses demonstrate operating margin to design limits, including MCPR, MAPLHGR, and LHGR. The approved nuclear design codes and methodologies are described in References 3-1, 3-2, and 3-4.

3.2.1 Fuel Rod Power History

Design Criteria

The nuclear design analysis must be consistent with the exposure dependent LHGR limit established during the mechanical design analysis for each fuel assembly design.

Two LHGR limits are established for each fuel design. One is a steady-state limit, the other a PAPT limit. Both limits are a function of fuel burnup. The transient LHGR design limit satisfies the strain and fuel overheating design criteria discussed in Section 2.2.2 and Section 2.2.12.4. The design margin between the steady-state and transient LHGR limits must be sufficient to account for increases in the LHGR during transients.

Bases

An exposure dependent LHGR limit is established for each fuel assembly design as part of the mechanical design analysis (Reference 2-6 and 2-9 with RODEX2A and Reference 2-14 with RODEX4).

With the RODEX2A methodology, the LHGR limit is consistent with the power histories established to perform the mechanical analyses. Therefore, operation of the fuel assembly within the LHGR limit is necessary to ensure that the power history assumption used in the mechanical design analyses remains valid. The specific mechanical design criteria are provided in Reference 2-10.

In the case of the RODEX4 methodology, the LHGR limit is validated by the analysis of power histories generated as part of the neutronic analyses. The neutronic power histories are created based on given margins to the LHGR limit. As part of the RODEX4 methodology, variations are introduced into the power histories to account for power uncertainties and reactor operator flexibility uncertainties. The overpower LHGR limit is evaluated using the fast transient methodology to demonstrate that slow transients satisfy the fuel melt and cladding strain criteria.

3.2.2 **Kinetics Parameters**

Design Criteria

The design criteria for the core reactivity coefficients are as follows:

- Void reactivity coefficient due to boiling in the active channel shall be negative
- Doppler coefficient shall be negative at all operating conditions
- Power coefficient shall be negative at all operating conditions.

Bases

Fuel assembly designs in which less moderation and/or higher temperatures reduce the core reactivity will therefore act as an automatic shutdown mechanism. Thus, prompt reactivity insertion events such as the control rod drop accident have an inherent shutdown mechanism. AREVA calculates the reactivity coefficients on a plant and cycle specific basis through application of the standard neutronics design and analysis methodology (References 3-1, 3-2, and 3-4).

3.2.3 **Stability**

Design Criteria

New fuel designs and new fuel design features must be stable (core decay ratio <1.0) and should exhibit channel decay ratio characteristics equivalent to existing NRC-approved AREVA fuel designs.

Bases

Determination of the effect of all fuel designs and design features on core stability is made on a cycle-specific basis. Associated with these calculations is confirmation of existing power / flow range exclusion regions or redefinition of the regions, as necessary.

AREVA uses the NRC-approved STAIF code (References 3-3 and 3-5) for stability evaluations. STAIF is a frequency domain code that simulates the dynamics of a BWR. AREVA performs cycle-specific analyses in order to establish reactor operating parameters that ensure stable operation throughout the cycle.

OPRM Set Points

For plants that have implemented Long Term Stability Option III or Enhanced Option III, analyses are performed to establish set points for the OPRM system to ensure that the SLMCPR is not violated should the plant experience unstable oscillations. AREVA uses the NRC-approved RAMONA5-FA code (Reference 3-6) in the OPRM set point analyses. For EO-III, the Reference 3-7 methodology is used.

3.2.4 Core Reactivity Control

Design Criteria

The design of the assembly shall be such that the technical specification shutdown margin will be maintained. Specifically, the assemblies and the core must be designed to remain subcritical by the technical specification margin with the highest reactivity worth control rod fully withdrawn and the remaining control rods fully inserted. Calculated shutdown margin is verified using startup critical data. At a minimum, this verification is performed at beginning-of-cycle (BOC) for each reactor.

Bases

Shutdown margin is calculated on a cycle-specific basis using NRC-approved methodology (References 3-1, 3-2, and 3-4). It is calculated at exposure points throughout the cycle in order to determine the minimum shutdown margin for a cycle. The calculated shutdown margin is reported on a plant and cycle specific basis as required in Reference 3-2. AREVA also confirms the worth of the standby liquid control system on a cycle specific basis using the technical specification values of boron concentration.

3.3 *NRC-Accepted Topical Report References*

The NRC has approved the following licensing topical reports that describe the methods and assumptions used by AREVA to demonstrate the adequacy of its fuel system nuclear design. These reports address nuclear design criteria and required fuel and thermal conditions used in licensing analyses. The purpose of each topical report and restrictions on the methods presented are described in the following sections.

3-1: 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, March 1983.

- Purpose: Development of BWR core analysis methodology which comprises codes for fuel neutronic parameters and assembly burnup calculations, reactor core simulation, diffusion theory calculations, core and channel hydrodynamic stability predictions, and producing input for nuclear plant transient analysis. Procedures for applying the codes for control rod drop, control rod withdrawal and fuel misloading events have been established.

- SER Restrictions: No restrictions

- Implementation of SER Restrictions:

  None

- Observations: Portions of this topical report have been superseded by subsequently approved codes or methodologies. Superseded and currently applicable portions are identified below:

  Superseded Portions:

  Fuel Assembly Depletion Model - XFYRE replaced with CASMO-4 (see Reference 3-4).

  Core Simulator - XTGBWR replaced with MICROBURN-B2 (see Reference 3-4).

  Diffusion Theory Model - XDT replaced with CASMO-4 (see Reference 3-4).

  Stability Analysis - COTRAN replaced with STAIF (see Reference 3-5).

  Control Rod Withdrawal - XTGBWR replaced with MICROBURN-B2 (see Reference 3-4).

Fuel Misloading Analysis – XFYRE replaced with CASMO-4 and XTGBWR replaced with MICROBURN-B2. These analyses are performed to verify that the offsite dose due to such events does not exceed a small fraction of 10 CFR 100 guidelines as described and approved in Reference 3-2.

Applicable Portions:

Control Rod Drop Accident – This analysis is performed using COTRAN.

Control Rod Withdrawal – This analysis determines the change in CPR (ΔCPR) for error rod patterns. In addition a check is made that the LHGR does not exceed the transient (PAPT) LHGR limit.

Neutronic Reactivity Parameters – These parameters are determined as described in the topical report but using the most recently approved codes.

Void Reactivity Coefficient – Method used to calculate core void reactivity coefficient is the same but MICROBURN-B2 is used instead of XTGBWR.

Doppler Reactivity Coefficient – Method used to calculate the core average Doppler coefficient is the same but CASMO-4 is used instead of XFYRE.

Scram Reactivity – Method used is the same except MICROBURN-B2 is used instead of XTGBWR.

Delayed Neutron Fraction – Calculated using CASMO-4 instead of XFYRE.

Prompt Neutron Lifetime – Calculated using CASMO-4 instead of XFYRE.

3-2: XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.

- Purpose: Summarize the types of BWR licensing analyses performed, identify each with approved computer codes and methodologies, and develop a reload reporting format.

- SER Restrictions: Conditions imposed were based on pending approvals of outstanding topical reports which have been subsequently approved.

- Implementation of SER Restrictions: This restriction is no longer applicable (because of subsequent approvals).

- Observations: Many of the codes and methodologies referenced have changed or have been replaced since the report was approved.

- Clarifications: AREVA provided a clarification related to the topical report in References 28 and 29. The clarification was associated with the use of power and flow dependent LHGR multipliers to establish LHGR limits that provide adequate margin during events initiated from off-rated conditions.

3-3: EMF-CC-074(P)(A) Volume 1, "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2 "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report," Siemens Power Corporation, July 1994.

- Purpose: Provide a methodology for the determination of the thermal-hydraulic stability of BWRs, including reactivity feedback effects.

- SER Restrictions:

  1. The core model must be divided into a minimum of 24 axial nodes.
  2. The core model must be divided into a series of radial nodes (i.e., thermal-hydraulic regions or channels) in such a manner that:
     a) No single region can be associated with more than 20 percent of the total core power generation. This requirement guarantees a good description of the radial power shape, especially for the high power channels.
     b) The core model must include a minimum of three regions for every bundle type that accounts for significant power generation.
     c) The model must include a hot channel for each significant bundle type with the actual conditions of the hot channel.
  3. Each of the thermal-hydraulic regions must have its own axial power shape to account for 3-D power distributions. For example, high power channels are likely to have more bottom peaked shapes.
  4. The collapsed 1-D cross sections must represent the actual conditions being analyzed as closely as possible, including control rod positions.
  5. The STAIF calculation must use the "shifted Nyquist" or complex pole search feature to minimize the error at low decay ratio conditions.

- Implementation of SER Restrictions: The SER restrictions are implemented in the code and the user's manual for STAIF. The requirements will automatically be satisfied if the code defaults are used and the MICROBURN-B2 STAIF guideline is followed.

- Observations: Stability analysis procedures described in Reference 3-1 were superseded by the approval of the STAIF code (References 3-3 and 3-5).

3-4: 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, October 1999.

- Purpose: Replace the MICBURN-3/CASMO-3G bundle depletion codes and the MICROBURN-B simulator code with the codes CASMO-4 and MICROBURN-B2, respectively.

- SER Restrictions:

1. The CASMO-4/MICROBURN-B2 code systems shall be applied in a manner that predicted results are within the range of the validation criteria (Tables 2.1 and 2.2) and measurement uncertainties (Table 2.3) presented in EMF-2158(P).

2. The CASMO-4/MICROBURN-B2 code system shall be validated for analyses of any new fuel design which departs from current orthogonal lattice designs and/or exceed gadolinia and U-235 enrichment limits.

3. The CASMO-4/MICROBURN-B2 code system shall only be used for BWR licensing analyses and BWR core monitoring applications.

4. The review of the CASMO-4/MICROBURN-B2 code system should not be construed as a generic review of the CASMO-4 or MICROBURN-B2 computer codes.

5. The CASMO-4/MICROBURN-B2 code system is approved as a replacement for the CASMO-3G/MICROBURN-B code system used in NRC-approved AREVA BWR licensing methodology and in AREVA BWR core monitoring applications. Such replacements shall be evaluated to ensure that each affected methodology continues to comply with its SER restrictions and/or conditions.

6. AREVA shall notify any customer who proposes to use the CASMO-4/MICROBURN-B2 code system independent of any AREVA fuel contract that conditions 1 through 4 above must be met. AREVA's notification

shall provide positive evidence to the NRC that each customer has been informed by AREVA of the applicable conditions for using the code system.

- Implementation of SER Restrictions: The SER restrictions relevant to methodology used by AREVA are implemented in engineering guidelines.

- Observations: None.

3-5: EMF-CC-074(P)(A) Volume 4, Revision 0, "BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000.

- Purpose: Document and justify enhancements to the STAIF code including the capability to accept input from the code MICROBURN-B2. Justify a modification to the approved stability criteria for STAIF in conjunction with input from both MICROBURN-B and MICROBURN-B2. The STAIF code is used to perform stability analysis for BWRs.

- SER Restrictions:

  The SER concludes that the STAIF code is acceptable for best-estimate decay ratio calculations. This conclusion applies to the three types of instabilities relevant to BWR operation, which are quantified by the hot-channel, core-wide, and out-of-phase decay ratios. The staff estimates that STAIF decay ratio calculations for the decay ratio range of 0.0 to 1.1 are accurate within:

  +/- .2 for the hot-channel decay ratio
  +/- .15 for the core-wide decay ratio
  +/- .2 for the out-of-phase decay ratio

  The staff concludes that the proposed modification of the E1A acceptance criteria for region-validation calculations is acceptable because it provides the intended protection against instabilities outside the E1A regions. The following E1A region-validation criteria are acceptable for the STAIF code:

  The calculated hot-channel decay ratio must be lower than .8.
  The calculated core-wide decay ratio must be lower than .85.
  The calculated out-of-phase decay ratio must be less than .8.

- Implementation of SER Restrictions: The SER restrictions are implemented in engineering guidelines.

- Observations: The NRC stated in Reference 36, that the revised stability criteria is applicable to calculations with the STAIF code with input from either MICROBURN-B or MICROBURN-B2.

3-6: BAW-10255PA Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," AREVA NP, May 2008.

- Purpose: Document and justify the use of the RAMONA5-FA code for performing cycle-specific DIVOM calculations.
- SER Restrictions:
  1. If a reduced scope of parameter variations is used to define the cycle-specific DIVOM slope as described in Section 7 of the TR, the scope must be justified and documented for NRC staff review.
  2. The NRC staff imposes a condition to perform a full code review of RAMONA5-FA, including constitutive relations, numerics, neutronic methods, and benchmarks before RAMONA5-FA can be used to calculate DIVOM curves in EFW operating domains without the 10 percent penalty on DIVOM slopes, as noted in Limitation and Condition No.3 below.
  3. The NRC staff imposes an interim 10 percent penalty on DIVOM slopes calculated using the RAMONA5-FA methodology under EFW conditions. This is an interim restriction that will be revised when the full RAMONA5-FA Code review is completed.
- Implementation of SER Restrictions: SER restriction 1 is implemented in engineering guidelines. SER restrictions 2 and 3 state that all DIVOM analyses performed in the EFW (MELLLA+) domain need to impose an interim 10% penalty on DIVOM slopes. This penalty was to remain until the NRC staff had the opportunity to perform a full code review of RAMONA5-FA. The NRC has since performed a full code review of RAMONA5-FA and has issued a new SE removing the 10% penalty on EFW DIVOM calculations. The new SE documenting the removal of this penalty is given in Reference 41.
- Observations: None.

3-7: ANP-10262PA Revision 0, "Enhanced Option III Long Term Stability Solution," AREVA NP, May 2008.

- Purpose: Document and justify the applicability of the Enhanced Option III Long Term Stability Solution methodology to the EPU/MELLLA+ operating domain.

- SER Restrictions:

  1. The NRC staff has not reviewed the hardware and software implementation of EO-III Long Term Stability Solution because it will be plant specific. AREVA has stated that implementation is not part of the generic EO-III Long Term Stability Solution, even though the EO-III Long Term Stability Solution implements an additional scram function (the channel-stability exclusion region) not present in the original Option III platforms. Plant implementations, including those using any original Option III platform, will require plant-specific reviews.

  2. The original Option III is already approved for plant operation up to 20 percent EPU. The EO-III Long Term Stability Solution is an extension of Option III, where the DIVOM correlation is guaranteed to be well-behaved by the channel-stability exclusion region. Thus, the EO-III Long Term Stability Solution is, in essence, an Option III implementation with the added channel-stability exclusion region scram. Therefore, the NRC staff finds that EO-III is a technically acceptable methodology for any reactor operating up to 20 percent EPU conditions.

  3. The confirmation analyses documented in Section 5 of TR ANP-10262(P), Revision 0, and the response to the NRC staff RAI, indicate that the EO-III Long Term Stability Solution methodology provides significant protection against MCPR criteria violations during anticipated instability events even under high-power-density conditions, including EPU and MELLLA+. Under all analyzed conditions, the loss of MCPR margin induced by the instability event

is compensated by the gain in MCPR margin induced by the reduction in flow, so that the net MCPR margin is positive. Based on this analysis, the NRC staff finds that the EO-III Long Term Stability Solution is a technically acceptable methodology for any reactor operating up to MELLLA+ conditions. Extension of operating domains beyond MELLLA+ have not been considered by the NRC staff and will require a re-evaluation of the EO-III Long Term Stability Solution scram effectiveness by the NRC staff.

4. Operation with feedwater heaters out of service (FWHOOS) is not anticipated in EFW like MELLLA+; therefore TR ANP-10262(P), Revision 0, specifies the use of equilibrium feedwater conditions. If a plant-specific application of the EO-III Long Term Stability Solution allows for a FWHOOS condition, two SPT regions will have to be calculated, with and without FWHOOS. TSs must enforce the change of SPT region settings when the FWHOOS condition is declared. Alternatively, a plant-specific application may choose to implement the more conservative of the two SPT regions.

5. The EO-III Long Term Stability Solution does not provide an integrated backup stability solution if the primary stability protection system is declared inoperable. Instead, it provides an example of TSs and rationale for their applicability in the responses to the NRC staff RAI. Therefore, the NRC staff review of the stability related TS requirements and/or a different backup stability solution must be performed on a plant-specific basis.

6. Plant-specific applications will include the specification of the backup stability protection. One possible EO-III Long Term Stability Solution backup stability protection is the SPT, which provides an automated scram upon entry on the channel-stability exclusion region. The SPT is an acceptable backup stability protection solution for up to 120 days (typical TS range) if the primary stability protection system is declared inoperable. However, the SPT scram region must include the natural circulation line. To be an acceptable backup stability

solution, the SPT must include the following scram conditions:
(1) recirculation pumps are tripped, or (2) the power flow inside the channel-stability exclusion region. Either of the two conditions should result in scram. In addition to the SPT, administrative interim corrective actions must be enforced with cycle-specific regions. The NRC staff finds that a SPT implemented with the above conditions would provide an acceptable backup stability implementation for up to 120 days under MELLLA+ conditions because it provides protection for the most likely scenarios where large amplitude oscillations could occur.

7. Plant specific applications will include an evaluation of the uncertainty induced by the presence of bypass voids on the OPRM and APRM readings. OPRM uncertainties will result in a set-down of the OPRM PBDA setpoint. APRM uncertainties will be applied to the SPT exclusion region.

- Implementation of SER Restrictions: The SER restrictions define requirements to be addressed for the initial licensing application of Enhanced Option III at each plant. These requirements will be addressed for each Technical Specification submittal for Enhanced Option III as described in the methodology Topical Report.

- Observations: The NRC stated that the Enhanced Option III methodology is technically acceptable for the EPU/MELLLA+ domain. Any extension of the operating window beyond MELLLA+ will require a resubmittal of the methodology.

4.0 THERMAL AND HYDRAULIC DESIGN

Thermal-hydraulic analyses of the fuel and core are performed to verify that design criteria are satisfied and to establish an appropriate value for the MCPR fuel cladding integrity safety limit.

4.1 *Regulatory Requirements*

The acceptance criteria of SRP Section 4.4 Thermal and Hydraulic Design are based on meeting the relevant requirements of General Design Criterion 10, as it relates to the reactor core design, with appropriate margin to assure that SAFDLs are not exceeded during normal operation and AOOs. Specific criteria are identified in Reference 2-10 and discussed below.

4.2 *Thermal and Hydraulic Design Analyses*

4.2.1 Hydraulic Compatibility

Design Criteria

The hydraulic flow resistance of the reload fuel assemblies shall be sufficiently similar to existing fuel in the reactor such that there is no significant degradation in total core flow or maldistribution of the flow among assemblies in the core.

Bases

The Standard Review Plan (Reference 1) does not contain an explicit criterion for fuel assembly hydraulic compatibility. However, flow differences between assembly types in a mixed core need to be accounted for in assuring that all design criteria are satisfied.

The component hydraulic resistances in the reactor core are determined by a combination of analytical techniques and experimental data. For example, the single-phase flow resistances of the inlet region, bare rod region, spacers, and upper tie plate of the AREVA fuel designs and co-resident designs are generally determined in single phase flow tests with full scale assemblies. The two-phase flow resistances of appropriate components are determined from the single-phase loss coefficients and two-phase flow models. The prediction of pressure drop by a combination of single-phase loss coefficients and two-phase flow models has been experimentally verified.

The AREVA thermal-hydraulic methodology implicitly includes the impact of assembly differences on the individual assembly flow. The overall criterion for acceptability is that individual fuel types must be in compliance with the thermal hydraulic limits. To assure this, for reload analyses, if there is more than a specified difference in assembly orifice flow for a given (or specified) assembly power at rated conditions (i.e., full flow and full power), additional core stability evaluations will be performed with the STAIF methodology (Reference 3-5). The purpose of these evaluations is to better define the core stability behavior with this mismatch in flow. The MCPR performance remains protected by compliance with the safety and operating limits.

4.2.2 Thermal Margin Performance

Design Criteria

The fuel design shall fall within the limits of applicability of the approved critical heat flux (CHF) or critical power correlations. New fuel assembly designs and/or changes in existing assembly designs shall minimize the likelihood of boiling transition during normal reactor operation and AOOs. The applicable critical power correlation will be used to determine the operating limits and, for consistency, will be used to monitor the core.

Bases

AREVA fuel and reload cores are designed so that operation within the technical specification limits ensures that 99.9% of the fuel rods are expected to avoid boiling transition during AOOs. An NRC-approved CHF or critical power correlation is used by AREVA to determine operating and safety limits during the design of a reload core, and, for consistency, the same CHF or critical power correlation is used to monitor the core during operation.

Operation of a BWR requires protection against fuel damage during normal reactor operation and AOOs. A rapid decrease in heat removal capacity associated with boiling transition could result in high temperatures in the cladding, which may cause cladding degradation and a loss of fuel rod integrity. Protection of the fuel against boiling transition assures that such degradation is avoided. This protection is accomplished by determining the operating limit minimum critical power ratio (OLMCPR) each cycle.

The AREVA thermal limits analysis methodology, THERMEX, is described in Reference 4-2. The thermal limits methodology in THERMEX consists of a series of related analyses which establish an OLMCPR. The OLMCPR is determined from two calculated values, the safety limit MCPR (SLMCPR) and the limiting transient ΔCPR. The overall methodology is comprised of four major segments: 1) reactor core hydraulic methodology, 2) a critical power correlation, 3) plant transient simulation methodology, and 4) critical power methodology.

AREVA fuel assembly pressure drop methodology is presented in Reference 4-1. This methodology is part of the calculational method used by AREVA to determine the assembly pressure drop that is used to calculate assembly flows for a BWR core. The pressure drop methodology determines the void fraction and the two-phase pressure losses, which are in turn used as input to the calculation of the assembly pressure drop using the XCOBRA computer code described in Reference 4-2.

The AREVA fuel assembly critical power performance is established by means of an empirical correlation based on results of boiling transition test programs (see References 4-5, 4-6, and 4-7). The critical power performance of co-resident fuel, which is not in the AREVA correlation development database, is determined using the methodology described in Reference 4-4.

The methodology and computer codes for AREVA BWR plant transient analyses are the XCOBRA-T code (Reference 5-6) and the COTRANSA2 code (Reference 5-7). The COTRANSA2 code is used to calculate BWR system behavior for steady-state and transient conditions. This behavior is then used to provide input to the XCOBRA-T and XCOBRA codes, from which critical power ratios are determined for limiting transients.

References 4-3 and 4-8 provide the basis for the AREVA methodologies for determining the SLMCPR which ensures that 99.9% of the fuel rods are expected to avoid boiling transition. The SLMCPR is determined by statistically combining calculational uncertainties and plant measurement uncertainties associated with the calculation of MCPR. This determination is carried out by a series of Monte Carlo calculations in which the variables affecting boiling transition are varied randomly and the total number of rods experiencing boiling transition is determined for each Monte Carlo trial. The AREVA CPR correlations depend on the core pressure, channel mass velocity, planar enthalpy, a local peaking function, radial and axial power, and channel geometry (channel bow). Power distribution uncertainties used in the calculation are those associated with the core monitoring system and are obtained from references such as Reference 3-4. The CPR correlation uncertainty is accounted for through the additive constant uncertainty. The additive constant uncertainties for specific fuel designs used in the determination of the SLMCPR are determined using the methodologies and values provided in References 4-5, 4-6, or 4-7.

Plant measurement uncertainties (such as pressure, core flow, feedwater temperature, etc.) are plant dependent and are obtained from the utility.

4.2.3 Fuel Centerline Temperature

Design Criteria

Fuel design and operation shall be such that fuel centerline melting is not predicted for normal operation and AOOs.

Bases

This design criterion is addressed during the specific mechanical design analysis performed for each fuel type. The bases are discussed in Section 2.2.12.4 of this document.

4.2.4 **Rod Bowing**

Design Criteria

The anticipated magnitude of fuel rod bowing under irradiation shall be accounted for in establishing thermal margins requirements.

Bases

The bases for rod bow are discussed in Section 2.2.6. Rod bow magnitude is determined during the mechanical design analyses done for each fuel type. The need for a thermal margin rod bow penalty is evaluated on a plant and cycle specific basis. Post-irradiation examinations of BWR fuel fabricated by AREVA show that the magnitude of fuel rod bowing is small and the potential effect of this bow on thermal margins is negligible. Rod bow at extended burnups does not affect thermal margins because of the lower powers experienced by high exposure assemblies.

4.2.5 **Bypass Flow**

Design Criteria

The bypass flow characteristics of the reload fuel assemblies shall not differ significantly from the existing fuel in order to provide adequate flow in the bypass region.

Bases

The Standard Review Plan (Reference 1) does not contain an explicit criterion for fuel assembly bypass flow characteristics. However, significant changes in bypass region flow may alter the response characteristics of the incore neutron detectors. In order to avoid altering the incore neutron detector response characteristics, AREVA evaluates bypass flow fraction on a plant and cycle specific basis to assure that the bypass flow characteristics are not significantly altered.

4.3 *NRC-Accepted Topical Report References*

The NRC has approved the following licensing topical reports that describe the methods and assumptions used by AREVA to demonstrate the adequacy of its thermal and hydraulic fuel system design analyses. These reports address thermal and hydraulic criteria and thermal conditions used in steady-state and transient licensing analyses. The purpose of each topical report and restrictions on the methods presented are described in the following sections.

4-1: XN-NF-79-59(P)(A), "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, November 1983.

- Purpose: Develop a methodology for determining the BWR assembly pressure drop which determines the assembly coolant flow and which varies with total recirculating flow and reactor power.
- SER Restrictions: No restrictions.
- Implementation of SER Restrictions: None.
- Observations: This methodology continues to be used and incorporates experimental pressure drop data for new fuel and spacer designs.

4-2: **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, January 1987.**

- Purpose: Provide an overall methodology for determining a MCPR operating limit. The methodology comprises CHF correlations, fuel hydraulic characteristics, safety limit analyses, AOO analyses, and statistical combination of uncertainties.

- SER Restriction: Monitoring systems other than POWERPLEX®*CMSS may be used provided that the associated power distribution uncertainties are identified and appropriate operating parameters compatible with ENC transient safety analyses are monitored. Whatever monitoring system is used should be specifically identified in plant submittals.

- Implementation of SER Restriction: The SER restriction is implemented in engineering guidelines.

- Observations: Although Reference 4-2 only discusses applications to ENC 8x8 and 9x9 fuel types, the overall methodology is applicable to other AREVA fuel designs when appropriate CHF correlations are implemented. Subsequent to the approval of this topical report, AREVA developed and the NRC approved the use of generic design criteria for new fuel designs (Reference 2-10). In the SER/TER for Reference 2-10, the NRC concurred with the continued applicability of the methodology in Reference 4-2 (with the exception of the CHF correlation) for demonstrating compliance with thermal hydraulic design criteria.

- Some of the computer codes referenced in the topical report have been superseded by other NRC-approved codes (e.g., COTRANSA with COTRANSA2, XTGBWR with MICROBURN-B2) and the XN-3 CHF correlation has been supplemented with the

---

* POWERPLEX is a trademark registered in the U.S. and various other countries.

NRC-approved SPCB CHF correlation (see Reference 4-5) and ACE critical power correlations (see Reference 4-6 and Reference 4-7).

The SER states “Based on the similarity of the computational models of the two codes (XCOBRA and XCOBRA-T) and the NRC approval of the XCOBRA-T code (Reference 5-6), we find the use of the steady-state code [XCOBRA] acceptable in this context.” XCOBRA continues to be applied for steady-state analyses.

4-3: ANF-524(P)(A) Revision 2 and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation, November 1990.

- Purpose: Provide a methodology for the determination of the SLMCPR.

- SER Restrictions:

  1. The NRC approved MICROBURN-B power distribution uncertainties should be used in the SLMCPR determination.

  2. Since the ANFB correlation uncertainties depend on fuel design, in plant-specific applications the uncertainty value used for the ANFB additive constants should be verified. [Note, ANFB was subsequently replaced in the methodology by the SPCB correlation, (Reference 4-5) and the ACE critical power correlations (Reference 4-6 and Reference 4-7).]

  3. The CPR channel bowing penalty for non-ANF fuel should be made using conservative estimates of the sensitivity of local power peaking to channel bow.

  4. The methodology for evaluating the effect of fuel channel bowing is not applicable to reused second-lifetime fuel channels.

- Implementation of SER Restrictions: SER restrictions 1 and 2 are implemented in engineering guidelines and automation tools. Restrictions 3 and 4 are implemented in engineering guidelines.

- Observations: The critical power methodology is a general methodology which may be used with all AREVA developed CHF correlations that include additive constants and additive constant uncertainties.

  Power distribution uncertainties for MICROBURN-B2 and other AREVA core simulator codes approved by the NRC will be used in the CPR methodology.

  As additive constants and additive constant uncertainties are fuel type specific, they do not change for each plant specific application, as noted in SER restriction 2.

4-4: EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.

- Purpose: Present and justify the use of AREVA critical power correlations to co-resident fuel (non-AREVA manufactured).
- SER Restrictions: Technology transfer to licensees who may be responsible for using these processes will be accomplished through AREVA and licensee procedures consistent with the requirements of GL 83-11, Supplement 1. This process includes the performance of an independent benchmarking calculation by AREVA for comparison to licensee-generated results to verify that the application of AREVA CHF correlations is properly applied for the first application by a licensee.
- Implementation of SER Restrictions: The SER restriction is implemented in engineering work practices.
- Observations: None.

4-5: EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," AREVA NP, September 2009.

- Purpose: Present and justify a critical power correlation applicable for the ATRIUM-9B and ATRIUM-10 fuel designs.

- SER Restrictions:

  1. The SPCB correlation is applicable to Framatome ANP, Inc. ATRIUM-9B and ATRIUM-10 fuel designs with a local peaking factor no greater than 1.5.
  2. If in the process of calculating the MCPR safety limit, the local peaking factor exceeds 1.5, an additional uncertainty of 0.026 for ATRIUM-9B and 0.021 for ATRIUM-10 will be imposed on a rod by rod basis.
  3. The SPCB correlation range of applicability is 571.4 to 1432.2 psia for pressure, 0.087 to 1.5 Mlb/hr-ft^2 for inlet mass velocity and 5.55 to 148.67 Btu/lbm for inlet subcooling.
  4. Technology transfer will be accomplished only through the process described in Reference 14, which includes the performance of an independent bench-marking calculation by FANP for comparison to the licensee-generated results to verify that the new CHF correlation (SPCB) is properly applied for the first application by the licensee.
  5. Application of this correlation and the proposed revisions to fuel designs other than the ATRIUM-9B and ATRIUM-10 designs require prior staff approval.

  Note, restrictions 1 – 4 are from Revision 1.

- Implementation of SER Restrictions: SER restrictions 1 and 5 are implemented in engineering guidelines. Restriction 2 is implemented in engineering guidelines and automation tools. Restriction 3 is directly implemented in engineering computer codes. Restriction 4 is implemented in engineering work practices.

- Observations: The purpose of Revision 2 was to modify the SPCB critical power correlation in the region of the uranium blanket at the top of the fuel. The purpose of Revision 3 was to make corrections to the correlation additive constants to account for an error in the KATHY loop operation.

- Clarifications: NRC concurrence with a clarification related to this topical report (Revision 1) was requested in References 31 and 32. The NRC concurrence with the clarification was provided in Reference 33. The clarification discusses the actions taken when the calculation values fall outside the correlation bounds.

4-6: ANP-10249P-A Revision 2 "ACE/ATRIUM-10 Critical Power Correlation," AREVA, March 2014.

- Purpose: Present and justify a new critical power correlation applicable for the ATRIUM-10 fuel design.

- SER Restrictions:

  1. The ACE/ATRIUM-10 methodology may only be used to perform evaluations of AREVA ATRIUM-10 fuel design. The ACE/ATRIUM-10 correlation may also be used to evaluate the performance of the co-resident fuel in mixed cores as discussed in Section 3.6 of the SE.

  2. ACE/ATRIUM-10 correlation shall not be used outside the range of applicability defined by the range of the test data prescribed in Table 2.1 of Reference 2.

  Note: Reference 2 from restriction 2 is ANP-10249P-A, Revision 2.

- Implementation of SER Restrictions:

  SER restriction 1 is implemented in engineering guidelines. Restriction 2 is directly implemented in the engineering software implementing the ACE correlation, ACELIB, for mass flow, pressure, and inlet subcooling. For maximum local peaking, the restriction is implemented via neutronics bundle design guidelines.

- Observations: The purpose of Revision 1 was to make corrections to the correlation additive constants to account for an error in the KATHY loop operation. The purpose of Revision 2 was to remove potentially non-physical behavior in the application of the K factor methodology.

4-7: ANP-10298P-A Revision 1 "ACE/ATRIUM 10XM Critical Power Correlation," AREVA, March 2014.

- Purpose: Present and justify a new critical power correlation applicable for the ATRIUM 10XM fuel design.

- SER Restrictions:

  1. The ACE/ATRIUM 10XM methodology may only be used to perform evaluations of AREVA ATRIUM 10XM fuel design. The ACE/ATRIUM 10XM correlation may also be used to evaluate the performance of the co-resident fuel in mixed cores as discussed in Section 3.4 of the SE.

  2. ACE/ATRIUM 10XM correlation shall not be used outside the range of applicability defined by the range of the test data prescribed in Table 2.1 of Reference 2.

     Note: Reference 2 from restriction 2 is ANP-10298P-A Revision 1

- Implementation of SER Restrictions:

  SER restriction 1 is implemented in engineering guidelines. Restriction 2 is directly implemented in the engineering software implementing the ACE correlation, ACELIB, for mass flow, pressure, and inlet subcooling. For maximum local peaking, the restriction is implemented via neutronics bundle design guidelines.

- Observations: The purpose of Revision 1 was to remove potentially non-physical behavior in the application of the K factor methodology.

4-8: ANP-10307PA Revision 0 "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.

- Purpose: Present and justify a new methodology for determining the safety limit minimum critical power ratio (SLMCPR) to incorporate the ACE critical power correlations (References 4.6 and 4.7) and a realistic fuel channel bow model approved in Reference 2-14.

- SER Restrictions:

  There are no specific SER restrictions

- Implementation of SER Restrictions:

- Observations: For non-typical situations, such as abnormal bow situations, transition cores, and new channel designs, the channel bow model will be applied in a conservative manner through use of [modified growth models or an increased bias and an increased uncertainty] according to Reference 2-14.

5.0 ACCIDENT ANALYSIS

This section addresses the methodologies used to perform the analyses of AOOs and postulated accidents in SRP Chapter 15 that are related to core reloads.

5.1 *Anticipated Operational Occurrences*

AOOs are evaluated to determine thermal operating limits to ensure applicable event acceptance criteria are met. Table 5-1 lists those AOOs analyzed with AREVA's approved methodologies.

Table 5-1 Anticipated Operational Occurrence Analyses

SRP No.	Chapter 15 AOO Analysis
15.1.1 – 15.1.3	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow
15.2.1 – 15.2.5	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)
15.2.7	Loss of Normal Feedwater Flow
15.3.1 – 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power
15.4.4 – 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a BWR Pressure Relief Valve

5.1.1 Regulatory Requirements

The specific criteria necessary to meet the requirements of the relevant GDCs 10, 15, and 26 for the AOOs listed in Table 5-1 (except SRP No. 15.4.2) are:

a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.
b) Fuel cladding integrity shall be maintained by ensuring that the CPR remains above the MCPR safety limit for BWRs based on acceptable CHF correlations (see SRP Section 4.4).
c) Cladding strain does not exceed 1%.
d) The event should not generate a more serious plant condition without other faults occurring independently.

The criteria necessary to meet GDCs 10, 20, and 25 for SRP 15.4.2 AOO are:

a) The thermal margin limits (MCPR) specified in SRP Section 4.4, II.1 are met.
b) Uniform cladding strain does not exceed 1%.

Analyses are performed to demonstrate that the fuel performs within design criteria during AOOs and to establish appropriate operating limits for the reactor. To protect the established safety limit MCPR, evaluations of AOOs are performed which produce the limiting transient ΔCPR, which when added to the safety limit MCPR, defines the operating limit MCPR. The methodologies used for the analysis of these events are found in References 3-1, 3-2, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7, 4-8, 5-4, 5-5, 5-6, 5-7, and 5-12.

5.1.2 Limiting Transient Events

The loading of fresh fuel, regardless of design, into a reactor core may alter the characteristics of both steady-state core performance and plant transient response throughout each subsequent cycle of operation. Limiting conditions for plant operations are established to assure that acceptable thermal operating margins are maintained during all anticipated operations. Application of AREVA's methodology provides a basis for the determination that plant operation will meet appropriate safety criteria.

The evaluation of anticipated operational occurrences considers events identified in the FSAR. These events are generally classified as:

- Decrease in core coolant temperature
- Increase in reactor pressure
- Decrease in reactor coolant flow rate
- Reactivity and power distribution anomalies
- Increase in reactor coolant inventory
- Decrease in reactor coolant inventory
- Increase in reactor coolant flow
- Increase in reactor core coolant temperature.

Primarily because of the strong void reactivity feedback characteristic of a boiling water reactor, AOOs involving a decrease in reactor coolant inventory, a decrease in core flow, or an increase in core coolant temperature do not result in a limiting ΔCPR.

A decrease in core coolant temperature may result in a gradual core heatup until the high neutron flux scram setpoint is exceeded. Since the power excursion is slow and the fuel thermal response does not significantly lag the neutronic response, this event can be evaluated with either a transient code or a steady-state code.

Rapid reactor pressure increases may result in a thermal margin limiting event for some designs and conditions. The severity of the event is strongly dependent upon the reactivity state of the core, the valve closure characteristics initiating the event, and the performance of the scram shutdown system. Thus, specific event sequences at some reactor conditions may emerge as consistently most limiting in nature. Each potentially limiting event is considered in the determination of cycle limiting conditions for operation.

Reactor and power distribution anomalies are localized reactivity additions that are usually initiated by operator error in selecting and withdrawing a control rod. While the event during refueling and reactor startup conditions are not limiting, the rod withdrawal error at power is potentially limiting and considered in the determination of the thermal operating limits.

The two event categories which involve increases in either core coolant flow rate or reactor coolant inventory are dependent upon plant design and conditions. Both involve potentially limiting conditions at partial power and flow conditions, where the augmentation of flow (either recirculation or feed) to the maximum physical capacity of equipment is greatest. Effective designs and/or reactor protection systems may substantially mitigate the rate and potential acceleration of power production in the core or terminate the transient prior to serious degradation of thermal margin.

Prior to the initial cycle that AREVA provides reload fuel, a disposition of events is performed to identify the FSAR events that may be affected by a change in fuel or core design. From the affected events, the potentially limiting events relative to thermal margins are identified and analyzed. The following AOOs are generally identified as being potentially limiting:

- Turbine/generator trip without bypass
- Feedwater controller failure to maximum demand
- Loss of feedwater heating
- Control rod withdrawal error
- Recirculating flow increase events

Once the applicable set of limiting transients for thermal margin has been identified for a specific reactor, the analysis of each event at reactor conditions at which it is potentially limiting provides the basis for determining the thermal operating limits.

5.1.3 **Pressurization Transient Analysis**

Events that result in significant reactor pressure increases are those that result in the closure of the steam isolation or turbine valves. There are several potential causes for the valve closure including loss of generator load, excessive turbine vibration and reaching a system set point (e.g. water level, low system pressure). The sudden reduction in steam flow causes an increase in reactor system pressure and core power. The event is usually terminated by reactor scram. In many cases, turbine bypass valves and safety relief valves operate to limit the system pressure rise. The turbine trip, generator load rejection and MSIV closure events are included in this classification. The feedwater controller failure event has many of the characteristics of these same events as it is a combination of an increase in coolant inventory and decrease in core coolant temperature event followed by an increase in reactor pressure event when the high water level trip setpoint is reached. The methodology used for the pressurization transient AOO analyses is presented in References 4-2, 5-6, and 5-7.

The plant transient AOO analysis methodology is also used in the overpressurization analyses to demonstrate compliance with the ASME pressure vessel code requirements.

5.1.4 Generic Loss of Feedwater Heating Methodology

The NRC has approved a generic AREVA methodology for evaluating the loss of feedwater heating (LFWH) transient in BWRs (Reference 5-12). The generic methodology is a parametric description of the critical power ratio response that was developed using the results of many applications of the previously approved plant and cycle specific methodology (Reference 3-1). Applying this methodology results in a conservative MCPR operating limit for the LFWH event.

5.1.5 Control Rod Withdrawal Error

During the control rod withdrawal error transient, the reactor operator is assumed to ignore the local power range monitor alarms and the rod block monitor alarms and continue to withdraw the control rod until the control rod motion is stopped by the control rod block. For this analysis the reactor is assumed to be in a normal mode of operation with the control rods being withdrawn in the proper sequence and all reactor parameters within technical specification limits and requirements. The most limiting case is when the reactor is operating at power with a high reactivity worth control rod fully inserted.

A detailed description of the AREVA control rod withdrawal error evaluation methodology is given in Reference 3-1. As noted in Reference 3-4, MICROBURN-B2 is approved for use in performing the analysis as a replacement to previously approved codes.

For BWR/6 reactors, the AREVA generic control rod withdrawal error analysis (Reference 5-4) is used. The generic analysis has been extended to cover maximum extended operating domain (MEOD) operation (Reference 5-5).

5.1.6 **Recirculation Flow Increase**

A slow flow excursion event assumes a failure of the recirculation flow control system such that the core flow increases slowly to the maximum flow physically attainable by the equipment. An uncontrolled increase in flow creates the potential for a significant increase in core power and heat flux. The analysis is performed using XCOBRA (Reference 4-2) to calculate the change in critical power ratio during the flow increase. Similar analyses are performed using MICROBURN-B2 (Reference 3-4) to determine the change in LHGR during a flow increase event.

The results of the slow flow excursion analyses are used to establish flow dependent MCPR ($MCPR_f$) limits and flow-dependent LHGR multipliers. The $MCPR_f$ limits ensure that the SLMCPR is protected if the recirculation flow is inadvertently increased to the maximum attainable value based on the plant equipment limitations.

5.1.7 **Determination of Thermal Limits**

The results of the evaluation of the anticipated operational occurrences at rated and off-rated power and flow conditions are used to establish power-dependent MCPR ($MCPR_p$) operating limits, including limits at rated power. As noted earlier, the results of the slow flow run-up event are used to establish the flow-dependent MCPR limits.

The results of reduced power and reduced flow analyses are used to ensure that the 1% strain and centerline melt criteria are met during anticipated operational occurrences. If adjustments to operating limits are needed, power and flow dependent

LHGR multipliers ($LHGRFAC_p$ and $LHGRFAC_f$) are established. The minimum of either the $LHGRFAC_p$ or $LHGRFAC_f$ multiplier is applied directly to the steady-state LHGR limit to determine the applicable LHGR operating limit to ensure that the 1% strain and centerline melt criteria are not violated during an AOO.

The scram insertion time used for the transient analyses may be based on either the technical specifications or plant measurement data. If plant measurement data are used to determine the scram performance assumed in the safety analyses, surveillance procedures are specified to determine the continued applicability of the data.

The core power and exposure distributions are monitored by the licensee throughout the cycle to assure that the end-of-cycle (EOC) axial power shape assumed in the licensing analysis will bound the actual EOC axial power shape.

5.2 ***Postulated Accidents***

Postulated accidents for BWRs evaluated for compliance with relevant GDCs are listed in Table 5-2 below.

Table 5-2 Postulated Accident Analyses

SRP No.	**Chapter 15 Accident Analysis**
15.3.3 – 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
15.4.9	Spectrum of Rod Drop Accidents (BWR)
15.4.9A	Radiological Consequences or Rod Drop Accident (BWR)
15.6.5	Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary
15.7.4	Radiological Consequences of Fuel Handling Accidents

5.2.1 Regulatory Requirements

The specific analytical criteria that are necessary to meet the requirements of the relevant GDCs for postulated accidents in Table 5-2 are:

SRP No. 15.3.3 – 15.3.4; GDCs 27, 28, and 31

a) Pressure in the reactor coolant and main steam systems should be maintained below design limits.

b) A small fraction of the fuel failures may occur, but these failures should not hinder the core coolability.

c) Radiological consequences should be a small fraction of 10 CFR 100 guidelines (generally < 10%).

d) The events should not generate a limiting fault or result in the consequential loss of the function of the reactor coolant system or containment barriers.

SRP No. 15.4.7; GDC 13

a) Offsite consequences due to fuel rod failure during this postulated accident should be a small fraction of 10 CFR 100 limits.

SRP No. 15.4.9; GDC 28

a) Reactivity excursions should not exceed a radially averaged fuel rod enthalpy greater than 280 cal/g at any axial location in any fuel rod.

b) The maximum reactor pressure should be less than “Service Limit C” defined in the ASME code (Reference 8).

c) The number of fuel rods predicted to reach assumed fuel failure thresholds and associated parameters such as the amount of fuel reaching melting conditions will be assessed in a radiological evaluation. The assumed failure thresholds are radially averaged fuel rod enthalpy greater than 170 cal/g at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions.

SRP No. 15.4.9A

a) Calculated exposure values should be less than 25% of the 10 CFR 100 exposure guideline values. The fission product source term used in the dose analysis is acceptable if it meets the guidelines of Regulatory Guide 1.77 (Reference 13).

SRP No. 15.6.5; GDC 35

a) Event-specific criteria are specified in: 10 CFR 50.46 and 10 CFR 50 Appendix K.

b) Regulatory Guide 1.3 (Reference 17) establishes a set of fission gas release fractions to be applied for radiological assessments. Radiological consequences are within the guidelines of 10 CFR 100 or 10 CFR 50.67.

SRP No. 15.7.4; GDC 61

a) Calculated exposure values should be less than 25% of the 10 CFR 100 exposure guideline values.

b) The model for calculating the whole-body and thyroid doses is acceptable if it incorporates the appropriate conservative measurements in Regulatory Guide 1.25 (Reference 18), with the exception of the guidelines for the atmospheric dispersion factors (χ/Q values). The acceptability of the χ/Q values is determined under SRP Section 2.3.4.

The methodologies used to analyze the hypothetical LOCAs and other postulated accidents are discussed in the following sections.

5.2.2 Pump Seizure

Recirculation pump seizure (RPS) event is considered an accident where an operating recirculation pump suddenly stops rotating. There are three parts to the RPS analysis - the simulation of the reactor system response, the determination of the number of failed fuel rods, and the radiological dose assessment.

The first part of the analysis uses the COTRANSA2 (Reference 5-7) and XCOBRA-T (Reference 5-6) codes to simulate the system and limiting assembly response. The key parameter determined is the ΔCPR for the limiting assembly during the event. The second part is the determination of the number of failed rods. The minimum CPR for the event is determined from the OLMCPR and the calculated ΔCPR. The AREVA MCPR safety limit methodologies (References 4-3 and Reference 4-8) are used to calculate the number of rods expected to experience boiling transition at the minimum CPR during the event. All rods that experience boiling transition are assumed to fail. This is a very conservative assumption because the minimum CPR occurs for a short period of time. The third part determines the dose from the number of rods which are calculated to fail. If the minimum CPR during the event remains above the safety limit MCPR, the dose calculation is not needed since operation at or above the safety limit MCPR meets the requirements of less than a small fraction of the 10 CFR 100 or 10 CFR 50.67 dose limits as appropriate.

Depending on the specific FSAR licensing requirements for a given reactor, RPS is specified as either an infrequent event or a limiting fault/design basis accident. For an infrequent event, the dose calculation result must remain below a small fraction (10%) of the 10 CFR 100 or 10 CFR 50.67 limits as appropriate. For a limiting fault/design basis accident, the dose calculation result must not exceed 10 CFR 100 or 10 CFR 50.67 limits as appropriate. If RPS is defined as a limiting fault/design basis accident, it is generally qualitatively dispositioned as mild and non-limiting as compared to a LOCA accident.

5.2.3 **Fuel Loading Error**

Two separate incidents are analyzed as part of the fuel misload analysis. The fuel mislocation error assumes a fuel assembly is placed in the wrong core location during refueling. The second incident, the fuel misorientation error, assumes that a fuel assembly is misoriented by rotation through 90° or 180° from the correct orientation when loaded into the reactor core. For both the fuel mislocation error and the fuel misorientation error, the assumption is made that the error is not discovered during the core verification and the reactor is operated during the cycle with a misloaded fuel assembly. Criteria for acceptability of the fuel misloading error analyses are that the off-site dose due to the event shall not exceed a small fraction of the 10 CFR 100 limits (Reference 4) as described in Reference 3-2.

The inadvertent misloading of a fuel assembly into an incorrect core location is analyzed with the MICROBURN-B2 methodology described in Reference 3-4. One approach to assuring that the 10 CFR 100 criteria are met is to calculate the minimum value of the MCPR in the misloaded core and the maximum LHGR in the mislocated fuel assembly. If the resulting minimum CPR is lower than the MCPR safety limit, the core configuration and power distribution are used to verify that at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition during full power operation with the misloaded fuel assembly. This prediction of the number of fuel rods in boiling transition is performed in accordance with the methodologies reported in Reference 4-3 or Reference 4-8.

The inadvertent rotation of a fuel assembly from its intended orientation is evaluated with the CASMO-4 methodology described in Reference 3-4. Similar to the analysis for misloaded fuel above, a minimum value of MCPR and a maximum LHGR associated with the orientation error are calculated. If the resulting minimum CPR is lower than the MCPR safety limit, the core configuration and power distribution associated with the misorientation error are used to verify that at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition during full power operation with the misoriented fuel assembly. This prediction of the number of fuel rods in boiling transition is performed in accordance with the methodology reported in Reference 4-3 or Reference 4-8. If an assessment of MCPR and LHGR show the potential for rod failures, a radiological evaluation may be needed to demonstrate that the off-site dose criterion (10 CFR 100) is met for both the fuel misload and fuel misorientation.

5.2.4 Control Rod Drop Accident Analysis

Analysis of the postulated CRDA is performed on a generic basis in Reference 3-1. Because the behavior of the fuel and core during such an event is not dependent upon system response, a generic CRDA parametric analysis can be applied to all BWR types.

The results of the generic CRDA analysis consist of deposited fuel enthalpy values parameterized as a function of effective delayed neutron fraction, Doppler coefficient, maximum (dropped) control rod worth, and four-bundle local peaking factor. For each cycle-specific application, values of each of the parameters are calculated and applied to the generic parametric analysis results and the resulting deposited fuel enthalpy is determined. The applicability of the generic analysis is verified for each application by comparison of the generic parameter range to the cycle-specific parameters, e.g., control rod worth, beta-eff and Doppler reactivity coefficient.

5.2.5 Loss of Coolant Accident Analysis

Plant specific ECCS analyses provide peak cladding temperature (PCT) and maximum local metal-water reaction (MWR) values and establish MAPLHGR limits for each fuel design. For the limiting single failure and limiting break, calculations are performed to determine the PCT and MWR values over the expected exposure lifetime of the fuel when operating at the MAPLHGR limit. The limiting break is determined by evaluating a spectrum of potential break locations, sizes, and single failures.

The limiting single failure of ECCS equipment is that failure which results in the minimum margin to the PCT criterion. The plant FSAR identifies potentially limiting ECCS single failures. AREVA analyzes those potentially limiting failures and identifies the worst single failure for the AREVA fuel design.

Evaluations and analyses to establish the location of the limiting break are performed. Analyses are performed for breaks on the suction and discharge sides of the recirculation pump. Non-recirculation line breaks are also evaluated but are generally non-limiting. The determination of the limiting location is based on minimum margin to the PCT criterion calculated for consistent fuel exposure conditions at each of the break locations. The MWR criterion is typically not challenged if the PCT limit is met, and is normally reported for the highest PCT case.

Analyses to establish the size of the limiting break are performed. Hypothetical split and guillotine piping system breaks are evaluated up to and including those with a break area equal to the cross-sectional area of the largest pipe in the recirculation system piping. As with the location spectrum, the determination of the limiting break size is based on the minimum margin to the PCT criterion.

The condition of the fuel during the LOCA analysis is conservatively based on exposure conditions which assure that the highest value of fuel stored energy is used. The condition of the fuel is based on fuel conditions associated with planar average exposure.

The AREVA Appendix K LOCA methodology is referred to as the EXEM BWR-2000 Evaluation Model (Reference 5-11). The reactor system and hot channel response is evaluated with RELAX (References 5-2, 5-8, and 5-9). Fuel assembly heatup during the LOCA is analyzed with HUXY (Reference 5-1) which incorporates approved cladding swelling and rupture models (Reference 5-3). Stored energy and fuel characteristics are determined with RODEX2 (Reference 2-3).

The use of Appendix K spray heat transfer coefficients for the ATRIUM-10 fuel design is justified in Reference 5-10.

5.2.6 Fuel Handling Accident During Refueling

The introduction of a new mechanical fuel design into a reactor core must be supported by an evaluation of the fuel handling accident for the new fuel design. When required, AREVA performs an incremental evaluation of the impact of the new fuel design on the fuel handling accident scenario defined in the FSAR. Using the boundary conditions and conservative assumptions given in the FSAR and the relevant characteristics of the new fuel design, AREVA calculates a conservative number of fuel rods expected to fail as a result of a fuel handling accident.

The radiological consequences of a fuel handling accident for a new mechanical fuel design are assessed based on the same reactor power history assumed in the evaluation of the existing fuel. The plenum activity for the new fuel is calculated based on the relative number of fuel rods per fuel assembly and relative maximum rod LHGR for the new and existing fuel designs.

5.3 ***NRC-Accepted Topical Report References***

The NRC-accepted topical reports for AOO and accident analyses are listed in the following sections.

5-1: XN-CC-33(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual," Exxon Nuclear Company, November 1975.

- Purpose: Develop a planar heat transfer model which includes rod-to-rod radiation. This code also includes the BULGEX model for the calculation of fuel rod strains and ballooning.

- SER Restrictions:

1. The staff, however, will require that a conservative reduction of 10% be made in the (spray heat transfer) coefficients specified in 10 CFR 50 Appendix K for 7x7 assemblies when applied to ENC 8x8 assemblies.

2. In each individual plant submittal employing the Exxon model the applicant will be required to properly take rod bowing in account.

3. Since GAPEX is not identical to HUXY in radial noding or solution scheme, it is required that the volumetric average fuel temperature for each rod be equal to or greater than that in the approved version of GAPEX. If it is not, the gap coefficient must be adjusted accordingly.

4. It has been demonstrated that the (2DQ local quench velocity) correlation gives hot plane quench time results that are suitably conservative with respect to the available data when a coefficient behind the quench front of 14000 Btu/(hr-ft^2-°F) is used.

5. It (Appendix K) requires that heat production from the decay of fission products shall be 1.2 times the value given by K. Shure as presented in ANS 5.1 and shall assume infinite operation time for the reactor.

6. It is to be assumed for all these heat sources (fission heat, decay of actinides and fission product decay) that the reactor has operated continuously at 102% of licensed power at maximum peaking factors allowed by Technical Specifications.

7. For small and intermediate size breaks, the applicability of the fission power curve used in the calculations will be justified on a case by case basis. This will include justification of the time of scram (beginning point in time of the fission power decrease) and the rate of fission power decrease due to voiding, if any.

8. The rate of (metal water) reaction must be calculated using the Baker-Just equation with no decrease in reaction rate due to the lack of steam. This rate equation must be used to calculate metal-water reactions both on the outside surface of the cladding, and if ruptured, on the inside surface of the cladding. The reaction zone must extend axially at least three inches.

9. The initial oxide thickness (that affects the zirconium-water reaction rate) used should be no larger than can be reasonably justified, including consideration of the effects of manufacturing processes, hot-functional testing and exposure.

10. Exxon has agreed to provide calculations on a plant by plant basis to demonstrate that the plane of interest assumed for each plant is the plane in which peak cladding temperatures occur for that plant.

- Implementation of SER Restrictions: SER restrictions 1, 2, 3, 4, 6, 7, 9, and 10 are implemented in engineering guidelines. Restrictions 5 and 8 are directly implemented in engineering computer codes.

- Observations: None.

5-2: XN-NF-80-19(P)(A) Volumes 2, 2A, 2B and 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, September 1982.

- Purpose: Provide an evaluation model methodology for licensing analyses of postulated LOCAs in jet pump BWRs. The methodology was developed to comply with 10 CFR 50.46 criteria and 10 CFR 50 Appendix K requirements.

- SER Restrictions: Counter-current flow limit correlation coefficients used in FLEX for new fuel designs that vary from fuel cooling test facility (FCTF) measured test configurations must be justified.

- Implementation of SER Restrictions: The FLEX computer code is no longer used. This was replaced in Reference 5-11.

- Observations: RELAX and FLEX, which are key computer codes in the methodology, have been subsequently modified as described in References 5-8 and 5-9, which documents the revised EXEM BWR Model, and in Reference 5-11 which documents EXEM BWR-2000 in which the RELAX code replaced FLEX. The EXEM BWR-2000 model supersedes the prior evaluation model.

5-3: XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, November 1982.

- Purpose: Incorporate the swelling and rupture models described in NUREG-0630 (Reference 12) which comply with 10 CFR 50 Appendix K requirements into the HUXY code (Reference 5-1).
- SER Restrictions: No restrictions.
- Implementation of SER Restrictions: None.
- Observations: The swelling and rupture model is currently applicable.

5-4: XN-NF-825(P)(A), "BWR/6 Generic Rod Withdrawal Error Analysis, MCPR$_p$," Exxon Nuclear Company, May 1986.

- Purpose: Modify approved control rod withdrawal error transient methodology (Reference 3-1) for application to BWR/6s or other BWRs with ganged control rods.

- SER Restrictions:

  1. The methodology and results are valid for operation within the power flow domain illustrated in Figure 4.1 of the topical report and for the fuel management scheme used for determining the operating states of the database. Use of other power-flow domains (e.g., the MEOD) or other fuel management schemes (e.g., the single rod sequence loading pattern) will require verification by analysis that the conclusions of this report are valid.

  2. Cycle specific analyses are not required if the operating power-flow region is bounded by that presented in the topical report and the core loading pattern and control rod patterns are consistent with the database used.

- Implementation of SER Restrictions:

  The SER restrictions are implemented in engineering guidelines.

- Observations: The original methodology, developed using the XTGBWR core simulator code which was superseded by MICROBURN-B2 (see Reference 3-4), is still applicable.

5-5: XN-NF-825(P)(A) Supplement 2, "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operations within the Extended Operating Domain," Exxon Nuclear Company, October 1986.

- Purpose: Extend the applicability of the Reference 5-4 licensing topical report to control rod withdrawal error transients for BWR/6 plants within the extended operating domain.

- SER Restrictions:

  1. The methodology and results are valid for operation within the power flow domain illustrated in Figure 3.1 of the topical report and for the fuel management scheme used for determining the operating states of the database for the MEOD. Other fuel management schemes will require verification by analysis that the conclusions of this report are valid.

  2. Cycle specific analyses are not required if the operating power-flow region is bounded by that presented in the topical report and the core loading pattern and control rod patterns are consistent with the database used.

- Implementation of SER Restrictions:

  The SER restrictions are implemented in engineering guidelines.

- Observations: The original methodology, developed using the XTGBWR core simulator code which was superseded with MICROBURN-B2 (see Reference 3-4), is still applicable.

5-6: XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.

- Purpose: Provide a capability to perform analyses of transient heat transfer behavior in BWR assemblies.

- SER Restrictions:

  1. XCOBRA-T was found acceptable for the analysis of only the following licensing basis transients:

     a) Load rejection without bypass
     b) Turbine trip without bypass
     c) Feedwater controller failure
     d) Steam isolation valve closure without direct scram
     e) Loss of feedwater heating or inadvertent high pressure coolant injection (HPCI) actuation
     f) Flow increase transients from low-power and low-flow operation

  2. XCOBRA-T analyses that result in any calculated downflow in the bypass region will not be considered valid for licensing purposes.

  3. XCOBRA-T licensing calculations must use NRC approved default options for void-quality relationship and two-phase multiplier correlations.

  4. The use of XCOBRA-T is conditional upon a commitment by ENC to a follow-up program to examine the XCOBRA-T void profile against experimental data from other sources.

- Implementation of SER Restrictions: SER restrictions 1, 2, and 3 are implemented in engineering guidelines. SER restriction 3 is also implemented through code controls (defaults, override warning messages). Restriction 4 was subsequently addressed in Reference 37 and no further action is required.

- Observations: None.

- Clarifications: NRC concurrence with an interpretation of the contents of the topical report was requested in References 24 and 25. The NRC concurrence with the interpretation was provided in Reference 26. The interpretation was with regard to a commitment to perform critical heat flux ratio evaluations at every node in the hot channel.

  NRC concurrence with clarifications related to SER and TER issues concerning the topical report was requested in References 28 and 29. The NRC concurrence with these clarifications was provided in Reference 30. These references clarify that XCOBRA-T is approved for the analysis of the following events:

SRP Section	Chapter 15 Analysis
15.1.1 – 15.1.3	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Demand
15.2.1 – 15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
15.2.7	Loss of Normal Feedwater Flow
15.3.1 – 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
15.3.3 – 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
15.4.4 – 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a PWR Pressure Relief Valve and BWR Pressure Relief Valve
15.8	Anticipated Transients Without Scram (the Initial Pressurization Only)

5-7: ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.

- Purpose: Develop an improved computer program for analyzing BWR system transients.
- SER Restrictions: The staff reviewed the subject safety evaluations and identified the following limitations that apply to COTRANSA2:
  1. Use of COTRANSA2 is subject to limitations set forth for methodologies described and approved for XCOBRA-T and COTRAN.
  2. The COTRANSA2 code is not applicable to the analysis of any transient for which lateral flow in a bundle is significant and non-conservative in the calculation of system response.
  3. For those analyses in which core bypass is modeled, the effect of a computed negative flow in the core bypass region should be shown to make no significant non-conservative contribution in the system response.
  4. Licensing applications referencing the COTRANSA2 methodology must include confirmation that sensitivity to the time step selection has been considered in the analysis.
- Implementation of SER Restrictions: SER restrictions 1, 2, and 4 are implemented in engineering guidelines. Restriction 3 is implemented in engineering guidelines and automation tools.
- Observations: The COTRANSA2 SER restrictions are similar to those for XCOBRA-T (Reference 5-6).

- Clarifications: NRC concurrence with clarifications related to SER and TER issues concerning the topical report was requested in References 28 and 29. The NRC concurrence with these clarifications was provided in Reference 30. These references clarify that COTRANSA2 is approved for the analysis of the following events:

SRP Section	Chapter 15 Analysis
15.1.1 – 15.1.3	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Demand
15.2.1 – 15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
15.2.7	Loss of Normal Feedwater Flow
15.3.1 – 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
15.3.3 – 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
15.4.4 – 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a PWR Pressure Relief Valve and BWR Pressure Relief Valve
15.8	Anticipated Transients Without Scram (the Initial Pressurization Only)

5-8: ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," Advanced Nuclear Fuels Corporation, January 1993.

- Purpose: Update the RELAX system blowdown code and FLEX refill code by reducing code instabilities and improving their predictive capabilities.

- SER Restrictions:

  1. The modified Dougall-Rohsenow heat transfer correlation has been shown to yield conservative results for many experimental measurements. The applicant used a suitable multiplier in the comparison calculations. Licensees will use this multiplier in licensing calculations.

  2. The revised model is valid within the range of applicability of the modified Dougall-Rohsenow heat transfer correlation.

  3. The staff requires that the revised evaluation model be protected with appropriate quality assurance procedures.

  4. The phase separation models will be limited to the models used in the topical report.

  5. The revised evaluation model will be limited to jet pump plant applications.

- Implementation of SER Restrictions: SER restrictions 1 and 2 are directly implemented in the RELAX engineering computer code. The revised model is valid within the range of applicability of the modified Dougall-Rohsenow heat transfer correlation – which was implemented in the coding of RELAX in terms of the logic for selecting the application of the correlation. The range of applicability, which is described in Section 3.3 of the NRC Safety Evaluation, is far wider than its application. Restriction 3 is implemented in the engineering software quality assurance procedures. With respect to restriction 4, the application of the phase separation models discussed in ANF-91-048 were revised in Reference 5-11. Restriction 5 was reiterated in Reference 5-11 and is implemented in the engineering guidelines.

- Observations: RELAX and FLEX, which are key computer codes in the methodology, have been subsequently modified as described in Reference 5-9, which documents the revised EXEM BWR Model, and in Reference 5-11 which documents EXEM BWR-2000 in which the RELAX code replaced FLEX. The EXEM BWR-2000 model supersedes the prior evaluation model.

5-9: ANF-91-048(P)(A) Supplements 1 and 2, "BWR Jet Pump Model Revision for RELAX," Siemens Power Corporation, October 1997.

- Purpose: Modify the jet pump model in the RELAX blowdown code to better predict jet pump performance for all ranges of LOCA conditions.
- SER Restrictions: No restrictions imposed.
- Implementation of SER Restrictions: None.
- Observations: RELAX and FLEX, which are key computer codes in the methodology, have been subsequently modified as described in Reference 5-11 which documents EXEM BWR-2000 in which the RELAX code replaced FLEX. The EXEM BWR-2000 model supersedes the prior evaluation model.

5-10 : EMF-2292(P)(A) Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000.

- Purpose: Justify the use of 10 CFR 50 Appendix K convective heat transfer coefficients during loss of coolant accident spray cooling for the ATRIUM-10 fuel design.
- SER Restrictions: None.
- Implementation of SER Restrictions: None.
- Observations: None.

5-11 : EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001.

- Purpose: Describes an evaluation model for licensing analyses of postulated LOCAs in jet pump BWRs. The methodology complies with 10 CFR 50.46 and 10 CFR 50 Appendix K.
- SER Restrictions: The staff concluded that the EXEM BWR-2000 Evaluation Model was acceptable for referencing in BWR LOCA analysis, with the limitation that the application of the revised evaluation model be limited to jet pump applications.
- Implementation of SER Restrictions: The SER restriction is implemented in engineering guidelines.
- Observations: Replace the FLEX code by the code RELAX in the BWR LOCA methodology.

NOTE: The Cross-string method of developing radiation heat transfer view factors used in the HUXY code has been replaced with the Ray-trace method which produces a direct finite element evaluation of view factors. This modification must be reported under 10 CFR 50.46 for each Licensee that uses the method or introduced as a plant specific methodology in a License Amendment Request if the EXEM BWR-2000 method is used.

5-12 : ANF-1358(P)(A) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005.

- Purpose: Develop a generic methodology for evaluating the loss of feedwater heating event.

- SER Restrictions:

  1. The methodology applies to BWR/3, BWR/4, BWR/5, and BWR/6 plants, and the fuel types which were part of the database (GNF-8X8, 9/9B and 11; ANF-8X8 and 9/9; and ATRIUM-9B and 10), provided that the exposure, the ratio of rated power and rated steam generation rate, rated feedwater temperature, and change in feedwater temperature are within the range covered by the data points presented in ANF-1358(P)(A), Revision 3.

  2. To confirm applicability of the correlation to fuel types outside the database, AREVA will perform additional calculations using the methodology, as described in Section 3.0 of the SER. In addition, AREVA calculations will be consistent with the methodology described in EMF-2158(P)(A), Revision 0 and comply with the guidelines and conditions identified in the associated NRC staff SE.

  3. The methodology applies only to the MCPR operating limit and the LHGR for the LFWH event.

- Implementation of SER Restrictions:

  The SER restrictions are implemented in engineering guidelines.

- Observations: The topical report includes results for GNF-8X8, -9/9B and -11; ANF-8X8, -9/9; and ATRIUM-9B and -10 fuel. Application of the correlation to fuel types outside the database needs to be verified according to SER Restriction Item 2.

6.0 CRITICALITY SAFETY ANALYSIS

In addition to reactor systems, AREVA performs criticality safety analyses of new fuel storage vaults and spent fuel storage pools. Storage array k-eff calculations are performed with the KENO.Va Monte Carlo code, which is part of the SCALE Modular Code System. The CASMO bundle depletion code (Reference 3-4) is used to calculate k_∞ values for fuel assemblies at beginning of life (new fuel storage) and as a function of exposure, void, and moderator temperature for both incore and in-rack (spent fuel storage) geometries.

The KENO.Va and the CASMO computer codes are widely used throughout the nuclear industry. They are used primarily for criticality safety and core physics calculations, respectively. AREVA has broad experience in the use of both of these codes. KENO.Va has been benchmarked by AREVA against critical experiment data to define appropriate reactivity biases and uncertainties.

AREVA performs criticality safety analyses consistent with the guidance given in References 19 - 23. The acceptance criteria (k-eff limit) for specific analyses are as defined in the plant Technical Specifications or from Chapters 9.1.1 (New Fuel Storage) or 9.1.2 (Spent Fuel Storage) of the Standard Review Plan NUREG-0800, References 19 and 20, respectively.

7.0 REFERENCES

1. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

2. FMM Revision 0 , *AREVA NP Fuel Sector Management Manual,* AREVA NP, effective April 2009.

3. "General Design Criteria for Nuclear Power Plants," *Code of Federal Regulations*, Title 10 "Energy," Part 50, Appendix A.

4. "Reactor Site Criteria," *Code of Federal Regulations*, Title 10 "Energy," Part 100.

5. "Accident Source Term," *Code of Federal Regulations*, Title 10 "Energy," Part 50.67.

6. *Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors*, Regulatory Guide 1.183, USNRC, July 2000.

7. "Domestic Licensing of Production and Utilization Facilities," *Code of Federal Regulations*, Title 10 "Energy," Part 50.

8. "Rules for Construction of Nuclear Power Plant Components," *ASME Boiler and Pressure Vessel Code*, Section III, 1977.

9. Swanson Analysis System, "ANSYS-Engineering Analysis System Theoretical Manual," 1977, and "ANSYS-User's Guide," 1979.

10. W. J. O'Donnell and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," *Nuc. Sci. Eng.*, 1964, 20:1.

11. JN-72-23 Revision 1, *Cladding Collapse Calculation Procedure*, Jersey Nuclear Company, Inc., November 1972.

12. *Cladding Swelling and Rupture Models for LOCA Analysis*, NUREG-0630, U.S. Nuclear Regulatory Commission, April 1980.

13. *Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors*, Regulatory Guide 1.77, U.S. Atomic Energy Commission, Washington, D.C., May 1974.

14. Letter, James F. Mallay (SPC) to Document Control Desk (NRC), “SER Condition for EMF-2209(P) Revision 1, ‘SPCB Critical Power Correlation’,” NRC:00:024, April 24, 2000.

15. NASA SP-221, *The NASTRAN Theoretical Manual*, 1969.

16. Letter, R. A. Copeland (Siemens Nuclear Power) to R. C. Jones (NRC), “No Subject,” RAC:050:91, May 13, 1992.

17. *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors*, Regulatory Guide 1.3 Revision 2, USNRC, June 1974.

18. *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors*, Regulatory Guide 1.25, U.S. Nuclear Regulatory Commission, March 1972.

19. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, NUREG-0800, Section 9.1.1 (New Fuel Storage), U.S. Nuclear Regulatory Commission, July 1981.

20. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, NUREG-0800, Section 9.1.2 (Spent Fuel Storage), U.S. Nuclear Regulatory Commission, July 1981.

21. *Spent Fuel Storage Facility Design Basis*, Regulatory Guide 1.13, Proposed Revision 2, U.S. Nuclear Regulatory Commission, December 1981.

22. *Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants*, ANSI/ANS American National Standard 57.2-1983, American Nuclear Society, October 1983.

23. *Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors*, ANSI/ANS American National Standard 8.17-1984, American Nuclear Society, January 1984.

24. Letter, James F. Mallay (SPC) to USNRC, “Clarification of SRP Chapter 15 Analyses Performed with XCOBRA-T and Checking of CHF Limits for Pump Seizure During SLO,” NRC:98:037, June 3, 1998.

25. Letter, Don Curet (SPC) to USNRC, “Equilibrium Quality Limits for Hench-Levy Limit Line Correlation,” NRC:98:044, June 25, 1998.

26. Letter, Cynthia A. Carpenter (NRC) to James F. Mallay (SPC), “Modification to Procedures for Use of XCOBRA-T,” June 10, 1999.

27. Letter, James F. Mallay (SPC) to Document Control Desk (NRC), “Assessment of Fuel Channel Design Calculations,” NRC:99:031, July 23, 1999.

28. Letter, James F. Mallay (SPC) to Document Control Desk (NRC), “Request for Concurrence on SER Clarifications,” NRC:99:030, July 28, 1999.

29. Letter, James F. Mallay (SPC) to Document Control Desk (NRC), “Revisions to Attachment 1 of Letter NRC:99:030, Request for Concurrence on SER Clarifications,” NRC:99:045, October 12, 1999.

30. Letter, Stuart Richards (NRC) to James F. Mallay (SPC), “Siemens Power Corporation Re: Request for Concurrence on Safety Evaluation Report Clarifications (TAC No. MA6160),” May 31, 2000.

31. Letter, James F. Mallay (Framatome ANP) to Document Control Desk (NRC), “BWR CHF Bounds Checking,” NRC:01:023, June 12, 2001.

32. Letter, James F. Mallay (Framatome ANP) to Document Control Desk (NRC), “BWR CHF Correlation Bounds Checking,” NRC:02:003, January 11, 2002.

33. Letter, William H. Ruland (NRC) to James F. Mallay (Framatome ANP), “Safety Evaluation for BWR CHF Correlation Bounds Checking Clarifications Relating to Topical Reports EMF-1997(P)(A) Revision 0 and EMF-2209(P)(A) Revision 1 (TAC No. MB3107),” July 2, 2002.

34. Letter, James F. Mallay (SPC) to Document Control Desk (NRC), “Clarification of Methodology for Analyzing Gadolinia-Bearing Fuel,” NRC:98:053, August 31, 1998.

35. Letter, Stuart A. Richards (NRC) to James F. Mallay (SPC), “Siemens Power Corporation Re: Request for Concurrence on Safety Evaluation Report Clarifications (MA6160),” November 3, 2000.

36. Letter, S. A. Richards (NRC) to James F. Mallay (SPC), “Supplement to Safety Evaluation and Technical Evaluation Report Clarifications for EMF-CC-074(P), Volume 4, Revision 0, ‘BWR Stability Analysis Assessment of STAIF with Input from MICROBURN-B2’ (TAC No. MA7221),” November 30, 2000.

37. XN-NF-84-105(P)(A) Volume 1 Supplement 4, *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis – Void Fraction Model Comparison to Experimental Data*, Advanced Nuclear Fuels Corporation, June 1988.

38. Letter, James F. Mallay (Framatome ANP) to Document Control Desk (NRC), “Clarification of Exposure Limit Applicable to Framatome ANP BWR Fuel,” NRC:02:029, June 27, 2002.

39. Letter, James F. Mallay (Framatome ANP) to Document Control Desk (NRC), “Clarification of Exposure Limit Applicable to Framatome ANP BWR Fuel,” NRC:02:041, September 4, 2002.

40. Letter, William H. Ruland (NRC) to James F. Mallay (Framatome ANP), “EMF-85-74(P)(A), Revision 0, Supplement 1 and Supplement 2, “RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model” – Clarification of Exposure Limit Applicable to Framatome ANP BWR Fuel (TAC No. MB6335),” December 17, 2002.

41. EMF-3028P-A Volume 2 Revision 4, RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain Volume 2: Theory Manual, AREVA NP, March 2013.

ATTACHMENT 7

Applicability of AREVA Methodology
(Proprietary)

ATTACHMENT 8

Mechanical Design Report
(Proprietary)

ATTACHMENT 9

Thermal-Hydraulic Design Report
(Proprietary)

ATTACHMENT 10

Fuel Rod Thermal-Mechanical Design Report
(Proprietary)

ATTACHMENT 11

Fuel Cycle Design Report
(Proprietary)

ATTACHMENT 12

Reload Safety Analysis Report
(Proprietary)

ATTACHMENT 13

LOCA Break Spectrum Analysis Report
(Proprietary)

ATTACHMENT 14

LOCA-ECCS Analysis MAPLHGR Limit Report
(Proprietary)

ATTACHMENT 15

Nuclear Fuel Design Report
(Proprietary)

ATTACHMENT 16

AREVA Affidavits

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA information contained in the report ANP-3338P, Revision 0, "Applicability of AREVA BWR Methods to the Dresden and Quad Cities Reactors Operating at Extended Power Uprate," dated November 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 6th

day of November, 2014.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/14/2016

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-14-16
STATE OF WASHINGTON

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA information contained in the report ANP-3305P, Revision 0, "Mechanical Design Report for Quad Cities and Dresden ATRIUM 10XM Fuel Assemblies Licensing Report," dated June 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 29th day of May, 2014.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/14/2016

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-14-16
STATE OF WASHINGTON

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA information contained in the report ANP-3287P, Revision 1, "Quad Cities Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM 10XM Fuel Assemblies," dated November 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 21st

day of November, 2014.

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-14-16
STATE OF WASHINGTON

Mary Anne Heilman
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 6/6/2016

A F F I D A V I T

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA information contained in the report ANP-3324P, Revision 0, "ATRIUM 10XM Fuel Rod Thermal-Mechanical Design for Quad Cities Unit 2 Cycle 24 Representative fuel Cycle Design," dated October 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 20th day of October, 2014.

Mary Anne Heilman
Commission Expires
Notary
Public
06-06-16
State of Washington

Mary Anne Heilman
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 6/6/2016

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA information contained in the report ANP-3293P, Revision 0, "Quad Cities Unit 2 Cycle 24 Representative Fuel Cycle Design," dated October 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 20th day of October, 2014.

Mary Anne Heilman
State of Washington
Commission Expires
Notary Public
06-06-16

Mary Anne Heilman
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 6/6/2016

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA information contained in the report ANP-3361P, Revision 0, "Quad Cities Unit 2 Cycle 24 Representative Cycle Design Reload Safety Analysis," dated December 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 16th day of December, 2014.

Mary Anne Heilman
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 6/6/2016

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-14-16
STATE OF WASHINGTON

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA information contained in the report ANP-3328P, Revision 0, "Quad Cities Units 1 and 2 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel," dated December 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 5th

day of December, 2014.

Mary Anne Heilman
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 6/6/2016

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-14-16
STATE OF WASHINGTON

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA information contained in the report ANP-3356P, Revision 0, "Quad Cities Units 1 and 2 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel," dated December 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 5th day of December, 2014.

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-14-16
STATE OF WASHINGTON

Mary Anne Heilman
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 6/6/2016

A F F I D A V I T

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA information contained in the report ANP-3294P, Revision 0, "Nuclear Fuel Design Report Quad Cities Unit 2 Cycle 24 Representative Cycle ATRIUM 10XM Fuel," dated July 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 1st

day of July, 2014.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/14/2016

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-14-16
STATE OF WASHINGTON

ATTACHMENT 17

Applicability of AREVA Methodology
(Non-Proprietary)

ANP-3338NP
Revision 0

Applicability of AREVA BWR Methods to the Dresden and Quad Cities Reactors Operating at Extended Power Uprate

November 2014

AREVA Inc.

ANP-3338NP
Revision 0

Applicability of AREVA BWR Methods to the Dresden and Quad Cities Reactors Operating at Extended Power Uprate

Nature of Changes

Item	Page	Description and Justification
1.	All	This is a new document.

Contents

Tables

Figures

Nomenclature

Acronym	**Definition**
ACE	AREVA's advanced critical power correlation **[ ]**
BWR	Boiling Water Reactor
CHF	Critical Heat Flux
CPR	Critical Power Ratio
DIVOM	Delta-over-Initial CPR Versus Oscillation Magnitude
EPU	Extended Power Uprate
KATHY	KArlstein Thermal HYdraulic test facility
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
LRNB	Load Reject with no Bypass
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
NRC	Nuclear Regulatory Commission
OLMCPR	Operating Limit Minimum Critical Power Ratio
SLMCPR	Safety Limit Minimum Critical Power Ratio
SPCB	AREVA (formerly Siemens Power Corporation) critical power correlation
WREM	Water Reactor Evaluation Model

1 Introduction

This document reviews the AREVA approved licensing methodologies to demonstrate that they are applicable to licensing and operation of the Dresden and Quad Cities Nuclear Power Generating Stations with the introduction of ATRIUM™* 10XM fuel. This confirmation of the applicability of AREVA methods to these plants includes the current operating domain as defined by the power/flow operating map in Figure 1-1.

All four BWR/3s (Dresden Units 2 and 3 and Quad Cities Units 1 and 2) are essentially the same since the core operational conditions (number of assemblies, rated thermal power, rated core flow), modeled geometry[†], safety system performance and ECCS parameters are either identical or have minor differences. The most significant difference between the units is the core loadings and corresponding core designs. The impact of the differences in core designs between units and cycles is addressed in the cycle specific reload report for each unit. Minor differences between the plants and units do not impact the application of AREVA's methodology as presented in this document.

---

* ATRIUM is a trademark of AREVA.

† Dresden Units 2 and 3 have an Isolation Condenser, but this is not credited in any licensing calculations.

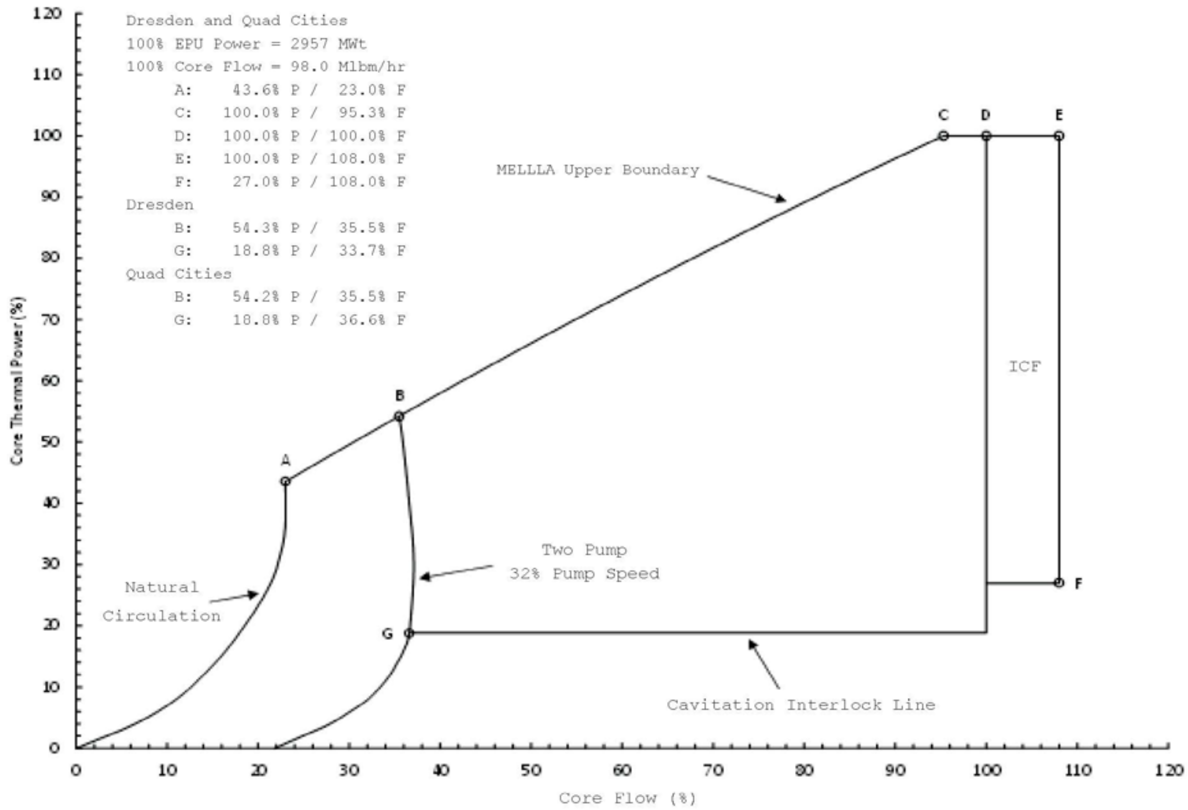


Figure 1-1 Dresden and Quad Cities Power Flow Operating Map

2 Overview

The first step in determining the applicability of current licensing methods to Dresden and Quad Cities operating conditions was a review of AREVA BWR topical reports listed in Table 2-1 to identify SER restrictions. This review identified that there are no SER restrictions on core power level or core flow for the AREVA topical reports. The review also indicated that there are no SER restrictions on the parameters most impacted by operation at EPU power level at any core flow rate: steam flow, feedwater flow, jet pump M-ratio, and core average void fraction.

The second step consisted of an evaluation of the core and reactor conditions experienced under Dresden and Quad Cities operating conditions to determine any challenges to the validity of the models. Operating margin for variations in the reactor power within the constraints of the power/flow map is mitigated to a large extent by variations in the limiting assembly radial power factor. A decrease in the limiting assembly radial power factor is necessary since the thermal operating limits (MCPR, MAPLHGR and LHGR) that restrict assembly power are dependent on the limiting assembly power but are fairly insensitive to the core thermal power.

Based on these fundamental characteristics each of the major analysis domains (thermal-hydraulics, mechanics, core neutronics, transient analysis, LOCA and stability) are assessed to determine any challenges to application. A description of the application of AREVA methodology to a mixed core is provided in Appendix A.

Table 2-1 AREVA Licensing Topical Reports

Document Number	Document Title
XN-NF-79-56(P)(A) Revision 1 and Supplement 1	"Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," Exxon Nuclear Company, November 1981
XN-75-32(P)(A) Supplements 1 through 4	"Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, October 1983. (Base document not approved.)
XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2	"RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984
XN-NF-81-51(P)(A)	"LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, May 1986
XN-NF-85-67(P)(A) Revision 1	“Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel,” Exxon Nuclear Company, September 1986
XN-NF-85-74(P)(A)	"RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model" Exxon Nuclear Company, August 1986
XN-NF-85-92(P)(A)	"Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986
ANF-89-98(P)(A) Revision 1 and Supplement 1	"Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995
ANF-90-82(P)(A) Revision 1	"Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995
EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A)	"RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998
EMF-93-177(P)(A) Revision 1	"Mechanical Design for BWR Fuel Channels," Framatome ANP, August 2005
BAW-10247PA Revision 0	"Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008
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, March 1983

Table 2-1 AREVA Licensing Topical Reports (Continued)

Document Number	Document Title
XN-NF-80-19(P)(A) Volume 4 Revision 1	"Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986
EMF-CC-074(P)(A) Volume 1	"STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2 "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report," Siemens Power Corporation, July 1994
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, October 1999
EMF-CC-074(P)(A) Volume 4, Revision 0	"BWR Stability Analysis Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000
BAW-10255PA Revision 2	"Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," AREVA NP, May 2008
EMF-3028P-A Volume 2 Revision 4	“RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain Volume 2: Theory Manual,” AREVA NP, May, 2013
XN-NF-79-59(P)(A)	"Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, November 1983
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, January 1987
EMF-2245(P)(A) Revision 0	"Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000
EMF-2209(P)(A) Revision 3	"SPCB Critical Power Correlation," AREVA NP, September 2009
ANP-10298PA Revision 1	"ACE/ATRIUM 10XM Critical Power Correlation," AREVA, March 2014
ANP-10307PA Revision 0	"AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011

Table 2-1 AREVA Licensing Topical Reports (Continued)

Document Number	Document Title
XN-CC-33(A) Revision 1	"HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual," Exxon Nuclear Company, November 1975
XN-NF-80-19(P)(A) Volumes 2, 2A, 2B and 2C	"Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, September 1982
XN-NF-82-07(P)(A) Revision 1	"Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, November 1982
XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2	"XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987
ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4	"COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990
ANF-91-048(P)(A)	"Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," Advanced Nuclear Fuels Corporation, January 1993
ANF-91-048(P)(A) Supplements 1 and 2	"BWR Jet Pump Model Revision for RELAX," Siemens Power Corporation, October 1997
EMF-2292(P)(A) Revision 0	"ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000
EMF-2361(P)(A) Revision 0	"EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001
ANF-1358(P)(A) Revision 3	“The Loss of Feedwater Heating Transient in Boiling Water Reactors,” Framatome ANP, September 2005

3 Thermal Hydraulic Analysis

AREVA assembly thermal-hydraulic methods are qualified and validated against full-scale heated bundle tests in the KATHY test facility in Karlstein, Germany. The KATHY tests are used to characterize the assembly two-phase pressure drop and CHF performance. This allows the hydraulic models to be verified for AREVA fuel designs over a wide range of hydraulic conditions prototypic of reactor conditions.

The standard matrix of test conditions for KATHY is compared to reactor conditions in Figure 3-1. This figure illustrates that the test conditions bound typical assembly conditions as well as anticipated operation for Dresden and Quad Cities. The data is based upon the projected operating conditions for the Dresden and Quad Cities reactors. Figure 3-1 also shows that the key physical phenomena (e.g. fluid quality and assembly flows) for Dresden and Quad Cities operating conditions are consistent with current reactor experience.

This similarity of assembly conditions is further enforced in AREVA analysis methodologies by the imposition of SPCB and ACE critical power correlation limits and, therefore, core designs must remain within the same parameter space. Since the bundle operating conditions for Dresden and Quad Cities are within the envelope of hydraulic test data used for model qualification and operating experience, the hydraulic models used to compute the core flow distribution and local void content remain valid for Dresden and Quad Cities operating conditions.

A more detailed discussion of the AREVA void quality correlations is presented in Appendix B.

Figure 3-1 Comparison of KATHY Two-Phase Pressure Drop and Void Fraction Test Matrices and Typical Dresden and Quad Cities Reactor Conditions

4 AREVA CHF/CPR Correlations

All AREVA CHF and CPR correlations are approved by the NRC staff to be applicable over specified ranges of assembly operating conditions. The NRC staff also approved specific corrective actions when the computed conditions fall outside of the approved range to assure that conservative calculations are obtained. For Dresden and Quad Cities operating conditions, some analyses can predict assembly conditions to be outside the approved range of specified conditions for the CHF correlations. Consequently, the AREVA licensing methods are programmed to determine whether the computed assembly conditions fall outside of the approved range of applicability for the CHF correlation and impose approved corrective actions as appropriate to conservatively assess the critical power margin for the assembly. The CPR correlation used for the ATRIUM 10XM fuel is the ACE/ATRIUM 10XM critical power correlation and the corrective actions for when the computed conditions fall outside the approved range are provided in Reference 2.

The application of AREVA critical power correlations to co-resident fuel is governed by Reference 7. **[**

]

Figure 4-1 [ ]

The overall statistics for the SPCB application to OPTIMA2 fuel is given in Table 4-1. [

]

Figure 4-2 shows the SPCB predicted critical power versus the Westinghouse critical power. The data shows a reasonable prediction of critical power without obvious trends.

Table 4-1 SPCB Application to OPTIMA2 Statistics

[

]

Figure 4-2 SPCB Critical Power Versus Westinghouse Critical Power for OPTIMA2

5 Safety Limit MCPR

The safety limit MCPR (SLMCPR) methodology is used to determine the Technical Specification SLMCPR value that ensures that 99.9% of the fuel rods are expected to avoid boiling transition during normal reactor operation and anticipated operation occurrences. The SLMCPR methodology for Dresden and Quad Cities is described in Reference 1. The SLMCPR is determined by statistically combining calculation uncertainties and plant measurement uncertainties that are associated with the calculation of MCPR. The thermal hydraulic, neutronic, and critical power correlation methodologies are used in the calculation of MCPR. The applicability of these methodologies for Dresden and Quad Cities operating conditions is discussed in other sections of this report.

AREVA calculates the SLMCPR on a cycle-specific basis to protect all allowed reactor operating conditions. The analysis incorporates the cycle-specific fuel and core designs. The initial MCPR distribution of the core is a major factor affecting how many rods are predicted to be in boiling transition. The MCPR distribution of the core depends on the neutronic design of the reload fuel and the fuel assembly power distributions in the core. AREVA SLMCPR methodology specifies that analyses be performed with a design basis power distribution that "… conservatively represents expected reactor operating states which could both exist at the MCPR operating limit and produce a MCPR equal to the MCPR safety limit during an anticipated operational occurrence." (Reference 1, Section 3.3.2).

[

]

[

]

The impact that a flatter core power distribution may have on the SLMCPR is explicitly accounted for by the methodology. EPU operation will lead to a flatter core power distribution;
[

]

[

]

Figure 5-1 Assembly Power Distribution for Limiting Case in Safety Limit MCPR Analysis

6 Mechanical Limits Methodology

The LHGR limit is established to support plant operation while satisfying the fuel mechanical design criteria. The methodology for performing the fuel rod evaluation is described in Reference 3. Fuel rod design criteria evaluated by the methodology are contained in References 3 and 4.

Fuel rod power histories are generated as part of the methodology for equilibrium cycle conditions as well as cycle-specific operation. These power histories include the impact of channel bow using the same model and limitations as previously described in Section 5 (Safety Limit MCPR). A comprehensive number of uncertainties are taken into account in the categories of operating power uncertainties, code model parameter uncertainties, and fuel manufacturing tolerances. In addition, adjustments are made to the power history inputs for possible differences in planned versus actual operation. Upper limits on the analysis results are obtained for comparison to the design limits for fuel melt, cladding strain, rod internal pressure and other topics as described by the design criteria.

Since the power history inputs, which include LHGR, fast neutron flux, reactor coolant pressure and reactor coolant temperature, are used as input to the analysis, the results explicitly account for conditions at EPU such as higher coolant voiding and offsets in axial power and neutron fast flux. The resulting LHGR limit is used to monitor the fuel so it is maintained within the same maximum allowable steady-state power envelope as analyzed.

7 Core Neutronics

The AREVA neutronic methodologies (Reference 19) are characterized by technically rigorous treatment of phenomena and are very well benchmarked (>100 cycles of operation plus gamma scan data for ATRIUM-10). Recent operating experience is tabulated in Table 7-1. These tables present the reactor operating conditions and in particular the average and hot assembly powers for both US and European applications. As can be seen from this information, the average and peak bundle powers in this experience base exceed that associated with the Dresden and Quad Cities application.

For Dresden and Quad Cities operation the high powered assemblies in uprated cores will be subject to the same LHGR, MAPLHGR, MCPR, and cold shutdown margin limits and restrictions as high powered assemblies in all other cores.

Detailed analysis of the neutronic methodology is presented in Appendix C. Specific applicability to Dresden and Quad Cities is addressed below.

7.1 *Shutdown Margin*

In order to accurately determine shutdown margins during transition cycles, AREVA typically performs detailed benchmarking analyses of the three to five cycles previous to insertion of AREVA fuel in that reactor. This benchmarking is performed with the CASMO-4/MICROBURN-B2 3-D core simulator code system. Hot depletions are performed using actually operated state conditions including as-loaded core configurations, as-operated control rod patterns, and operating power, pressure, flow, and inlet subcooling. To confirm the validity of the hot depletions, comparison of eigenvalue trends and predicted versus measured TIP distributions for the benchmark cycles are performed. These results are used to establish the hot-operating target k-eff for design of the first transition cycle. All cold critical measurements taken during the benchmarking cycles are also modeled in MICROBURN-B2 by restarting from the hot cycle depletions discussed above. The results from the cold critical benchmarks are used to define a cold critical k-eff target. A typical design target is 1% Δk/k which ensures that the transition loading fuel design will support the 0.38% Δk/k technical specification cold shutdown margin requirement with additional margin to cover the uncertainty in the design target chosen based on the benchmarking results. Past AREVA experience indicates that the variation in the target cold critical k-eff when transitioning to AREVA fuel is small.

During the design of each transition cycle, shutdown margin is computed by performing restart solutions based on a shuffled core from a short window previous cycle condition. This means that the previous cycle is assumed to shutdown earlier than the nominal planned shutdown for the cycle. The short window shutdown of the previous cycle results in additional carryover reactivity for the shutdown margin analysis of the cycle being designed. Setting the gadolinia design of the fresh fuel and the loading plan to meet the design shutdown margin based on the assumed short window shutdown of the previous cycle assures that adequate shutdown margin exists for the entire cycle at the design stage. At startup, when each designed cycle reaches cold critical conditions, comparison to the predicted point of criticality to the actual point of criticality is made. High accuracy of the predicted versus actual critical eigenvalue demonstrates the validity of the shutdown margin design for that cycle.

The initial critical and any subsequent cold critical data points achieved in each transition and follow-on cycle are fed back into the cold critical eigenvalue database for the reactor unit, and the target is revised as needed for the design of the subsequent cycle. This method assures continued accuracy in predicting the cold shutdown margin as new fuel is transitioned into the reactor core during the first and second transition cycles and all subsequent cycles.

As part of the design process for developing the fuel/core design for Dresden and Quad Cities it is necessary to establish a target cold critical eigenvalue. Benchmarking of the previous Dresden and Quad Cities cycles that contained Westinghouse OPTIMA2* fuel resulted in the cold critical data presented in Figure 7-1. The target cold critical eigenvalues used for the reference cycle design focused on a subset of this data. Specifically, only the Quad Cities benchmark data was used since the reference cycle is based upon Quad Cities Unit 2 Cycle 24. Furthermore, the eigenvalue selection concentrated more heavily on the later operating cycles which is typical of the selection process. The resulting target cold critical eigenvalues used for the Quad Cities reference design is shown in Table 7-2 and Figure 7-2. This determination of the target, together with a conservatively chosen design goal, ensures conservative

* Early cycles of the benchmark analysis include mixed cores containing AREVA 9X9 and GNF GE-14 fuel.

determination of shutdown margin for the design. A similar approach will be used for cold target determination for the initial Dresden transition cycles.

7.2 *LHGR Monitoring of Advanced Fuel Designs*

Through various interactions between AREVA and the NRC, the NRC has requested verification that certain detailed models available with MICROBURN-B2 are utilized in the modeling of advanced fuel designs. These models include the impact of LPRM detectors (instrument tube) on the surrounding fuel rods and the impact of modeling the plenum region above the end of the heated portion of the part-length rods. The explicit LPRM model is used in the core monitoring to account for perturbations to the local peaking factors of rods surrounding the LPRM, hence rod power biases due to the presence of LPRM detectors are accounted for in the monitoring of LHGR limits. Monitoring for conformance with the operating limit LHGR will include explicit modeling of the fission gas plena in the node directly above the top of PLFR active fuel length. This provides the confirmation that the NRC has requested.

7.3 *Bypass Boiling*

The level of bypass boiling for a given state-point is a direct result of the hydraulic solution. The potential for boiling increases as the power/flow ratio increases or the inlet subcooling decreases. While the licensing methodology utilizes a **[**

] to estimate the potential for localized bypass boiling. This **[**

] to specifically determine a bounding local void distribution in the bypass. The model is conservative in that it **[**

]. Review of the edit of bypass channel exit void for a Dresden and Quad Cities equilibrium cycle case identified a few assemblies with minimal (< 0.005 void fraction) bypass channel exit void at a cycle exposure of 13,500 MWd/MTU. To force more boiling in the bypass the inlet subcooling was set to a value of 20.1 BTU/lbm (compared to the typical value of 25.463 for this statepoint) at the 13,500 MWd/MTU statepoint to demonstrate the capability of this model to predict localized bypass boiling. The results are presented in Figure 7-3 through Figure 7-5.

Figure 7-3 presents the average void fraction for the channel bypass and Figure 7-4 presents the core exit bypass channel void fraction. One of the more significant impacts of voiding in the bypass is the impact on the LPRM reading. The average void fraction of the four channels surrounding any LPRM location is presented in Figure 7-5. Since no boiling is observed at any LPRM location for normal operating conditions, there is no impact on LPRM readings.

7.4 *Normal Operation*

From a neutronic perspective, moderator density (void fraction) and exposure cause the greatest variation in cross sections. Reactor conditions for Dresden and Quad Cities are not significantly different from that of current experience and are bounded by the experience for the important parameters. Dresden and Quad Cities operating conditions (Figure 7-6, Figure 7-7, and Figure 7-8) can be compared to the equivalent data of the topical report EMF-2158(P)(A). Comparison of Figure 7-6 vs. Figure C-29 and Figure 7-7 vs. Figure C-30 shows that Dresden and Quad Cities operation is within the range of the original methodology approval for assembly power and exit void fraction.

The axial profile of the power and void fraction of the limiting assembly and core average values are presented in Figure 7-8 for a Dresden and Quad Cities design. These profiles demonstrate that the core average void fraction and the maximum assembly power void fractions are bounded by the topical report data and are consistent with recent experience on other reactors.

Figure 7-9 presents a histogram of the void fraction for Dresden and Quad Cities conditions. This histogram was taken at the point of maximum exit void fraction expected during the cycle. The distribution of voids is shifted slightly toward the 70 -80 % void fraction levels. The population of nodes experiencing 85 -90% voids is still small.

The neutronic and thermal hydraulic conditions predicted for the Dresden and Quad Cities operation are bounded by the data provided in the topical report EMF-2158(P)(A). Concerns about Pu production with high voids are not relevant as the isotopic validation presented in the topical report continues to be applicable to Dresden and Quad Cities operation.

Table 7-1 CASMO-4/MICROBURN-B2 Operating Experience

Reactor	Reactor Size, #FA	Power, MWt (% Uprated)*	Ave. Bundle Power, MWt/FA	Approximate Peak Bundle Power, MWt/FA
A	592	2575 (0.0)	4.4	7.2
B	592	2575 (0.0)	4.4	7.4
C	532	2292 (0.0)	4.3	7.3
D	840	3690 (0.0)	4.4	7.5
E	500	2500 (15.7)	5.0	8.0
F	444	1800 (5.9)	4.1	7.3
G	676	2928 (8.0)	4.3	7.6
H	700	3300 (9.3)	4.7	8.0
I	784	3840 (0.0)	4.9	8.1
J	624	3237 (11.9)	5.2	7.8
K	648	3600 (14.7)	5.6	8.6
L	648	2500 (10.1)	3.9	6.9
M	624	3091 (6.7)	5.0	7.7
N	800	3898 (1.7)	4.9	7.7
O	764	3489 (5.0)	4.6	7.2
P	560	2923 (20.0)	5.2	8.0
Q	764	3952 (20.0)	5.2	7.7
R‡	724	2957 (17.8)	4.1	6.6

* Latest power uprates.
‡ Dresden and Quad Cities

Table 7-2 Dresden and Quad Cities Target Cold Critical Eigenvalue

Cycle Exposure (MWd/MTU)	k-eff
0.0	0.9970
6,000.0	0.9950
20,000.0	0.9950

Figure 7-1 Dresden and Quad Cities Cold Critical Data

Figure 7-2 Quad Cities Cold Critical Data Used For Cold Target Determination

Figure 7-3 MICROBURN-B2 Multi-Channel Average Bypass Void Distribution from a Dresden and Quad Cities Equilibrium Cycle Design

[ ]

Figure 7-4 MICROBURN-B2 Multi-Channel Exit Bypass Void Distribution from a Dresden and Quad Cities Equilibrium Cycle Design

[ ]

[ ]

Figure 7-5 MICROBURN-B2 Multi-Channel Bypass Void at an LPRM Location from a Dresden and Quad Cities Equilibrium Cycle Design

Figure 7-6 Maximum Assembly Power in a Dresden and Quad Cities Design

Figure 7-7 Maximum Exit Void Fraction in a Dresden and Quad Cities Design

Figure 7-8 Dresden and Quad Cities Design Axial Profile of Power and Void Fraction

Figure 7-9 Dresden and Quad Cities Design Nodal Void Fraction Histogram

8 Transient Analysis

The core phenomena of primary interest for limiting transients in BWRs are void fraction/quality relationships, determination of CHF, pressure drop, reactivity feedbacks and heat transfer correlations. One fundamental validation of the core hydraulic solution is separate effects testing against KATHY transient CHF measurements. The transient benchmark to time of dryout for prototypic Load Reject with no Bypass (LRNB) and pump trip transients encompass the transient integration of the continuity equations (including the void-quality closure relation), heat transfer, and determination of CHF. Typical benchmarks to KATHY (Figure 8-1) illustrate that the transient hydraulic solution and application of ACE (AREVA Critical Power Correlation) result in conservative predictions of the time of dryout. The measured data is taken from ATRIUM 10XM tests.

Outside of the core, the system simulation relies primarily on solutions of the basic conservation equations and equations of state. The models associated with predicting the pressure wave are general and have no limitation within the range of variation associated with Dresden and Quad Cities EPU operation.

The reactivity feedbacks are validated by a variety of means including initial qualification of advanced fuel design lattice calculations to Monte Carlo results as required by SER restrictions, steady-state monitoring of reactor operation (power distributions and eigenvalue), and the Peach Bottom 2 turbine trip benchmarks that exhibited a minimum of 2% conservatism in the calculation of integral power.

From these qualifications and the observation that the nodal hydraulic conditions during EPU operation are expected to be within the current operating experience, the transient analysis methods remain valid.

Appendix D provides additional information on the transient cross section treatment in the COTRANSA2 transient simulator for both EPU and pre-EPU reactor conditions.

Appendix F provides a summary of the impact of thermal conductivity degradation on transient analysis and corrective actions taken in the Dresden and Quad Cities analyses.

[ ]

Figure 8-1 Typical Hydraulic Benchmarks to KATHY Transient Simulations (time to dryout)

9 LOCA Analysis

LOCA results are strongly dependent on local power and are weakly dependent on core average power. As discussed in previous sections, maximum local power is not significantly changed due to EPU because the core is still constrained by the same thermal limits. The parameters associated with EPU that may impact LOCA results at each of the core flow rates in the operating domain are: increased core average initial stored energy, decreased initial coolant inventory, relative flow distribution between highest power and average power assemblies, and increased core decay heat.

BWR LOCA analyses are not sensitive to initial stored energy. During the blowdown phase the heat transfer remains high and the stored energy is removed prior to the start of the heatup phase. Initial inventory differences may impact LOCA event timing and the minimum inventory during blowdown prior to refill of the reactor vessel. However, any impact on event timing or minimum inventory would be smaller than the impact associated with the different size breaks that are already considered in the break spectrum analyses. At the elevated powers associated with EPU conditions, the difference in flow between the highest power assembly and the average power assembly is reduced. Therefore, these parameters do not change the range of conditions encountered or the capability of the codes to model LOCA at EPU conditions.

The potential impact of the EPU on LOCA analyses is thus primarily associated with the increase in decay heat levels in the core. For the EXEM BWR-2000 LOCA methodology the decay heat is conservatively modeled. The 11 decay equation curve fit to the 1971 draft ANS standard for fission product decay heat from the WREM model is used to calculate fission product decay heat during blowdown. The draft ANS standard values are used for spray cooling and reflood. The required multiplier of 1.2 is applied to the fission product decay heat throughout the LOCA scenario. The models used for decay heat calculations are valid for EPU.

From the above discussion and the observation that nodal thermal-hydraulic conditions during EPU are expected to be within the current operating experience, the LOCA methods remain applicable for EPU conditions.

Independent of EPU, additional modifications have been made to the approved EXEM BWR-2000 LOCA methodology to more accurately model advanced fuel designs and to address regulatory concerns with the approved methodology. These modifications are described in Appendix E. Appendix F summarizes the assessment of thermal conductivity degradation in the Dresden and Quad Cities LOCA analyses.

10 Stability Analysis

10.1 *Linear Stability*

The flatter radial power profile characteristic of EPU core designs will tend to decrease the first azimuthal eigenvalue separation and result in slightly higher regional decay ratios. These effects are computed by STAIF as it directly computes the channel, global, and regional decay ratio and does not rely on a correlation to protect the regional mode.

STAIF has been benchmarked against full assembly tests (in KATHY facility) to validate the channel hydraulics from a decay ratio of approximately 0.4 to limit cycles. These tests or benchmarks exceed the bounds of allowed operation. These benchmarks include prototypical ATRIUM-10 assemblies. From a reactor perspective, STAIF is benchmarked to both global and regional reactor data, and includes current reactor cycle and fuel design elements. This strong benchmarking qualification and the direct computation of the regional mode assure that the impact of the EPU core designs are reflected in the stability analysis.

10.2 *DIVOM*

RAMONA5-FA has been generically approved to calculate DIVOM for EPU operation (Reference 5 and 6).

11 ATWS

11.1 *ATWS General*

The COTRANSA2 computer code is the primary code used for the ATWS overpressurization analysis. The ATWS overpressurization event is not used to establish operating limits for critical power; therefore, the critical power correlation(s) pressure limit is not a factor in the analysis.

Dryout conditions are not expected to occur for the core average channel that is modeled in COTRANSA2 for the ATWS overpressurization analysis. Dryout might occur in the limiting (high power) channels of the core during the ATWS event; however, these channels are not modeled in COTRANSA2 analyses. For the ATWS overpressurization analysis, ignoring dryout for the hot channels is conservative in that it maximizes the heat transferred to the coolant and results in a higher calculated pressure.

The ATWS event is not limiting relative to acceptance criteria identified in 10 CFR 50.46. The core remains covered and adequately cooled during the event. Following the initial power increase during the pressurization phase, the core returns to natural circulation conditions after the recirculation pumps trip and fuel cladding temperatures are maintained at acceptably low levels. The ATWS event is significantly less limiting than the loss of coolant accident relative to 10 CFR 50.46 acceptance criteria.

11.2 *Void Quality Correlation Bias*

AREVA performs cycle-specific ATWS analyses of the short-term reactor vessel peak pressure using the COTRANSA2 computer code. The ATWS peak pressure calculation is a core-wide pressurization event that is sensitive to similar phenomenon as other pressurization transients. Bundle design is included in the development of input for the coupled neutronic and thermal-hydraulic COTRANSA2 core model. Important inputs to the COTRANSA2 system model are biased in a conservative direction.

The AREVA analysis methods and the correlations used by the methods are applicable for both pre-EPU and EPU conditions. The transient analysis methodology is a deterministic, bounding approach that contains sufficient conservatism to offset biases and uncertainties in individual phenomena. The void-quality correlation is robust as discussed in Appendix B for past and present fuel designs. For future fuel designs the void-quality correlation would be reviewed for applicability, which may involve additional verification and validation.

A s nsitivity study was performed for the limiting ATWS pressurization event for a proposed BWR/4 cycle with EPU to assess the bias between the ATRIUM-10 test data and the void-quality correlation. The event was a pressure regulator failure-open (PRFO), which is a depressurization event, followed by pressurization due to main steam line isolation valve (MSIV) closure. The neutronics input included the impact of the fuel depleted with the changes in the void-quality correlation. To remove the bias in the MICROBURN-B2 neutronics input, the **[ ]** void-quality correlation was modified. To address the bias in the Ohkawa-Lahey void-quality correlation for the COTRANSA2 code, the void-quality relationship was changed to a **[ ]**. Additionally, the sensitivity study was repeated without depleting the fuel with the changes in the void-quality correlation (the change in the void-quality correlation was instantaneous at the exposure of interest).

The reference ATWS case had a peak vessel pressure of 1477 psig. The change in the void-quality correlations resulted in a **[ ]** increase in the peak vessel pressure. The results for an instantaneous change in the void-quality correlation showed the same impact.

A study was also performed for the ASME overpressure event for the same BWR/4 cycle with EPU. The event was the MSIV closure. The change in the void-quality correlations resulted in a **[ ]** increase in the peak vessel pressure.

The impact of a change in the bias of the void-quality correlations on peak pressure is expected to be more than offset by the model conservatisms. Until quantitative values of the conservatisms can be demonstrated, AREVA has imposed that a **[ ]** increase to the peak vessel pressure for the ATWS overpressure analysis and a **[ ]** increase to the peak vessel pressure for the ASME overpressure analysis be included in analyses results.

11.3 *ATWS Containment Heatup / Long-Term Evaluation*

Fuel design differences may impact the power and pressure excursion experienced during the ATWS event. This in turn may impact the amount of steam discharged to the suppression pool and containment. **[**

]

[

]

Table 11-1 [ ]

[ ]

12 Summary

This review concluded that there are no SER restrictions on AREVA methodology that impact the transition to AREVA fuel at Dresden and Quad Cities. Since the core and assembly conditions for the Dresden and Quad Cities units are bounded by core and assembly conditions of other plants for which the methodology was benchmarked, the AREVA methodology (including uncertainties) remains applicable for conditions at the Dresden and Quad Cities Units.

More specifically:

a) The steady state and transient neutronics and thermal-hydraulic analytical methods and code systems supporting Dresden and Quad Cities are within NRC approved applicability ranges because the conditions for Dresden and Quad Cities application are bounded by existing core and assembly conditions in other plants for which the AREVA methodology was benchmarked.

b) The calculational and measurement uncertainties applied in Dresden and Quad Cities applications are valid because the conditions for Dresden and Quad Cities application are bounded by existing core and assembly conditions for which the AREVA methodology was benchmarked.

c) The assessment database and uncertainty of models used to simulate the plant response at Dresden and Quad Cities conditions are bounded by core and assembly conditions for which the AREVA methodology was benchmarked.

Sections 4, 5, 7, 9, and 11 summarize methodology or application enhancements specifically for Dresden and Quad Cities:

a) SPCB CPR correlation applied to OPTIMA2

b) **[ ]**

c) **[ ]**

d) LOCA radiation view factors and **[ ]**

e) Thermal conductivity degradation

f) Void quality correlation biases

13 References

1. ANP-10307PA Revision 0, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors*, AREVA NP, June 2011.

2. ANP-10298PA Revision 1, *ACE/ATRIUM 10XM Critical Power Correlation*, AREVA, March 2014.

3. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, AREVA NP, February 2008.

4. ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995.

5. EMF-3028P-A Volume 2 Revision 4, *RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain Volume 2: Theory Manual*, AREVA NP, March, 2013.

6. BAW-10255PA Revision 2, *Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code*, AREVA NP, May 2008.

7. EMF-2245(P)(A) Revision 0, *Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel*, Siemens Power Corporation, August 2000.

8. ANF-913 (P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses*, Advanced Nuclear Fuels Corporation, August 1990.

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

10. P. Coddington and R. Macian, "A Study of the Performance of Void Fraction Correlations Used In the Context of Drift-Flux Two-Phase Flow Models," Nuclear Engineering and Design, 215, 199-216, June 2002.

11. **[ ]**

12. K. Ohkawa and R. T. Lahey, Jr., "The Analysis of CCFL Using Drift-Flux Models," Nuclear Engineering and Design, 61, 1980.

13. S. Misu et al., "The Comprehensive Methodology for Challenging BWR Fuel Assembly and Core Design used at FANP," proceedings on CD-ROM, PHYSOR 2002, Seoul, Korea, October 7-10, 2002.

14. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis*, Exxon Nuclear Company, February 1987.

15. XN-NF-84-105(P)(A) Volume 1 Supplement 4, *XCOBRA-T: A Computer Code For BWR Transient Thermal-Hydraulic Core Analysis, Void Fraction Model Comparison to Experimental Data*, Advanced Nuclear Fuels Corporation, June 1988.

16. O. Nylund et al., “Hydrodynamic and Heat Transfer Measurements on A Full-Scale Simulated 36-Rod Marviken Fuel Element with Non-Uniform Radial Heat Flux Distribution,” FRIGG-3, R-494/RL-1154, November 1969.

17. J. Skaug et al., “FT-36b, Results of void Measurements,” FRIGG-PM-15, May 1968.

18. O. Nylund et al., “Hydrodynamic and Heat Transfer Measurements on A Full-Scale Simulated 36-Rod Marviken Fuel Element with Uniform Heat Flux Distribution,” FRIGG-2, R-447/RTL-1007, May 1968.

19. 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, October 1999.

20. EMF-2361(P)(A) Revision 0, *EXEM BWR-2000 ECCS Evaluation Model*, Framatome ANP, May 2001.

21. XN-CC-33(A) Revision 1, *HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual*, Exxon Nuclear Company, November 1975.

22. Letter, P. Salas (AREVA) to Document Control Desk, U.S. Nuclear Regulatory Commission, “Proprietary Viewgraphs and Meeting Summary for Closed Meeting on Application of the EXEM BWR-2000 ECCS Evaluation Methodology,” NRC:11:096, September 22, 2011.

23. Letter, T.J. McGinty (NRC) to P. Salas (AREVA),” Response to AREVA, Inc. (AREVA) Proposed Analysis Approach for Its EXEM Boiling Water Reactor (BWR)-2000 Emergency Core Cooling System (ECCS) Evaluation Model,” July 5, 2012.

24. Letter, R. Gardner to NRC, “Informational Transmittal Regarding Requested White Paper on the Treatment of Exposure Dependent Fuel Thermal Conductivity Degradation in RODEX Fuel Performance Codes and Methods,” NRC:09:069, ML092010160, July 14, 2009.

25. Letter, T. J. McGinty (NRC) to P. Salas (AREVA), “Nuclear Fuel Thermal Conductivity Degradation Evaluation for Light Water Reactors Using AREVA Codes and Methods (TAC No. ME5178),” March 23, 2012.

26. Letter, P. Salas (AREVA) to NRC, “Response to NRC Letter Regarding Nuclear Fuel Thermal Conductivity Degradation Evaluation for Light Water Reactors Using AREVA Codes and Methods,” NRC:12:023, April 27, 2012.

27. ANP-10300P, Revision 0, *AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios*, (ML100040158) December 2009.

28. 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, January 1987.

29. Correspondence, R.A. Copeland (Siemens) to R.C. Jones (NRC), “ATRIUM-10 Presentations,” RAC:95:080, May 4, 1995 (38-9091703-000).

30. EMF-2209(P)(A) Revision 3, *SPCB Critical Power Correlation*, AREVA NP, September 2009.

31. R.T. Lahey Jr. and F.J. Moody, *The Thermal-Hydraulics of a Boiling Water Nuclear Reactor, Second Edition*, American Nuclear Society, Hinsdale, IL, 1993.

32. Y. Aounallah, “Boiling Suppression in Convective Flow,” Proceedings of ICAPP ’04, Pittsburgh, PA, June 13–17, 2004, Paper 4251.

Appendix A. Application of AREVA Methodology for Mixed Cores

A.1 *Discussion*

AREVA has considerable experience analyzing fuel design transition cycles and has methodology and procedures to analyze mixed cores composed of multiple fuel types. For each core design, analyses are performed to confirm that all design and licensing criteria are satisfied. The analyses performed explicitly include each fuel type in the core. The analyses consider the cycle-specific core loading and use input data appropriate for each fuel type in the core. The mixed core analyses are performed using generically approved methodology **[ ]** in a manner consistent with NRC approval of the methodology. Based on results from the analyses, operating limits are established for each fuel type present in the core. During operation, each fuel type is monitored against the appropriate operating limits.

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 other vendor co-resident fuel to minimize variability due to methods.

For core design and nuclear safety analyses, the neutronic cross-section data is developed for each fuel type in the core using CASMO-4. MICROBURN-B2 is used to design the core and provide input to safety analyses (core neutronic characteristics, power distributions, etc.). 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 AREVA and co-resident 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. The thermal mechanical limits (steady-state and transient) for the co-resident fuel are provided to AREVA by the utility. AREVA performs design and licensing analyses to demonstrate that the core design meets steady-state limits and that transient limits are not exceeded during anticipated operational occurrences.

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. Fuel type specific correlation coefficients for AREVA fuel are based on data from the AREVA critical power test facility. Consistent with Reference 7 **[ ]** the SPCB critical power correlation will be used for monitoring OPTIMA2 fuel present in transition cycles of operation at Dresden and Quad Cities. The critical power ratio (CPR) correlation used for the ATRIUM 10XM fuel is the ACE/ATRIUM 10XM critical power correlation described in Reference 2. The ACE CPR correlation uses K-factor values to account for rod local peaking, rod location and bundle geometry effects.

Analyses performed to determine the safety limit MCPR 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). CPR is evaluated for each assembly using fuel type specific correlation coefficients. Plant and fuel type specific uncertainties are considered in the statistical analysis performed to determine the safety limit MCPR. The safety limit MCPR analysis is performed each cycle and uses the cycle specific core configuration.

An operating limit MCPR is established for each fuel type in the core. For fast transients the COTRANSA2 code (Reference 8) is used to determine the overall system response. The core nuclear characteristics used in COTRANSA2 are obtained from MICROBURN-B2 and reflect the actual core loading pattern. Boundary conditions from COTRANSA2 are used with an XCOBRA-T core model. In the XCOBRA-T model, a hot channel with appropriate geometric and thermal hydraulic characteristics is modeled for each fuel type present 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 operating limit MCPR for each fuel type.

For transient events that are sufficiently slow such that the heat transfer remains in phase with changes in neutron flux during the transient, evaluations are performed with steady state codes such as MICROBURN-B2 in accordance with NRC approval. Such slow transients are modeled by performing a series of steady state solutions with appropriate boundary conditions using the cycle specific design core loading plan. Each fuel assembly type in the core is explicitly modeled. The change in CPR between the initial and final condition after the transient is determined, and if the CPR change is more severe than those determined from fast transient analyses, the slow transient result is used to determine the MCPR operating limit.

Stability analyses to establish OPRM setpoints and backup stability exclusion regions are performed using the cycle-specific core loading pattern. The stability analyses performed with RAMONA5-FA and STAIF explicitly model each fuel type in the core. Each fuel type is modeled using appropriate geometric, thermal hydraulic and nuclear characteristics determined as described above. The stability OPRM setpoints and exclusion region boundaries are established based on the predicted performance of the actual core composition.

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. While system analyses are typically performed on an equilibrium core basis, the thermal hydraulic characteristics of all fuel assemblies in the core are considered to ensure the LOCA analysis results are applicable to mixed core configurations. Results from the hot channel analysis provide boundary conditions to the HUXY computer code. The HUXY model includes fuel type specific input such as dimensions and local power peaking for each fuel rod.

The core monitoring system will monitor each fuel assembly in the core. Each assembly is modeled with geometric, thermal hydraulic, neutronic, and CPR correlation input data appropriate for the specific fuel type. Each assembly in the core will be monitored relative to thermal limits that have been explicitly developed for each fuel type.

In summary, AREVA methodology is used consistent with NRC approval to perform design and licensing analyses for mixed cores. The cycle design and licensing analyses explicitly consider each fuel type in mixed core configurations. Limits are established for each fuel type and operation within these limits is verified by the monitoring system during operation.

Appendix B. Void-Quality Correlations

B.1 *AREVA Void Quality Correlations*

The Zuber-Findlay drift flux model (Reference 9) is utilized in the AREVA nuclear and safety analysis methods for predicting vapor void fraction in the BWR system. The model has a generalized form that may be applied to two phase flow by defining an appropriate correlation for the void concentration parameter, Co, and the drift flux, Vgj. The model parameters account for the radially non-uniform distribution of velocity and density and the local relative velocity between the phases, respectively. This model has received broad acceptance in the nuclear industry and has been successfully applied to a host of different applications, geometries, and fluid conditions through the application of different parameter correlations (Reference 10).

Two different correlations are utilized at AREVA to describe the drift flux parameters for the analysis of a BWR core. The correlations and treatment of uncertainties are as follows:

- The nuclear design, frequency domain stability, nuclear AOO transient and accident analysis methods use the **[ ]** void correlation (Reference 11) to predict nuclear parameters. Uncertainties are addressed at the overall methodology and application level rather than individually for the individual correlations of each method. The overall uncertainties are determined statistically by comparing predictions using the methods against measured operating data for the reactors operating throughout the world.

- The thermal-hydraulic design, system AOO transient and accident analysis, and loss of coolant accident (only at specified junctions) methods use the Ohkawa-Lahey void correlation (Reference 12). This correlation is not used in the direct computation of nuclear parameters in any of the methods. Uncertainties are addressed at the overall methodology level through the use of conservative assumptions and biases to assure uncertainties are bounded.

The **[ ]** void correlation was developed for application to multi-rod geometries operating at typical BWR operating conditions using multi-rod data and was also validated against simple geometry data available in the public domain. The correlation was defined to be functionally dependent on the mass flux, hydraulic diameter, quality, and fluid properties.

The multi-rod database used in the **[**

]. As a result, the multi-rod database and prediction uncertainties are not available to AREVA. However, the correlation has been independently validated by AREVA against public domain multi-rod data and proprietary data collected for prototypical ATRIUM-10 and ATRIUM 10XM test assemblies. Selected results for the ATRIUM-10 test assembly are reported in the public domain in Reference 13.

The Ohkawa-Lahey void correlation was developed for application in BWR transient calculations. In particular, the correlation was carefully designed to predict the onset of counter current flow limit (CCFL) characteristics during the occurrence of a sudden inlet flow blockage. The correlation was defined to be functionally dependent on the mass flux, quality, and fluid properties.

Independent validation of the Ohkawa-Lahey correlation was performed by AREVA at the request of the NRC during the NRC review of the XCOBRA-T code. The NRC staff subsequently reviewed and approved Reference 15, which compared the code to a selected test from the FRIGG experiments (Reference 16). More recently the correlation has been independently validated by AREVA against additional public domain multi-rod data and proprietary data collected for prototypic ATRIUM-10 and ATRIUM 10XM test assemblies, as described below.

The characteristics of the AREVA multi-rod void fraction validation database are listed in Table B-1.

The FRIGG experiments have been included in the validating database because of the broad industry use of these experiments in benchmarking activities, including TRAC, RETRAN, and S-RELAP5. The experiments include a wide range of pressure, subcooling, and quality from which to validate the general applicability of a void correlation. However, the experiments do not contain features found in modern rod bundles such as part length fuel rods and mixing vane grids. The lack of such features makes the data less useful in validating correlations for modern fuel designs. Also the reported instrument uncertainty for these tests is provided in Table B-1 based on mockup testing. However, the total uncertainty of the measurements (including power and flow uncertainties) is larger than the indicated values.

Because of its prototypical geometry, the ATRIUM-10 and ATRIUM 10XM void data collected at KATHY was useful in validating void correlation performance in modern rod bundles that include part length fuel rods, mixing vane grids, and prototypic axial/radial power distributions. Void measurements were made at one of three different elevations in the bundle for each test point: just before the end of the part length fuel rods, midway between the last two spacers, and just before the last spacer.

As shown in Figure 3-1, the range of conditions for the ATRIUM void data are valid for typical reactor conditions. This figure compares the equilibrium quality at the plane of measurement for the ATRIUM 10XM void data with the exit quality of bundles in the EMF-2158 benchmarks and Quad Cities operating conditions. As seen in the figure, the data at the measurement plane covers nearly the entire range of reactor conditions. However, calculations of the exit quality from the void tests show the overall test conditions actually envelope the reactor conditions.

Figure B-1 and Figure B-2 provide comparisons of predicted versus measured void fractions for both the FRIGG data and the AREVA multi-rod void fraction validation database using the **[ ]** correlation. These figures show the predictions fall within ±0.05 (predicted – measured) error bands with good reliability and with very little bias. Also, there is no observable trend of uncertainty as a function of void fraction.

Figure B-3 and Figure B-4 provide comparisons of predicted versus measured void fractions for both the FRIGG data and the AREVA multi-rod void fraction validation database using the Ohkawa-Lahey correlation. In general, the correlation predicts the void data with a scatter of about ±0.05 (predicted – measured), but a bias in the prediction is evident for voids between 0.5 and 0.8. The observed under prediction is consistent with the observations made in Reference 17.

In conclusion, validation using the AREVA multi-rod void fraction validation database has shown that both drift flux correlations remain valid for modern fuel designs. Furthermore, there is no observable trend of uncertainty as a function of void fraction. This shows there is no increased uncertainty in the prediction of nuclear parameters at EPU conditions within the nuclear methods when applied to the Dresden and Quad Cities.

B.2 *Void Quality Correlation Uncertainties*

The AREVA analysis methods and the correlations used by the methods are applicable for modern fuel designs in both pre-EPU and EPU conditions. The approach for addressing the void-quality correlation bias and uncertainties remains unchanged and is applicable for EPU conditions.

The OLMCPR is determined based on the safety limit MCPR (SLMCPR) methodology and the transient analysis (ΔCPR) methodology. Void-quality correlation uncertainty is not a direct input to either of these methodologies; however, the impact of void-correlation uncertainty is inherently incorporated in both methodologies as discussed below.

The SLMCPR methodology explicitly considers important uncertainties in the Monte Carlo calculation performed to determine the number of rods in boiling transition. One of the uncertainties considered in the SLMCPR methodology is the bundle power uncertainty. This uncertainty is determined through comparison of calculated to measured core power distributions. Any miscalculation of void conditions will increase the error between the calculated and measured power distributions and be reflected in the bundle power uncertainty. Therefore, void-quality correlation uncertainty is an inherent component of the bundle power uncertainty used in the SLMCPR methodology.

The transient analysis methodology is not a statistical methodology and uncertainties are not directly input to the analyses. The transient analysis methodology is a deterministic, bounding approach that contains sufficient 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% 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.

Based on the above discussions, the impact of void-quality correlation uncertainty is inherently incorporated in the analytical methods used to determine the OLMCPR. Biasing of important input parameters in licensing calculations provides additional conservatism in establishing the OLMCPR. No additional adjustments to the OLMCPR are required to address void-quality correlation uncertainty.

B.3 *Biasing of the Void-Quality Correlation*

AREVA has performed studies to determine the OLMCPR sensitivity to biases approaching the upper and lower extremes of the data comparisons shown in Figure B-1 through Figure B-4.

For one of these studies, the transient ΔCPR impact was determined by propagating void-quality biases through three main computer codes: MICROBURN-B2, COTRANSA2, and XCOBRA-T.

The **[ ]** correlation in MICROBURN-B2 was modified to correct the mean to match the measured ATRIUM-10 void fraction data shown in Figure B-2. The modified **[ ]** correlation parameters were then modified to generate two bounding correlations for the ATRIUM-10 of ±0.05 void. The results of this modified correlation are presented in Figure B-5.

COTRANSA2 does not have the **[ ]** correlation. For COTRANSA2 the modified **[ ]** correlations in MICROBURN-B2 were approximated in COTRANSA2 with **[ ]**. Figure B-6 shows a comparison of the **[ ]** ratio results compared to the ATRIUM-10 test data. This approach created equivalent void fractions as the **[ ]** correlation modifications.

The thermal hydraulic methodology incorporates the effects of subcooled boiling through use of the Levy model. The Levy model predicts a critical subcooling that defines the onset of boiling. The critical subcooling is used with a profile fit model to determine the total flow quality that accounts for the presence of subcooled boiling. The total flow quality is used with the void-quality correlation to determine the void fraction. This void fraction explicitly includes the effects of subcooled boiling. Application of the Levy model results in a continuous void fraction distribution at the boiling boundary.

Like COTRANSA2, XCOBRA-T does not have the [ ] correlation. Unlike COTRANSA2, XCOBRA-T does not have [ ]. For the other void scenarios, no correction was done in XCOBRA-T. Not modifying the void-quality correlation for the other void scenarios results in a very small difference in ΔCPR.

The ransient response was assessed relative to a limiting uprated BWR plant transient calculation. The impact of the change in the void correlations was also captured in the burn history of the fuel (the results are not for an instantaneous change in the void correlations). The SLMCPR response was also assessed with the new input corresponding to the three different void scenarios. The results are provided in Table B-2.

The major influence that the void-quality models have on scram reactivity worth is through the predicted axial power shape. The void-quality models, used for ATRIUM fuel, result in a very good prediction of the axial power shape.

As seen in the results in Table B-2, modifying the void-quality correlations to correct the mean to match the measured ATRIUM-10 void fraction data results in a very small increase in ΔCPR, a very small decrease in SLMCPR, and a very small increase in OLMCPR for this study; therefore, the impact of the correlation bias is insignificant.

The +0.05 void scenarios show an increase in the OLMCPR; however, as mentioned previously, the transient analysis methodology is a deterministic, bounding approach that contains sufficient conservatism to offset uncertainties in individual phenomena. Conservatism is incorporated in the models to bound results on an integral basis relative to benchmark tests. For licensing calculations, important input parameters are biased in a conservative direction. In addition, the licensing calculations include a 110% multiplier to the calculated integral power to provide additional conservatism to offset uncertainties in the transient analysis methodology (which includes the void-quality correlation). Even with an extreme bias in the void correlation of +0.05, the conservatism introduced by the 110% multiplier is alone sufficient to offset the increase in results presented in Table B-2. For the study, the conservatism of the 110% multiplier was [ ]. These calculations demonstrate that the overall methodology has sufficient conservatism to account for both the bias and the uncertainty in the void-quality correlation.

To provide a more accurate assessment of the impact of a +0.05 void bias, AREVA would need to re-evaluate the Peach Bottom transient benchmarks; it is likely that the +0.05 void scenario would show overconservatism in the benchmarks. Likewise, the pressure drop correlations and core monitoring predictions of power will likely show a bias relative to measured data. Correcting the models to new benchmarks and measured data would further reduce the OLMCPR sensitivity.

B.4 *Void-Quality Correlation Uncertainty Summary*

Integral power is a parameter obtainable from test measurements that is directly related to ΔCPR and provides a means to assess code uncertainty. The COTRANSA transient analysis methodology was a predecessor to the COTRANSA2 methodology. The integral power figure of merit was introduced with the COTRANSA methodology as a way to assess (not account for) code uncertainty impact on ΔCPR. From COTRANSA analyses of the Peach Bottom turbine trip tests, the mean of the predicted to measured integral power was 99.7% with a standard deviation of 8.1%. AREVA (Exxon Nuclear at the time) initially proposed to treat integral power as a statistical parameter. However, following discussions with the NRC, it was agreed to apply a deterministic 110% integral power multiplier (penalty) on COTRANSA calculations for licensing analyses. That increase was sufficient to make the COTRANSA predicted to measured integral power conservative for all of the Peach Bottom turbine trip tests.

COTRANSA2 is not a statistical methodology and uncertainties are not directly input to the analyses. The methodology is a deterministic bounding approach that contains sufficient 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. Justification that the integrated effect of all the conservatisms in COTRANSA2 licensing analyses is adequate for EPU operation at Dresden and Quad Cities is provided below.

The COTRANSA2 methodology results in predicted power increases that are bounding ([ ] on average) relative to Peach Bottom benchmark tests. In addition, for licensing calculations a 110% multiplier is applied to the calculated integral power to provide additional conservatism. This approach adds significant conservatism to the calculated OLMCPR as discussed previously.

Biasing of important input parameters in licensing calculations provides additional conservatism in establishing the OLMCPR. The Peach Bottom turbine trips were performed assuming the measured performance of important input parameters such as control rod scram speed and turbine valve closing times. For licensing calculations, these (and other) parameters are biased in a conservative bounding direction. These conservative assumptions are not combined statistically; assuming all parameters are bounding at the same time produces very conservative results.

With the ATRIUM 10XM void fraction benchmarks presented in Figure B-2 and Figure B-4, the applicability of the void-quality correlation at high void fractions is confirmed and the uncertainty associated with the application of the correlation to the ATRIUM 10XM design is demonstrated to be equivalent to the data used in the bias assessment. Therefore, the sensitivity studies and conclusions drawn from the study are equally applicable to EPU operation of the ATRIUM 10XM fuel at Dresden and Quad Cities.

Table B-1 AREVA Multi-Rod Void Fraction Validation Database

	FRIGG-2 (Reference 18)	FRIGG-3 (Reference 16 & 17)	ATRIUM-10 KATHY	ATRIUM 10XM KATHY
Axial Power Shape	**uniform**	**uniform**	[ ]	[ ]
Radial Power Peaking	uniform	mild peaking	[ ]	[ ]
Bundle Design	circular array with 36 rods + central thimble	circular array with 36 rods + central thimble	[ ]	[ ]
Pressure (psi)	725	725, 1000, and 1260	[ ]	[ ]
Inlet Subcooling (^{0}F)	4.3 to 40.3	4.1 to 54.7	[ ]	[ ]
Mass Flow Rate (lbm/s) *(Based on mass flux assuming ATRIUM-10 inlet area)*	14.3 to 31.0	10.1 to 42.5	[ ]	[ ]
Equilibrium Quality at Measurement Plane (fraction)	-0.036 to 0.203	-0.058 to 0.330	[ ]	[ ]
Max Void at Measurement Plane (fraction)	0.828	0.848	[ ]	[ ]
Reported Instrument Uncertainty (fraction)	0.025	0.016	[ ]	[ ]
Number of Data	27 tests, 174 points	39 tests, 157 points	[ ]	[ ]

Table B-2 Void Sensitivity Results

Parameter	Reference Calculation	Modified V-Q (0.0)	Modified V-Q (+0.05)	Modified V-Q (−0.05)
ΔCPR	0.305	0.307	0.321	0.305
SLMCPR	1.09	1.09	1.09	1.09
ΔSLMCPR	NA	−0.001	−0.002	+0.002
OLMCPR	1.395	1.396	1.409	1.397

[ ]

Figure B-1 Validation of [ ] using FRIGG-2 and FRIGG-3 Void Data

[ ]

Figure B-2 Validation of [ ] using ATRIUM-10 and ATRIUM 10XM Void Data

Figure B-3 Validation of Ohkawa-Lahey using FRIGG-2 and FRIGG-3 Void Data

Figure B-4 Validation of Ohkawa-Lahey using ATRIUM-10 and ATRIUM 10XM Void Data

Figure B-5 Modified Void Fraction Correlation Comparison to ATRIUM-10 Test Data

Figure B-6 [ ] Void Fraction Comparison to ATRIUM-10 Test Data

Appendix C. Neutronic Methods

C.1 *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. The library has cross sections for **[ ]** materials including **[ ]** heavy metals. Depletion is performed with a predictor-corrector approach in each fuel or absorber rod. The two-dimensional transport solution is based upon the **[ ]**. The solution provides pin-by-pin power and as macro-scopic cross sections. Discontinuity factors are determined from the solution. **[ ]** gamma transport calculation are performed. The code has the ability to perform **[ ]** calculations with different mesh spacings. Reflector calculations are easily performed.

MICROBURN-B2 performs microscopic fuel depletion on a nodal basis. The neutron diffusion equation is solved with a full two energy group method. A modern nodal method solution using discontinuity factors is used along with a **[ ]**. The flux discontinuity factors are **[ ]**. A multilevel iteration technique is employed for efficiency. MICROBURN-B2 treats a total of **[ ]** heavy metal nuclides to account for the primary reactivity components. Models for nodal **[ ]** are used to improve the accurate representation of the in-reactor configuration. A full three-dimensional pin power reconstruction method is utilized. TIP (neutron and gamma) and LPRM response models are included to compare calculated and measured instrument responses. Modern steady state thermal hydraulics models define the flow distribution among the assemblies. **[ ]** based upon CASMO-4 calculations are used for the in-channel fluid conditions as well as in the bypass and water rod regions. Modules for the calculation of CPR, LHGR and MAPLHGR are implemented for direct comparisons to the operating limits.

MICROBURN-B2 determines the nodal macroscopic cross sections by summing the contribution of the various nuclides.

$$\Sigma_x(\rho,\Pi,E,R) = \sum_{i=1}^{I} N_i \sigma_x^i(\rho,\Pi,E,R) + \Delta\Sigma_x^b(\rho,\Pi,E,R)$$

where:

Σ_x	= nodal macroscopic cross section
$\Delta\Sigma_x^b$	= background nodal macroscopic cross section $(D, \Sigma_f, \Sigma_a, \Sigma_r)$
N_i	= nodal number density of nuclide "i"
σ_x^i	= microscopic cross section of nuclide "i"
I	= total number of explicitly modeled nuclides
ρ	= nodal instantaneous coolant density
Π	= nodal spectral history
E	= nodal exposure
R	= control fraction

The functional representations of σ_x^i and $\Delta\Sigma_x^b$ come from 3 void depletion calculations with CASMO-4. Instantaneous branch calculations at alternate conditions of void and control state are also performed. The result is a multi-dimensional table of microscopic and macroscopic cross sections that are shown in Figure C-1 and Figure C-2 for a representative lattice and each lattice will have specific cross section data.

At BOL the relationship is fairly simple; the cross section is only a function of void fraction (water density) and the reason for the variation is the change in the spectrum due to the water density variations. 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. This fit is shown in Figure C-3 and Figure C-4.

Detailed CASMO-4 calculations confirm that a quadratic fit accurately represents the cross sections and diffusion coefficient as shown in Figure C-5, Figure C-6, and Figure C-7.

With depletion the isotopic changes cause other spectral changes. Cross sections change due to the spectrum changes. Cross sections also change due to self-shielding as the concentrations change. These are accounted for by the void (spectral) history and exposure parameters. Exposure variations utilize a piecewise linear interpolation over tabulated values at

[ ] exposure points. The four dimensional representation can be reduced to three dimensions (see Figure C-8) by looking at a single exposure.

Quadratic interpolation is performed in each direction independently for the most accurate representation. Considering the case at 70 GWd/MTU with an instantaneous void fraction of 70% and a historical void fraction of 60%, Figure C-9 and Figure C-10 illustrate the interpolation process. The table values from the library at 0, 40 and 80 % void fractions are used to generate 3 quadratic curves representing the behavior of the cross section as a function of the historical void fraction for each of the tabular instantaneous void fractions (0, 40 and 80 %).

The intersection of the three quadratic lines with the historical void fraction of interest are then used to create another quadratic fit in order to obtain the resultant cross section as shown in Figure C-10.

The results of this process for all isotopes and all cross sections in MICROBURN-B2 were compared for an independent CASMO-4 calculation with continuous operation at 20, 60 and 90% void and are presented in Figure C-11. Branch calculations at 90% void from a 40% void depletion were performed for multiple exposures. The results show very good agreement for the whole exposure range as shown in Figure C-12.

At the peak reactivity point, multiple comparisons were made (Figure C-13) to show the results for various instantaneous void fractions.

[

]

Void fraction has been used for the previous illustrations; however MICROBURN-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. 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.) The Doppler feedback due to the fuel temperature is modeled by accumulating the Doppler broadening of microscopic cross sections of each nuclide.

$$\Delta\Sigma_x = \left(\sqrt{T_{eff}} - \sqrt{T_{ref}}\right)\sum_{i}^{I} \frac{\partial \sigma_x^i}{\partial \sqrt{T_{eff}}} N_i$$

where:

T_{eff} = Effective Doppler Fuel Temperature

T_{ref} = Reference Doppler Fuel Temperature

σ_x^i = Microscopic Cross Section (fast and thermal absorption) of nuclide “i”

N_i = Density of nuclide “i”

The partial derivatives are determined from branch calculations performed with CASMO-4 at various exposures and void fractions for each void history depletion. The tables of cross sections include data for **[** **]** states. The process is the same for **[** **]** states. Other important feedbacks to nodal cross sections are lattice **[** **]** and instantaneous **[** **]** between lattices of different **[** **]**. These feedbacks are modeled in detail.

The methods used in CASMO-4 are state of the art. The methods used in MICROBURN-B2 are state of the art. The methodology accurately models a wide range of thermal hydraulic conditions including EPU and extended power/flow operating map conditions.

C.2 *Applicability of Uncertainties*

The TIPs directly measure the local neutron flux from the surrounding four fuel assemblies. Thus, the calculated bundle power distribution uncertainty will be closely related to the calculated TIP uncertainty. However, the bundle powers in the assemblies surrounding a TIP are not independent. If a bundle is higher in power, neutronic feedback increases the power in the nearby assemblies, thus producing a positive correlation between nearby bundles. The gamma scan data provides the means to determine this correlation according to the EMF-2158(P)(A) (Reference 19) methodology. A smaller correlation coefficient implies that there is less correlation between nearby bundle powers, thus, there would be a larger bundle power distribution uncertainty.

The EMF-2158(P)(A) data was re-evaluated by looking at the deviations between measured and calculated TIP response for each axial level. The standard deviation of these deviations at each axial plane are presented in Figure C-15 and demonstrate that there is no significant trend vs. axial position, which indicates no significant trend vs. void fraction. This same data was evaluated for trends based upon the core conditions at the time of each TIP scan. The core parameters of interest that were evaluated include core thermal power, the core average void fraction and the ratio between core power and core flow. The 2D standard deviations for “C” and “D” lattice plants are presented in Figure C-16 through Figure C-21, while the 3D standard deviations are presented in Figure C-22 through Figure C-27. This evaluation of the data indicates that there is no significant trend in the data associated with these plant parameters.

Parameters such as fuel density, part length rods, active fuel length, fuel pellet diameter and fuel cladding diameter are all inputs to the methodology. The methodology explicitly accounts for such changes in design parameters. The changes in these parameters for the ATRIUM 10 XM fuel are insignificant relative to the changes that have been included in the validation of the methodology that demonstrate the methodology’s capability to evaluate these parameters. Fuel designs including 7X7, 8X8, 9X9 and 10X10 with corresponding changes in pellet and cladding diameters were presented in Reference 19.

The Quad Cities assembly gamma scan data was used to determine the correlation coefficient which accounts for the correspondence between the assembly powers of adjacent assemblies. This correspondence is quantified by a conservative multiplier to the uncertainty in the TIP measurements. In order to conservatively account for this correspondence, the bundle power uncertainty is increased due to the radial TIP uncertainty by a multiplier based on the correlation coefficient. The correlation coefficient is statistically calculated and shown in Figure 9.1 and Figure 9.2 of EMF-2158(P)(A). It indicates a less than perfect correlation between powers of neighboring bundles. The conservative multiplier is calculated as follows:

The calculated TIP uncertainty would normally be expected to be slightly larger than the calculated power uncertainty due to the TIP model. The Quad Cities gamma scan comparison shows the 2-D radial power uncertainty of [ ] (see Section 9.6 of EMF-2158(P)(A)). The D-Lattice plant calculated radial TIP uncertainty is [ ]. The data indicates that the calculated TIP uncertainty is indeed larger than the calculated power uncertainty. The use of the correlation coefficient to increase the calculated power uncertainty is a very conservative approach resulting from the statistical treatment. The types of fuel bundles (8x8, 9x9, or 10x10) loaded in the core has no effect on the reality of the physical model which precludes the possibility of the calculated power uncertainty to be larger than the calculated TIP uncertainty. The accuracy of the MICROBURN-B2 models is demonstrated by comparisons between measured and calculated TIP's as well as comparison of calculated and measured Ba-140 density distribution. The accuracy of the MICROBURN-B2 models was further validated with detailed axial pin by pin gamma scan measurements of 9X9-1 and ATRIUM-10 fuel assemblies in the reactor designated as KWU-S. These measurements demonstrated the continued accuracy of the MICROBURN-B2 models with modern fuel assemblies. Details of these measurements are provided in Section 8.2.2 of the topical report, EMF-2158(P)(A). Reference 19 Figures 8.18 through 8.31 showed very good comparisons between the calculated and measured relative Ba-140 density distributions of actual irradiated assemblies for both radial and axial values.

The AREVA SAFLIM3D code is used to calculate the number of expected rods in boiling transition (BT) for a specified value of the SLMCPR (i.e., SLMCPR is an input, not a calculated result). The extremes of the two correlation coefficients from the Quad Cities assembly gamma scan data sets [ ] discussed above were used for a sensitivity study of the MCPR safety limit. An analysis of the safety limit was performed with SAFLIM3D using an input SLMCPR of 1.0658 and the base RPF and nodal power uncertainties. The number of boiling transition (BT) rods was calculated to be 50 from this analysis. The analysis was repeated in a series of SAFLIM3D calculations using the increased RPF and nodal power uncertainties and performed by iterating on the input value of SLMCPR. Different values for the SLMCPR input were used until the number of BT rods calculated by SAFLIM3D was the same as the base case (50 rods). A SLMCPR input value of [ ] resulted in 50 rods in BT when the increased RPF and nodal power uncertainties was input. The difference in SLMCPR input [ ] for the two cases that resulted in the same number of BT rods is a measure of the safety limit sensitivity to the increased RPF and nodal power uncertainties.

The only input parameters that changed between the two SAFLIM3D analyses were the SLMCPR, the RPF, and nodal power uncertainties. For each analysis, 1000 Monte Carlo trials were performed. To minimize statistical variations in the sensitivity study, the same random number seed was used and all bundles were analyzed for both analyses. As discussed above, 50 rods were calculated to be in BT in both analyses.

This sensitivity study was performed to quantify the sensitivity of SLMCPR to an increase in RPF and nodal power uncertainties and did not follow the standard approach used in SLMCPR licensing analyses. In standard licensing calculations, the SLMCPR is not input at a precision greater than the hundredths decimal place. As a result, the increased RPF and nodal power uncertainties would result **[ ]** in SLMCPR licensing analyses depending on how close the case was to the acceptance criterion prior to the increase in RPF uncertainty.

Gamma scanning provides data on the relative gamma flux from the particular spectrum associated with La140 gamma activity. The relative gamma flux corresponds to the relative La140 concentration. Based upon the time of shutdown and the time of the gamma scan the Ba140 relative distribution at the time of shutdown is determined. This Ba140 relative distribution is thus correlated to the pin or assembly power during the last few weeks of operation. The data presented in the topical report, EMF-2158(P)(A), includes both pin and assembly Ba140 relative density data. The assembly gamma scan data was taken at Quad Cities after the operation of cycles 2, 3 and 4. Some of this data also included individual pin data. This data was from 7X7 and 8X8 fuel types. Additional fuel pin gamma scan data was taken at the Gundremingen plant for ATRIUM-9 and ATRIUM-10 fuel. This data is also presented in the topical report.

Pin-by-pin Gamma scan data is used for verification of the local peaking factor uncertainty. Quad Cities measurements presented in the topical report EMF-2158(P)(A) have been re-evaluated to determine any axial dependency. Figure C-28 presents the raw data including measurement uncertainty and demonstrates that there is no axial dependency. The more recent Gamma scans performed by KWU, presented in the topical report EMF-2158(P)(A) and re-arranged by axial level in Table C-1, indicate no axial dependency. Full axial scans were performed on 16 fuel rods. Comparisons to calculated data show excellent agreement at all axial levels. The dip in power associated with spacers, observed in the measured data, is not modeled in MICROBURN-B2. There is no indication of reduced accuracy at the higher void fractions. Details of the gamma scan process is described in Section C.4.

CASMO-4 and MCNP calculations have been performed to compare the fission rate distribution statistics to Table 2-1 of the topical report EMF-2158(P)(A) which is shown in Table C-2. The fission rate differences at various void fractions demonstrate that CASMO-4 calculations have very similar uncertainties relative to the MCNP results for all void fractions. These fission rate differences also meet the criteria of the topical report EMF-2158(P)(A) for all void fractions.

Data presented in these figures and tables demonstrate that the AREVA methodology is capable of accurately predicting reactor conditions for fuel designs operated under the current operating strategies and core conditions.

C.3 *Fuel Cycle Comparisons*

AREVA has reviewed the data presented in EMF-2158(P)(A) with regard to the maximum assembly power (Figure C-29) and maximum exit void fraction (Figure C-30) to determine the range of data previously benchmarked.

Fuel loading patterns and operating control rod patterns are constrained by the minimum critical power ratio (MCPR) limit, which consequently limits the assembly power and exit void fraction regardless of the core power level. Operating data from several recent fuel cycle designs have been evaluated and compared to that in the topical report EMF-2158(P)(A). Maximum assembly power and maximum void fraction are presented in Figure C-31 and Figure C-32.

In order to evaluate some of the details of the void distribution a current design calculation was reviewed in more detail. Figure C-33 and Figure C-34 present the parameters shown below at the point of the highest exit void fraction (at 9336 MWd/MTU cycle exposure) in cycle core design for a BWR-6 reactor (high power density plant) with ATRIUM-10 fuel. Additional details of the thermal hydraulic conditions is the population distribution of the void fractions. These are representative figures for a high power density plant.

- Core Average and peak assembly axial power profile

- Core average void axial profile
- Axial void profile of the peak assembly
- Histogram of the nodal void fractions in core

The actual core designs used for each cycle will have slightly different power distributions and reactivity characteristics than any other cycle. Conclusions from analyses that are dependent on the core design (loading pattern, control rod patterns, fuel types) are re-confirmed as part of the reload licensing analyses performed each cycle. Cycle-specific reload licensing calculations will continue to be performed for all future cycles using NRC approved methodologies consistent with the current processes.

The AREVA methodology **[ ]** the reactivity coefficients that are used in the transient analysis. Conservatisms in the methodology are used to produce results that bound the uncertainties in the reactivity coefficients. Data presented in these referenced figures indicate that there are no significant differences between EPU and non- EPU conditions that have an impact on the reactivity coefficients.

C.3.1 <u>Bypass Voiding</u>

The core bypass water is modeled in the AREVA steady-state core simulator, transient simulator, LOCA and stability codes as **[ ]**.

The steady-state core simulator, MICROBURN-B2, explicitly models the assembly specific flow paths through the lower tie-plate flow holes and the channel seals in addition to a **[ ]** through the core support plate. The numerical solution for the individual flow paths is computed based on a general parallel channel hydraulic solution that imposes a constant pressure drop across the core fuel assemblies and the bypass region. This solution scheme incorporates **[ ]** that is dependent on the **[ ]**.

The MICROBURN-B2 state-point specific solution for bypass flow rate and **[ ]** is then used as initial conditions in the transient and LOCA analyses. When the reactor operates on high rod-lines at low flow conditions, the in-channel pressure drop decreases to a point where a solid column of water cannot be supported in the bypass region, and voiding occurs in the core bypass. For these conditions (in the region of core stability concerns) the neutronic feedback of bypass voiding [

]

Bypass voiding is of greatest concern for stability analysis due to its direct impact on the fuel channel flow rates and the axial power distributions. The reduced density head in the core bypass due to boiling results in a higher bypass flow rate and consequently a lower hot channel flow rate. This lower hot channel flow rate and a more bottom-peaked power distribution (due to lower reactivity in the top of the core due to boiling in the bypass region) destabilize the core through higher channel decay ratios. These effects are small compared to the general conditions of low flow and high power that dominate the stability regime. Never the less, AREVA stability methods directly model these phenomena to assure that the core stability is accurately predicted.

CASMO-4 has the capability to specify the density of the moderator in the bypass and in-channel water rods, [

] .

Significant bypass voiding is not encountered during full power, steady-state EPU operation for Dresden and Quad Cities so there is no impact on steady-state analyses. For transient conditions it is conservative to ignore the density changes as additional voiding aids in shutting down the power generation.

For Dresden and Quad Cities, a 100% power / 85% flow statepoint (120% of the original licensed thermal power) was assessed even though this statepoint is outside of the power/flow operating map to cover a wider range of flow. Even with the conservative multi-channel model, there was minimal localized bypass boiling at the EPU power level. This assessment assures that the limiting transients at the uprated thermal power are not adversely affected by bypass boiling. As the flow is reduced along the 100% power line, the decrease in flow is compensated by increased subcooling which compensates for the decrease in flow. When flow is further reduced along the highest rod line, more significant boiling in the bypass is calculated to begin. This is in the area of stability concerns where the boiling in the bypass is modeled explicitly. For normal operation at 100% power minimal boiling in the bypass is expected to occur, so there is no impact on the lattice local peaking or the LPRM response.

C.3.2 <u>Fuel Assembly Design</u>

No fuel design modifications have been made for EPU operation, neither mechanical nor thermal hydraulic. The maximum allowed enrichment level of any fuel pellet is 4.95 wt% U-235.

All new and spent fuel at Dresden and Quad Cities is stored in the Spent Fuel Storage Pool (SFSP) and in accordance with Technical Specification 4.3.1.1 must maintain a subcritical multiplication factor (keff) of less than 0.95 when flooded with non-borated water. The customer has chosen another vendor to perform the required criticality safety analyses so AREVA methods are not utilized.

C.4 *Gamma Scans*

The gamma scanning process for irradiated fuel uses germanium semi-conductor detectors for gamma radiation energy spectral analysis. Gamma rays deposit their energy in the germanium and produce free electrons and holes (vacancies where the electrons were located in the crystalline germanium). The amount of charge collected is correlated with the amount of energy deposited in the detector and therefore with the energy of the gamma ray that caused it. The detectors are used with single channel analyzers to sort the pulses according to pulse height. This means that if multiple gamma-ray energies are being analyzed simultaneously, the germanium detector will separate them cleanly. A single-channel analyzer (SCA) uses two discriminators. The discriminators are called upper and lower level discriminators. Pulses from the amplifier are fed to the analyzer, and if the pulse height falls between the lower and upper discriminators the usual logic is to allow such a pulse to be recorded (counted). The voltage levels of the two discriminators are adjustable so that the gap between them corresponds to a group of pulse heights within a fixed energy interval. Even though the gamma rays from a specific decay transition are of a discrete energy, there is a statistical spread of pulses coming from the detector and associated electronics so that the gap between the discriminators must be large enough to include most of such pulses. By varying the voltage levels of each of the discriminators, it is possible to measure gamma rays of different energies.

Power measurements for irradiated fuel target the gamma spectrum associated with Lanthanum (La) 140. La140 is a decay product of Barium (Ba) 140 which is a direct fission product. The half-life of Ba140 is 12.8 days and the half-life of La140 is 40 hours. La140 activity is, therefore, directly related to the density of Ba140. The Ba140 density is representative of the integrated fissions over the last 25 days due to its short half-life. Gamma scan measurements are taken shortly after reactor shutdown (within 25 days) before the Ba140 decays to undetectable levels. Gamma scan measurements may be performed on individual fuel rods removed from assemblies using a high-purity germanium (HPGe) detector and an underwater collimator assembly or on entire fuel assemblies where the collimator has a broad opening to capture the gamma radiation from all of the pins in the assembly.

To compare core physics models to the gamma scan results, the calculated pin power distribution is converted into a Ba140 density distribution. A rigorous mathematical process using CASMO-4 pin nuclide inventory and MICROBURN-B2 nodal nuclide inventory is used.

Table C-1 KWU-S Gamma Scan Benchmark Results from EMF-2158(P)(A)

Table C-2 Comparison of CASMO-4 and MCNP results for ATRIUM-10 Design

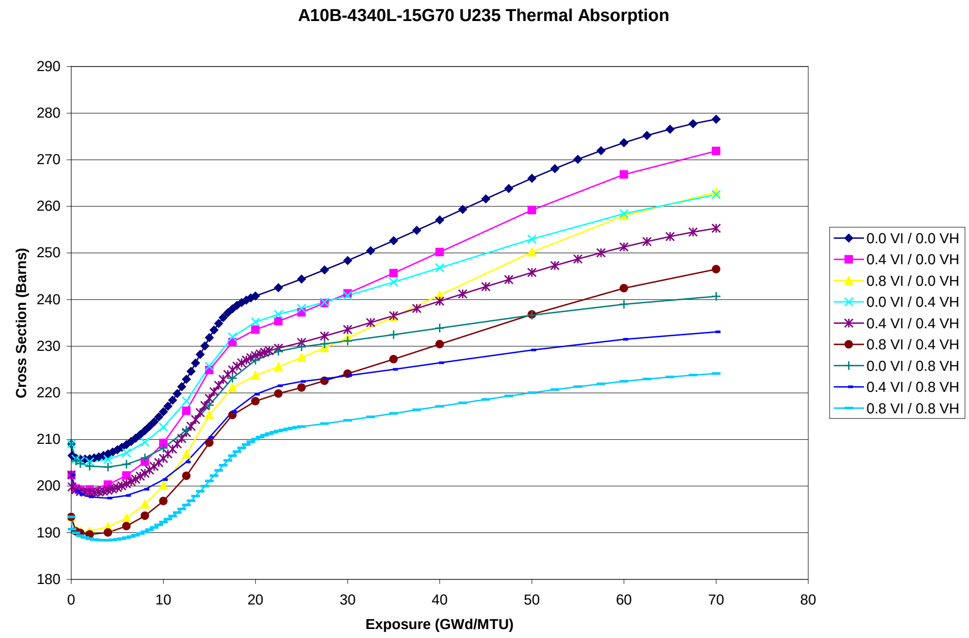


Figure C-1 Microscopic Thermal Cross Section of U-235 from Base Depletion and Branches

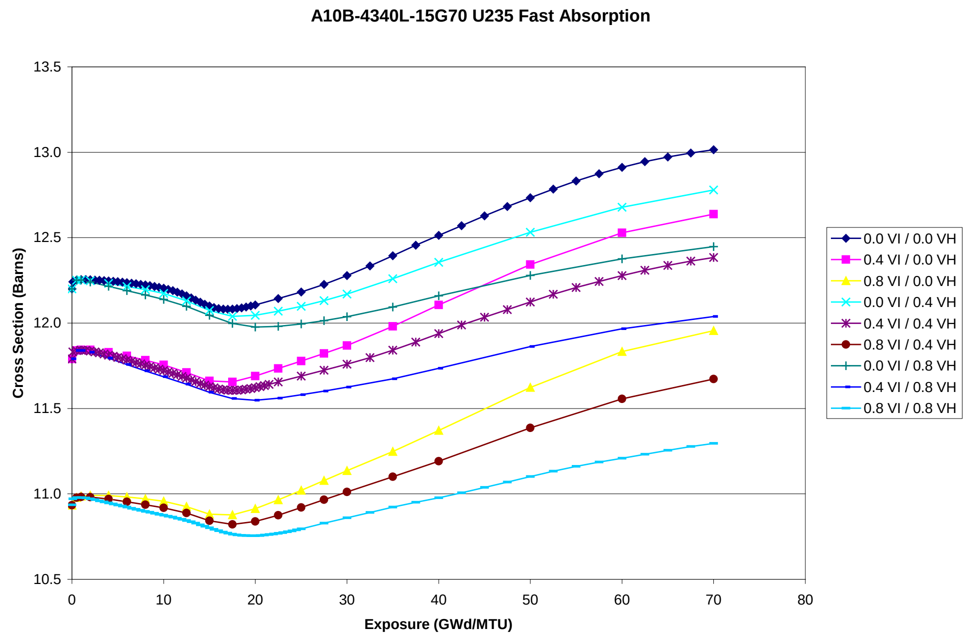


Figure C-2 Microscopic Fast Cross Section of U-235 from Base Depletion and Branches

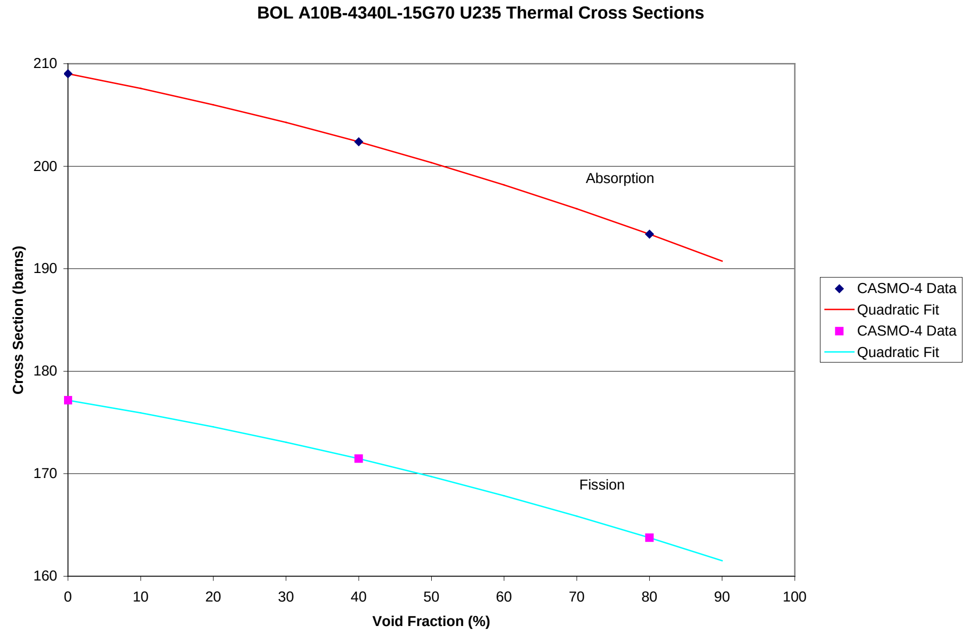


Figure C-3 Microscopic Thermal Cross Section of U-235 at Beginning of Life

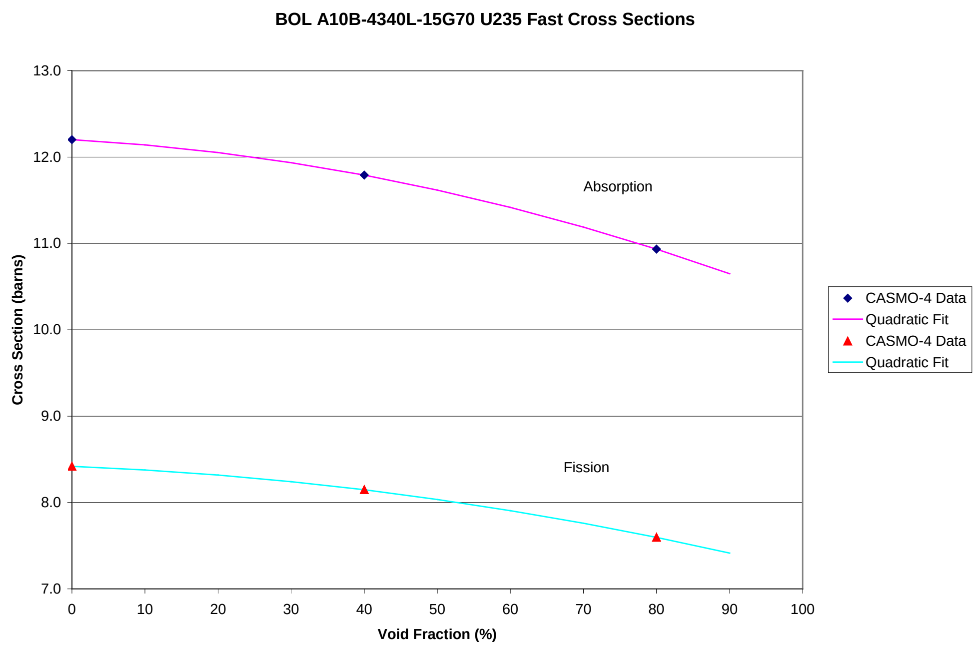


Figure C-4 Microscopic Fast Cross Section of U-235 at Beginning of Life

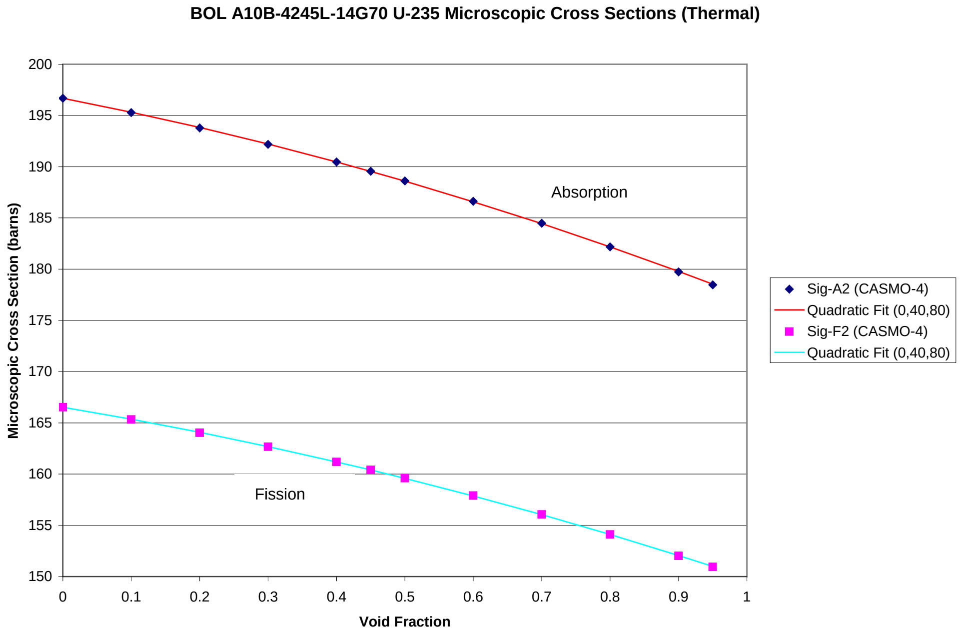


Figure C-5 Microscopic Thermal Cross Section of U-235 Comparison of Quadratic Fit with Explicit Calculations at Various Void Fractions

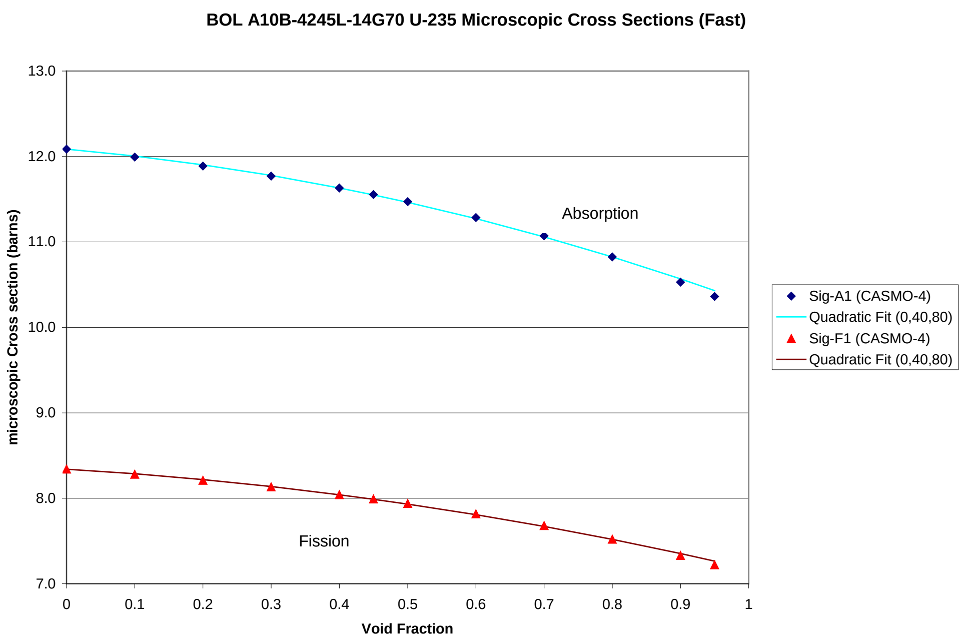


Figure C-6 Microscopic Fast Cross Section of U-235 Comparison of Quadratic Fit with Explicit Calculations at Various Void Fractions

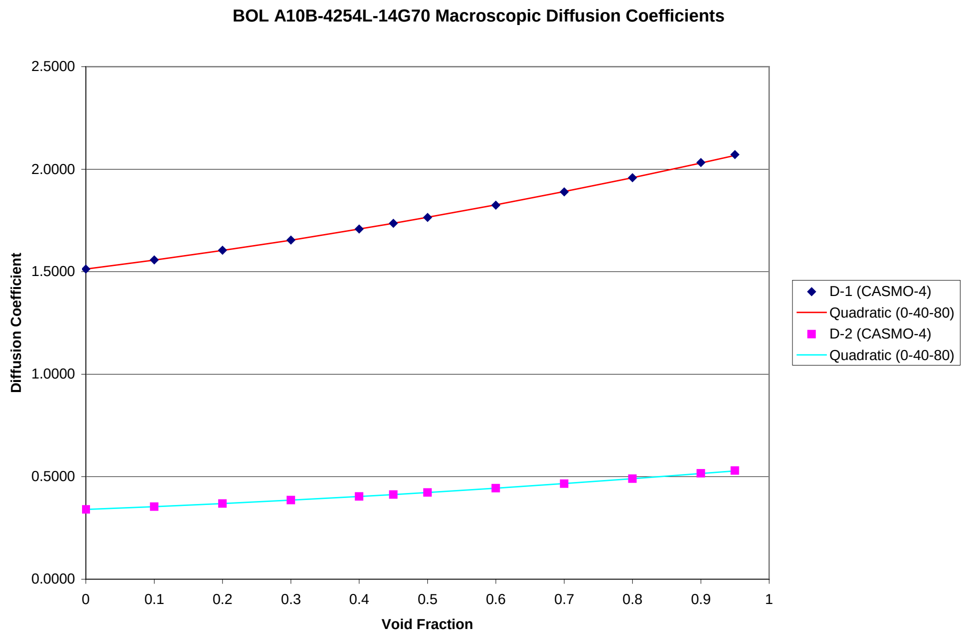


Figure C-7 Macroscopic Diffusion Coefficient (Fast and Thermal) Comparison of Quadratic Fit with Explicit Calculations at Various Void Fractions

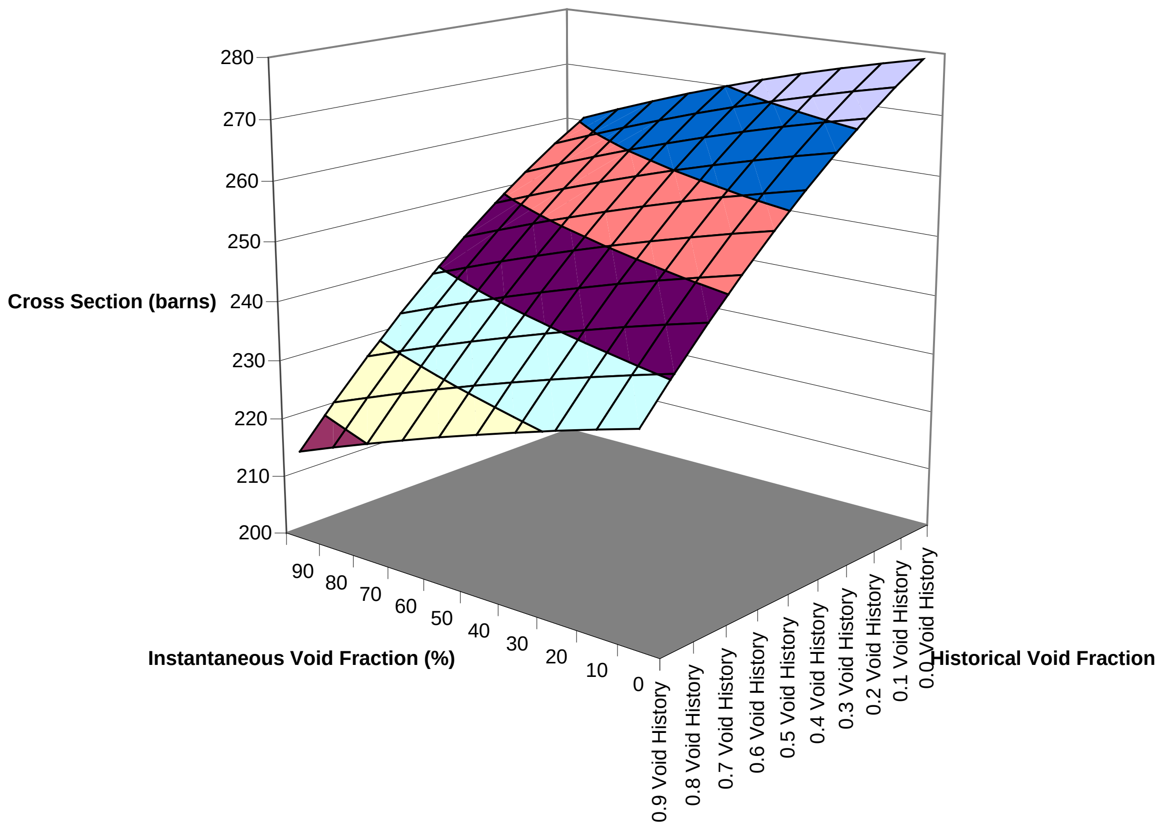


Figure C-8 Microscopic Thermal Cross Section of U-235 at 70 GWd/MTU

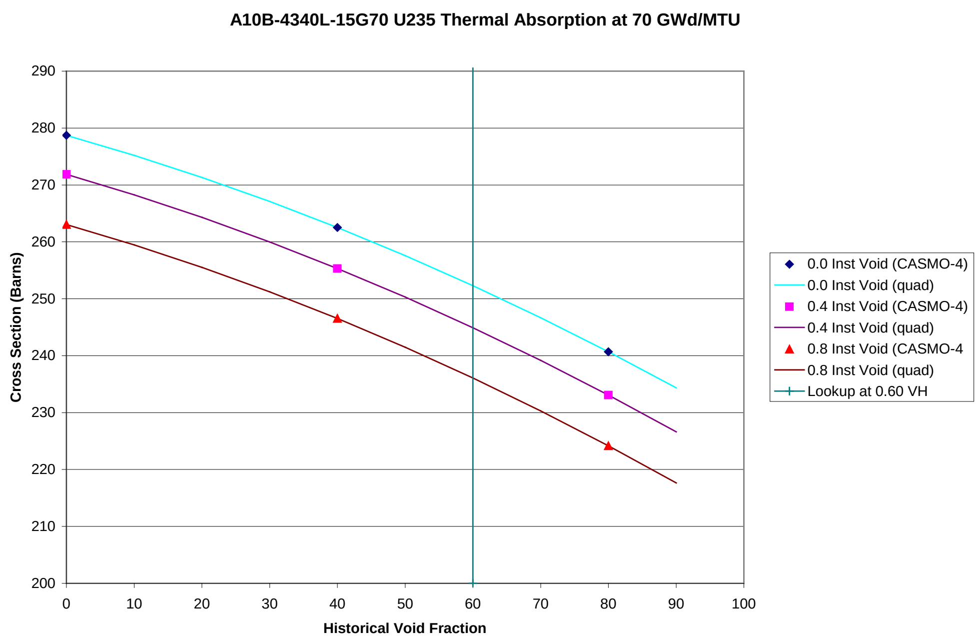


Figure C-9 Quadratic Interpolation Illustration of Microscopic Thermal Cross Section of U-235

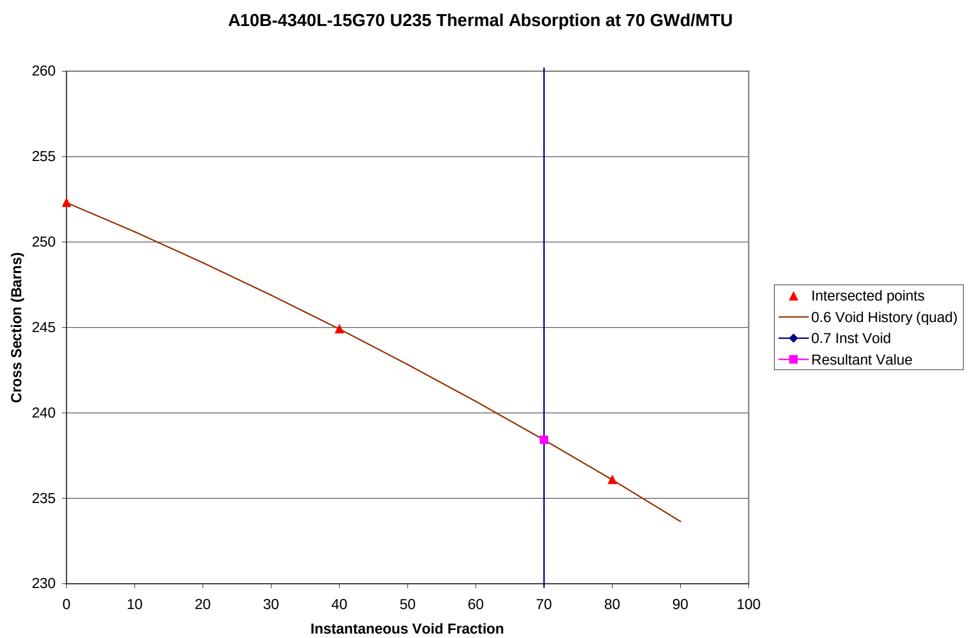


Figure C-10 Illustration of Final Quadratic Interpolation for Microscopic Thermal Cross Section of U-235

[ ]

Figure C-11 Comparison of k-infinity from MICROBURN-B2 Interpolation Process with CASMO-4 Calculations at Intermediate Void Fractions of 0.2, 0.6 and 0.9

[ ]

Figure C-12 Comparison of k-infinity from MICROBURN-B2 Interpolation Process with CASMO-4 Calculations at 0.4 Historical Void Fractions and 0.9 Instantaneous Void Fraction

[ ]

Figure C-13 Delta k-infinity from MICROBURN-B2 Interpolation Process with CASMO-4 Calculations at 0.4 Historical Void Fraction and 0.9 Instantaneous Void Fraction

[ ]

Figure C-14 Comparison of Interpolation Process Using Void Fractions of 0.0, 0.4 and 0.8 and Void Fractions of 0.0, 0.45 and 0.9

Figure C-15 EMF-2158(P)(A) TIP Statistics by Axial Level

Figure C-16 EMF-2158(P)(A) 2-D TIP Statistics for C-Lattice Plants vs. Core Power

Figure C-17 EMF-2158(P)(A) 2-D TIP Statistics for C-Lattice Plants vs. Core Average Void Fraction

Figure C-18 EMF-2158(P)(A) 2-D TIP Statistics for C-Lattice Plants vs. Core Power/Flow Ratio

Figure C-19 EMF-2158(P)(A) 2-D TIP Statistics for D-Lattice Plants vs. Core Power

Figure C-20 EMF-2158(P)(A) 2-D TIP Statistics for D-Lattice Plants vs. Core Average Void Fraction

Figure C-21 EMF-2158(P)(A) 2-D TIP Statistics for D-Lattice Plants vs. Core Power/Flow Ratio

Figure C-22 EMF-2158(P)(A) 3-D TIP Statistics for C-Lattice Plants vs. Core Power

Figure C-23 EMF-2158(P)(A) 3-D TIP Statistics for C-Lattice Plants vs. Core Average Void Fraction

Figure C-24 EMF-2158(P)(A) 3-D TIP Statistics for C-Lattice Plants vs. Core Power/Flow Ratio

Figure C-25 EMF-2158(P)(A) 3-D TIP Statistics for D-Lattice Plants vs. Core Power

Figure C-26 EMF-2158(P)(A) 3-D TIP Statistics for D-Lattice Plants vs. Core Average Void Fraction

Figure C-27   EMF-2158(P)(A) 3-D TIP Statistics for D-Lattice Plants vs. Core Power/Flow Ratio

Figure C-28   Quad Cities Unit 1 Pin by Pin Gamma Scan Results

Figure C-29 Maximum Assembly Power in Topical Report EMF-2158(P)(A)

Figure C-30 Maximum Exit Void Fraction in Topical Report EMF-2158(P)(A)

Figure C-31 Maximum Assembly Power Observed from Recent Operating Experience

Figure C-32 Void Fractions Observed from Recent Operating Experience

Figure C-33 Axial Power and Void Profile Observed from Recent Design Experience

Figure C-34 Nodal Void Fraction Histogram Observed from Recent Design Experience

Appendix D. Transient Methods

D.1 *COTRANSA2*

D.1.1 Conservatism

Integral power is a parameter obtainable from test measurements that is directly related to ΔCPR and provides a means to assess code uncertainty by increasing heat flux during the event. The COTRANSA transient analysis methodology was a predecessor to the COTRANSA2 methodology. The integral power figure of merit was introduced with the COTRANSA methodology as a way to assess (not account for) code uncertainty impact on ΔCPR. From COTRANSA analyses of the Peach Bottom turbine trip tests, the mean of the predicted to measured integral power was 99.7% with a standard deviation of 8.1%. AREVA (Exxon Nuclear at the time) initially proposed to treat integral power as a statistical parameter. However, following discussions with the NRC, it was agreed to apply a deterministic 110% integral power multiplier (penalty) on COTRANSA calculations for licensing analyses. That increase was sufficient to make the COTRANSA predicted to measure integral power conservative for all of the Peach Bottom turbine trip tests.

COTRANSA2 (Reference 8) was developed and approved as a replacement for COTRANSA in the AREVA thermal limits methodology (Reference 28). Initially it was not planned to use the 110% integral power multiplier with the COTRANSA2 methodology. COTRANSA2 predictions of integral power were conservative for all Peach Bottom turbine trip tests. The minimum conservatism was **[ ]** and the mean of the predicted to measured integral power was **[ ]**. The comparisons to the Peach Bottom turbine trip tests demonstrated that the 110% integral power multiplier was not needed for COTRANSA2. However, because the thermal limits methodology that was approved independently of COTRANSA2 included discussion of the 110% integral power multiplier, the use of the multiplier was retained for COTRANSA2 licensing calculations. With the 110% multiplier, the COTRANSA2 predicted to measured mean integral power is **[ ]** for the Peach Bottom turbine trip tests. Applying a **[ ]** integral power multiplier provides an OLMCPR conservatism of

[ ] . The 110% integral power multiplier is just one part of the conservatism in the COTRANSA2 methodology and application process that covers methodology uncertainties.

The 110% integral thermal power multiplier is applied to the output of COTRANSA2 that is used as the input to XCOBRA-T; therefore, the 110% integral power multiplier is included in the transient analyses. Important input parameters are biased in a conservative direction in licensing calculations. For Technical Specification (TS) controlled input parameters, the biasing is either the limiting value allowable by TS, or an analytical limit that is beyond the limiting value allowable by TS. If a particular equipment out-of-service is applicable to a particular transient event, the transient analysis is performed with the limiting plant configuration for the allowable equipment out-of-service.

D.1.2 COTRANSA2 Cross Section Representation

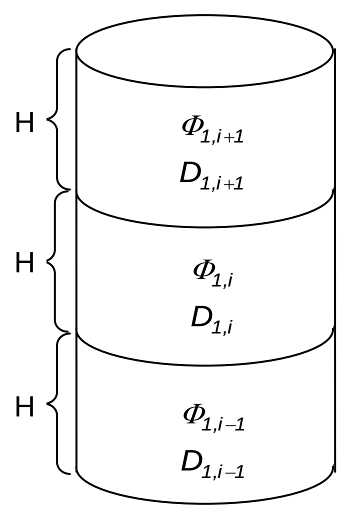
The COTRANSA2 transient simulator solves the one-dimensional neutron diffusion equation to predict the core average power response. In order to accurately capture the core reactivity characteristics, a series of MICROBURN-B2 calculations are performed. These successive calculations are:

[ ]

The 1½ energy group diffusion equation in steady-state can be written as

$$\nabla \cdot D_1 \nabla \Phi_1 - \left(\Sigma_{a1} + \frac{\Sigma_{1-2}}{\Sigma_{a2}} \cdot \Sigma_{a2} \right) \Phi_1 + \frac{\left(\nu\Sigma_{f2} + \frac{\Sigma_{1-2}}{\Sigma_{a2}} \cdot \nu\Sigma_{f2} \right) \Phi_1}{k_{eff}} = 0$$

The first term is a leakage. This equation is integrated over the cylindrical node depicted in the following figure.



The leakage term is approximated as:

$$-\sum_{j=1}^{3}\frac{2D_{1,i}D_{1,j}(\Phi_{1,i}-\Phi_{1,j})}{(D_{1,i}+D_{1,j})}\frac{A}{HV}$$

where

$D_{1,i}$ = D for plane of interest

$D_{1,j}$ = D for the nodes adjacent to the plane of interest

$\Phi_{1,i}$ = flux in the plane of interest

$\Phi_{1,j}$ = flux in the regions adjacent to the plane of interest

A = surface area between nodes i and j

H = distance between nodes i and nodes j

V = node volume

D.2 *XCOBRA-T*

D.2.1 Axial Geometry Changes

XCOBRA-T calculates the fuel rod surface heat flux using a fuel rod heat conduction model, the power generated in the fuel rod, and the fluid conditions at the surface of the rod. The power generated in the fuel rod is described in Reference 14 Section 2.5.5. The power generated in each axial section of a fuel rod is calculated using Equation 2.130 from Reference 14. Although Reference 14 states that Equation 2.130 is calculated for each axial node, the equation itself does not denote which variables are axially dependent. Because the equation is for each axial node, the variables for heat generation rate, axial peaking factor, and number of rods are axial dependent. At the time Reference 14 was prepared, the number of rods at each axial plane was a constant for the fuel designs being supplied. For the ATRIUM-10 and ATRIUM 10XM fuel design with part-length fuel rods (PLFRs), the number of rods became axial dependent and the code was modified to make application of Equation 2.130 correct and consistent with the NRC-approved Reference 14. For application to current fuel designs, a better definition of the variable Nr in Equation 2.130 would be "number of *heated* rods per assembly *at the axial plane*" (italic indicates added text).

For bundles with part-length fuel rods (PLFRs), the rod heat flux calculation begins by computing the time-dependent heat flux generation rate at each axial section in the fuel rod. The updated equation, corresponding to Equation 2.130 of Reference 14 is:

$$q''(t) = \frac{P(t)}{\pi D_{rod,i}} \frac{1}{L N_a N_{ri}} \left(f_f + f_c \right) F_{ri} F_{li} F_a$$

where

P(t)	=	transient reactor power
f_f	=	fraction of power produced in the fuel
f_c	=	fraction of power produced in the cladding
N_a	=	total number of assemblies in the core
N_{ri}	=	total number of heated rods for type i assembly at the axial plane
F_{ri}	=	radial peaking factor of type i assembly
F_{li}	=	local peaking factor of type i assembly
F_a	=	axial peaking factor at the axial plane
$D_{rod,i}$	=	fuel rod diameter of type i assembly
L	=	axial heated length

This equation differs from that in Reference 14 by replacing the initial reactor power in the denominator with π. In addition, the variable definitions have been modified to identify that the total number of heated rods is dependent on both the assembly type and axial elevation and the definition of L has been corrected to the axial heated length of the assembly. This equation is substituted into equations 2.129a and 2.129b in Section 2.5.5 of Reference 14 to define the volumetric heat deposition rate for the fuel pellet and cladding, respectively. This volumetric heat deposition rate is used in the right-hand side of equation 2.85 of Reference 14 to iteratively solve the transient heat conduction equation and the hydraulic conservation equations for the new time step temperatures and surface heat flux. The heat flux is introduced into the channel energy equation (2.2 of Reference 14) through the term q'. This linear heat deposition rate is a summation of the energy added by direct energy deposition and surface heat flux:

$$q'(t) = \left\{ \frac{P(t)}{N_a L} f_{cool} \, F_{ri} F_a + H_{surf} \cdot (T_{NodesT} - T_{fluid}) \cdot \pi \cdot D_{rod,i} \cdot N_{ri} \right\} N_i$$

where

f_{cool}	=	fraction of power produced in the coolant
H_{surf}	=	film heat transfer coefficient at the axial plane
T_{NodesT}	=	cladding surface temperature at the axial plane
T_{fluid}	=	fluid temperature at the axial plane
N_i	=	number of fuel assemblies in channel i

In addition to axially varying number of heated rods, proper modeling of PLFRs also requires axial variations in the active flow area, the heated perimeter, and the wetted perimeter and these parameters are now defined as axially dependent quantities in AREVA methods. Consequently, all references to these parameters or parameters derived from the basic geometry data in the approved topical reports should be interpreted as being axially dependent variables. The pressure drop due to the area expansion at the end of the PLFRs (or anywhere in the active flow path) is modeled using the specific volume for momentum as expressed in Equations 2.78 and 2.79 of Reference 14. For current designs, area contractions occur in the single phase region, but the coding was generalized to address area contractions in the two-phase region based on a solution of the two-phase Bernoulli equation.

D.2.2 Power

The decay heat is calculated by COTRANSA2 and is included in the total core power versus time provided as a boundary condition to XCOBRA-T. The decay heat model used in COTRANSA2 is a curve fit (11 groups) to the 1973 ANS standard decay heat model. The COTRANSA2 core power boundary condition includes the decay heat contribution based on the core average power density. The decay heat power remains essentially constant during the transient. Therefore, the decay heat during the transient is primarily a function of initial power density. Application of power peaking factors (axial, radial, local) to the COTRANSA2 average power properly accounts for local decay heat in the XCOBRA-T hot channel analysis.

Gamma smearing does not affect the XCOBRA-T hot channel calculation. The hot channel calculation models an average fuel rod (the average is not affected by flattening of the distribution). The calculation process for determining the peak transient LHGR is equivalent **[**

] and is not dependent of the actual rod local peaking factor.

The total core power calculated by COTRANSA2 is distributed between the fuel rod, the active channel coolant, and the core bypass coolant. The fraction deposited in each component is based on fuel type specific calculations performed with the CASMO computer code. Power fractions calculated by CASMO are used in XCOBRA-T.

D.2.3 Default Models

The Dresden and Quad Cities transient analyses used the default models of XCOBRA-T. The default models include Levy subcooled boiling model, the Martinelli-Nelson two phase friction multipliers, the two phase component loss multiplier, and the heated wall viscosity correction model. **[**

] as discussed with the NRC on May 4, 1995 (Reference 29). Thermo-dynamic properties from the ASME steam tables were used. The code provides a message if the default models are not used. Per AREVA's licensing analyses requirements, use of default models is required.

The Martinelli-Nelson two phase friction multiplier has been confirmed to be applicable in the annular flow regime by verifying the AREVA hydraulic models against two-phase full-scale heated bundle tests in the KATHY test facility in Karlstein, Germany. The range of assembly conditions at EPU are bounded by the tested two-phase flow conditions. Many of the tested conditions are in the annular flow regime.

The Levy subcooled boiling model does not directly predict void fraction in subcooled boiling. Instead, the model predicts a critical subcooling that defines the onset of boiling. The critical subcooling is used in conjunction with a profile fit model to determine the local flow quality that accounts for the presence of subcooled boiling. The local flow quality is then used in the Ohkawa-Lahey correlation to predict the void fraction in subcooled boiling.

D.2.4 Bounds Checking

Bounds checking is provided in the XCOBRA-T coding to ensure the conditions provided to the CPR correlations are within the correlation limits. Should any of the condition limits be violated, the behavior will be as specified in Section 2.6 of Reference 30 and Section 5.13 of Reference 2. In the specific case where the pressure limit is exceeded, XCOBRA-T will write an appropriate error message and terminate the calculation.

With respect to the remaining parameters, the behavior for transient calculations is summarized in Table D-1.

The out-of-bounds corrections affect the **[ ]** used in the evaluation of the transient LHGR. Therefore, the corrections do not impact the evaluation of the thermal-mechanical performance.

The critical power calculations for Dresden and Quad Cities fuel are made with the ACE and SPCB critical power correlations. The range of applicability of these parameters is sufficiently broad to cover the ranges of conditions encountered during the licensing calculations. Correlation bounds checking is incorporated in the XCOBRA-T critical power calculations. The bounds checking routine does not allow a calculation outside the range of applicability of these parameters except as described in References 2 and 30.

The transient code, XCOBRA-T, evaluates the Reynolds number for each node for each step of the calculation. If the flow becomes negative at any node, the code stops the calculation.

D.2.5 Heat Transfer Correlations

The thermocouples used for measuring temperature data in full scale critical power tests are **[**

] measure heat transfer coefficients associated with pre-CHF heat transfer in the range of mass and heat fluxes associated with BWRs. As a result, no relevant qualification studies of the Dittus-Boelter and Thom heat transfer correlations can be performed from the test data.

As noted in Reference 31, fully developed nucleate boiling is relatively insensitive to flow rate and quality. However, “boiling suppression” may occur in high quality annular flow that provides very high heat transfer coefficients, resulting in decreasing wall temperature as the heat flux increases.

Extracted heat transfer information from experiments in a tube for a range of pressure, flow, and quality that is relevant to BWRs is reported in Reference 32. This reference shows the relative insensitivity of heat transfer coefficient to flow rate and quality and that boiling suppression does not become significant until quality reaches approximately 0.47, which is well above the range of interest to BWRs. **[**

] Therefore, it is concluded that liquid entrainment and droplet redeposition does not have an impact on boiling heat transfer for flow conditions that are applicable to an operating BWR at EPU.

D.2.6 <u>Axial Power Shape</u>

The initial axial power shape is determined from the COTRANSA2 calculation based on the cross section data for the core exposure considered in the analysis. The cross section data is obtained from the MICROBURN-B2 computer code. The MICROBURN-B2 calculations used to generate cross section data for COTRANSA2 licensing calculations are typically performed assuming that all control rods are fully withdrawn. Assuming all control rods are fully withdrawn results in a significant conservatism in calculated scram reactivity for exposure conditions with some control rods partially inserted.

During pressurization transients, the axial power shape shifts due to the void collapse in the top of core (void reactivity), the core flow increase in the bottom of core (void reactivity), and control rod insertion in the bottom of the core (scram reactivity). The coupled 1-D neutronic and thermal-hydraulic core model in COTRANSA2 determines a **[** **]** including the impact of void collapse and scram. This **[** **]** is used in XCOBRA-T. The assembly **[**

] and the assembly radial peaking factor input to XCOBRA-T.

The change in axial nodal power, **[**

] results in a change in fuel rod surface heat flux and the energy transferred to the coolant at each axial node. Both COTRANSA2 and XCOBRA-T have fuel rod heat transfer models that determine the fuel rod surface heat flux based on the nodal power history and the coolant conditions at each axial node.

Both COTRANSA2 and XCOBRA-T have thermal-hydraulic models that are used to calculate the flow at each axial node in the core and the hot channel during the pressurization transient. The energy equation captures the effect of changes in fuel rod surface heat flux on coolant conditions. The mass and momentum equations, with applicable correlations, are used to determine the local coolant flow rate during the pressurization transient. During the initial phase of the pressurization transient, these models predict a decrease in flow near the top of the core and an increase in flow near the bottom of the core. Note, although the flow decreases in the upper portion of the hot assembly, the assembly flow does not stagnate during the pressurization phase of an AOO or ATWS. Local fluid conditions (enthalpy and flow) calculated from the thermal-hydraulic model are used to determine local dryout conditions.

D.2.7 Thermal Mechanical Performance

XCOBRA-T was used to demonstrate acceptable fuel rod thermal-mechanical performance during transients (AOOs). The fuel rod models in XCOBRA-T are consistent with RODEX2 and the fuel rod gap conductance values input to XCOBRA-T are obtained from RODEX2 analyses. The gap conductance includes the effect of pellet geometry changes (densification, swelling, etc.).

Table D-1 Bounds Checking

Figure D-1 Comparison of Scram Bank Worth for [ ]

Appendix E. LOCA Modifications

E.1 *LOCA Analysis*

The AREVA LOCA methodology applied at Dresden and Quad Cities differs from the approved methodology in three aspects:

E.1.1. Radiation View Factors

In the Safety Evaluation for Reference 20 the NRC approved the AREVA EXEM BWR-2000 ECCS evaluation model. The HUXY code (Reference 21) is the part of this model that performs the heatup calculations and provides PCT and local clad oxidation at the axial plane of interest. The code evaluates the radiation heat transfer between the fuel rod of interest and other fuels rods, the internal water canisters, and the fuel channel. AREVA has implemented an automated approach for calculating radiation view factors within the HUXY computer program.

The original approach was based on the method of cross-strings as described in Section 2.3 of Reference 21. This resulted in the derivation and programming of analytical expressions as a function of fuel rod diameters for the radiation view factors between each fuel rod and its predominant neighbors. The view factors were then internally computed throughout the HUXY heatup analyses based on these analytical expressions and the time dependent evolution of the fuel rod dimensions.

This analytical approach was well suited for the 7x7 and 8x8 designs analyzed at the time the method was originally implemented. With the evolution of design features such as larger internal water structures and part-length rods, the assembly lattice has become more heterogeneous and the derivation of the analytical expressions for the radiation view factors has become more complex. To automate the computation of the view factors for current and future fuel design concepts, a numerical computation of the view factors has been introduced utilizing a straight forward finite element method. The numerical computation, achieved through a ray-tracing algorithm, provides a direct accounting of the geometry to compute the view factor from each fuel rod to all other fuel rods, internal water structures, and the external fuel channel without the simplifying assumptions associated with the previous analytical approach.

Therefore, not only is the calculation of view factors automated, but the resulting values are more precise. The view factors calculated with the ray-tracing algorithm are then utilized identically to the way view factors developed from the cross-string method were utilized in the HUXY heatup analysis.

E.1.2. [ ]

[ ]

[

]

E.1.3. Thermal Conductivity Degradation

The EXEM BWR-2000 ECCS evaluation model uses the RODEX2 fuel rod models and therefore, underpredicts the impact of thermal conductivity degradation with exposure. The evaluation of thermal conductivity degradation and impact on PCT for Dresden and Quad Cities are presented in F.3.2.

[

]

Figure E-1 [ ]

Appendix F. Fuel Conductivity Degradation

F.1 *Introduction*

The U.S. Nuclear Regulatory Commission (NRC) issued Information Notice (IN) 2009-23 (No. ML091550527), dated October 8, 2009, for concerns regarding the use of historical fuel thermal conductivity models in the safety analysis of operating reactor plants. IN 2009-23 discusses how historical fuel thermal mechanical codes may overpredict 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. The NRC staff concluded that the use of the older legacy fuel models will result in predicted fuel pellet conductivities that are higher than the expected values.

This appendix summarizes the impact and treatment of fuel conductivity degradation for licensing safety analyses supporting operation at Dresden and Quad Cities.

F.2 *Disposition of Licensing Safety Analysis for Dresden and Quad Cities ATRIUM 10XM Fuel*

RODEX2 and RODEX2A codes were approved by the NRC in the early and mid-1980's, respectively. At that time, thermal conductivity degradation (TCD) with exposure was not well characterized by irradiation tests or post-irradiation specific-effects tests at high burnups. The fuel 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. Likewise, conductivity models in the transient codes COTRANSA2 and XCOBRA-T do not account for thermal conductivity degradation.

RODEX4 (Reference 3) 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 Dresden and Quad Cities are performed with RODEX4 and therefore explicitly account for thermal conductivity degradation. No additional assessment is needed for those analyses. For thermal-hydraulic and safety analyses an evaluation is needed. The following analysis methodologies use RODEX2 and/or include a separate UO_2 thermal conductivity correlation:

- Anticipated Operational Occurrence (AOO) analysis based on COTRANSA2/RODEX2/XCOBRA-T codes;
- Loss of Coolant Accidents (LOCA) analyses based on RELAX/RODEX2/HUXY codes;
- Overpressurization analyses based on COTRANSA2/RODEX2 codes;
- Stability analyses based on STAIF/RAMONA5-FA codes.

F.3 *Assessment of Analyses for Dresden and Quad Cities Operations*

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 (Reference 24). The white paper presented results of an extensive evaluation; for BWRs the assessments consisted primarily of ATRIUM-10 fuel.

The NRC reviewed Reference 24 and provided requests for information in Reference 25. AREVA provided responses in Reference 26. Items relevant from References 25 and 26 are also discussed in the following subsections.

F.3.1 Anticipated Operational Occurrence Analyses

The computer codes COTRANSA2 and XCOBRA-T are used in AOO analyses. Both codes use 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.

The assessment of Reference 24 demonstrated that COTRANSA2 uses several conservative assumptions, which results in conservatism relative to the Peach Bottom turbine trip qualification database. The COTRANSA2 methodology results in predicted integral power increases that are bounding relative to the Peach Bottom benchmark tests. With the 110% integral power multiplier used in the methodology, the COTRANSA2 predicted to measured mean integral power is **[ ]** for the Peach Bottom turbine trip tests. The COTRANSA2 benchmark testing was performed using the same UO_2 conductivity model as used in the current licensing analyses. Therefore, the benchmarking comparisons inherently include any impact of UO_2 conductivity degradation with exposure exhibited in the Peach Bottom tests.

The prior assessment was based on fuel designs current at the time of the Peach Bottom tests. To supplement the assessment with modern fuel, calculations were performed using the as-submitted AURORA-B code (Reference 27). AURORA-B is built from previous NRC approved methods. These methods include codes RODEX4, MICROBURN-B2, and S-RELAP5; UO_2 thermal conductivity degradation is correctly modeled. It is noted that the AURORA-B methodology and application have not yet been reviewed by the NRC; however, the staff accepted its use for sensitivity calculations for this assessment (Reference 25). The AURORA-B sensitivity studies show that the impact of fuel thermal conductivity degradation with exposure results in a decrease in the ΔCPR of **[ ]** increase in the transient LHGR excursion.

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.

The application of the methodology for EPU operation does not change the conservatisms nor invalidate the sensitivity; therefore, the AOO methodology remains applicable for Dresden and Quad Cities. It should be noted that transient LHGR analyses are performed with the RODEX4 code for Dresden and Quad Cities ATRIUM 10XM fuel, which correctly accounts for thermal conductivity degradation.

F.3.2 Loss of Coolant Accident Analyses

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. Clad ballooning has a small impact on Peak Cladding Temperature (PCT) and metal water reaction (MWR), but clad rupture can have a significant impact on PCT, depending on event timing.

The LOCA event is divided into two phases: the blowdown and refill/reflood phases. During the initial or blowdown portion of a LOCA, good cooling conditions exist, and the initial stored energy in the fuel is removed. While a decrease in the thermal conductivity increases the overall thermal resistance, heat transfer conditions remain sufficient to remove the initial stored energy. Numerous sensitivity studies have been performed to demonstrate that BWR LOCA analyses are insensitive to initial stored energy. After the initial phase of a LOCA, the heat transfer coefficient at the cladding surface is degraded due to the loss of coolant (low flow and high quality). As a result, the heat transfer from the fuel is primarily controlled by the surface heat flux, and the temperature profile across the pellet is very flat. When compared to the rod surface thermal resistance, the pellet thermal conductivity is not a significant portion of the fuel rod total thermal resistance. Therefore, LOCA calculations are not sensitive to the UO_2 thermal conductivity used in RELAX and HUXY.

To demonstrate that limiting LOCA calculations are not sensitive to UO_2 thermal conductivity, assessments were performed for multiple BWRs. Most LOCA analyses of record are limiting at beginning of life (BOL) conditions. Thermal conductivity degradation may impact calculated PCTs and oxidation as exposure increases; however, since the MAPLHGR limit decreases linearly at higher burnups, significant margin is gained that offsets any decrease in margin associated with thermal conductivity degradation. For these cases, increases in PCT at later exposures remained non-limiting.

Assessments of the potential impact of exposure-dependent degradation of UO_2 thermal conductivity on the fuel mechanical parameters were made using the RODEX4 computer code. The RODEX4 code explicitly incorporates the impact of UO_2 thermal conductivity degradation with exposure. RODEX4 calculations were performed with and without the models which account for TCD. The differences in these RODEX4 results were used to adjust the RODEX2 data which is input to HUXY. Evaluations were performed for plants with MAPLHGR limits that were constant from BOL to an exposure break point (generally 15 GWd/MTU planar average exposure) and then decreasing at higher exposure. PCT drops significantly after the exposure break point. At the highest exposure with the maximum MAPLHGR limit, accounting for TCD as described above resulted in PCT increases of 4°F and 18°F. The results of these evaluations were summarized to the NRC in References 24 and 26.

The impact of TCD was incorporated in the Dresden and Quad Cities ATRIUM 10XM HUXY analysis. An input option was added to RODEX4 to allow the analyst to turn off the models for thermal conductivity degradation with exposure. The impact of TCD was determined by running the nominal RODEX4 fuel rod depletions and then repeating them with the input option selected to turn off TCD. These RODEX4 results were used to increase the stored energy (average pellet temperature) calculated by RODEX2 prior to their input to HUXY. As shown below, this is a very conservative method for evaluating the impact of TCD since RODEX2 was developed to calculate conservatively high stored energy in support of its use as part of the AREVA Appendix K BWR LOCA methodology.

After the NRC approval of RODEX2, more Halden tests were performed with fuel centerline temperature monitoring. As with the RODEX4 submittal, this extended temperature database was used to benchmark RODEX2 over the approved burnup range. The extended temperature benchmarking for RODEX2 shows centerline temperature remains conservative to at least 10 GWd/MTU (Reference 24). Even so, the increase in stored energy predicted with the TCD models in RODEX4 was applied to the RODEX2 calculated stored energy for all nonzero exposures.

The ATRIUM 10XM PCT results with the impact of TCD will be presented in the MAPLHGR report that will be included in the Exelon Licensing Amendment Request to transition to AREVA fuel. Each cycle the MAPLHGR limit will be analyzed for any new neutronic lattice designs. The impact of TCD will be analyzed if warranted by the exposure dependent PCT results for the new lattice.

F.3.2.1 Responses to NRC Requests

From the NRC's review of Reference 24, additional information was requested in Reference 25. The information requests and responses are provided as follows:

A detailed explanation of the source of the heat transfer coefficients utilized in the HUXY calculation

This request is answered in Reference 26 and this answer is applicable to Dresden and Quad Cities.

A description of how LOCA analyses are initialized in terms of power distribution; specifically, how thermal limits (such as MLHGR or OLMCPR) are considered in the initialization

This request is answered in Reference 26. This response explains that AREVA's goal is to establish an MAPLHGR limit that is less restrictive than the LHGR limit and satisfies 10 CFR 50.46 acceptance criteria. Because it's a goal and not a requirement, MAPLHGR limits can be more restrictive than LHGR limits.

A characterization of the PCT sensitivity to fuel conductivity for plants where early boiling transition is predicted to occur during the early stages of LOCA

LOCA break spectrum analyses for ATRIUM 10XM fuel show boiling transition occurring after 5 seconds in the limiting two-loop operation analysis and as early as 1 second in the limiting single-loop analysis. These calculations used a conservatively high initial stored energy. The highest stored energy occurs at early exposure when the density of the pellet reaches a maximum. The maximum stored energy for Quad Cities ATRIUM 10XM fuel was calculated with RODEX2 at 2 GWd/MTU. The time of boiling transition was calculated based on a conservatively high stored energy.

The change in stored energy from UO_2 thermal conductivity degradation is of primary concern for the LOCA analyses; however, it is important to note that maximum stored energy would occur between 0 – 15 GWd/MTU. Within this range, maximum stored energy occurs from pellet densification when the gap between the cladding and pellet is at its maximum. This usually occurs before 5 GWd/MTU – an exposure region that is not an issue for conductivity degradation. At later exposures where conductivity degradation is significant, the reduction in power associated with the decreasing MAPLHGR limit would prevent this exposure region from being limiting in terms of stored energy. The RELAX system and RELAX hot channel analyses are performed with stored energy determined from the earlier exposure region. As noted in Reference 24, RODEX2 has an over-prediction of fuel centerline temperature to at least 10 GWd/MTU, therefore stored energy used in the RELAX analyses is conservative.

F.3.3 Overpressurization Analyses

The COTRANSA2 code is used to perform analyses to demonstrate that the reactor vessel pressure will not exceed the ASME vessel pressure limit during specified events. COTRANSA2 is also used to demonstrate that the vessel pressure does not exceed the overpressure acceptance criterion for an anticipated transient without scram (ATWS) event.

Analyses using COTRANSA2 are potentially affected by UO_2 thermal conductivity degradation with exposure, as described in Section F.3.1 for AOO analyses. As discussed in Reference 24, the impact on overpressurization analysis was assessed in two ways: using AURORA-B to assess the relative impact of using UO_2 thermal conductivity degradation with exposure; and decreasing the core average thermal conductivity input into COTRANSA2 to account for the effects of exposure. Reference 24 summarized the increase in pressure as less than a **[ ]** pressure rise (peak pressure – initial pressure) for the AURORA-B assessment and a pressure rise of **[ ]** for COTRANSA2 when the core average thermal conductivity assumed a 30% reduction. The Reference 24 evaluations concluded that the impact of UO_2 thermal conductivity degradation with exposure on the peak vessel pressure in overpressurization analyses was a small increase, the increase is less than the existing margins to the acceptance criteria.

The impact of TCD will be accounted for in ASME and ATWS overpressurization analyses performed for Dresden and Quad Cities by reducing the core average thermal conductivity in COTRANSA2 to account for the effects of exposure. The reduction will be calculated based on the exposure of the fuel in the core.

F.3.3.1 Responses to NRC Requests

From the NRC's review of Reference 24, additional information was requested in Reference 25. The requests and responses to the requests are provided as follows:

> *A comprehensive list of the identified nonconservative biases in the AREVA overpressure analysis methods*
>
> The comprehensive list of items was provided in Reference 26. The biases applicable for Dresden and Quad Cities are summarized as follows. These biases are addressed for each cycle to ensure that the pressure limits are not exceeded.
>
> <u>Void-Quality Correlation</u>: The bias is **[ ]** for ASME and **[ ]** for ATWS calculations.

Thermal Conductivity Degradation: In Reference 24 AREVA evaluated the impact of TCD for ATRIUM-10 fuel in two ways: using the AURORA-B code (Reference 27) to assess the relative impact of using UO_2 thermal conductivity with exposure degradation; and decreasing the core average thermal conductivity input into COTRANSA2 to account for the effects of exposure degradation. It was noted that changing the UO_2 thermal conductivity model provides a conservative estimate of the impact of exposure degradation on calculated peak vessel pressure. The limiting results obtained for the plants assessed in support of Reference 24 were reported as follows. For ASME, the increase in peak reactor pressure is expected to be less than **[ ]** of the pressure rise (peak pressure – initial pressure). For ATWS, the increase in pressure rise was **[ ]**.

Doppler Model Mismatch Between MICROBURN-B2 and COTRANSA2: The bias is **[ ]** of the calculated pressure rise from steady-state conditions for the ASME calculation and **[ ]** for the ATWS calculation.

Verification that the nonconservative biases are considered in an integral sense in the safety analyses.

Reference 26 demonstrated that it is conservative to add the biases together from separate effect assessments. The integral study demonstrated a decrease in total bias pressure.

F.3.4 Stability Analyses

As summarized in Reference 24, the computer codes STAIF and RAMONA5-FA are used in stability analyses. Both of these codes have fuel models that include UO_2 thermal conductivity degradation with exposure. Therefore, there is no impact on AREVA stability analyses.

Appendix G. Propagation of Errors Analysis for OPTIMA2 Quadrant Flow Uncertainty

The approved SAFLIM3D methodology described in Reference 1 addresses uncertainties in the assembly power/flow calculations. However, the existence of the water cross in the OPTIMA2 design introduces an additional source of uncertainty. The water cross divides the assembly into separate flow quadrants which introduces additional uncertainty into the SLMCPR calculation due to the quadrant power and flow calculations. The use of these additional uncertainties represents a plant specific extension of the approved Reference 1 methodology.

The quadrant flow uncertainty is determined by recognizing that the uncertainties in flow is determined by the uncertainties in the pressure drop prediction. The uncertainties in inlet pressure drop, two-phase pressure drop, assembly loss coefficients, and assembly flow area were provided for the OPTIMA2 fuel. A total flow uncertainty is calculated using a propagation of errors analysis.

ATTACHMENT 18

Mechanical Design Report
(Non-Proprietary)

Mechanical Design Report for Quad Cities and Dresden ATRIUM 10XM Fuel Assemblies

ANP-3305NP
Revision 0

Licensing Report

June 2014

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
AFC	Advanced fuel channel
AOO	Anticipated operational occurrences
ASME	American Society of Mechanical Engineers
B&PV	Boiler and pressure vessel
BWR	Boiling water reactor
CRDA	Control rod drop accident
EOL	End of life
LOCA	Loss-of-coolant accident
LTP	Lower tie plate
MWd/kgU	Megawatt-days per kilogram of Uranium
NRC	U. S. Nuclear Regulatory Commission
PLFR	Part-length fuel rods
psi	Pounds per square inch
S_m	Design stress intensity
SRA	Stress relief annealed
SRP	Standard review plan
S_u	Ultimate stress
S_y	Yield stress
UTP	Upper tie plate

1.0 INTRODUCTION

This report provides a design description, mechanical design criteria, fuel structural analysis results, and test results for the ATRIUM™ 10XM* fuel assembly and 100/75 Advanced Fuel Channel (AFC) designs supplied by AREVA Inc. (AREVA) for use at the Quad Cities and Dresden nuclear generating plants beginning with Quad Cities Unit 2 Cycle 24 and Dresden Unit 3 Cycle 25.

The scope of this report is limited to an evaluation of the structural design of the fuel assembly and fuel channel. The fuel assembly structural design evaluation is not cycle-specific so this report is intended to be referenced for each cycle where the fuel design is in use. Minor changes to the fuel design and cycle-specific input parameters will be dispositioned for future reloads. AREVA will confirm the continued applicability of this report prior to delivery of each subsequent reload of ATRIUM 10XM fuel at Quad Cities and Dresden in a cycle specific compliance document.

The fuel assembly design was evaluated according to the AREVA boiling water reactor (BWR) generic mechanical design criteria (Reference 1). The fuel channel design was evaluated to the criteria given in fuel channel topical report (Reference 2). The generic design criteria have been approved by the U.S. Nuclear Regulatory Commission (NRC) and the criteria are applicable to the subject fuel assembly and channel design.

Mechanical analyses have been performed using NRC-approved design analysis methodology (References 1, 2, 3 and 4). The methodology permits maximum licensed assembly and fuel channel exposures of 54 MWd/kgU (Reference 3).

* ATRIUM is a trademark of AREVA Inc.

2.0 DESIGN DESCRIPTION

This report documents the structural evaluation of the ATRIUM 10XM fuel assembly and fuel channel described below. Reload-specific design information is available in the design package provided by AREVA for each reload delivery.

2.1 *Overview*

This ATRIUM 10XM fuel bundle geometry consists of a 10x10 fuel lattice with a square internal water channel that displaces a 3x3 array of rods.

Table 2-1 lists the key design parameters of the ATRIUM 10XM fuel assembly.

2.2 *Fuel Assembly*

The ATRIUM 10XM fuel assembly consists of a lower tie plate (LTP) and upper tie plate (UTP), 91 fuel rods, 9 spacer grids, a central water channel [ ] and miscellaneous assembly hardware. Of the 91 fuel rods, 12 are PLFRs. The structural members of the fuel assembly include the tie plates, spacer grids, water channel, and connecting hardware. The structural connection between the LTP and UTP is provided by the central water channel. The lowest of the nine spacer grids is located just above the LTP to restrain the lower ends of the fuel rods.

The fuel assembly is accompanied by a fuel channel, as described later in this section.

Table 2-1 lists the main fuel assembly attributes, and an illustration of the fuel bundle assembly is provided in the appendix.

2.2.1 Spacer Grid

[

]

Table 2-1 lists the main spacer grid attributes, and an illustration of the spacer grid is provided in the appendix.

2.2.2 Water Channel

[

]

† ULTRAFLOW is a trademark of AREVA Inc.

[

]

Table 2-1 lists the main water channel attributes and the appendix provides an illustration of a section of the water channel.

2.2.3 Lower Tie Plate

[

]

The appendix provides an illustration of the LTP.

‡ FUELGUARD is a trademark of AREVA Inc.

2.2.4 Upper Tie Plate and Connecting Hardware

[

]

The appendix provides an illustration of the UTP and locking components.

2.2.5 Fuel Rods

[

]

[

]

Table 2-1 lists the main fuel rod attributes, and the appendix provides an illustration of the full length and part length fuel rods.

2.3 *Fuel Channel and Components*

[

] .

Table 2-2 lists the fuel channel component attributes. The fuel channel and fuel channel fastener are depicted in the appendix.

Table 2-1 Fuel Assembly and Component Description

Table 2-2 Fuel Channel and Fastener Description

[ ]

3.0 FUEL DESIGN EVALUATION

A summary of the mechanical methodology and results from the structural design evaluations is provided in this section. Results from the mechanical design evaluation demonstrate that the design satisfies the mechanical criteria to the analyzed exposure limit.

3.1 *Objectives*

The objectives of designing fuel assemblies (systems) to specific criteria are to provide assurance that:

- The fuel assembly (system) shall not fail as a result of normal operation and anticipated operational occurrences (AOOs). The fuel assembly (system) dimensions shall be designed to remain within operational tolerances, and the functional capabilities of the fuels shall be established to either meet or exceed those assumed in the safety analysis.
- Fuel assembly (system) damage shall never prevent control rod insertion when it is required.
- The number of fuel rod failures shall be conservatively estimated for postulated accidents.
- Fuel coolability shall always be maintained.
- The mechanical design of fuel assemblies shall be compatible with co-resident fuel and the reactor core internals.
- Fuel assemblies shall be designed to withstand the loads from handling and shipping.

The first four objectives are those cited in the Standard Review Plan (SRP). The latter two objectives are to assure the structural integrity of the fuel and the compatibility with the existing reload fuel. To satisfy these objectives, the criteria are applied to the fuel rod and the fuel assembly (system) designs. Specific component criteria are also necessary to assure compliance. The criteria established to meet these objectives include those given in Chapter 4.2 of the SRP.

3.2 *Fuel Rod Evaluation*

The mechanical design report documents the fuel structural analyses only. The fuel rod evaluation will be documented in Quad Cities and Dresden plant specific fuel rod thermal-mechanical report.

3.3 *Fuel System Evaluation*

The detailed fuel system design evaluation is performed to ensure the structural integrity of the design under normal operation, AOO, faulted conditions, handling operations, and shipping. The analysis methods are based on fundamental mechanical engineering techniques—often employing finite element analysis, prototype testing, and correlations based on in-reactor performance data. Summaries of the major assessment topics are described in the sections that follow.

3.3.1 Stress, Strain, or Loading Limits on Assembly Components

The structural integrity of the fuel assemblies is assured by setting design limits on stresses and deformations due to various handling, operational, and accident or faulted loads. AREVA uses Section Ill of the ASME B&PV Code as a guide to establish acceptable stress, deformation, and load limits for standard assembly components. These limits are applied to the design and evaluation of the UTP, LTP, spacer grids, springs, and load chain components, as applicable. 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.

All significant loads experienced during normal operation, AOOs, and under faulted conditions are evaluated to confirm the structural integrity of the fuel assembly components. Outside of faulted conditions, most structural components are under the most limiting loading conditions during fuel handling. See Section 3.3.9 for a discussion of fuel handling loads and Section 3.4.4 for the structural evaluation of faulted conditions. Although normal operation and AOO loads are often not limiting for structural components, a stress evaluation may be performed to confirm the design margin and to establish a baseline for adding accident loads. The stress calculations use conventional, open-literature equations. A general-purpose, finite element stress analysis code, such as ANSYS, may be used to calculate component stresses.

[

]

See Table 3-1 for results from the component strength evaluations.

3.3.2 Fatigue

Fatigue of structural components is generally [

] .

3.3.3 Fretting Wear

Fuel rod failures due to grid-to-rod fretting shall not occur. [ ].

Fretting wear is evaluated by testing, as described in Section 4.5. The testing is conducted by [ ]. The inspection measurements for wear are documented. The lack of significant wear demonstrates adequate rod restraint geometry at the contact locations. Also, the lack of significant wear at the spacer cell locations, relaxed to end of life (EOL) conditions, provides further assurance that no significant fretting will occur at higher exposure levels.

[ ] and has operated successfully without incidence of grid-to-rod fretting in more than 20,000 fuel assemblies.

3.3.4 Oxidation, Hydriding, and Crud Buildup

Because of the low amount of corrosion on fuel assembly structural components, [ ].

3.3.5 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 AREVA design criterion for fuel rod bowing is that [

]

Rod bow is calculated using the approved model described in Reference 4. The model has been shown to be conservative for application to the ATRIUM-10 fuel design. Less rod bow is predicted for the ATRIUM 10XM compared to the ATRIUM-10 due to a larger diameter fuel rod and a reduced distance between most spacer grids. [

] .

The predicted rod-to-rod gap closure due to bow is assessed for impact on thermal margins.

3.3.6 Axial Irradiation Growth

Fuel assembly components, including the fuel channel, shall maintain clearances and engagements, as appropriate, throughout the design life. Three specific growth calculations are considered for the ATRIUM 10XM design:

- Minimum fuel rod clearance between the LTP and UTP
- Minimum engagement of the fuel channel with the LTP seal spring
- External interfaces (e.g., channel fastener springs)

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.

The SRA fuel rod growth correlation was established from [ ]. Assembly growth is dictated by the water channel growth. The growth of the water channel and the fuel channel is based on [ ]. These data and the resulting growth correlations are described in Reference 3. The [ ], as appropriate, are used to obtain EOL growth values.

The minimum EOL rod growth clearance and EOL fuel channel engagement with the seal spring are listed in Table 3-1. The channel fastener spring axial compatibility is reported in Table 3-3.

3.3.7 Rod Internal Pressure

This will be addressed in the Quad Cities and Dresden fuel rod thermal-mechanical reports.

3.3.8 Assembly Lift-off

Fuel assembly lift-off is evaluated under both normal operating conditions (including AOOs) and under faulted conditions. The fuel shall not levitate under normal operating or AOO 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 net axial force acting on the fuel assembly is calculated by adding the loads from gravity, hydraulic resistance from coolant flow, difference in fluid flow entrance and exit momentum, and buoyancy. The calculated net force is confirmed to be in the downward direction, indicating no assembly lift-off. Maximum hot channel conditions are used in the calculation because the greater two-phase flow losses produce a higher uplift force.

Mixed core conditions for assembly lift-off are considered on a cycle-specific basis, as determined by the plant and other fuel types. Analyses to date indicate a large margin to assembly lift-off under normal operating conditions. Therefore, fuel lift-off in BWRs under normal operating conditions is considered to be a small concern.

For faulted conditions, [

]. The uplift is limited to be less than the axial engagement such that the fuel assembly neither becomes laterally displaced nor blocks insertion of the control blade.

3.3.9 Fuel Assembly Handling

The fuel assembly shall withstand, without permanent deformation, all normal axial loads from shipping and fuel handling operations. Analysis or testing shall show that the fuel is capable of [

].

The fuel assembly structural components are assessed for axial fuel handling loads by testing. To demonstrate compliance with the criteria, the test is performed by loading a test assembly to an axial tensile force greater than [ ]. An acceptable test shows no yielding after loading. The testing is described further in Section 4.1.

There are also handling requirements for the fuel rod plenum spring which are addressed in the Quad Cities and Dresden fuel rod thermal-mechanical reports.

3.3.10 Miscellaneous Component Criteria

3.3.10.1 Compression Spring Forces

The ATRIUM 10XM has a single large compression spring mounted on the central water channel. The compression spring serves the same function as previous designs by providing support for the UTP and fuel channel. The spring force is calculated based on the deflection and specified spring force requirements. Irradiation-induced relaxation is taken into account for EOL conditions. The minimum compression spring force at EOL is shown to be greater than the combined weight of the UTP and fuel channel (including channel fastener hardware). Since the compression spring does not interact with the fuel rods, no consideration is required for fuel rod buckling loads.

3.3.10.2 LTP Seal Spring

The LTP seal spring shall limit the bypass coolant leakage rate between the LTP and fuel channel. The seal spring shall accommodate expected channel deformation while remaining in contact with the fuel channel. Also, the seal spring shall have adequate corrosion resistance and be able to withstand the operating stresses without yielding.

Flow testing is used to confirm acceptable bypass flow characteristics. The seal spring is designed with adequate deflection to accommodate the maximum expected channel bulge while maintaining acceptable bypass flow. [ ] is selected as the material because of its high strength at elevated temperature and its excellent corrosion resistance. Seal spring stresses are analyzed using a finite element method.

3.4 *Fuel Coolability*

For accidents in which severe fuel damage might occur, core coolability and the capability to insert control blades are essential. Chapter 4.2 of the SRP provides several specific areas important to fuel coolability, as discussed below.

3.4.1 Cladding Embrittlement

The LOCA evaluation is addressed in the Quad Cities and Dresden LOCA MAPLHGR analysis for ATRIUM 10XM fuel reports.

3.4.2 Violent Expulsion of Fuel

Results for the CRDA analysis are presented in the Quad Cities and Dresden ATRIUM 10XM fuel transition report and the subsequent cycle-specific reload licensing reports.

3.4.3 Fuel Ballooning

The LOCA evaluation is addressed in the Quad Cities and Dresden LOCA MAPLHGR analysis for ATRIUM 10XM fuel reports.

3.4.4 Structural Deformations

Deformations 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 Section 4.2, Appendix A. The limits for each ATRIUM 10XM structural component are derived from analyses and/or component load tests.

Testing is performed to obtain the dynamic characteristics of the fuel assembly and spacer grids. The stiffness, natural frequencies and damping values derived from the tests are used as inputs for analytical models of the fuel assembly and fuel channel. [

]. See Section 1.0 for descriptions of the testing.

The methodology for analyzing the channeled fuel assembly under the influence of accident loads is described in Reference 2. 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 design was analyzed under a limiting top guide and core support time history supplied by Exelon for Quad Cities Units 1 and 2 and Dresden Units 2 and 3. [

]. Tables 3-1 and 3-2 list the minimum design margins for the fuel assembly structural components and fuel channel.

Assembly liftoff under accident conditions is described in Section 3.3.8.

3.4.4.1 Fuel Storage Seismic Qualification

The New and Spent Fuel Storage Racks analysis accounts for the fuel as added mass in calculating the structural integrity under postulated seismic loads. The weights of legacy fuel assembly designs at Quad Cities and Dresden encompass the weight of the ATRIUM 10XM fuel design. Therefore, the fuel storage racks remain qualified with the introduction of the ATRIUM 10XM fuel design.

3.5 *Fuel Channel and Fastener*

The fuel channel and fastener design criteria are summarized below, and evaluation results are summarized in Table 3-2 and Table 3-3. The analysis methods are described in detail in Reference 2.

3.5.1 Design Criteria for Normal Operation

Steady-State Stress Limits. The stress limits during normal operation are obtained from the ASME B&PV Code, Section III, Division 1, Subsection NG for Service Level A. The calculated stress intensities are due to the differential pressure across the channel wall. The pressure loading includes the normal operating pressure plus the increase during AOO. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation (Reference 8).

As an alternative to the elastic analysis stress intensity limits, a plastic analysis may be performed as permitted by paragraph NB-3228.3 of the ASME B&PV Code.

In the case of AOOs, the amount of bulging is limited to that value which will permit control blade movement. During normal operation, any significant permanent deformation due to yielding is precluded by restricting the maximum stresses at the inner and outer faces of the channel to be less than the yield strength.

Fuel Channel Fatigue. Cyclic changes in power and flow during operation impose a duty loading on the fuel channel. The cyclic duty from pressure fluctuations is limited to less than the fatigue lifetime of the fuel channel. The fatigue life is based on the O'Donnell and Langer curve (see Reference 5), which includes a factor of 2 on stress amplitude or a factor of 20 on the number of cycles, whichever is more conservative.

Corrosion and Hydrogen Concentration. Corrosion reduces the material thickness and results in less load-carrying capacity. The fuel channels have thicker walls than other components (e.g., fuel rods), and the normal amounts of oxidation and hydrogen pickup are not limiting provided: the alloy composition and impurity limits are carefully selected; the heat treatments are also carefully chosen; and the water chemistry is controlled. **[**

] .

Long-Term Creep Deformation. Changes to the geometry of the fuel channel occur due to creep deformation during the long term exposure in the reactor core environment. Overall deformation of the fuel channel occurs from a combination of bulging and bowing. Bulging of the side walls occurs because of the differential pressure across the wall. Lateral bowing of the channel is caused primarily from the neutron flux and thermal gradients. Too much deflection may prevent normal control blade maneuvers and it may increase control blade insertion time above the Technical Specification limits. The total channel deformation must not stop free movement of the control blade.

3.5.2 Design Criteria for Accident Conditions

Fuel Channel Stresses and Limit Load. The criteria are based on the ASME B&PV Code, Section III, Appendix F, for faulted conditions (Service Level D). Component support criteria for elastic system analysis are used as defined in paragraphs F-1332.1 and F-1332.2. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation.

Stresses are alternatively addressed by the plastic analysis collapse load criteria given in paragraph F-1332.2(b). For the plastic analysis collapse load, the permanent deformation is limited to twice the deformation the structure would undergo had the behavior been entirely elastic.

The amount of bulging remains limited to that value which will permit control blade insertion.

Fuel Channel Gusset Load Rating. [

] .

Table 3-1 Results for ATRIUM 10XM Fuel Assembly

Criteria Section	Description	Criteria	Results
3.3	**Fuel System Criteria**		
3.3.1	Stress, strain and loading limits on assembly components	The ASME B&PV Code Section III is used to establish acceptable stress levels or load limits for assembly structural components. The design limits for accident conditions are derived from Appendix F of Section III.	[ ]
	Water channel	[ ]	[ ]
3.3.2	Fatigue	[ ]	[ ]
3.3.3	Fretting wear	[ ]	[ ]
3.3.4	Oxidation, hydriding, and crud buildup	[ ]	[ ]
3.3.5	Rod bow	Protect thermal limits.	[ ]

(Table continued on next page)

Table 3-1 Results for ATRIUM 10XM Fuel Assembly *(Continued)*

Criteria Section	Description	Criteria	Results
3.3	**Fuel System Criteria** *(Continued)*		
3.3.6	Axial irradiation growth		
	Upper end cap clearance	Clearance always exists.	[ ]
	Seal spring engagement	Remains engaged with channel.	[ ]
3.3.7	Rod internal pressure	N/A	Not covered in structural report
3.3.8	Assembly liftoff		
	Normal operation (including AOOs)	No liftoff from fuel support.	[ ]
	Postulated accident	No disengagement from fuel support. No liftoff from fuel support.	[ ]
3.3.9	Fuel assembly handling	[ ]	[ ]
3.3.10	Miscellaneous components		
3.3.10.1	Compression spring forces	Support weight of UTP and fuel channel throughout design life.	The design criteria are met.
3.3.10.2	LTP seal spring	Accommodate fuel channel deformation, adequate corrosion, and withstand operating stresses.	The design criteria are met.

(Table continued on next page)

Table 3-1 Results for ATRIUM 10XM Fuel Assembly *(Continued)*

Criteria Section	Description	Criteria	Results
3.4	**Fuel Coolability**		
3.4.1	Cladding embrittlement	N/A	Not covered in structural report.
3.4.2	Violent expulsion of fuel	N/A	Not covered in structural report.
3.4.3	Fuel ballooning	N/A	Not covered in structural report.
3.4.4	Structural deformations	Maintain coolable geometry and ability to insert control blades. SRP 4.2, App. A, and ASME Section III, App. F.	See results below for individual components. [ ]
	Fuel rod stresses	[ ]	[ ]
	Spacer grid lateral load	[ ]	[ ]
	Water channel load	[ ]	[ ]
	UTP lateral load	[ ]	[ ]
	LTP lateral load	[ ]	[ ]

Table 3-2 Results for Advanced Fuel Channel

Criteria Section	Description	Criteria	Results
3.5.1	**Advanced Fuel Channel – Normal Operation**		
	Stress due to pressure differential	The pressure load including AOO is limited to [ ] according to ASME B&PV Code, Section III. The pressure load is also limited such that [ ]	The deformation during AOO remains within functional limits for normal control blade operation and the collapse load requirement is met with [ ] There is no significant plastic deformation during normal operation [ ]
	Fatigue	Cumulative cyclic loading to be less than the design cyclic fatigue life for Zircaloy. [ ]	Expected number of cycles [ ] is less than allowable.
	Oxidation and hydriding	Oxidation shall be accounted for in the stress and fatigue analyses.	The maximum expected oxidation is low in relation to the wall thickness. Oxidation was accounted in the stress and fatigue analyses.
	Long-term deformation (bulge creep and bow)	Bulge and bow shall not interfere with free movement of the control blade.	Margin to a stuck control blade remains positive.

(Table continued on next page)

Table 3-2 Results for Advanced Fuel Channel *(Continued)*

Criteria Section	Description	Criteria	Results
3.5.2	**Advanced Fuel Channel – Accident Conditions**		
	Fuel channel stresses and load limit	The pressure load is limited to [ ] according to ASME B&PV Code, Section III, App. F. The pressure load is also limited such that [ ]	The deformation during blowdown does not interfere with control blade insertion [ ]
	Channel bending from combined horizontal excitations	Allowable bending moment based on ASME Code, Section III, Appendix F plastic analysis collapse load.	[ ]
	Fuel channel gusset strength	ASME allowable load rating of one gusset is [ ]	[ ]

Table 3-3 Results for Channel Fastener

Criteria Section	Description	Criteria	Results
3.5	**Channel Fastener**		
	Compatibility	Spring height must extend to the middle of the control cell to ensure contact with adjacent spring.	All compatibility requirements are met. The spring will extend beyond the cell mid-line.
		Spring axial location must be sufficient to ensure alignment with adjacent spring at all exposures.	The axial location of the spring flat will always be in contact with an adjacent spring; even if a fresh ATRIUM 10XM is placed adjacent to an EOL co-resident assembly.
	Strength	Spring must meet ASME stress criteria and not yield beyond functional limit. Cap screw must meet ASME criteria for threaded fastener.	All ASME stress criteria are met for the spring and cap screw. In addition, the spring will not yield under the maximum deflection.

4.0 MECHANICAL TESTING

Prototype testing is an essential element of AREVA's methodology for demonstrating compliance with structural design requirements. Results from design verification testing may directly demonstrate compliance with criteria or may be used as input to design analyses. Test results and corresponding analyses confirm that the acceptance criteria are met.

Testing performed to qualify the mechanical design or evaluate assembly characteristics includes:

- Fuel assembly axial load structural strength test
- Spacer grid lateral impact strength test
- Tie plate lateral load strength tests and LTP axial compression test
- Debris filter efficiency test
- Fuel assembly fretting test
- Fuel assembly static lateral deflection test
- Fuel assembly lateral vibration tests
- Fuel assembly impact tests

Summary descriptions of the tests are provided below.

4.1 *Fuel Assembly Axial Load Test*

An axial load test was conducted by applying an axial tensile load between the LTP grid and UTP handle of a fuel assembly cage specimen. The load was slowly applied while monitoring the load and deflection. No significant permanent deformation was detected for loads in excess [

] .

4.2 *Spacer Grid Lateral Impact Strength Test*

Spacer grid impact strength was determined by a [

] .

The maximum force prior to the onset of buckling was determined from the testing. The results were adjusted to reactor operating temperature conditions to establish an allowable lateral load.

4.3 *Tie Plate Strength Tests*

In addition to the axial tensile tests described above, a lateral load test is performed on the UTP and LTP, and a compressive load test is performed on the LTP.

The UTP lateral load test was conducted on a test machine which applied an increasing load until a measurable amount of plastic deformation was detected. This provides a limiting lateral load for accident conditions. [

] .

For the Improved FUELGUARD LTP compressive load test, the tie plate was supported by the nozzle to simulate the fuel support conditions. A uniform, compressive axial load was applied to the grid. [

] .

To determine a limiting lateral load for accident conditions, the LTP lateral load test was conducted by attaching the grid of the tie plate to a rigid vertical plate and applying a side load to the cylindrical part of the nozzle. [

] .

4.4 *Debris Filter Efficiency Test*

Debris filtering tests were performed for the Improved FUELGUARD lower tie plate to evaluate its debris filtering efficiency. These tests evaluated the ability of the Improved FUELGUARD to protect the fuel rod array from a wide set of debris forms. In particular, testing was performed using small filamentary debris (e.g., wire brush debris) as this form is known to be a cause of debris fretting fuel failures. When testing the small filamentary debris forms, the debris filter is placed in a hydraulic test loop above a debris chamber. After insertion of a debris set in the debris chamber, the loop pump is started to circulate water in the loop for a given amount of time. Multiple pump shutdowns and restarts are then simulated. At the end of the test, the location of all debris is recorded and the filtering efficiency is determined. These debris filtering tests demonstrate that the Improved FUELGUARD is effective at protecting the fuel rod array from all high-risk debris forms.

4.5 *Fuel Assembly Fretting Test*

A fretting test was conducted on a full-size test assembly to evaluate the ATRIUM 10XM fuel rod support design. Spacer springs were relaxed in selected locations to simulate irradiation relaxation. [

] . After the test, the assembly was inspected for signs of fretting wear. No significant wear was found on fuel rods in contact with spacer springs relaxed to EOL conditions. The results agree with past test results on BWR designs where no noticeable wear was found on the fuel rods or other interfacing components following exposure to coolant flow conditions.

4.6 *Fuel Assembly Static Lateral Deflection Test*

A lateral deflection test was performed to determine the fuel assembly stiffness, both with and without the fuel channel. The stiffness is obtained by supporting the fuel assembly at the two ends in a vertical position, applying a side displacement at the central spacer location, and measuring the corresponding force.

4.7 *Fuel Assembly Lateral Vibration Tests*

The lateral vibration testing consists of both a free vibration test and a forced vibration test [

].

The test setup for the free vibration test is similar to the lateral deflection test described above. The fuel assembly is deflected to a specific displacement and then released. Displacement data are recorded at several spacer locations. The assembly natural frequencies and damping ratios are derived from the recorded motion. The test is repeated for several initial displacements.

The forced vibration testing [

].

4.8 *Fuel Assembly Impact Tests*

Impact testing was performed in a similar manner to the lateral deflection test. The unchanneled assembly is supported in a vertical position with both ends fixed. The assembly is displaced a specified amount and then released. A load cell is fixed to a rigid structure and located adjacent to a mid-assembly spacer. The fuel assembly impacts the load cell and the resulting impact force is recorded as a function of the initial displacement. The measured impact loads are used in establishing the spacer grid stiffness.

5.0 CONCLUSION

The fuel assembly and channel meet all the mechanical design requirements identified in References 1 and 2. Additionally, the fuel assembly and channel meet the mechanical compatibility requirements for use in Quad Cities and Dresden. This includes compatibility with both co-resident fuel and the reactor core internals.

6.0 REFERENCES

1. ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs,* Advanced Nuclear Fuels Corporation, May 1995.

2. EMF-93-177(P)(A) Revision 1*, Mechanical Design for BWR Fuel Channels,* Framatome ANP Inc., August 2005.

3. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model,* Siemens Power Corporation, February 1998.

4. XN-NF-75-32(P)(A) Supplements 1 through 4, *Computational Procedure for Evaluating Fuel Rod Bowing*, Exxon Nuclear Company, October 1983. (Base document not approved.)

5. W. J. O'Donnell and B. F. Langer, *Fatigue Design Basis for Zircaloy Components*, Nuclear Science and Engineering, Volume 20, January 1964.

6. XN-NF-81-51(P)(A), *LOCA - Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly*, Exxon Nuclear Company, May 1986.

7. XN-NF-84-97(P)(A), LOCA - Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly, Exxon Nuclear Company, August 1986.

8. Huan, P. Y., Mahmood, S. T., and R. Adamson, R. B. "Effects of Thermomechanical Processing on In-Reactor Corrosion and Post-Irradiation Properties of Zircaloy-2", *Zirconium in the Nuclear Industry: Eleventh International Symposium*, ASTM STP 1295, E. R. Bradley and G. P. Sabol, Eds., American Society for Testing and Materials, 1996, pp. 726-757.

APPENDIX A ILLUSTRATIONS

The following table lists the fuel assembly and fuel channel component illustrations in this section:

These illustrations are for descriptive purpose only. Please refer to the current reload design package for product dimensions and specifications.

Figure A-1 ATRIUM 10XM Fuel Assembly

(not to scale)

Figure A-2 UTP with Locking Hardware

Figure A-3 Improved FUELGUARD LTP

Figure A-4 ATRIUM 10XM ULTRAFLOW Spacer Grid

Figure A-5 Full and Part-Length Fuel Rods

(not to scale)

Figure A-6 Advanced Fuel Channel

Figure A-7 Fuel Channel Fastener Assembly

ATTACHMENT 19

Thermal-Hydraulic Design Report
(Non-Proprietary)

AREVA

ANP-3287NP
Revision 1

Quad Cities Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM™ 10XM Fuel Assemblies

November 2014

AREVA Inc.

ANP-3287NP
Revision 1

Quad Cities Units 1 and 2
Thermal-Hydraulic Design Report
for ATRIUM™ 10XM Fuel Assemblies

AREVA Inc.

ANP-3287NP
Revision 1

Nature of Changes

Item	Page	Description and Justification
1.	3-13	Adjust OPTIMA2 CPR value for the radial peaking equal to 1.5
2.	3-15	Adjust OPTIMA2 CPR values for the radial peaking equal to 1.5 in all core loadings as well as the radial peaking equal to 1.0 for the full core of OPTIMA2 loading.

Changed items are further identified
by yellow highlighting.

Contents

Tables

Figures

Nomenclature

AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BWR	boiling water reactor
CHF	critical heat flux
CPR	critical power ratio
CRDA	control rod drop accident
LOCA	loss-of-coolant accident
LTP	lower tie plate
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
NRC	Nuclear Regulatory Commission, U.S.
OD	outside diameter
PLFR	part-length fuel rod
RPF	radial peaking factor
UTP	upper tie plate

1.0 Introduction

The results of Quad Cities thermal-hydraulic analyses are presented to demonstrate that AREVA ATRIUM™ 10XM* fuel is hydraulically compatible with the previously loaded Westinghouse fuel, OPTIMA2. This report also provides the hydraulic characterization of the ATRIUM 10XM and the co-resident OPTIMA2 design for both Quad Cities Units.

The generic thermal-hydraulic design criteria applicable to the design have been reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) in the topical report ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1). In addition, thermal-hydraulic criteria applicable to the design have also been reviewed and approved by the NRC in the topical report XN-NF-80-19(P)(A) Volume 4 Revision 1 (Reference 2).

* ATRIUM is a trademark of AREVA Inc.

2.0 Summary and Conclusions

ATRIUM 10XM fuel assemblies have been determined to be hydraulically compatible with the co-resident OPTIMA2 fuel design in the Quad Cities reactors for the entire range of the licensed power-to-flow operating map. Detailed calculation results supporting this conclusion are provided in Section 3.2 and Table 3.4 to Table 3.8.

The ATRIUM 10XM fuel design is geometrically different from the co-resident OPTIMA2 design, but the designs are hydraulically compatible. **[**

]

Core bypass flow (defined as leakage flow through the lower tie plate (LTP) flow holes, channel seal, core support plate, and LTP-fuel support interface) is not adversely affected by the introduction of the ATRIUM 10XM fuel design. Analyses at rated conditions show core bypass flow varying between **[** **]** of rated flow for transition core configurations ranging from a full OPTIMA2 core, transition cores with both OPTIMA2 and ATRIUM 10XM fuel, and a full ATRIUM 10XM core.

Analyses demonstrate the thermal-hydraulic design and compatibility criteria discussed in Section 3.0 are satisfied for the Quad Cities transition cores consisting of ATRIUM 10XM and OPTIMA2 fuel for the expected core power distributions and core power/flow conditions encountered during operation. It is noted that some of the generic fuel rod mechanical issues are addressed in the fuel rod thermal and mechanical design reports (as specified in Table 3.1).

All analyses supporting this report were performed consistent with the safety evaluation report limitations and conditions associated with the NRC-approved methods.

3.0 Thermal-Hydraulic Design Evaluation

Thermal-hydraulic analyses are performed to verify that design criteria are satisfied and to help establish thermal operating limits with acceptable margins of safety during normal reactor operation and anticipated operational occurrences (AOOs). The design criteria that are applicable to the ATRIUM 10XM fuel design are described in Reference 1. To the extent possible, these analyses are performed on a generic fuel design basis. However, due to reactor and cycle operating differences, many of the analyses supporting these thermal-hydraulic operating limits are performed on a plant- and cycle-specific basis and are documented in plant- and cycle-specific reports. It is noted that some of the generic fuel rod mechanical issues are addressed in the mechanical and fuel rod thermal-mechanical design reports.

The thermal-hydraulic design criteria are summarized below:

- **Hydraulic compatibility.** 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 or the flow distribution among assemblies in the core. This criterion evaluation is addressed in Sections 3.1 and 3.2.
- **Thermal margin performance.** Fuel assembly geometry, including spacer design and rod-to-rod local power peaking, should minimize the likelihood of boiling transition during normal reactor operation as well as during AOOs. The fuel design should fall within the bounds of the applicable empirically based boiling transition correlation approved for AREVA reload fuel. Within other applicable mechanical, nuclear, and fuel performance constraints, the fuel design should achieve good thermal margin performance. The thermal-hydraulic design impact on steady-state thermal margin performance is addressed in Section 3.3. Additional thermal margin performance evaluations dependent on the cycle-specific design are addressed in the reload licensing report.
- **Fuel centerline temperature.** Fuel design and operation shall be such that fuel centerline melting is not projected for normal operation and AOOs. This criterion evaluation is addressed in the fuel rod thermal and mechanical evaluation report.
- **Rod bow.** The anticipated magnitude of fuel rod bowing under irradiation shall be accounted for in establishing thermal margin requirements. This criterion evaluation is addressed in Section 3.4.
- **Bypass flow.** The bypass flow characteristics of the reload fuel assemblies shall not differ significantly from the existing fuel in order to provide adequate flow in the bypass region. This criterion evaluation is addressed in Section 3.5.
- **Stability.** Reactors fueled with new fuel designs must be stable in the approved power and flow operating region. The stability performance of new fuel designs will be equivalent to, or better than, existing (approved) AREVA fuel designs. This criterion evaluation is addressed in Section 3.6. Additional core stability evaluations dependent on the cycle-specific design are addressed in the reload licensing report.

- **Loss-of-coolant accident (LOCA) analysis.** LOCAs are analyzed in accordance with Appendix K modeling requirements using NRC-approved models. The criteria are defined in 10 CFR 50.46. LOCA analysis results are presented in the break spectrum and MAPLHGR reports.
- **Control rod drop accident (CRDA) analysis.** The deposited enthalpy must be less than 280 cal/gm for fuel coolability. The fuel failure limit for deposited enthalpy is 170 cal/gm. The evaluation for these criteria is addressed in the reload licensing report.
- **ASME overpressurization analysis.** ASME pressure vessel code requirements must be satisfied. This criterion evaluation is addressed in the reload licensing report.
- **Seismic/LOCA liftoff.** Under accident conditions, the assembly must remain engaged in the fuel support. This criterion evaluation will be addressed in the mechanical design report.

A summary of the thermal-hydraulic design evaluations is given in Table 3.1.

3.1 *Hydraulic Characterization*

Basic geometric parameters for the ATRIUM 10XM and OPTIMA2 fuel designs are summarized in Table 3.2. Component loss coefficients for the ATRIUM 10XM are based on tests and are presented in Table 3.3. These loss coefficients include modifications to the test data reduction process **[**

] The bare rod friction, ULTRAFLOW™* spacer, UTP and LTP losses for ATRIUM 10XM are based on tests performed at AREVA's Portable Hydraulic Test Facility.
[
]

The primary resistance for the leakage flow through the ATRIUM 10XM LTP flow holes is
[

] The resistances for the leakage paths are shown in Table 3.3.

The local component (LTP, spacer, UTP, and LTP flow holes) loss coefficients for the OPTIMA2 fuel is based on pressure drop and flow distribution information provided by Westinghouse.

* ULTRAFLOW is a trademark of AREVA Inc.

3.2 *Hydraulic Compatibility*

The thermal-hydraulic analyses were performed in accordance with the AREVA thermal-hydraulic methodology for BWRs. The methodology and constitutive relationships used by AREVA for the calculation of pressure drop in BWR fuel assemblies are presented in Reference 3 and are implemented in the XCOBRA code. The XCOBRA code predicts steady-state thermal-hydraulic performance of the fuel assemblies of BWR cores at various operating conditions and power distributions. XCOBRA received NRC approval in Reference 4. The NRC reviewed the information provided in Reference 5 regarding inclusion of water rod models in XCOBRA and accepted the inclusion in Reference 6.

Hydraulic compatibility, as it relates to the relative performance of the ATRIUM 10XM and OPTIMA2 fuel designs, has been evaluated. Detailed Quad Cities analyses were performed for full core OPTIMA2 and full core ATRIUM 10XM configurations. Analyses for mixed cores with ATRIUM 10XM and OPTIMA2 fuel were also performed.

The hydraulic compatibility analysis is based on **[**

]

Table 3.4 summarizes the input conditions for the analyses. These conditions reflect two of the state points considered in the analyses: 100% power / 100% flow and 55% power / 38.5% flow. Table 3.4 also defines the core loading for the transition core configurations. Input for other core configurations is similar in that core operating conditions remain the same and the same axial power distribution is used. Evaluations were made with the bottom-, middle-, and top-peaked axial power distributions presented in Figure 3.1. Results presented in this report are for the middle-peaked power distribution. Results for bottom- and top-peaked axial power distributions show similar trends.

Table 3.5 and Table 3.6 provide a summary of calculated thermal-hydraulic results using the first transition core configuration. Table 3.7 and Table 3.8 provide a summary of results for all core configurations evaluated. Core average results and the differences between the

ATRIUM 10XM and OPTIMA2 results at rated power are within the range considered compatible. Similar agreement occurs at lower power levels. As shown in Table 3.5, **[**

]

Table 3.6 shows that **[**

]

Differences in assembly flow between the ATRIUM 10XM and OPTIMA2 fuel designs as a function of assembly power level are shown in Figure 3.2 and Figure 3.3.

Core pressure drop and core bypass flow fraction are also provided for the configurations evaluated. Based on the reported changes in pressure drop and assembly flow caused by the transition from a full core of OPTIMA2 fuel to a full core of ATRIUM 10XM, the ATRIUM 10XM design is considered hydraulically compatible with the co-resident fuel design based on the analysis results.

3.3 *Thermal Margin Performance*

Relative thermal margin analyses were performed in accordance with the thermal-hydraulic methodology for AREVA's XCOBRA code. The calculation of the fuel assembly critical power ratio (CPR) (thermal margin performance) is established by means of an empirical correlation based on results of boiling transition test programs. The CPR methodology is the approach used by AREVA to determine the margin to thermal limits for BWRs.

CPR values for ATRIUM 10XM are calculated with the ACE/ATRIUM 10XM critical power correlation (Reference 7) while the CPR values for the OPTIMA2 fuel are calculated with the SPCB critical power correlation (Reference 8) **[**

]. The NRC-approved methodology to demonstrate the acceptability of using the SPCB correlation for computing OPTIMA2 fuel CPR is presented in Reference 9 **[**

]. Assembly design features are incorporated in the CPR calculation through the K-factor term for the ACE correlation and the F-eff term for the SPCB correlation. The K-factors and F-effs are made up of two parts which are added together. The first part depends on the local power peaking in the fuel assembly, which depends on the nuclear design and is a function of void fraction and exposure. The second part is called an additive constant, which is determined for each rod position based on critical power testing and calculated using the methods approved in References 7 and 8.

For the compatibility evaluation, steady-state analyses evaluated ATRIUM 10XM and OPTIMA2 assemblies with radial peaking factors (RPFs) between **[**

]

Table 3.5 and Table 3.6 show CPR results of the ATRIUM 10XM and OPTIMA2 fuel. Table 3.7 and Table 3.8 show similar comparisons of CPR and assembly flow for the various core configurations evaluated. Analysis results indicate ATRIUM 10XM fuel will not cause thermal margin problems for the co-resident fuel designs.

3.4 *Rod Bow*

The bases for rod bow are discussed in the mechanical design report. Rod bow magnitude is determined during the fuel-specific mechanical design analyses. Rod bow has been measured during post-irradiation examinations of BWR fuel fabricated by AREVA.

[

]

3.5 *Bypass Flow*

Total core bypass flow is defined as leakage flow through the LTP flow holes, channel seal, core support plate, and LTP-fuel support interface. Table 3.7 shows that total core bypass flow (excluding water rod flow) fraction at rated conditions changes from **[** **]** of rated core flow during the transition from a full OPTIMA2 core loading to a full ATRIUM 10XM core (middle-peaked power shape).

In summary, adequate bypass flow will be available with the introduction of the ATRIUM 10XM fuel design.

3.6 *Stability*

Each new fuel design is analyzed to demonstrate that the stability performance is equivalent to or better than an existing (NRC-approved) AREVA fuel design. The stability performance is a function of the core power, core flow, core power distribution, and to a lesser extent, the fuel design. **[**

] A comparative stability analysis was performed with the NRC-approved STAIF code (Reference 10). The study shows that the ATRIUM 10XM fuel design has decay ratios equivalent to or better than other approved AREVA fuel designs.

[

]

As stated above, the stability performance of a core is strongly dependent on the core power, core flow, and power distribution in the core. Therefore, core stability is evaluated on a cycle-specific basis and addressed in the reload licensing report. The cycle-specific stability evaluations are performed consistent with the BWR Owner's Group Option III solution.

Table 3.1 Design Evaluation of Thermal and Hydraulic Criteria for the ATRIUM 10XM Fuel Assembly

Report Section	Description	Criteria	Results or Disposition
	Thermal and Hydraulic Criteria		
3.1 / 3.2	Hydraulic compatibility	Hydraulic flow resistance shall be sufficiently similar to existing fuel such that there is no significant impact on total core flow or flow distribution among assemblies.	Verified on a plant-specific basis. ATRIUM 10XM evaluated to be compatible with OPTIMA2 fuel. [ ]
3.3	Thermal margin performance	Fuel design shall be within the limits of applicability of an approved CHF correlation.	ACE/ATRIUM 10XM critical power correlation is applied to the ATRIUM 10XM fuel. SPCB critical power correlation is applied to the OPTIMA2 fuel.
		< 0.1% of rods in boiling transition.	Verified on cycle-specific basis for Chapter 15 analyses.
	Fuel centerline temperature	No centerline melting.	Plant- and fuel-specific analyses are performed.
3.4	Rod bow	Rod bow must be accounted for in establishing thermal margins.	The lateral displacement of the fuel rods due to fuel rod bowing is not of sufficient magnitude to impact thermal margins.
3.5	Bypass flow	Bypass flow characteristics shall be similar among assemblies to provide adequate bypass flow.	Verified on a plant-specific basis. Analysis results demonstrate that adequate bypass flow is provided and no boiling occurs in the bypass region.

**Table 3.1 Design Evaluation of
Thermal and Hydraulic Criteria for the
ATRIUM 10XM Fuel Assembly** *(Continued)*

Report Section	Description	Criteria	Results or Disposition
	Thermal and Hydraulic Criteria *(Continued)*		
3.6	Stability	New fuel designs are stable in the approved power and flow operating region, and stability performance will be equivalent to (or better than) existing (approved) AREVA fuel designs. [ ]	ATRIUM 10XM channel and core decay ratios have been demonstrated to be equivalent to or better than other approved AREVA fuel designs. Core stability behavior is evaluated on a cycle-specific basis. [ ]
	LOCA analysis	LOCA analyzed in accordance with Appendix K modeling requirements. Criteria defined in 10 CFR 50.46.	Approved Appendix K LOCA model. Plant- and fuel-specific analysis with cycle-specific verifications.
	CRDA analysis	< 280 cal/gm for coolability. 170 cal/gm fuel failure limit	Cycle-specific analysis is performed.
	ASME over-pressurization analysis	ASME pressure vessel core requirements shall be satisfied.	Cycle-specific analysis is performed.
	Seismic/LOCA liftoff	Assembly remains engaged in fuel support.	Criterion will be discussed in the mechanical design report.

**Table 3.2 Comparative Description for
Quad Cities Units 1 and 2
ATRIUM 10XM and OPTIMA2 Fuel Types**

Fuel Parameter	ATRIUM 10XM	OPTIMA2
Number of fuel rods		
Full-length fuel rods	79	84
PLFRs	12	8 (2/3 length) 4 (1/3 length)
Fuel clad OD, in	0.4047	0.3874
Number of spacers	9	8
Active fuel length, in		
Full-length fuel rods	145.24	145.28
PLFRs	75	99.6 (2/3 length) 50.4 (1/3 length)
Hydraulic resistance characteristics	Table 3.3	Table 3.3
Number of water rods	1	1*
Total Water rod flow area, in^2	1.7228	1.677*

* The OPTIMA2 fuel is modeled with 1 water rod. The water flow area includes the flow area of the OPTIMA2 water wings.

Table 3.3 Hydraulic Characterization Comparison for Quad Cities Units 1 and 2 ATRIUM 10XM and OPTIMA2 Fuel

[

]

[

]

**Table 3.4 Quad Cities Units 1 and 2
Thermal-Hydraulic Design Conditions**

Reactor Conditions	100%P / 100%F	55%P / 38.5%F
Core power level, MWt	2957.0	1626.3
Core exit pressure, psia	1015.0	927.05
Core inlet enthalpy, Btu/lbm	521.2	493.0
Total core coolant flow, Mlbm/hr	98.0	37.7
Axial power shape	Middle-peaked (Figure 3.1)	Middle-peaked (Figure 3.1)

	Number of Assemblies	
	Central Region	Peripheral Region
Full Core OPTIMA2 Core Loading		
[		]
First Transition Core Loading		
[		]
[		]
Second Transition Core Loading		
[		]
[		]
Full Core ATRIUM 10XM Core Loading		
[		]

Table 3.5 Quad Cities Units 1 and 2
First Transition Core Thermal-Hydraulic Results at
Rated Conditions (100%P / 100%F)

[

]

[

]

**Table 3.6 Quad Cities Units 1 and 2
First Transition Core Thermal-Hydraulic Results at
Off-Rated Conditions (55%P / 38.5%F)**

[

]

[

]

Table 3.7 Quad Cities Units 1 and 2 Thermal-Hydraulic Results at Rated Conditions (100%P / 100%F) for Transition to ATRIUM 10XM Fuel

[

]

Table 3.8 Quad Cities Units 1 and 2 Thermal-Hydraulic Results at Off-Rated Conditions (55.0%P / 38.5%F) for Transition to ATRIUM 10XM Fuel

[

]

[

]

Figure 3.1 Axial Power Shapes

[

]

Figure 3.2 First Transition Core:
Hydraulic Demand Curves
100%P / 100%F

[

]

Figure 3.3 First Transition Core:
Hydraulic Demand Curves
55%P / 38.5%F

4.0 References

1. ANF-89-98(P)(A) Revision 1 and Supplement 1*, Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995.

2. XN-NF-80-19(P)(A) Volume 4 Revision 1, *Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads*, Exxon Nuclear Company, June 1986.

3. XN-NF-79-59(P)(A), *Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies*, Exxon Nuclear Company, November 1983.

4. 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, January 1987.

5. Letter, R.A. Copeland (ANF) to R.C. Jones (USNRC), "Explicit Modeling of BWR Water Rod in XCOBRA," RAC:002:90, January 9, 1990.

6. Letter, R.C. Jones (USNRC) to R.A. Copeland (ANF), no subject (regarding XCOBRA water rod model), February 1, 1990.

7. ANP-10298P-A Revision 1, *ACE/ATRIUM 10XM Critical Power Correlation*, AREVA, March 2014.

8. EMF-2209(P)(A) Revision 3, *SPCB Critical Power Correlation*, AREVA NP, September 2009.

9. EMF-2245(P)(A) Revision 0, *Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel*, Siemens Power Corporation, August 2000.

10. EMF-CC-074(P)(A) Volume 1, *STAIF – A Computer Program for BWR Stability Analysis in the Frequency Domain;* and Volume 2, *STAIF – A Computer Program for BWR Stability Analysis in the Frequency Domain – Code Qualification Report*, Siemens Power Corporation, July 1994.

11. ANP-3338P Revision 0, *Applicability of AREVA BWR Methods to the Dresden and Quad Cities Reactors Operating at Extended Power Uprate*, AREVA, November 2014.

ATTACHMENT 20

Fuel Rod Thermal-Mechanical Design Report
(Non-Proprietary)

ATRIUM 10XM Fuel Rod Thermal-Mechanical Design for Quad Cities Unit 2 Cycle 24 Representative Fuel Cycle Design

ANP-3324NP
Revision 0

Licensing Report

October 2014

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
AOO	anticipated operational occurrences
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BOL	beginning of life
BWR	boiling water reactor
CRWE	control rod withdrawal error
CUF	cumulative usage factor
EOL	end of life
FDL	fuel design limit
ID	inside diameter
MWd/kgU	megawatt days per kilogram of initial uranium
LHGR	linear heat generation rate
NRC	Nuclear Regulatory Commission, U. S.
OD	outside diameter
OOS	Equipment Out Of Service
PCI	pellet-to-cladding-interaction
PLFR	part length fuel rod
ppm	parts per million
S-N	stress amplitude versus number of cycles
SRS	Single Rod Sequence

1.0 INTRODUCTION

Results of the fuel rod thermal-mechanical analyses are presented to demonstrate that the applicable design criteria are satisfied. The analyses are for the AREVA Inc. (AREVA) ATRIUM™ 10XM* fuel that will be inserted for operation in Quad Cities Unit 2 Cycle 24 as reload batch QCI2-24. The evaluations are based on methodologies and design criteria approved by the U. S. Nuclear Regulatory Commission (NRC). Equilibrium cycle conditions as well as Cycle 24 conditions are included in the analyses.

The analysis results are evaluated according to the generic fuel rod thermal and mechanical design criteria contained in ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1) along with design criteria provided in the RODEX4 fuel rod thermal-mechanical topical report (Reference 2). The cladding external oxidation limit defined by Reference 2 is 130 μm, however the reduced value of 85 μm was used to match a regulatory commitment made to the NRC when RODEX4 was first implemented in the U.S.

The RODEX4 fuel rod thermal-mechanical analysis code is used to analyze the fuel rod for fuel centerline temperature, cladding strain, rod internal pressure, cladding collapse, cladding fatigue and external oxidation. The code and application methodology are described in the RODEX4 topical report (Reference 2). The cladding steady-state stress and plenum spring design methodology are summarized in Reference 1.

The fuel rod design is very similar to the ATRIUM 10XM design currently supplied in reload quantities to two U.S. BWR/4 units except the fuel column length is shorter by 4.76 inches for compatibility with a BWR/3 core height. The ATRIUM 10XM fuel rod design is based on the ATRIUM-10 design in a way that preserves the nearly 20 years of extensive operating experience and performance history of the ATRIUM-10 rod design.

The following sections describe the fuel rod design, design criteria and methodology with reference to the source topical reports. Results from the analyses are summarized for comparison to the design criteria.

---

* ATRIUM is a trademark of AREVA Inc.

2.0 SUMMARY AND CONCLUSIONS

Key results are shown in Table 2-1 in comparison to each of the design criterion. Results are presented for the limiting cases. Additional RODEX4 results from different cases are given in Section 3.0.

The analysis methodology supports a maximum fuel rod discharge exposure of 62 MWd/kgU.

Fuel rod criteria applicable to the design are summarized in Section 3.0. Analyses show the criteria are satisfied when the fuel is operated at or below the LHGR (linear heat generation rate) limit presented in Figure 2-1.

Table 2-1 Summary of Fuel Rod Design Evaluation Results

Criteria Section*	Description	Criteria	Result, Margin[†] or Comment
3.2	**Fuel Rod Criteria**		
3.2.1	Internal hydriding	[	]
(3.1.1)	Cladding collapse	[	]
(3.1.2)	Overheating of fuel pellets	No fuel melting margin to fuel melt > 0. °C	[ ]
3.2.5	Stress and strain limits		
(3.1.1) (3.1.2)	Pellet-cladding interaction	[	]
3.2.5.2	Cladding stress	[	]
3.3	**Fuel System Criteria**		
(3.1.1)	Fatigue	[	]
(3.1.1)[‡]	Oxidation, hydriding, and crud buildup	[	]
(3.1.1) (3.1.2)	Rod internal pressure	[	]
3.3.9	Fuel rod plenum spring (fuel handling)	Plenum spring to [	]

* Numbers in the column refer to paragraph sections in the generic design criteria document, ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1). A number in parentheses is the paragraph section in the RODEX4 fuel rod topical report (Reference 2).

† Margin is expressed as (limit – result)

‡ The cladding external oxidation limit is restricted to the reduced value of [ ] µm.

§ All values except the oxidation margin are taken from calculations which conservatively assume that the oxidation limit of [ ] µm will be reached at EOL. The oxidation margin is taken from standard runs and represents the actual expected margin for EOL oxidation.

[

]

Figure 2-1 LHGR Limit (Normal Operation)

3.0 FUEL ROD DESIGN EVALUATION

Summaries of the design criteria and methodology are provided in this section along with analysis results in comparison to criteria. Both the fuel rod criteria and fuel system criteria as directly related to the fuel rod analyses are covered.

The fuel rod analyses cover normal operating conditions and AOOs (anticipated operational occurrences). The fuel centerline temperature analysis (overheating of fuel) and cladding strain analysis take into account slow transients at rated operating conditions.

Other fuel rod related topics on overheating of cladding, cladding rupture, fuel rod mechanical fracturing, rod bow, axial irradiation growth, cladding embrittlement, violent expulsion of fuel and fuel ballooning are evaluated as part of the respective fuel assembly structural analysis, thermal hydraulic analyses, or LOCA analyses and are reported elsewhere. The evaluation of fast transients and transients at off-rated conditions also are reported separate from this report.

3.1 *Fuel Rod Design*

[

] plenum spring on the upper end of

the fuel column [

].

Table 3-1 lists the main parameters for the fuel rod and components.

3.2 *Summary of Fuel Rod Design Evaluation*

Results from the analyses are listed in Table 3-2 through Table 3-4. Summaries of the methods and codes used in the evaluation are provided in the following paragraphs. The design criteria also are listed along with references to the sections of the design criteria topical reports (References 1 and 2).

The fuel rod thermal and mechanical design criteria are summarized as follows.

- **Internal Hydriding**. The fabrication limit [ ] to preclude cladding failure caused by internal sources of hydrogen (Section 3.2.1 of Reference 1).

- **Cladding Collapse**. Clad creep collapse shall be prevented. [ ] (Section 3.1.1 of Reference 2).

- **Overheating of Fuel Pellets**. The fuel pellet centerline temperature during anticipated transients shall remain below the melting temperature (Section 3.1.2 of Reference 2).

- **Stress and Strain Limits**. [ ] during normal operation and during anticipated transients (Sections 3.1.1 and 3.1.2 of Reference 2).

  Fuel rod cladding steady-state stresses are restricted to satisfy limits derived from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code (Section 3.2.5.2 of Reference 1).

- **Cladding Fatigue**. The fatigue cumulative usage factor for clad stresses during normal operation and design cyclic maneuvers shall be below [ ] (Section 3.1.1 of Reference 2).

- **Cladding Oxidation, Hydriding and Crud Buildup**. Section 3.1.1 of Reference 2 limits the maximum cladding oxidation to less than [ ] µm to prevent clad corrosion failure. The oxidation limit is further reduced to [ ] µm consistent with a regulatory commitment made to the NRC during the first application of the RODEX4 methodology in the U.S.

- **Rod Internal Pressure**. The rod internal pressure is limited [ ] to assure that significant outward clad creep does not occur and unfavorable hydride reorientation on cooldown does not occur (Section 3.1.1 of Reference 2).

- **Plenum Spring Design (Fuel Handling)**. The rod plenum spring must maintain a force against the fuel column stack [ ] (Section 3.3.9 of Reference 1).

The cladding collapse, overheating of fuel, cladding transient strain, cladding cyclic fatigue, cladding oxidation, and rod pressure are evaluated [ ]. Cladding stress and the plenum spring are evaluated on a design basis.

3.2.1 Internal Hydriding

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and formation of hydride platelets. Careful moisture control during fuel fabrication reduces the potential for hydrogen absorption on the inside of the cladding. The fabrication limit [ ] is verified by quality control inspection during fuel manufacturing.

3.2.2 Cladding Collapse

Creep collapse of the cladding and the subsequent potential for fuel failure is avoided in the design by limiting the gap formation due to fuel densification subsequent to pellet-clad contact. The size of the axial gaps which may form due to densification following first pellet-clad contact shall be less than [ ].

The evaluation is performed using RODEX4. The design criterion and methodology are described in Reference 2. RODEX4 takes into account the [ ]. A brief overview of RODEX4 and the statistical methodology is provided in the next section.

Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.3 Overheating of Fuel Pellets

Fuel failure from the overheating of the fuel pellets is not allowed. The centerline temperature of the fuel pellets must remain below melting during normal operation and AOOs. The melting point of the fuel includes adjustments for gadolinia content. AREVA establishes an LHGR limit to protect against fuel centerline melting during steady-state operation and during AOOs.

Fuel centerline temperature is evaluated using the RODEX4 code (Reference 2) for both normal operating conditions and AOOs. A brief overview of the code and methodology follow.

RODEX4 evaluates the thermal-mechanical responses of the fuel rod surrounded by coolant. The fuel rod model considers the fuel column, gap region, cladding, gas plena and the fill gas and released fission gases. The fuel rod is divided into axial and radial regions with conditions computed for each region. The operational conditions are controlled by the [ ]

[

].

The heat conduction in the fuel and clad is [

].

Mechanical processes include [

].

As part of the methodology, fuel rod power histories are generated [

]

[

].

Since RODEX4 is a best-estimate code, uncertainties [

]. Uncertainties taken into account in the analysis are summarized as:

- Power measurement and operational uncertainties – [

]
- Manufacturing uncertainties – [ ].
- Model uncertainties – [

]

[

].

Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.4 Stress and Strain Limits

3.2.4.1 Pellet/Cladding Interaction

Cladding strain caused by transient-induced deformations of the cladding is calculated using the RODEX4 code and methodology as described in Reference 2. See Section 3.2.3 for an overview of the code and method. [

].

Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.4.2 Cladding Stress

Cladding stresses are calculated using solid mechanics elasticity solutions and finite element methods. The stresses are conservatively calculated for the individual loadings and are categorized as follows:

Category	Membrane	Bending
Primary	[	]
Secondary	[	]

Stresses are calculated at the cladding outer and inner diameter in the three principal directions for both beginning of life (BOL) and end of life (EOL) conditions. At EOL, the stresses due to mechanical bow and contact stress are decreased due to irradiation relaxation. The separate stress components are then combined, and the stress intensities for each category are compared to their respective limits.

The cladding-to-end cap weld stresses are evaluated for loadings from differential pressure, differential thermal expansion, rod weight, and plenum spring force.

The design limits are derived from the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel (B&PV) Code Section III (Reference 3) and the minimum specified material properties.

Table 3-4 lists the results in comparison to the limits for hot, cold, BOL and EOL conditions.

3.2.5 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. Although there are no explicit criteria for fuel densification and swelling, the effect of these phenomena are included in the RODEX4 fuel rod performance code.

3.2.6 Fatigue

[

]. The CUF (cumulative usage factor) is summed for all of the axial regions of the fuel rod using Miner's rule. The axial region with the highest CUF is used in the subsequent [

] is determined. The maximum CUF for the cladding must remain below [ ] to satisfy the design criterion.

Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.7 Oxidation, Hydriding, and Crud Buildup

Cladding external oxidation is calculated using RODEX4. Section 3.2.3 includes an overview of the code and method. The corrosion model 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 model uncertainty is included as part of the [

].

In the event abnormal crud is observed for a plant, a specific analysis is required to address the higher crud level. An abnormal level of crud is defined by a formation that increases the calculated fuel average temperature by 25°C above the design basis calculation. The formation of crud is not calculated within RODEX4. Instead, an upper bound of expected crud is input by the use of the crud heat transfer coefficient. The corrosion model also takes into consideration the effect of the higher thermal resistance from the crud on the corrosion rate. A higher corrosion rate is therefore included as part of the abnormal crud evaluation. A similar specific analysis is required if a plant experiences higher corrosion instead of crud.

The maximum calculated oxide on the fuel rod cladding shall not exceed [ ] µm. Previously, a [ ] µm limit was approved as part of the RODEX4 methodology (Reference 2). Concerns were raised on the effect of non-uniform corrosion, such as spallation, and localized hydride formations on the ductility limit of the cladding. As a result, a regulatory commitment was made for the first U.S. application to reduce the limit to [ ] µm. While not formally required, this reduced value has been conservatively adopted for Quad Cities analyses.

The current measurements of crud at Quad Cities Unit 2 indicate normal low crud levels. However, in order to address the potential impact of changing water chemistry conditions, the input parameters have been conservatively selected in order to generate corrosion in excess of the current operating experience at Quad Cities. The values given in this report are the results from these conservative calculations, and as shown, meet all design criteria.

Currently, there is [ ]. However, as mentioned above, the [ ] µm was established, in part, as a means of [ ].

The oxide limit is evaluated such that greater than [ ].

Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.8 Rod Internal Pressure

Fuel rod internal pressure is calculated using the RODEX4 code and methodology as described in Reference 2. Section 3.2.3 provides an overview of the code and methodology. The maximum rod pressure is calculated under steady-state conditions and also takes into account slow transients. Rod internal pressure is limited to [ ]. The expected upper bound of rod pressure [ ] is calculated for comparison to the limit.

Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.9 Plenum Spring Design (Fuel Assembly Handling)

The plenum spring must maintain a force against the fuel column to [ ]. This is accomplished by designing and verifying the spring force in relation to the fuel column weight. The plenum spring is designed such that the [ ].

Table 3-1 Key Fuel Rod Design Parameters

[

]

Table 3-2 RODEX4 Fuel Rod Results for Equilibrium Cycle Conditions

Criteria Topic	Limit	Margin* to Limit		
		Steady-State	[ ]	[ ]
[				]
[				]
[				]
[		]		
[		]		
[		]		

* Margin is defined as (limit – result).

† [

]

Table 3-3 RODEX4 Fuel Rod Results for Quad Cities Unit 2 Cycle 24*

Criteria Topic	Limit	Margin[†] to Limit		
		Steady-State	[ ][‡]	[ ]
[				]
[				]
[				]
[		]		
[		]		
[		]		

* Note that cycle-specific results are provided up to the end of cycle.

† Margin is defined as (limit – result).

‡ Note that Quad Cities Unit 2 intends to operate cycle 24 with a single rod sequence (SRS) that limits the number of control rods inserted in the core. For this cycle none of the ATRIUM-10XM fuel assemblies reside in a SRS control cell. As such, control rod withdrawal error transients are not part of the scope of this calculation.

Table 3-4 Cladding and Cladding-End Cap Steady-State Stresses

Description, Stress Category	Criteria‡	Result		
Cladding stress		[		]
P_m (primary membrane stress)	[			]
$P_m + P_b$ (primary membrane + bending)	[			]
P + Q (primary + secondary)	[			]
Cladding-End Cap stress				
$P_m + P_b$	[			]

‡ In all analyzed conditions, the value of S_u is less than twice the value of S_y. As such, the S_u criteria are more limiting than the S_y criteria and only the stress result ratios to S_u have been reported.

4.0 REFERENCES

1. ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995.

2. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, AREVA NP Inc., February 2008.

3. *ASME Boiler and Pressure Vessel Code*, Section III, “Rules for Construction of Nuclear Power Plant Components,” 1983.

4. O’Donnell, W.J., and B. F. Langer, “Fatigue Design Basis for Zircaloy Components,” *Nuclear Science and Engineering*, Vol. 20, 1964.

ATTACHMENT 21

Fuel Cycle Design Report
(Non-Proprietary)

Quad Cities Unit 2 Cycle 24 Representative Fuel Cycle Design

ANP-3293NP
Revision 0

October 2014

AREVA Inc.

Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial release.

Contents

Tables

Figures

Nomenclature

ACE	AREVA's advanced critical power correlation [ ]
BOC	beginning of cycle
BOL	beginning of life
BWR	boiling water reactor
CSDM	cold shutdown margin
EOC	end of cycle
EOFP	end of full power capability
FFTR	final feedwater temperature reduction
GWd/MTU	gigawatt days per metric ton of initial uranium
HEXR	hot excess reactivity
ICF	increased core flow
LAR	license amendment request
LHGR	linear heat generation rate
MCPR	minimum critical power ratio
MICROBURN-B2	AREVA Inc. advanced BWR core simulator methodology with PPR capability
MWd/MTU	megawatt days per metric ton of initial uranium
NRC	Nuclear Regulatory Commission, U. S.
PLFR	part-length fuel rod
PPR	Pin Power Reconstruction. The PPR methodology accounts for variation in local rod power distributions due to neighboring assemblies and control state. The local rod power distributions are reconstructed based on the actual flux solution for each statepoint.
R Value	the larger of zero or the shutdown margin at BOC minus the minimum calculated shutdown margin in the cycle
SLC	standby liquid control
SPCB	AREVA (formerly Siemens Power Corporation) critical power correlation

1.0 Introduction

AREVA Inc. (AREVA) has performed a representative fuel cycle design and fuel management calculations based on expected Cycle 24 operation of the Quad Cities Unit 2 BWR as defined in References 3 and 4. The core design presented within is the basis for the safety analyses supporting the License Amendment Request (LAR) for the inclusion of AREVA's methodology in the Tech Specs for both Dresden and Quad Cities. This methodology addition supports the introduction of the ATRIUM™* 10XM fuel design.

These analyses have been performed with the approved AREVA neutronics methodology (Reference 1). The CASMO-4 lattice depletion code was used to generate nuclear data including cross sections and local power peaking factors. The MICROBURN-B2 three dimensional core simulator code, combined with the application of the applicable critical power correlations, was used to model the core. The MICROBURN-B2 pin power reconstruction (PPR) model was used to determine the thermal margins presented in this report. The ACE critical power correlation (Reference 5) was utilized for the ATRIUM 10XM fuel assemblies while the co-resident Westinghouse OPTIMA-2 fuel assemblies will utilize the SPCB critical power correlation (Reference 6) with appropriate additive constants developed consistent with the NRC approved co-resident fuel topical report (Reference 7). The following MICROBURN-B2 modeling features are included in this analysis:

- Version 2 of MICROBURN-B2
- Explicit control blade modeling
- Control blade ^{10}B depletion
- Explicit neutronic treatment of the spacer grids
- Explicit thermal-hydraulic modeling of the water rod flow
- Explicit modeling of the plenum/spring region above the PLFRs

Design results for the representative Cycle 24 reactor core loading including projected control rod patterns and evaluations of thermal and reactivity margins are presented. The representative Cycle 24 results are based on Cycle 22 and 23 core operational history and/or projections as summarized in Table 2.1.

---

* ATRIUM is a trademark of AREVA Inc.

2.0 Summary

The representative Cycle 24 fresh batch size (248 assemblies) and batch average enrichment **[ ]** were determined to meet the energy requirements provided by Exelon (Reference 4). For a complete description of the fresh reload assemblies, see Reference 2. The loading of the representative Cycle 24 fuel as described in this report results in a projected full power energy capability (including ICF out to 102% of rated) of 1,948±38 GWd (15,485±300 MWd/MTU). Beyond the full power capability, the cycle has been designed to achieve 110 GWd of additional energy via Constant Pressure Power Coastdown operation.

In order to obtain optimum operating flexibility, the projected control rod patterns were developed to be consistent with a conservative margin to thermal limits. The cycle design calculations also demonstrate adequate hot excess reactivity and cold shutdown margin throughout the cycle. Key results from the design analysis are summarized in Table 2.1. Table 2.2 summarizes the assembly identification range by nuclear fuel type batch for the Cycle 24 design. Figure 2.1 and Figure 2.2 provide a summary of the cycle design step-through projection.

Table 2.1 Quad Cities Unit 2 Representative Cycle 24 Energy and Key Results Summary

Cycle Energy, GWd (Cycle Exposure, MWd/MTU)	
Cycle 22	
• Core follow through October 8, 2013	1,570 (12,590)
• Best estimate depletion to Nominal EOC 22	2,082 (16,690)
Cycle 23	
• Best estimate depletion to Nominal EOC 23	2,013 (16,132)
• Short window EOC 23	1,959 (15,700)
Cycle 24	
• EOFP Energy	1,948±38 (15,485±300)
• Constant Pressure Power Coastdown Energy	110 (878)
• EOC Energy	2,058±38 (16,362±300)

Key Results	
BOC CSDM, %Δk/k (based on short EOC 23)	1.60
Minimum CSDM, %Δk/k (based on short EOC 23)	1.60
Cycle Exposure of Minimum CSDM, MWd/MTU (short basis)	0
Moderator Temperature of Minimum CSDM, °F (short basis)	68
Cycle R Value, %Δk/k (short basis)	0.00
Minimum SLC SDM, %Δk/k (based on short EOC 23)	6.54
Cycle Exposure of Minimum SLC SDM, MWd/MTU (short basis)	0
BOC HEXR, %Δk/k (based on nominal EOC 23)	1.23
Maximum HEXR, %Δk/k (based on nominal EOC 23)	1.39
Cycle Exposure of Maximum HEXR, MWd/MTU (nominal basis)	11,950.0
Minimum MAPLHGR Margin, %	13.0
Exposure of Minimum MAPLHGR Margin, MWd/MTU	9,351.0
Minimum LHGR Margin, %	14.1
Exposure of Minimum LHGR Margin, MWd/MTU	250.0
Minimum CPR Margin, %	13.6
Exposure of Minimum CPR Margin, MWd/MTU	15,250.0
Minimum Pellet Exposure Ratio Margin, %	3.31
Minimum Rod (including PLFRs) Average Exposure Ratio Margin, %	3.11
Minimum Assembly Average Exposure Ratio Margin, %	11.75
EOC Core Average Exposure (CAVEX), MWd/MTU	37,064.4

Table 2.2 Quad Cities Unit 2 Representative Cycle 24 Fuel Cycle Design Assembly ID Range by Nuclear Fuel Type

Nuclear Fuel Type	Number of Assemblies	Assembly ID Range
19	84	QBD001-QBD088
20	8	QBD089-QBD096
21	96	QBD097-QBD200
22	48	QBD201-QBD248
23	104	QBE001-QBE104
24	80	QBE105-QBE184
25	56	QBE185-QBE240
26	96	XRA001-XRA096
27	64	XRA097-XRA160
28	88	XRA161-XRA248

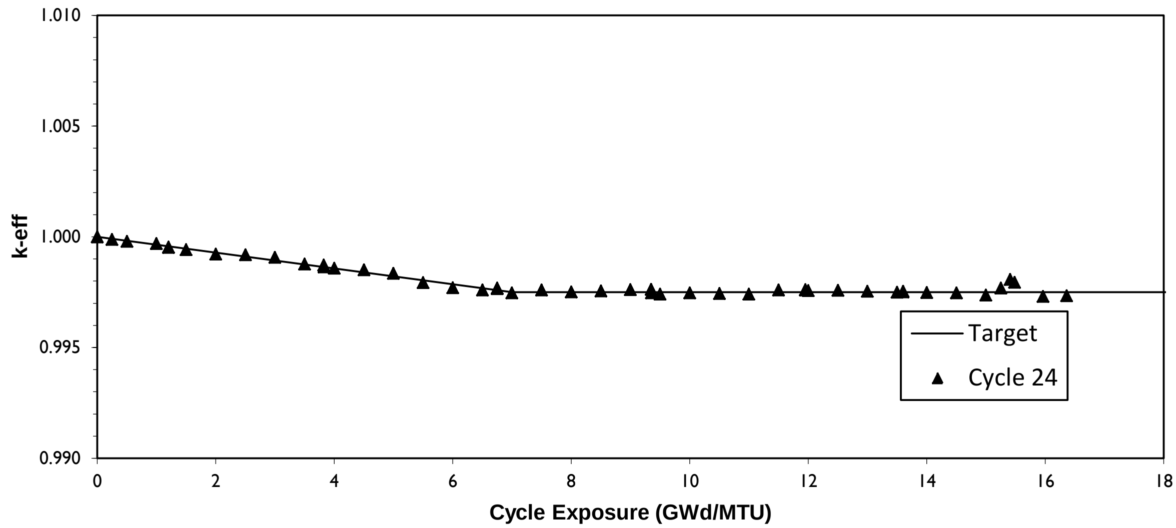


**Figure 2.1 Quad Cities Unit 2 Representative Cycle 24 Design
Step-Through k-eff versus Cycle Exposure**

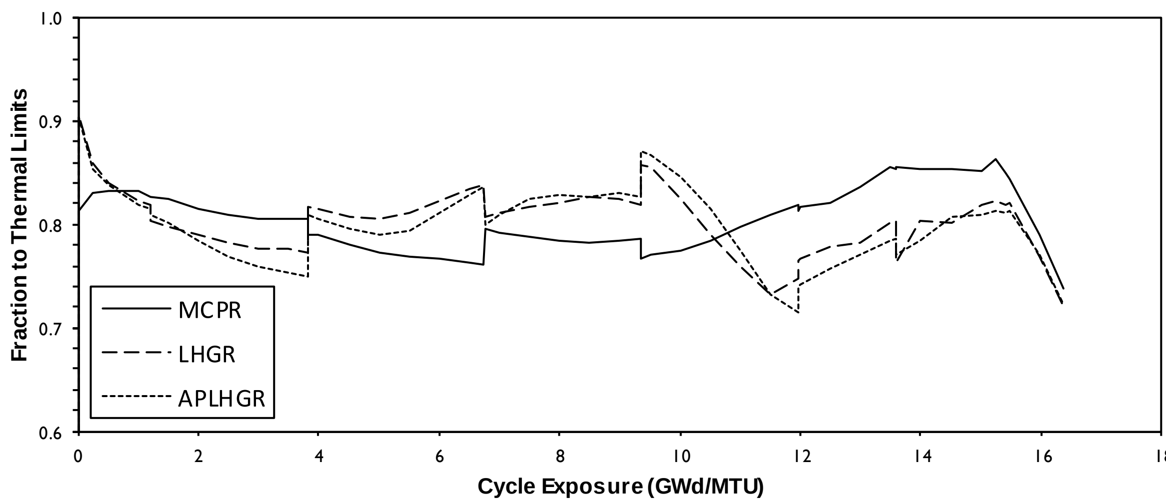


**Figure 2.2 Quad Cities Unit 2 Representative Cycle 24 Design
Margin to Thermal Limits versus Cycle Exposure**

3.0 Representative Cycle 24 Fuel Cycle Design

3.1 *General Description*

The assembly design for the representative Quad Cities Unit 2 Cycle 24 fresh reload fuel is described in detail in Reference 2. Elevation views of the fresh reload fuel design axial enrichment and gadolinia distributions are shown in Appendix B, Figures B.1 through B.3. The loading pattern maintains octant symmetry (with the exception of a limited number of locations near the core periphery to maintain exposure limit margin and SIL320 compliance) within a Single Rod Sequence (SRS) fuel management scheme. This loading in conjunction with the control rod patterns presented in Appendix A shows acceptable power peaking and associated margins to limits for projected operation. The analyses supporting this fuel cycle design were based on the core parameters shown in Table 3.1. Figure 3.1 through Figure 3.5, along with Table 3.1 define the reference loading pattern used in the fuel cycle design. The specific core location of the fresh assemblies is provided in Appendix C. Key results for the cycle are summarized in Table 2.1.

3.2 *Control Rod Patterns and Thermal Limits*

Projected control rod patterns and resultant key operating parameters including thermal margins are shown in Appendix A. The thermal margins presented in this report were determined using the MICROBURN-B2 3D core simulator PPR model to provide adequate margin to the thermal limits. The limits used to calculate thermal margins are representative and based on best-estimates from the thermal-mechanical, LOCA and safety analyses. A detailed summary of the core parameters resulting from the step-through projection analysis is provided in Tables A.1 and A.2. Limiting results from the step-through are summarized in Table 2.1 and in Figure 2.2. The hot operating target k-eff versus cycle exposure which was determined to be appropriate for the representative Cycle 24 is shown in Table 3.2 (Reference 8). The k-eff and margin to limits results from the design cycle depletion are presented graphically in Figure 2.1 and Figure 2.2. The k-eff values presented in Figure 2.1 and in Appendix A are not bias corrected. Selected exposure and radial power distributions from the design step-through are presented in Appendix D.

3.3 *Hot Excess Reactivity and Cold Shutdown Margin*

The cycle design calculations demonstrate adequate hot excess reactivity, SLC shutdown margin, and cold shutdown margin throughout the cycle. Key shutdown margin and R-Value results are presented in Table 2.1. The shutdown margin is in conformance with the Technical Specification limit of 0.38 + R %Δk/k at BOC. The cold target k-eff versus exposure determined to be appropriate for calculation of cold shutdown margin is shown in Table 3.3 (Reference 8). Calculations have been performed for a range of increased temperatures from 100 °F to 220 °F (in addition to 68 °F) to verify the most limiting shutdown margin was calculated. The core hot excess reactivity was calculated at full power with all rods out, 98.0 Mlb/hr core flow, with equilibrium xenon. Table 3.4 summarizes reactivity margins versus cycle exposure, including the SLC shutdown margin for the representative cycle.

Table 3.1 Representative Cycle 24 Core Composition and Design Parameters

Fuel Description	Cycle Loaded	Nuclear Fuel Type	Number of Assemblies
OPTIMA-2 **[ ]**	22	19	84
OPTIMA-2 **[ ]**	22	20	8
OPTIMA-2 **[ ]**	22	21	96
OPTIMA-2 **[ ]**	22	22	48
OPTIMA-2 **[ ]**	23	23	104
OPTIMA-2 **[ ]**	23	24	80
OPTIMA-2 **[ ]**	23	25	56
ATRIUM 10XM **[ ]**	24	26	96
ATRIUM 10XM **[ ]**	24	27	64
ATRIUM 10XM **[ ]**	24	28	88

Number of Fuel Assemblies in Core	724
Total Number of Fresh Assemblies	248
Total Core Mass, MTU	125.81
Rated Thermal Power Level, MW_t	2,957
Rated Core Flow, Mlb/hr	98.0
Reference Pressure, psia	1,015*
Reference Inlet Subcooling, Btu/lbm	24.21†

* Value is representative of MICROBURN-B2 input for dome pressure at rated conditions and varies depending on core state point.

† Value shown is from the 100% Power, 100% Flow condition and is determined by MICROBURN-B2 using a heat balance method based on nominal feedwater temperature and other parameters identified in the cycle specific plant parameters document.

Table 3.2 Hot Operating Target k-eff versus Cycle Exposure

Cycle Exposure (MWd/MTU)	Hot Operating k-eff*
0.0	1.0000
7,000.0	0.9975
20,000.0	0.9975

Table 3.3 Cold Critical Target k-eff versus Cycle Exposure

Cycle Exposure (MWd/MTU)	Cold Critical k-eff*
0.0	0.9970
6,000.0	0.9950
20,000.0	0.9950

* Values are linearly interpolated between cycle exposure points.

Table 3.4 Quad Cities Unit 2 Representative Cycle 24 Reactivity Margin Summary

Cycle Exposure (MWd/MTU)	Cold Shutdown Margin* (% Δk/k)	SLC Cold Shutdown Margin† (% Δk/k)	Hot Excess Reactivity‡ (% Δk/k)
0	1.60	6.54	1.23
250	1.66	6.83	1.14
1,200	1.93	7.16	1.05
2,500	2.49	7.42	1.08
3,825	2.93	7.66	1.07
5,000	3.26	7.81	1.08
6,000	3.34	7.97	1.12
7,000	3.32	8.05	1.17
8,000	3.29	8.16	1.19
9,000	3.22	8.22	1.24
10,000	3.13	8.26	1.28
11,000	2.98	8.28	1.34
11,950	**2.72**	8.31	1.39
13,000	**2.39**	8.38	1.31
14,000	**2.24**	8.72	0.98
15,000	**2.27**	9.33	0.37
15,405	**2.38**	9.65	0.06
15,485	**2.40**	9.71	--
16,362	**2.74**	10.51	--

NOTE: Values shown for cold shutdown margin are shown in **BOLD** if the most reactive temperature is greater than 68 °F.

* Based on short window EOC 23.

† Based on short window EOC 23, calculated at 358.3 °F ARO conditions which correspond to the saturation temperature for the RHR suction low pressure permissive (Reference 9).

‡ Based on nominal EOC 23.

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29
60											QBD017	QBD009	QBD049	QBD025	QBD153
58										QBD089	QBD033	QBD137	QBD081	QBD059	QBD073
56							QBD161	QBD185	QBD065	QBD121	QBD001	QBD233	QBD217	QBD097	QBD209
54						QBD177	QBD057	QBD105	QBE193	QBD145	QBE187	XRA218	XRA210	XRA194	XRA186
52					QBD041	QBD129	QBD113	QBD227	QBE219	XRA242	QBE233	XRA202	QBE097	XRA178	QBE201
50				QBD179	QBD131	QBD193	QBD201	QBE225	XRA234	XRA154	XRA146	XRA122	XRA170	XRA082	XRA074
48			QBD163	QBD195	QBD011	QBD203	QBD243	XRA226	QBE209	XRA138	QBE017	QBE057	XRA114	QBE033	QBE025
46			QBD187	QBD107	QBD225	QBE227	XRA225	QBE139	XRA130	XRA090	QBE065	QBE041	XRA106	XRA050	QBE081
44			QBD067	QBE195	QBE217	XRA233	QBE211	XRA129	QBE123	XRA162	QBE049	XRA066	QBE113	XRA042	QBE089
42		QBD091	QBD123	QBD147	XRA241	XRA153	XRA137	XRA089	XRA161	QBE121	XRA058	QBE129	XRA034	QBE105	XRA026
40	QBD019	QBD035	QBD003	QBE185	QBE235	XRA145	QBE019	QBE067	QBE051	XRA057	QBE153	QBE169	QBE009	XRA018	QBE073
38	QBD099	QBD139	QBD115	XRA217	XRA201	XRA121	QBE059	QBE043	XRA065	QBE131	QBE171	QBE155	XRA098	QBE001	QBE161
36	QBD051	QBD083	QBD235	XRA209	QBE099	XRA169	XRA113	XRA105	QBE115	XRA033	QBE011	XRA097	QBE137	XRA010	QBE177
34	QBD027	QBD241	QBD219	XRA193	XRA177	XRA081	QBE035	XRA049	XRA041	QBE107	XRA017	QBE003	XRA009	QBE147	XRA002
32	QBD155	QBD075	QBD211	XRA185	QBE203	XRA073	QBE027	QBE083	QBE091	XRA025	QBE075	QBE163	QBE179	XRA001	QBE145
30	QBD157	QBD077	QBD213	XRA192	QBE205	XRA080	QBE029	QBE085	QBE093	XRA032	QBE077	QBE165	QBE181	XRA008	QBE151
28	QBD029	QBD247	QBD221	XRA200	XRA184	XRA088	QBE037	XRA056	XRA048	QBE109	XRA024	QBE005	XRA016	QBE149	XRA007
26	QBD053	QBD085	QBD101	XRA216	QBE101	XRA176	XRA120	XRA112	QBE117	XRA040	QBE013	XRA104	QBE143	XRA015	QBE183
24	QBD013	QBD141	QBD117	XRA224	XRA208	XRA128	QBE061	QBE045	XRA072	QBE133	QBE173	QBE157	XRA103	QBE007	QBE167
22	QBD021	QBD037	QBD005	QBE191	QBE237	XRA152	QBE021	QBE069	QBE053	XRA064	QBE159	QBE175	QBE015	XRA023	QBE079
20		QBD093	QBD125	QBD149	XRA248	XRA160	XRA144	XRA096	XRA168	QBE127	XRA063	QBE135	XRA039	QBE111	XRA031
18			QBD069	QBE197	QBE223	XRA240	QBE213	XRA136	QBE125	XRA167	QBE055	XRA071	QBE119	XRA047	QBE095
16			QBD189	QBD109	QBD231	QBE229	XRA232	QBE141	XRA135	XRA095	QBE071	QBE047	XRA111	XRA055	QBE087
14			QBD165	QBD197	QBD015	QBD205	QBD245	XRA231	QBE215	XRA143	QBE023	QBE063	XRA119	QBE039	QBE031
12				QBD181	QBD133	QBD199	QBD207	QBE231	XRA239	XRA159	XRA151	XRA127	XRA175	XRA087	XRA079
10					QBD047	QBD135	QBD239	QBD229	QBE221	XRA247	QBE239	XRA207	QBE103	XRA183	QBE207
8						QBD183	QBD063	QBD111	QBE199	QBD151	QBE189	XRA223	XRA215	XRA199	XRA191
6							QBD167	QBD191	QBD071	QBD127	QBD007	QBD119	QBD237	QBD223	QBD215
4										QBD095	QBD039	QBD143	QBD087	QBD061	QBD079
2											QBD023	QBD103	QBD055	QBD031	QBD159

Figure 3.1 Quad Cities Unit 2 Representative Cycle 24 Reference Loading Pattern

	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
60	QBD154	QBD026	QBD050	QBD010	QBD018										
58	QBD074	QBD060	QBD082	QBD138	QBD034	QBD090									
56	QBD210	QBD098	QBD218	QBD114	QBD002	QBD122	QBD066	QBD186	QBD162						
54	XRA187	XRA195	XRA211	XRA219	QBE188	QBD146	QBE194	QBD106	QBD058	QBD178					
52	QBE202	XRA179	QBE098	XRA203	QBE234	XRA243	QBE220	QBD228	QBD234	QBD130	QBD042				
50	XRA075	XRA083	XRA171	XRA123	XRA147	XRA155	XRA235	QBE226	QBD202	QBD194	QBD132	QBD180			
48	QBE026	QBE034	XRA115	QBE058	QBE018	XRA139	QBE210	XRA227	QBD244	QBD204	QBD116	QBD196	QBD164		
46	QBE082	XRA051	XRA107	QBE042	QBE066	XRA091	XRA131	QBE140	XRA228	QBE228	QBD226	QBD108	QBD188		
44	QBE090	XRA043	QBE114	XRA067	QBE050	XRA163	QBE124	XRA132	QBE212	XRA236	QBE218	QBE196	QBD068		
42	XRA027	QBE106	XRA035	QBE130	XRA059	QBE122	XRA164	XRA092	XRA140	XRA156	XRA244	QBD148	QBD124	QBD092	
40	QBE074	XRA019	QBE010	QBE170	QBE154	XRA060	QBE052	QBE068	QBE020	XRA148	QBE236	QBE186	QBD004	QBD036	QBD020
38	QBE162	QBE002	XRA099	QBE156	QBE172	QBE132	XRA068	QBE044	QBE060	XRA124	XRA204	XRA220	QBD236	QBD140	QBD012
36	QBE178	XRA011	QBE138	XRA100	QBE012	XRA036	QBE116	XRA108	XRA116	XRA172	QBE100	XRA212	QBD220	QBD084	QBD052
34	XRA003	QBE148	XRA012	QBE004	XRA020	QBE108	XRA044	XRA052	QBE036	XRA084	XRA180	XRA196	QBD100	QBD242	QBD028
32	QBE146	XRA004	QBE180	QBE164	QBE076	XRA028	QBE092	QBE084	QBE028	XRA076	QBE204	XRA188	QBD212	QBD076	QBD156
30	QBE152	XRA005	QBE182	QBE166	QBE078	XRA029	QBE094	QBE086	QBE030	XRA077	QBE206	XRA189	QBD214	QBD078	QBD158
28	XRA006	QBE150	XRA013	QBE006	XRA021	QBE110	XRA045	XRA053	QBE038	XRA085	XRA181	XRA197	QBD222	QBD248	QBD030
26	QBE184	XRA014	QBE144	XRA101	QBE014	XRA037	QBE118	XRA109	XRA117	XRA173	QBE102	XRA213	QBD102	QBD086	QBD054
24	QBE168	QBE008	XRA102	QBE158	QBE174	QBE134	XRA069	QBE046	QBE062	XRA125	XRA205	XRA221	QBD120	QBD142	QBD014
22	QBE080	XRA022	QBE016	QBE176	QBE160	XRA061	QBE054	QBE070	QBE022	XRA149	QBE238	QBE192	QBD006	QBD038	QBD022
20	XRA030	QBE112	XRA038	QBE136	XRA062	QBE128	XRA165	XRA093	XRA141	XRA157	XRA245	QBD150	QBD126	QBD094	
18	QBE096	XRA046	QBE120	XRA070	QBE056	XRA166	QBE126	XRA133	QBE214	XRA237	QBE224	QBE198	QBD070		
16	QBE088	XRA054	XRA110	QBE048	QBE072	XRA094	XRA134	QBE142	XRA229	QBE230	QBD232	QBD110	QBD190		
14	QBE032	QBE040	XRA118	QBE064	QBE024	XRA142	QBE216	XRA230	QBD246	QBD206	QBD118	QBD198	QBD166		
12	XRA078	XRA086	XRA174	XRA126	XRA150	XRA158	XRA238	QBE232	QBD208	QBD200	QBD134	QBD182			
10	QBE208	XRA182	QBE104	XRA206	QBE240	XRA246	QBE222	QBD230	QBD016	QBD136	QBD048				
8	XRA190	XRA198	XRA214	XRA222	QBE190	QBD152	QBE200	QBD112	QBD064	QBD184					
6	QBD216	QBD224	QBD238	QBD240	QBD008	QBD128	QBD072	QBD192	QBD168						
4	QBD080	QBD062	QBD088	QBD144	QBD040	QBD096									
2	QBD160	QBD032	QBD056	QBD104	QBD024										

Figure 3.1 Quad Cities Unit 2 Representative Cycle 24 Reference Loading Pattern *(Continued)*

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29
60											19 44.5	19 41.8	19 43.5	19 43.4	21 43.7
58	Nuclear Fuel Type BOC Exposure (GWd/MTU)									20 41.6	19 43.2	21 42.8	19 42.8	19 41.7	19 43.8
56							21 44.9	21 44.2	19 44.1	21 42.1	19 41.8	22 41.8	22 40.8	21 40.7	22 40.1
54						21 44.6	19 43.4	21 42.6	25 19.5	21 39.2	25 18.6	28 0.0	28 0.0	28 0.0	28 0.0
52					19 44.6	21 43.6	21 41.6	22 36.1	25 19.5	28 0.0	25 20.3	28 0.0	23 22.2	28 0.0	25 19.9
50				21 44.5	21 43.6	21 43.1	22 36.6	25 17.5	28 0.0	27 0.0	27 0.0	27 0.0	28 0.0	26 0.0	26 0.0
48			21 44.9	21 43.1	19 41.7	22 36.5	22 35.7	28 0.0	25 20.0	27 0.0	23 22.4	23 23.6	27 0.0	23 22.9	23 22.9
46			21 44.1	21 42.5	22 36.2	25 17.5	28 0.0	24 20.9	27 0.0	26 0.0	23 23.0	23 23.2	27 0.0	26 0.0	23 22.9
44			19 44.1	25 19.5	25 19.5	28 0.0	25 20.0	27 0.0	24 21.6	28 0.0	23 22.6	26 0.0	24 21.3	26 0.0	23 22.0
42		20 34.3	21 42.0	21 39.1	28 0.0	27 0.0	27 0.0	26 0.0	28 0.0	24 21.6	26 0.0	24 22.5	26 0.0	24 20.0	26 0.0
40	19 44.4	19 43.1	19 41.8	25 18.6	25 20.3	27 0.0	23 22.5	23 23.0	23 22.6	26 0.0	24 23.3	24 23.6	23 21.9	26 0.0	23 22.9
38	21 40.6	21 42.7	21 41.5	28 0.0	28 0.0	27 0.0	23 23.6	23 23.2	26 0.0	24 22.6	24 23.6	24 23.4	27 0.0	23 20.8	24 23.5
36	19 43.7	19 42.7	22 39.8	28 0.0	23 22.2	28 0.0	27 0.0	27 0.0	24 21.3	26 0.0	23 22.0	27 0.0	24 21.0	26 0.0	24 22.9
34	19 43.3	22 42.8	22 40.7	28 0.0	28 0.0	26 0.0	23 23.0	26 0.0	26 0.0	24 20.0	26 0.0	23 20.8	26 0.0	24 22.1	26 0.0
32	21 43.7	19 43.8	22 40.0	28 0.0	25 20.0	26 0.0	23 22.9	23 22.9	23 22.0	26 0.0	23 22.9	24 23.5	24 23.0	26 0.0	24 22.1

[

]

Figure 3.2 Quad Cities Unit 2 Representative Cycle 24 Upper Left Quarter Core Layout by Fuel Type

Nuclear Fuel Type
BOC Exposure (GWd/MTU)

	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
60	21	19	19	19	19										
	43.7	43.3	43.5	41.8	44.5										
58	19	19	19	21	19	20									
	43.7	41.8	42.8	42.8	43.1	41.5									
56	22	21	22	21	19	21	19	21	21						
	40.2	40.7	40.8	41.6	41.9	42.2	44.0	44.1	44.9						
54	28	28	28	28	25	21	25	21	19	21					
	0.0	0.0	0.0	0.0	18.6	39.2	19.5	42.7	43.4	44.6					
52	25	28	23	28	25	28	25	22	22	21	19				
	20.0	0.0	22.2	0.0	20.3	0.0	19.4	36.1	41.7	43.7	44.4				
50	26	26	28	27	27	27	28	25	22	21	21	21			
	0.0	0.0	0.0	0.0	0.0	0.0	0.0	17.5	36.6	43.0	43.7	44.6			
48	23	23	27	23	23	27	25	28	22	22	21	21	21		
	22.9	22.9	0.0	23.6	22.4	0.0	20.0	0.0	35.7	36.5	41.5	43.0	44.9		
46	23	26	27	23	23	26	27	24	28	25	22	21	21		
	22.8	0.0	0.0	23.2	23.0	0.0	0.0	20.9	0.0	17.5	36.1	42.6	44.1		
44	23	26	24	26	23	28	24	27	25	28	25	25	19		
	22.0	0.0	21.3	0.0	22.6	0.0	21.5	0.0	20.0	0.0	19.5	19.5	44.0		
42	26	24	26	24	26	24	28	26	27	27	28	21	21	20	
	0.0	20.0	0.0	22.5	0.0	21.6	0.0	0.0	0.0	0.0	0.0	39.2	42.1	34.4	
40	23	26	23	24	24	26	23	23	23	27	25	25	19	19	19
	22.9	0.0	21.9	23.6	23.3	0.0	22.6	23.0	22.4	0.0	20.3	18.6	41.9	43.1	44.4
38	24	23	27	24	24	24	26	23	23	27	28	28	22	21	19
	23.5	20.8	0.0	23.3	23.6	22.5	0.0	23.2	23.6	0.0	0.0	0.0	39.8	42.7	41.7
36	24	26	24	27	23	26	24	27	27	28	23	28	22	19	19
	22.9	0.0	21.0	0.0	21.9	0.0	21.3	0.0	0.0	0.0	22.2	0.0	40.8	42.7	43.7
34	26	24	26	23	26	24	26	26	23	26	28	28	21	22	19
	0.0	22.0	0.0	20.8	0.0	20.0	0.0	0.0	22.9	0.0	0.0	0.0	40.6	42.7	43.3
32	24	26	24	24	23	26	23	23	23	26	25	28	22	19	21
	22.1	0.0	22.9	23.5	22.9	0.0	22.0	22.8	22.9	0.0	20.0	0.0	40.1	43.7	43.7

[

]

Figure 3.3 Quad Cities Unit 2 Representative Cycle 24 Upper Right Quarter Core Layout by Fuel Type

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29
30	21 43.7	19 43.8	22 40.0	28 0.0	25 20.0	26 0.0	23 22.9	23 22.9	23 22.1	26 0.0	23 22.9	24 23.5	24 22.9	26 0.0	24 22.1
28	19 43.3	22 42.7	22 40.8	28 0.0	28 0.0	26 0.0	23 23.0	26 0.0	26 0.0	24 20.0	26 0.0	23 20.8	26 0.0	24 22.0	26 0.0
26	19 43.7	19 42.7	21 40.6	28 0.0	23 22.2	28 0.0	27 0.0	27 0.0	24 21.3	26 0.0	23 21.9	27 0.0	24 21.0	26 0.0	24 22.9
24	19 41.7	21 42.7	21 41.5	28 0.0	28 0.0	27 0.0	23 23.6	23 23.2	26 0.0	24 22.5	24 23.6	24 23.3	27 0.0	23 20.8	24 23.5
22	19 44.4	19 43.1	19 41.9	25 18.6	25 20.3	27 0.0	23 22.5	23 23.0	23 22.6	26 0.0	24 23.3	24 23.6	23 22.0	26 0.0	23 22.9
20		20 34.4	21 42.1	21 39.1	28 0.0	27 0.0	27 0.0	26 0.0	28 0.0	24 21.6	26 0.0	24 22.6	26 0.0	24 20.0	26 0.0
18			19 44.1	25 19.5	25 19.5	28 0.0	25 20.0	27 0.0	24 21.5	28 0.0	23 22.6	26 0.0	24 21.3	26 0.0	23 22.1
16			21 44.2	21 42.5	22 36.1	25 17.5	28 0.0	24 20.9	27 0.0	26 0.0	23 23.0	23 23.2	27 0.0	26 0.0	23 22.9
14			21 44.9	21 43.0	19 41.8	22 36.5	22 35.8	28 0.0	25 20.0	27 0.0	23 22.5	23 23.6	27 0.0	23 23.0	23 22.9
12				21 44.5	21 43.7	21 43.0	22 36.6	25 17.5	28 0.0	27 0.0	27 0.0	27 0.0	28 0.0	26 0.0	26 0.0
10					19 44.6	21 43.7	22 41.7	22 36.1	25 19.4	28 0.0	25 20.4	28 0.0	23 22.2	28 0.0	25 20.0
8						21 44.6	19 43.4	21 42.7	25 19.5	21 39.2	25 18.6	28 0.0	28 0.0	28 0.0	28 0.0
6							21 44.9	21 44.1	19 44.1	21 42.1	19 42.0	21 41.6	22 39.8	22 40.8	22 40.2
4	Nuclear Fuel Type BOC Exposure (GWd/MTU)									20 41.5	19 43.1	21 42.8	19 42.8	19 41.7	19 43.8
2											19 44.5	21 40.7	19 43.5	19 43.4	21 43.7

[

]

Figure 3.4 Quad Cities Unit 2 Representative Cycle 24 Lower Left Quarter Core Layout by Fuel Type

	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
30	24 22.1	26 0.0	24 22.9	24 23.5	23 22.9	26 0.0	23 22.0	23 22.8	23 22.9	26 0.0	25 20.0	28 0.0	22 40.1	19 43.8	21 43.7
28	26 0.0	24 22.1	26 0.0	23 20.8	26 0.0	24 20.0	26 0.0	26 0.0	23 23.0	26 0.0	28 0.0	28 0.0	22 40.7	22 42.8	19 43.3
26	24 22.9	26 0.0	24 21.0	27 0.0	23 22.0	26 0.0	24 21.3	27 0.0	27 0.0	28 0.0	23 22.2	28 0.0	21 40.6	19 42.7	19 43.8
24	24 23.5	23 20.8	27 0.0	24 23.3	24 23.6	24 22.6	26 0.0	23 23.2	23 23.6	27 0.0	28 0.0	28 0.0	21 41.6	21 42.7	19 41.7
22	23 22.9	26 0.0	23 22.0	24 23.6	24 23.3	26 0.0	23 22.6	23 23.0	23 22.5	27 0.0	25 20.4	25 18.6	19 41.9	19 43.1	19 44.4
20	26 0.0	24 20.0	26 0.0	24 22.6	26 0.0	24 21.6	28 0.0	26 0.0	27 0.0	27 0.0	28 0.0	21 39.2	21 42.0	20 34.4	
18	23 22.0	26 0.0	24 21.3	26 0.0	23 22.6	28 0.0	24 21.6	27 0.0	25 20.0	28 0.0	25 19.5	25 19.5	19 44.1		
16	23 22.8	26 0.0	27 0.0	23 23.3	23 23.0	26 0.0	27 0.0	24 21.0	28 0.0	25 17.5	22 36.2	21 42.5	21 44.1		
14	23 22.9	23 23.0	27 0.0	23 23.6	23 22.5	27 0.0	25 20.0	28 0.0	22 35.8	22 36.5	21 41.5	21 43.0	21 44.9		
12	26 0.0	26 0.0	28 0.0	27 0.0	27 0.0	27 0.0	28 0.0	25 17.5	22 36.6	21 43.1	21 43.6	21 44.6			
10	25 20.0	28 0.0	23 22.2	28 0.0	25 20.4	28 0.0	25 19.5	22 36.1	19 41.8	21 43.7	19 44.6				
8	28 0.0	28 0.0	28 0.0	28 0.0	25 18.6	21 39.3	25 19.5	21 42.7	19 43.4	21 44.6					
6	22 40.2	22 40.8	22 39.8	22 41.7	19 42.0	21 42.2	19 44.1	21 44.2	21 44.9						
4	19 43.8	19 41.8	19 42.8	21 42.8	19 43.2	20 41.6									
2	21 43.7	19 43.4	19 43.5	21 40.7	19 44.5										

Nuclear Fuel Type
BOC Exposure (GWd/MTU)

[

]

Figure 3.5 Quad Cities Unit 2 Representative Cycle 24 Lower Right Quarter Core Layout by Fuel Type

4.0 References

1. 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, October 1999.

2. ANP-3294P Revision 0, *Nuclear Fuel Design Report Quad Cities Unit 2 Cycle 24 Representative Cycle ATRIUM 10XM Fuel,* AREVA Inc., July 2014.

3. 38-2201197-000, TODI NF1300233, *AREVA Representative Cycle - Cycle Design Inputs and Requirements (CDIR)*, January 2014.

4. 38-2201177-000, TODI NF1300232, *AREVA Representative Cycle - Energy Utilization Plan (EUP)*, October 2013.

5. ANP-10298P-A Revision 1, *ACE/ATRIUM 10XM Critical Power Correlation,* AREVA Inc., March 2014.

6. EMF-2209(P)(A) Revision 3, *SPCB Critical Power Correlation,* AREVA NP, September 2009.

7. EMF-2245(P)(A) Revision 0, *Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel,* Siemens Power Corporation, August 2000.

8. 12-9215611-000, *Quad Cities Unit 2 Cycle 24 Transition Core Design Target K-Effective Selection*, AREVA NP, December, 2013.

9. 38-2201170-003, TODI ES1300015, *Quad Cities Unit 2 Plant Parameters Document for Transition*, October 2014.

Appendix A Quad Cities Unit 2 Representative Cycle 24 Step-Through Depletion Summary, Control Rod Patterns and Core Average Axial Power and Exposure Distributions

Table A.1 Quad Cities Unit 2 Representative Cycle 24 Design Depletion Summary

Cycle Exposure (GWd/MT)	Calculated K-eff	Control Rod Density	Total Core Power MWt	Total Core Flow (Mlb/hr)	Ref. Pressure (psia)	Inlet Sub-Cooling (Btu/lb)	Void Fraction	Core Minimum CPR	Core Maximum LHGR (kW/ft)	Core Maximum APLHGR (kW/ft)
0.000	1.00000	4.28	2957.0	96.04	1015.00	24.74	0.407	1.793	12.33	9.10
0.250	0.99988	4.19	2957.0	96.04	1015.00	24.74	0.400	1.758	11.63	8.57
0.500	0.99979	3.48	2957.0	96.04	1015.00	24.74	0.404	1.755	11.54	8.91
1.000	0.99969	3.46	2957.0	96.04	1015.00	24.74	0.401	1.752	11.37	8.84
1.200	0.99953	3.48	2957.0	96.04	1015.00	24.74	0.401	1.766	11.38	8.86
1.201	0.99952	4.05	2957.0	96.04	1015.00	24.74	0.389	1.765	11.16	8.40
1.500	0.99941	4.07	2957.0	96.04	1015.00	24.74	0.388	1.770	11.03	8.36
2.000	0.99922	4.10	2957.0	96.04	1015.00	24.74	0.387	1.792	10.82	8.31
2.500	0.99918	4.10	2957.0	96.04	1015.00	24.74	0.384	1.791	10.59	8.25
3.000	0.99907	4.10	2957.0	96.04	1015.00	24.74	0.382	1.788	10.40	8.26
3.500	0.99877	4.12	2957.0	96.04	1015.00	24.74	0.382	1.788	10.45	8.38
3.825	0.99872	4.12	2957.0	96.04	1015.00	24.74	0.381	1.788	10.45	8.44
3.826	0.99864	3.58	2957.0	96.04	1015.00	24.74	0.395	1.848	11.25	9.06
4.000	0.99857	3.58	2957.0	96.04	1015.00	24.74	0.395	1.851	11.23	9.06
4.500	0.99851	3.58	2957.0	96.04	1015.00	24.74	0.393	1.859	11.16	9.08
5.000	0.99834	3.60	2957.0	96.04	1015.00	24.74	0.392	1.865	11.17	9.14
5.500	0.99792	3.72	2957.0	96.04	1015.00	24.74	0.393	1.873	11.27	9.30
6.000	0.99770	3.77	2957.0	96.04	1015.00	24.74	0.393	1.880	11.36	9.49
6.500	0.99760	3.81	2957.0	96.04	1015.00	24.74	0.393	1.888	11.46	9.70
6.750	0.99768	3.81	2957.0	96.04	1015.00	24.74	0.393	1.892	11.52	9.80
6.751	0.99765	4.47	2957.0	96.04	1015.00	24.74	0.378	1.811	10.92	9.36
7.000	0.99746	4.52	2957.0	96.04	1015.00	24.74	0.379	1.819	11.01	9.48
7.500	0.99759	4.52	2957.0	96.04	1015.00	24.74	0.379	1.828	11.08	9.64
8.000	0.99751	4.57	2957.0	96.04	1015.00	24.74	0.379	1.834	11.07	9.70
8.500	0.99755	4.61	2957.0	96.04	1015.00	24.74	0.378	1.841	11.02	9.67
9.000	0.99761	4.66	2957.0	96.04	1015.00	24.74	0.376	1.835	10.95	9.72
9.350	0.99763	4.71	2957.0	96.04	1015.00	24.74	0.374	1.832	10.82	9.67
9.351	0.99746	4.73	2957.0	96.04	1015.00	24.74	0.383	1.877	11.51	10.18
9.500	0.99741	4.76	2957.0	96.04	1015.00	24.74	0.383	1.871	11.46	10.14
10.000	0.99746	4.80	2957.0	96.04	1015.00	24.74	0.378	1.857	10.95	9.76
10.500	0.99743	4.90	2957.0	96.04	1015.00	24.74	0.373	1.833	10.37	9.30
11.000	0.99740	5.18	2957.0	96.04	1015.00	24.74	0.368	1.807	9.71	8.75
11.500	0.99759	5.23	2957.0	96.04	1015.00	24.74	0.361	1.780	9.12	8.22
11.950	0.99760	5.27	2957.0	96.04	1015.00	24.74	0.355	1.761	9.16	8.09
11.951	0.99761	5.11	2957.0	96.04	1015.00	24.74	0.345	1.768	9.45	8.38
12.000	0.99757	5.13	2957.0	96.04	1015.00	24.74	0.344	1.764	9.47	8.40
12.500	0.99758	5.13	2957.0	96.04	1015.00	24.74	0.337	1.755	9.60	8.53
13.000	0.99753	5.06	2957.0	96.04	1015.00	24.74	0.329	1.722	9.60	8.60
13.500	0.99750	4.94	2957.0	96.04	1015.00	24.74	0.320	1.719	9.66	8.69
13.600	0.99753	4.92	2957.0	96.04	1015.00	24.74	0.319	1.722	9.68	8.71
13.601	0.99751	4.17	2957.0	96.04	1015.00	24.74	0.333	1.719	9.41	8.47
14.000	0.99748	3.91	2957.0	96.04	1015.00	24.74	0.324	1.721	9.52	8.55
14.500	0.99746	2.38	2957.0	96.04	1015.00	24.74	0.323	1.721	9.56	8.71
15.000	0.99736	1.93	2957.0	96.04	1015.00	24.74	0.310	1.725	9.57	8.65
15.250	0.99768	0.49	2957.0	96.04	1015.00	24.74	0.316	1.701	9.79	8.79
15.405	0.99807	0.00	2957.0	98.00	1015.00	24.21	0.312	1.732	9.74	8.74
15.485	0.99794	0.00	2957.0	99.96	1015.00	23.70	0.308	1.741	9.78	8.78
15.963	0.99731	0.00	2735.9	99.96	1015.00	22.15	0.289	1.862	9.23	8.28
16.362	0.99733	0.00	2513.4	99.96	1015.00	20.61	0.270	1.996	8.64	7.74

Table A.2 Quad Cities Unit 2 Representative Cycle 24 Design Depletion Thermal Margin Summary

Cycle Exposure (GWd/MT)	Calculated K-eff	Control Rod Density	Core Limiting CPR	Fraction of Limiting CPR	Core Limiting LHGR (kW/ft)	Fraction of Limiting LHGR	Core Limiting APLHGR (kW/ft)	Fraction of Limiting APLHGR
0.000	1.00000	4.284	1.793	0.814	10.48	0.905	9.10	0.903
0.250	0.99988	4.190	1.758	0.831	9.58	0.859	8.57	0.854
0.500	0.99979	3.484	1.755	0.832	9.32	0.841	8.05	0.838
1.000	0.99969	3.460	1.752	0.833	9.01	0.822	7.86	0.818
1.200	0.99953	3.484	1.766	0.827	8.94	0.819	7.84	0.815
1.201	0.99952	4.049	1.765	0.827	8.78	0.804	7.92	0.809
1.500	0.99941	4.073	1.770	0.825	8.65	0.798	7.80	0.801
2.000	0.99922	4.096	1.792	0.815	8.64	0.790	7.62	0.785
2.500	0.99918	4.096	1.804	0.809	8.46	0.782	7.45	0.768
3.000	0.99907	4.096	1.788	0.805	8.32	0.777	7.34	0.759
3.500	0.99877	4.120	1.788	0.805	8.22	0.776	7.28	0.754
3.825	0.99872	4.120	1.788	0.805	8.13	0.773	7.23	0.750
3.826	0.99864	3.578	1.848	0.790	8.73	0.817	7.83	0.809
4.000	0.99857	3.578	1.851	0.789	8.67	0.815	7.79	0.806
4.500	0.99851	3.578	1.869	0.781	8.49	0.807	7.67	0.795
5.000	0.99834	3.602	1.865	0.772	8.37	0.805	7.59	0.789
5.500	0.99792	3.719	1.873	0.769	8.30	0.812	9.30	0.794
6.000	0.99770	3.766	1.880	0.766	8.31	0.822	9.49	0.811
6.500	0.99760	3.814	1.888	0.763	8.33	0.834	9.70	0.829
6.750	0.99768	3.814	1.892	0.761	8.34	0.839	9.80	0.837
6.751	0.99765	4.473	1.811	0.795	8.06	0.807	9.36	0.800
7.000	0.99746	4.520	1.819	0.791	7.85	0.811	9.48	0.810
7.500	0.99759	4.520	1.828	0.788	7.82	0.817	9.64	0.824
8.000	0.99751	4.567	1.834	0.785	7.80	0.820	9.70	0.829
8.500	0.99755	4.614	1.841	0.782	7.78	0.827	9.67	0.827
9.000	0.99761	4.661	1.835	0.785	7.67	0.825	9.72	0.831
9.350	0.99763	4.708	1.832	0.786	7.55	0.818	9.67	0.826
9.351	0.99746	4.732	1.877	0.767	8.09	0.858	10.18	0.870
9.500	0.99741	4.755	1.871	0.770	8.03	0.855	10.14	0.868
10.000	0.99746	4.802	1.857	0.775	7.66	0.824	9.76	0.846
10.500	0.99743	4.896	1.833	0.785	7.14	0.789	9.30	0.815
11.000	0.99740	5.179	1.807	0.797	6.80	0.759	8.75	0.775
11.500	0.99759	5.226	1.780	0.809	6.53	0.733	8.22	0.733
11.950	0.99760	5.273	1.761	0.818	6.91	0.747	8.09	0.715
11.951	0.99761	5.108	1.768	0.814	7.33	0.765	8.35	0.739
12.000	0.99757	5.132	1.764	0.817	7.34	0.767	8.37	0.741
12.500	0.99758	5.132	1.755	0.820	7.37	0.778	8.53	0.758
13.000	0.99753	5.061	1.722	0.836	7.33	0.782	8.58	0.770
13.500	0.99750	4.944	1.719	0.855	7.51	0.801	8.68	0.785
13.600	0.99753	4.920	1.722	0.854	7.52	0.803	8.68	0.786
13.601	0.99751	4.167	1.719	0.855	6.80	0.763	8.47	0.771
14.000	0.99748	3.908	1.721	0.854	7.97	0.804	8.55	0.785
14.500	0.99746	2.378	1.721	0.854	7.14	0.802	8.71	0.807
15.000	0.99736	1.930	1.725	0.852	7.58	0.819	8.65	0.810
15.250	0.99768	0.494	1.701	0.864	7.36	0.823	8.78	0.814
15.405	0.99807	0.000	1.732	0.849	7.29	0.818	8.71	0.811
15.485	0.99794	0.000	1.741	0.844	7.30	0.820	8.74	0.814
15.963	0.99731	0.000	1.862	0.789	6.76	0.768	8.19	0.771
16.362	0.99733	0.000	1.996	0.737	6.41	0.721	7.61	0.723

```
Cycle:                            24                   Core Average Exposure: MWd/MTU    20701.2
Exposure: MWd/MTU (GWd)          0.0 (    0.00 )
Delta E: MWd/MTU, (GWd)          0.0 (    0.00 )             Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)         N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                    Top 24  0.141   3.641    19  0.376 0.643  55 40
Inlet Subcooling: Btu/lbm     -24.74                        23  0.259   8.406    20  0.333 0.348   3 42
Flow: Mlb/hr                   96.04 ( 98.00 %)             22  0.547  16.276    21  0.455 0.801  53 42
                                                            21  0.658  19.868    22  0.712 0.864  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             20  0.739  21.758    23  1.319 1.504  27 38
59                -- -- -- -- --                 59         19  0.794  22.672    24  1.324 1.548  29 38
55          -- -- -- -- -- -- -- -- -- -- --     55         18  0.831  22.587    25  1.118 1.305  13 44
51       -- -- -- -- -- -- -- -- -- -- -- -- --  51         17  0.884  22.593    26  1.301 1.428  27 36
47    -- -- 34 -- -- -- 20 -- -- -- 34 -- --     47         16  0.920  22.785    27  1.281 1.378  23 36
43    -- -- -- -- -- -- -- -- -- -- -- -- -- --  43         15  0.954  23.190    28  1.129 1.394  17 42
39 -- -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 39       14  0.984  23.788
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35       13  1.079  23.089
31 -- -- -- 20 -- -- -- -- 12 -- -- -- 20 -- -- -- 31       12  1.137  23.463
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27       11  1.193  24.076
23 -- -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 23       10  1.246  24.479
19    -- -- -- -- -- -- -- -- -- -- -- -- -- --  19          9  1.302  24.620
15    -- -- 34 -- -- -- 20 -- -- -- 34 -- --     15          8  1.364  25.091
11       -- -- -- -- -- -- -- -- -- -- -- -- --  11          7  1.408  25.438
 7          -- -- -- -- -- -- -- -- -- -- --      7          6  1.475  25.487
 3                -- -- -- -- --                  3          5  1.542  25.554*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              4  1.558* 24.637
                                                             3  1.488  22.197
Control Rod Density: %          4.28                         2  1.169  16.960
                                                      Bottom 1  0.328   5.075
k-effective:                 1.00000
Void Fraction:                 0.407              % AXIAL TILT -26.752  -7.822
Core Delta-P: psia            21.961          AVG BOT 8ft/12ft  1.1967  1.0860
Core Plate Delta-P: psia      17.528
Coolant Temp: Deg-F            543.9
In Channel Flow: Mlb/hr        85.42   Active Channel Flow: Mlb/hr        81.53
Total Bypass Flow (%):          11.1   (of total core flow)
Total Water Rod Flow (%):        4.1   (of total core flow)
Source Convergence           0.00042
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.548	24	29	38	1.793	0.814	23	23	34	9.10	0.903	24.9	24	31	24	3	10.48	0.905	29.6	24	31	26	3
1.547	24	23	32	1.794	0.814	24	25	32	9.08	0.902	24.8	24	37	30	3	10.27	0.904	31.7	24	31	24	3
1.516	24	29	36	1.795	0.813	24	31	26	8.80	0.883	22.2	23	31	22	3	10.43	0.903	29.8	24	25	32	3
1.514	24	25	32	1.797	0.813	23	33	24	8.78	0.881	22.3	23	39	30	3	10.24	0.902	31.7	24	23	32	3
1.504	23	27	38	1.803	0.810	24	23	32	8.82	0.871	23.0	24	31	26	3	10.04	0.890	32.5	23	33	24	4
1.504	23	23	34	1.804	0.809	24	31	24	8.79	0.869	23.1	24	35	30	3	10.03	0.889	32.5	23	23	34	4
1.501	23	29	40	1.862	0.784	23	31	40	8.47	0.860	26.9	23	27	24	4	12.33	0.875	0.0	26	39	34	4
1.500	23	21	32	1.838	0.783	26	27	36	8.48	0.860	26.8	23	37	28	4	12.33	0.875	0.0	26	33	40	4
1.458	24	25	36	1.864	0.783	23	39	32	8.25	0.833	25.9	23	21	16	4	9.84	0.861	31.0	23	31	22	4
1.449	24	27	34	1.842	0.782	26	35	28	8.22	0.831	26.0	23	45	22	4	9.69	0.861	32.8	23	39	32	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.1 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 0.0 MWd/MTU

```
Cycle:                            24
Exposure: MWd/MTU (GWd)        250.0 (  31.45 )
Delta E: MWd/MTU, (GWd)        250.0 (  31.45 )
Power: MWt                    2957.0 (100.00 %)
Core Pressure: psia           1015.0
Inlet Subcooling: Btu/lbm     -24.74
Flow: Mlb/hr                   96.04 ( 98.00 %)

    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                -- -- -- -- --                59
55          -- -- -- -- -- -- -- -- --          55
51       -- -- -- -- -- -- -- -- -- -- --       51
47    -- -- 34 -- -- -- 20 -- -- -- 34 -- --    47
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43
39 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- 20 -- -- -- 12 -- -- -- 20 -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 23
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19
15    -- -- 34 -- -- -- 20 -- -- -- 34 -- --    15
11       -- -- -- -- -- -- -- -- -- -- --       11
 7          -- -- -- -- -- -- -- -- --           7
 3                -- -- -- -- --                 3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58

Control Rod Density: %          4.19

k-effective:                 0.99988
Void Fraction:                 0.400
Core Delta-P: psia            21.857
Core Plate Delta-P: psia      17.424
Coolant Temp: Deg-F            543.8
In Channel Flow: Mlb/hr        85.48   Active Channel Flow: Mlb/hr        81.62
Total Bypass Flow (%):          11.0   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00048
```

Core Average Exposure: MWd/MTU 20951.2

	Axial Profile	
N(PRA)	Power	Exposure
Top 24	0.153	3.672
23	0.280	8.476
22	0.592	16.425
21	0.713	20.048
20	0.802	21.960
19	0.850	22.889
18	0.881	22.814
17	0.929	22.826
16	0.960	23.022
15	0.987	23.435
14	1.011	24.041
13	1.098	23.353
12	1.148	23.742
11	1.195	24.368
10	1.239	24.784
9	1.285	24.935
8	1.337	25.416
7	1.369	25.773
6	1.423	25.838
5	1.474	25.921*
4	1.475*	25.008
3	1.397	22.552
2	1.096	17.239
Bottom 1	0.307	5.154
% AXIAL TILT	-22.879	-8.002
AVG BOT 8ft/12ft	1.1751	1.0869

Edit	Radial Power			
Zone	Avg.	Max.	IR	JR
19	0.377	0.642	55	40
20	0.334	0.349	3	42
21	0.456	0.799	53	42
22	0.711	0.866	13	48
23	1.322	1.514	27	38
24	1.337	1.554	29	38
25	1.114	1.300	13	44
26	1.301	1.431	27	36
27	1.273	1.386	23	36
28	1.119	1.387	17	42

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.554	24	29	38	1.758	0.831	23	23	34	8.57	0.854	25.5	24	31	24	3	9.58	0.859	33.8	24	31	26	4
1.553	24	23	32	1.761	0.829	23	33	24	8.56	0.853	25.5	24	37	30	3	9.55	0.857	33.9	24	25	32	4
1.523	24	29	36	1.767	0.826	24	25	32	8.25	0.834	26.1	23	31	22	4	9.36	0.857	36.1	24	31	24	4
1.521	24	25	32	1.767	0.826	24	31	26	8.23	0.833	26.2	23	39	30	4	9.34	0.855	36.1	24	23	32	4
1.514	23	27	38	1.799	0.811	24	23	30	8.24	0.828	26.8	24	31	26	4	9.51	0.848	33.2	23	33	24	4
1.514	23	23	34	1.799	0.811	24	29	38	8.22	0.826	26.9	24	35	30	4	9.50	0.847	33.2	23	23	34	4
1.506	23	31	40	1.830	0.798	23	31	40	8.05	0.819	27.5	23	27	24	4	11.63	0.825	0.8	26	39	34	4
1.506	23	21	32	1.832	0.797	23	39	32	8.05	0.819	27.4	23	37	28	4	11.63	0.825	0.8	26	33	40	4
1.470	24	25	36	1.838	0.794	24	29	30	7.83	0.793	26.5	23	21	16	4	9.20	0.821	33.4	23	31	40	4
1.457	24	27	34	1.857	0.786	24	19	34	7.81	0.791	26.6	23	45	22	4	9.19	0.821	33.5	23	39	32	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.2 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 250.0 MWd/MTU

```
Cycle:                           24                  Core Average Exposure: MWd/MTU    21201.2
Exposure: MWd/MTU (GWd)       500.0 (  62.90 )
Delta E: MWd/MTU, (GWd)       250.0 (  31.45 )             Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)       N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                 Top 24  0.153   3.706    19  0.372 0.632  55 40
Inlet Subcooling: Btu/lbm    -24.74                     23  0.279   8.551    20  0.328 0.342   3 42
Flow: Mlb/hr                  96.04 ( 98.00 %)          22  0.589  16.586    21  0.453 0.790  53 42
                                                        21  0.707  20.242    22  0.719 0.960  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58         20  0.794  22.179    23  1.315 1.513  27 38
59                -- -- -- -- --                59      19  0.842  23.122    24  1.346 1.551  29 38
55             -- -- -- -- -- -- -- -- --       55      18  0.874  23.055    25  1.117 1.330  13 44
51          -- -- -- -- -- -- -- -- -- -- --    51      17  0.918  23.071    26  1.300 1.437  27 36
47       -- -- -- -- -- -- 20 -- -- -- -- -- -- 47      16  0.942  23.269    27  1.275 1.383  23 36
43       -- -- -- -- -- -- -- -- -- -- -- -- -- 43      15  0.963  23.690    28  1.123 1.391  17 42
39 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 39      14  0.979  24.301
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35      13  1.059  23.622
31 -- -- -- 20 -- -- -- 16 -- -- -- 20 -- -- -- 31      12  1.104  24.023
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27      11  1.149  24.661
23 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 23      10  1.194  25.087
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19       9  1.248  25.247
15    -- -- -- -- -- -- 20 -- -- -- -- -- --    15       8  1.316  25.735
11       -- -- -- -- -- -- -- -- -- -- --       11       7  1.378  26.099
 7          -- -- -- -- -- -- -- -- --           7       6  1.455  26.177
 3             -- -- -- -- --                    3       5  1.526  26.272*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58          4  1.543* 25.360
                                                         3  1.478  22.885
Control Rod Density: %         3.48                      2  1.176  17.500
                                                  Bottom 1  0.334   5.228
k-effective:                0.99979
Void Fraction:                0.404             % AXIAL TILT -24.173  -8.131
Core Delta-P: psia           21.899         AVG BOT 8ft/12ft  1.1777  1.0876
Core Plate Delta-P: psia     17.469
Coolant Temp: Deg-F           543.9
In Channel Flow: Mlb/hr       85.46   Active Channel Flow: Mlb/hr       81.58
Total Bypass Flow (%):         11.0   (of total core flow)
Total Water Rod Flow (%):       4.0   (of total core flow)
Source Convergence          0.00030
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.551	24	29	38	1.755	0.832	24	25	32	8.05	0.838	25.9	25	17	14	4	9.32	0.841	34.5	24	31	26	4
1.551	24	23	32	1.755	0.832	24	31	26	8.05	0.837	26.0	25	47	44	4	9.29	0.839	34.6	24	25	32	4
1.529	24	29	36	1.760	0.830	23	23	34	8.18	0.836	29.3	24	31	24	4	9.11	0.839	36.7	24	31	24	4
1.527	24	25	32	1.763	0.828	23	33	24	8.17	0.835	29.2	24	37	30	4	9.09	0.837	36.8	24	23	32	4
1.513	23	27	38	1.780	0.820	24	29	30	8.09	0.821	27.0	23	21	16	4	9.28	0.833	33.9	23	33	24	4
1.513	23	23	34	1.790	0.815	24	29	38	8.09	0.820	26.7	23	31	22	4	9.27	0.832	33.9	23	23	34	4
1.501	23	31	40	1.791	0.815	24	23	30	8.06	0.819	27.1	23	45	22	4	9.10	0.821	34.5	23	39	18	4
1.500	23	21	32	1.809	0.796	26	27	36	8.07	0.819	26.8	23	39	30	4	11.54	0.819	1.5	28	41	18	4
1.472	24	27	34	1.836	0.795	23	31	40	8.04	0.817	28.4	24	17	18	4	9.06	0.818	34.7	23	17	40	4
1.472	24	25	36	1.838	0.794	23	39	32	8.00	0.812	26.9	23	21	18	4	11.54	0.818	1.4	28	17	20	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.3 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 500.0 MWd/MTU

```
Cycle:                            24                   Core Average Exposure: MWd/MTU    21701.2
Exposure: MWd/MTU (GWd)       1000.0 ( 125.81 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )               Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)         N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                   Top 24  0.158   3.774    19  0.366 0.623  55 40
Inlet Subcooling: Btu/lbm     -24.74                       23  0.288   8.701    20  0.322 0.336   3 42
Flow: Mlb/hr                   96.04 ( 98.00 %)            22  0.603  16.906    21  0.446 0.783   7 20
                                                           21  0.722  20.627    22  0.711 0.952  13 14
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            20  0.809  22.613    23  1.315 1.520  23 28
59                -- -- -- -- -- --               59       19  0.858  23.582    24  1.348 1.549  23 30
55          -- -- -- -- -- -- -- -- -- --         55       18  0.889  23.534    25  1.110 1.329  13 18
51       -- -- -- -- -- -- -- -- -- -- -- --      51       17  0.933  23.555    26  1.312 1.454  27 26
47    -- -- -- -- -- -- 20 -- -- -- -- -- --      47       16  0.956  23.754    27  1.282 1.406  23 26
43    -- -- -- -- -- -- -- -- -- -- -- -- --      43       15  0.974  24.185    28  1.124 1.408  17 20
39 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 39       14  0.988  24.805
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35       13  1.063  24.141
31 -- -- -- 20 -- -- -- 16 -- -- -- 20 -- -- -- 31       12  1.104  24.564
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27       11  1.145  25.224
23 -- -- -- -- -- 12 -- -- -- 10 -- -- -- -- -- 23       10  1.188  25.673
19    -- -- -- -- -- -- -- -- -- -- -- -- --      19        9  1.240  25.852
15    -- -- -- -- -- -- 20 -- -- -- -- -- --      15        8  1.305  26.362
11       -- -- -- -- -- -- -- -- -- -- -- --      11        7  1.365  26.756
 7          -- -- -- -- -- -- -- -- -- --          7        6  1.439  26.870
 3                -- -- -- -- -- --                3        5  1.506  26.999*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             4  1.518* 26.095
                                                            3  1.453  23.589
Control Rod Density: %          3.46                        2  1.162  18.060
                                                     Bottom  1  0.334   5.388
k-effective:                 0.99969
Void Fraction:                 0.401              % AXIAL TILT -22.992  -8.412
Core Delta-P: psia            21.878          AVG BOT 8ft/12ft  1.1712  1.0890
Core Plate Delta-P: psia      17.448
Coolant Temp: Deg-F            543.9
In Channel Flow: Mlb/hr        85.47   Active Channel Flow: Mlb/hr        81.60
Total Bypass Flow (%):          11.0   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00039
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.549	24	23	30	1.752	0.833	24	25	32	7.86	0.818	27.0	25	17	14	4	9.01	0.822	35.8	24	31	26	4
1.548	24	29	24	1.753	0.833	24	31	26	7.86	0.817	27.1	25	47	44	4	8.98	0.820	35.9	24	25	32	4
1.528	24	29	26	1.759	0.830	23	23	28	7.94	0.816	30.4	24	31	24	4	8.80	0.819	38.0	24	31	24	4
1.527	24	25	30	1.762	0.829	23	27	24	7.93	0.815	30.4	24	37	30	4	8.78	0.818	38.0	24	23	32	4
1.520	23	23	28	1.773	0.823	24	29	30	7.91	0.807	28.2	23	39	16	4	9.00	0.816	35.2	23	33	24	4
1.519	23	27	24	1.784	0.818	24	23	30	7.89	0.807	29.5	24	43	18	4	8.99	0.815	35.2	23	23	34	4
1.502	23	21	30	1.786	0.818	24	29	24	7.89	0.805	28.3	23	45	22	4	8.87	0.809	35.8	23	39	18	4
1.501	23	29	22	1.778	0.810	26	27	26	7.89	0.804	27.9	23	31	22	4	8.83	0.807	36.0	23	17	40	4
1.482	24	25	26	1.779	0.810	26	25	28	7.88	0.803	27.9	23	39	30	4	11.37	0.807	2.9	28	19	18	4
1.477	24	27	28	1.817	0.792	26	21	28	7.83	0.799	28.1	23	39	18	4	11.37	0.806	2.9	28	17	20	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.4 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 1,000.0 MWd/MTU

```
Cycle:                          24                    Core Average Exposure: MWd/MTU    21901.2
Exposure: MWd/MTU (GWd)     1200.0 ( 150.97 )
Delta E: MWd/MTU, (GWd)      200.0 (  25.16 )                Axial Profile    Edit   Radial Power
Power: MWt                  2957.0 (100.00 %)          N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia         1015.0                    Top 24  0.158   3.802    19  0.365 0.622  55 40
Inlet Subcooling: Btu/lbm   -24.74                        23  0.289   8.763    20  0.320 0.334   3 42
Flow: Mlb/hr                 96.04 ( 98.00 %)             22  0.603  17.037    21  0.444 0.780  53 42
                                                          21  0.719  20.784    22  0.709 0.947  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58           20  0.805  22.790    23  1.314 1.508  27 38
59                -- -- -- -- -- --              59       19  0.852  23.770    24  1.344 1.539  29 38
55          -- -- -- -- -- -- -- -- -- --        55       18  0.886  23.728    25  1.109 1.324  13 44
51       -- -- -- -- -- -- -- -- -- -- -- --     51       17  0.932  23.752    26  1.316 1.451  27 36
47    -- -- -- -- -- -- 20 -- -- -- -- -- --     47       16  0.957  23.951    27  1.286 1.394  23 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --     43       15  0.976  24.386    28  1.127 1.402  17 42
39 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 39       14  0.990  25.009
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35       13  1.064  24.349
31 -- -- -- 20 -- -- -- 16 -- -- -- 20 -- -- -- 31       12  1.105  24.781
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27       11  1.146  25.449
23 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 23       10  1.188  25.905
19    -- -- -- -- -- -- -- -- -- -- -- -- --     19        9  1.240  26.092
15    -- -- -- -- -- -- 20 -- -- -- -- -- --     15        8  1.306  26.611
11       -- -- -- -- -- -- -- -- -- -- -- --     11        7  1.365  27.016
 7          -- -- -- -- -- -- -- -- -- --         7        6  1.440  27.144
 3                -- -- -- -- -- --               3        5  1.507  27.286*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            4  1.518* 26.384
                                                           3  1.453  23.866
Control Rod Density: %        3.48                         2  1.166  18.282
                                                    Bottom 1  0.336   5.452
k-effective:               0.99953
Void Fraction:               0.401             % AXIAL TILT -23.085  -8.510
Core Delta-P: psia          21.884         AVG BOT 8ft/12ft  1.1723  1.0895
Core Plate Delta-P: psia    17.454
Coolant Temp: Deg-F          543.9
In Channel Flow: Mlb/hr      85.47   Active Channel Flow: Mlb/hr        81.60
Total Bypass Flow (%):        11.0   (of total core flow)
Total Water Rod Flow (%):      4.0   (of total core flow)
Source Convergence         0.00039
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.539	24	29	38	1.766	0.827	24	25	32	7.84	0.815	27.5	25	17	14	4	8.94	0.819	36.3	24	31	26	4
1.539	24	23	32	1.767	0.826	24	31	26	7.83	0.814	27.6	25	47	44	4	8.91	0.817	36.4	24	25	32	4
1.521	24	29	36	1.768	0.826	23	23	34	7.89	0.812	30.9	24	31	24	4	8.73	0.816	38.5	24	31	24	4
1.519	24	25	32	1.772	0.824	23	33	24	7.88	0.811	30.8	24	37	30	4	8.71	0.815	38.5	24	23	32	4
1.508	23	27	38	1.784	0.818	24	29	30	7.87	0.808	29.9	24	17	18	4	8.93	0.814	35.7	23	33	24	4
1.508	23	23	34	1.798	0.812	24	29	38	7.88	0.806	28.6	23	21	16	4	8.93	0.813	35.7	23	23	34	4
1.494	23	29	40	1.799	0.812	24	23	30	7.86	0.804	28.7	23	45	22	4	8.83	0.809	36.3	23	39	18	4
1.494	23	21	32	1.799	0.801	26	27	36	7.86	0.802	28.3	23	31	22	4	11.38	0.807	3.5	28	19	18	4
1.470	24	27	34	1.803	0.799	26	25	28	7.84	0.801	28.4	23	39	30	4	8.79	0.807	36.5	23	17	40	4
1.469	24	25	36	1.848	0.790	23	31	40	7.81	0.798	28.5	23	21	18	4	11.37	0.807	3.5	28	17	20	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.5 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 1,200.0 MWd/MTU

```
Cycle:                             24                Core Average Exposure: MWd/MTU    21902.2
Exposure: MWd/MTU (GWd)      1201.0 ( 151.10 )
Delta E: MWd/MTU, (GWd)         1.0 (   0.13 )               Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                     Top 24  0.165   3.802    19  0.368 0.618  55 40
Inlet Subcooling: Btu/lbm    -24.74                         23  0.303   8.764    20  0.319 0.333  57 42
Flow: Mlb/hr                  96.04 ( 98.00 %)              22  0.633  17.038    21  0.444 0.768  53 42
                                                            21  0.759  20.785    22  0.713 0.935  47 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             20  0.852  22.791    23  1.273 1.490  39 36
59                -- -- -- -- --                59          19  0.908  23.770    24  1.416 1.541  39 38
55          -- -- -- -- -- -- -- -- --          55          18  0.950  23.729    25  1.096 1.282  47 44
51       -- -- -- -- -- -- -- -- -- -- --       51          17  1.009  23.753    26  1.337 1.422  31 28
47    -- -- -- -- 24 -- -- -- 24 -- -- -- --    47          16  1.047  23.952    27  1.245 1.481  37 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43          15  1.077  24.387    28  1.121 1.390  43 42
39 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 39          14  1.095  25.010
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35          13  1.155  24.350
31 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 31          12  1.153  24.782
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27          11  1.160  25.450
23 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 23          10  1.175  25.907
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19           9  1.205  26.093
15    -- -- -- -- 24 -- -- -- 24 -- -- -- --    15           8  1.251  26.612
11       -- -- -- -- -- -- -- -- -- -- --       11           7  1.294  27.018
 7          -- -- -- -- -- -- -- -- --           7           6  1.353  27.145
 3                -- -- -- -- --                 3           5  1.401* 27.287*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              4  1.393  26.386
                                                             3  1.317  23.867
Control Rod Density: %          4.05                         2  1.044  18.283
                                                     Bottom  1  0.300   5.452
k-effective:                 0.99952
Void Fraction:                 0.389           % AXIAL TILT -17.055  -8.510
Core Delta-P: psia            21.743       AVG BOT 8ft/12ft  1.1513  1.0895
Core Plate Delta-P: psia      17.311
Coolant Temp: Deg-F            543.7
In Channel Flow: Mlb/hr        85.56   Active Channel Flow: Mlb/hr        81.72
Total Bypass Flow (%):          10.9   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00038


                          Top Ten Thermal Limits Summary - Sorted by Margin

     Power                 MCPR                         APLHGR                              LHGR
Value FT   IR JR    Value Margin FT   IR JR    Value Margin Exp. FT   IR JR   K    Value Margin Exp. FT   IR JR   K
1.541  24 39 38    1.765  0.827  23 25 40    7.92  0.809 29.3  24 37 22   4     8.78  0.804 36.3  23 39 26  4
1.540  24 37 40    1.766  0.827  23 39 26    7.91  0.809 29.4  24 39 24   4     8.74  0.802 36.5  24 39 24  4
1.539  24 37 38    1.799  0.812  24 25 36    7.89  0.806 29.3  24 37 24   4     8.75  0.801 36.2  23 25 40  4
1.523  24 39 40    1.811  0.806  24 41 38    7.75  0.792 29.2  24 39 22   4     8.92  0.801 34.1  23 31 18  4
1.513  24 29 30    1.812  0.806  24 23 24    7.75  0.787 27.0  23 31 16   4     8.70  0.799 36.5  24 23 40  4
1.493  24 35 36    1.814  0.805  24 37 42    7.74  0.786 27.1  23 45 30   4     8.87  0.798 34.1  23 17 32  4
1.490  23 39 36    1.814  0.805  24 37 30    7.71  0.786 29.0  24 31 30   4     8.87  0.794 33.6  23 31 16  4
1.489  23 35 40    1.816  0.804  24 29 24    7.54  0.774 29.4  23 35 22   4    11.16  0.792  3.1  27 25 24  4
1.481  27 37 36    1.817  0.804  24 23 40    7.53  0.773 29.4  23 39 26   4     8.95  0.791 32.2  24 37 24  3
1.480  24 41 38    1.818  0.803  24 39 24    7.62  0.773 27.0  23 31 18   4    11.15  0.791  3.1  27 23 26  4

  * LHGR calculated with pin-power reconstruction
  * CPR calculated with pin-power reconstruction & CPR limit type 3
```

Figure A.6 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 1,201.0 MWd/MTU

```
Cycle:                           24                     Core Average Exposure: MWd/MTU    22201.2
Exposure: MWd/MTU (GWd)      1500.0 ( 188.71 )
Delta E: MWd/MTU, (GWd)       299.0 (  37.62 )                Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)          N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                      Top 24  0.166   3.846    19  0.365 0.614  55 40
Inlet Subcooling: Btu/lbm    -24.74                          23  0.305   8.861    20  0.317 0.331   3 42
Flow: Mlb/hr                  96.04 ( 98.00 %)               22  0.635  17.244    21  0.441 0.766   7 42
                                                             21  0.758  21.032    22  0.711 0.933  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              20  0.849  23.069    23  1.272 1.487  21 26
59                -- -- -- -- --               59            19  0.907  24.067    24  1.411 1.535  23 40
55          -- -- -- -- -- -- -- -- -- --      55            18  0.951  24.040    25  1.095 1.282  13 44
51       -- -- -- -- -- -- -- -- -- -- -- --   51            17  1.013  24.071    26  1.342 1.427  19 40
47    -- -- -- -- 24 -- -- -- 24 -- -- -- --   47            16  1.053  24.274    27  1.251 1.484  23 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --   43            15  1.082  24.718    28  1.125 1.397  17 42
39 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 39           14  1.099  25.347
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35           13  1.158  24.688
31 -- -- -- -- -- 10 -- -- --  8 -- -- -- -- -- 31           12  1.154  25.120
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27           11  1.159  25.790
23 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 23           10  1.174  26.251
19    -- -- -- -- -- -- -- -- -- -- -- -- --   19             9  1.203  26.442
15    -- -- -- -- 24 -- -- -- 24 -- -- -- --   15             8  1.249  26.969
11       -- -- -- -- -- -- -- -- -- -- -- --   11             7  1.292  27.386
 7          -- -- -- -- -- -- -- -- -- --       7             6  1.351  27.530
 3                -- -- -- -- --                3             5  1.398* 27.687*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               4  1.388  26.783
                                                              3  1.312  24.243
Control Rod Density: %         4.07                           2  1.043  18.580
                                                       Bottom  1  0.301   5.538
k-effective:                0.99941
Void Fraction:                0.388              % AXIAL TILT -16.877  -8.572
Core Delta-P: psia           21.744          AVG BOT 8ft/12ft  1.1511  1.0899
Core Plate Delta-P: psia     17.312
Coolant Temp: Deg-F           543.7
In Channel Flow: Mlb/hr       85.56   Active Channel Flow: Mlb/hr        81.73
Total Bypass Flow (%):         10.9   (of total core flow)
Total Water Rod Flow (%):       4.0   (of total core flow)
Source Convergence          0.00032
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.535	24	23	40	1.770	0.825	23	25	40	7.80	0.801	30.0	24	37	22	4	8.65	0.798	37.0	23	39	26	4
1.535	24	21	38	1.783	0.819	23	21	26	7.79	0.800	30.0	24	39	24	4	8.81	0.797	34.8	23	31	18	4
1.533	24	23	38	1.801	0.811	24	25	36	7.77	0.798	30.0	24	37	24	4	8.61	0.794	37.0	23	25	40	4
1.518	24	21	40	1.820	0.802	24	23	24	7.65	0.784	29.8	24	39	22	4	8.59	0.794	37.2	24	39	24	4
1.507	24	29	30	1.820	0.802	24	19	38	7.63	0.782	29.7	24	31	30	4	8.77	0.793	34.9	23	17	32	4
1.490	24	25	36	1.795	0.802	26	19	40	7.68	0.782	27.7	23	31	16	4	8.55	0.790	37.2	24	23	40	4
1.487	23	21	26	1.822	0.802	24	23	40	7.66	0.781	27.7	23	45	30	4	8.75	0.789	34.4	23	31	16	4
1.486	23	25	40	1.797	0.801	26	21	42	7.55	0.769	27.6	23	31	18	4	8.71	0.785	34.5	23	15	32	4
1.484	27	23	36	1.822	0.801	24	37	42	7.55	0.768	27.6	23	43	30	4	11.03	0.782	3.9	27	25	24	4
1.483	27	25	24	1.822	0.801	24	23	30	7.46	0.767	30.0	23	35	22	4	11.02	0.782	3.9	27	23	26	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.7 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 1,500.0 MWd/MTU

```
Cycle:                           24                      Core Average Exposure: MWd/MTU    22701.2
Exposure: MWd/MTU (GWd)      2000.0 ( 251.62 )
Delta E: MWd/MTU, (GWd)       500.0 (  62.90 )                 Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                      Top 24  0.168   3.919    19  0.361 0.609  55 40
Inlet Subcooling: Btu/lbm    -24.74                          23  0.308   9.025    20  0.312 0.325  57 42
Flow: Mlb/hr                  96.04 ( 98.00 %)               22  0.639  17.589    21  0.435 0.760  53 42
                                                             21  0.759  21.445    22  0.705 0.925  47 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              20  0.848  23.533    23  1.271 1.477  35 40
59                -- -- -- -- --               59            19  0.907  24.563    24  1.404 1.522  37 40
55          -- -- -- -- -- -- -- -- --         55            18  0.954  24.561    25  1.091 1.280  47 44
51       -- -- -- -- -- -- -- -- -- -- --      51            17  1.018  24.605    26  1.352 1.434  39 42
47    -- -- -- -- 24 -- -- -- 24 -- -- -- --   47            16  1.060  24.816    27  1.261 1.484  35 38
43    -- -- -- -- -- -- -- -- -- -- -- -- --   43            15  1.089  25.276    28  1.131 1.405  43 42
39 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 39           14  1.105  25.913
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35           13  1.160  25.256
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31           12  1.154  25.685
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27           11  1.158  26.358
23 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 23           10  1.172  26.826
19    -- -- -- -- -- -- -- -- -- -- -- -- --   19             9  1.200  27.025
15    -- -- -- -- 24 -- -- -- 24 -- -- -- --   15             8  1.246  27.564
11       -- -- -- -- -- -- -- -- -- -- --      11             7  1.289  28.002
 7          -- -- -- -- -- -- -- -- --          7             6  1.348  28.174
 3                -- -- -- -- --                3             5  1.394* 28.353*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               4  1.380  27.444
                                                              3  1.302  24.868
Control Rod Density: %         4.10                           2  1.038  19.078
                                                      Bottom  1  0.302   5.683
k-effective:                0.99922
Void Fraction:                0.387               % AXIAL TILT -16.520  -8.669
Core Delta-P: psia           21.744           AVG BOT 8ft/12ft  1.1498  1.0906
Core Plate Delta-P: psia     17.312
Coolant Temp: Deg-F           543.6
In Channel Flow: Mlb/hr       85.56   Active Channel Flow: Mlb/hr        81.73
Total Bypass Flow (%):         10.9   (of total core flow)
Total Water Rod Flow (%):       4.0   (of total core flow)
Source Convergence          0.00047
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.522	24	37	40	1.792	0.815	23	25	40	7.62	0.785	31.1	24	37	22	4	8.64	0.790	36.0	23	31	18	4
1.520	24	39	38	1.803	0.810	23	39	26	7.62	0.784	31.1	24	39	24	4	8.44	0.787	38.2	23	39	26	4
1.518	24	37	38	1.818	0.803	24	25	36	7.59	0.782	31.1	24	37	24	4	8.60	0.786	36.1	23	17	32	4
1.506	24	39	40	1.797	0.801	26	21	42	7.54	0.775	30.7	24	31	30	4	8.41	0.784	38.2	23	25	40	4
1.498	24	29	30	1.800	0.800	26	19	40	7.56	0.773	28.8	23	31	16	4	8.37	0.782	38.4	24	39	24	4
1.484	27	35	38	1.828	0.799	24	37	42	7.55	0.772	28.8	23	45	30	4	8.58	0.781	35.6	23	31	16	4
1.481	27	37	36	1.834	0.796	24	41	38	7.49	0.771	30.9	24	39	22	4	8.34	0.779	38.4	24	23	40	4
1.478	24	35	36	1.844	0.792	24	23	40	7.46	0.763	28.7	23	31	18	4	8.53	0.778	35.7	23	15	32	4
1.477	23	35	40	1.845	0.791	24	29	24	7.45	0.762	28.7	23	43	30	4	8.28	0.771	38.1	24	41	24	4
1.473	23	39	36	1.846	0.791	24	39	24	7.33	0.756	31.1	23	35	22	4	8.41	0.768	36.0	24	37	24	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.8 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 2,000.0 MWd/MTU

```
Cycle:                            24                  Core Average Exposure: MWd/MTU    23201.6
Exposure: MWd/MTU (GWd)       2500.0 ( 314.52 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )             Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)        N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                  Top 24  0.172   3.995    19  0.355 0.602  55 40
Inlet Subcooling: Btu/lbm     -24.74                      23  0.315   9.192    20  0.306 0.320  57 42
Flow: Mlb/hr                   96.04 ( 98.00 %)           22  0.650  17.939    21  0.429 0.753  53 42
                                                          21  0.769  21.861    22  0.698 0.918  47 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58           20  0.857  23.999    23  1.270 1.471  35 40
59                -- -- -- -- -- --                 59    19  0.916  25.061    24  1.400 1.511  37 40
55          -- -- -- -- -- -- -- -- -- --           55    18  0.963  25.086    25  1.086 1.277  47 44
51       -- -- -- -- -- -- -- -- -- -- -- --        51    17  1.027  25.144    26  1.364 1.443  31 28
47    -- -- -- -- 24 -- -- -- 24 -- -- -- --        47    16  1.070  25.364    27  1.270 1.489  35 38
43    -- -- -- -- -- -- -- -- -- -- -- -- --        43    15  1.098  25.839    28  1.135 1.413  43 42
39 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 39       14  1.110  26.483
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35       13  1.161  25.824
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31       12  1.151  26.250
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27       11  1.153  26.924
23 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 23       10  1.165  27.399
19    -- -- -- -- -- -- -- -- -- -- -- -- --        19     9  1.193  27.605
15    -- -- -- -- 24 -- -- -- 24 -- -- -- --        15     8  1.239  28.156
11       -- -- -- -- -- -- -- -- -- -- -- --        11     7  1.282  28.615
 7          -- -- -- -- -- -- -- -- -- --            7     6  1.341  28.814
 3                -- -- -- -- -- --                  3     5  1.386* 29.014*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            4  1.368  28.099
                                                           3  1.288  25.485
Control Rod Density: %          4.10                       2  1.028  19.570
                                                   Bottom  1  0.300   5.827
k-effective:                 0.99918
Void Fraction:                 0.384             % AXIAL TILT -15.777  -8.745
Core Delta-P: psia            21.734         AVG BOT 8ft/12ft  1.1457  1.0912
Core Plate Delta-P: psia      17.302
Coolant Temp: Deg-F            543.6
In Channel Flow: Mlb/hr        85.57  Active Channel Flow: Mlb/hr        81.75
Total Bypass Flow (%):          10.9  (of total core flow)
Total Water Rod Flow (%):        4.0  (of total core flow)
Source Convergence           0.00044
```

```
                           Top Ten Thermal Limits Summary - Sorted by Margin

     Power                  MCPR                        APLHGR                              LHGR
Value FT   IR JR      Value Margin FT   IR JR     Value Margin Exp. FT   IR JR  K     Value Margin Exp. FT   IR JR  K
1.511  24 37 40       1.804  0.809  23 25 40      7.45  0.768 32.2  24 37 22  4      8.46  0.782 37.2  23 31 18  4
1.510  24 39 38       1.815  0.805  23 39 26      7.44  0.768 32.2  24 39 24  4      8.42  0.778 37.3  23 17 32  4
1.507  24 37 38       1.791  0.804  26 21 42      7.44  0.767 31.8  24 31 30  4      8.22  0.776 39.4  23 39 26  4
1.497  24 39 40       1.794  0.803  26 19 40      7.42  0.765 32.2  24 37 24  4      8.40  0.773 36.8  23 31 16  4
1.494  24 31 32       1.825  0.800  24 25 36      7.43  0.764 29.8  23 31 16  4      8.20  0.773 39.3  23 25 40  4
1.489  27 35 38       1.836  0.795  24 37 42      7.42  0.763 29.8  23 45 30  4      8.16  0.770 39.5  24 39 24  4
1.486  27 37 36       1.842  0.793  24 41 38      7.34  0.757 32.0  24 39 22  4      8.35  0.769 36.8  23 15 32  4
1.472  24 35 36       1.856  0.786  24 29 24      7.35  0.756 29.8  23 31 18  4      8.12  0.767 39.6  24 23 40  4
1.471  23 35 40       1.857  0.786  24 23 40      7.35  0.755 29.7  23 43 30  4      8.07  0.760 39.2  24 41 24  4
1.467  23 39 36       1.860  0.785  24 39 24      7.20  0.744 32.1  23 35 22  4      8.32  0.758 35.7  24 31 30  4

  * LHGR calculated with pin-power reconstruction
  * CPR calculated with pin-power reconstruction & CPR limit type 3
```

Figure A.9 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 2,500.0 MWd/MTU

```
Cycle:                            24              Core Average Exposure: MWd/MTU    23701.7
Exposure: MWd/MTU (GWd)       3000.0 ( 377.42 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )
Power: MWt                    2957.0 (100.00 %)
Core Pressure: psia           1015.0
Inlet Subcooling: Btu/lbm     -24.74
Flow: Mlb/hr                   96.04 ( 98.00 %)

    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                -- -- -- -- --                59
55          -- -- -- -- -- -- -- -- --          55
51       -- -- -- -- -- -- -- -- -- -- --       51
47    -- -- -- -- 24 -- -- -- 24 -- -- -- --    47
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43
39 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 23
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19
15    -- -- -- -- 24 -- -- -- 24 -- -- -- --    15
11       -- -- -- -- -- -- -- -- -- -- --       11
 7          -- -- -- -- -- -- -- -- --           7
 3                -- -- -- -- --                 3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58

Control Rod Density: %          4.10

k-effective:                 0.99907
Void Fraction:                 0.382
Core Delta-P: psia            21.729
Core Plate Delta-P: psia      17.297
Coolant Temp: Deg-F            543.6
In Channel Flow: Mlb/hr        85.58   Active Channel Flow: Mlb/hr        81.76
Total Bypass Flow (%):          10.9   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00047
```

	Axial Profile	
N(PRA)	Power	Exposure
Top 24	0.175	4.072
23	0.320	9.363
22	0.659	18.295
21	0.777	22.283
20	0.863	24.469
19	0.921	25.563
18	0.969	25.614
17	1.034	25.688
16	1.077	25.917
15	1.104	26.405
14	1.113	27.055
13	1.161	26.393
12	1.149	26.813
11	1.149	27.488
10	1.160	27.969
9	1.187	28.182
8	1.233	28.745
7	1.276	29.224
6	1.338	29.452
5	1.381*	29.673*
4	1.360	28.749
3	1.277	26.096
2	1.019	20.057
Bottom 1	0.299	5.971
% AXIAL TILT	-15.231	-8.805
AVG BOT 8ft/12ft	1.1426	1.0918

Edit	Radial Power			
Zone	Avg.	Max.	IR	JR
19	0.350	0.595	55	40
20	0.301	0.314	57	42
21	0.423	0.747	53	42
22	0.692	0.910	47	48
23	1.269	1.465	35	40
24	1.395	1.500	37	40
25	1.081	1.274	47	44
26	1.375	1.453	31	28
27	1.279	1.493	35	38
28	1.140	1.420	43	42

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR						LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.500	24	37	40	1.788	0.805	26	21	42	7.34	0.759	32.8	24	31	30	4	8.32	0.777	38.4	23	31	18	4
1.499	24	39	38	1.791	0.804	26	19	40	7.34	0.757	30.9	23	31	16	4	8.27	0.773	38.4	23	17	32	4
1.496	24	37	38	1.818	0.803	23	25	40	7.33	0.756	30.9	23	45	30	4	8.25	0.767	37.9	23	31	16	4
1.493	27	35	38	1.828	0.798	23	39	26	7.28	0.753	33.2	24	37	22	4	8.02	0.765	40.5	23	39	26	4
1.490	27	37	36	1.834	0.796	24	25	36	7.28	0.752	33.2	24	39	24	4	8.21	0.764	38.0	23	15	32	4
1.488	24	31	32	1.846	0.791	24	37	42	7.28	0.750	30.8	23	31	18	4	8.00	0.762	40.5	23	25	40	4
1.488	24	39	40	1.852	0.788	24	41	38	7.25	0.750	33.2	24	37	24	4	7.95	0.758	40.6	24	39	24	4
1.466	24	35	36	1.838	0.783	26	27	36	7.27	0.750	30.8	23	43	30	4	7.92	0.755	40.7	24	23	40	4
1.465	23	35	40	1.839	0.783	26	35	28	7.19	0.743	33.0	24	39	22	4	8.18	0.753	36.8	24	31	30	4
1.460	23	39	36	1.840	0.783	26	35	20	7.11	0.735	32.5	23	31	14	4	7.88	0.750	40.4	24	41	24	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.10 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 3,000.0 MWd/MTU

```
Cycle:                           24
Exposure: MWd/MTU (GWd)      3500.0 ( 440.33 )
Delta E: MWd/MTU, (GWd)       500.0 (  62.90 )
Power: MWt                   2957.0 (100.00 %)
Core Pressure: psia          1015.0
Inlet Subcooling: Btu/lbm    -24.74
Flow: Mlb/hr                  96.04 ( 98.00 %)

    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                -- -- -- -- --                59
55          -- -- -- -- -- -- -- -- --          55
51       -- -- -- -- -- -- -- -- -- -- --       51
47    -- -- -- -- 24 -- -- -- 24 -- -- -- --    47
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43
39 -- -- -- 24 -- -- --  8 -- -- -- 24 -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 23
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19
15    -- -- -- -- 24 -- -- -- 24 -- -- -- --    15
11       -- -- -- -- -- -- -- -- -- -- --       11
 7          -- -- -- -- -- -- -- -- --           7
 3                -- -- -- -- --                 3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58

Control Rod Density: %         4.12

k-effective:                0.99877
Void Fraction:                0.382
Core Delta-P: psia           21.738
Core Plate Delta-P: psia     17.306
Coolant Temp: Deg-F           543.5
In Channel Flow: Mlb/hr       85.58   Active Channel Flow: Mlb/hr        81.76
Total Bypass Flow (%):         10.9   (of total core flow)
Total Water Rod Flow (%):       4.0   (of total core flow)
Source Convergence          0.00041
```

Core Average Exposure: MWd/MTU 24201.2

	Axial Profile	
N(PRA)	Power	Exposure
Top 24	0.176	4.150
23	0.321	9.535
22	0.660	18.654
21	0.775	22.706
20	0.858	24.940
19	0.918	26.067
18	0.967	26.144
17	1.035	26.233
16	1.080	26.472
15	1.108	26.974
14	1.115	27.628
13	1.160	26.962
12	1.147	27.376
11	1.147	28.051
10	1.158	28.537
9	1.186	28.757
8	1.232	29.332
7	1.277	29.832
6	1.341	30.089
5	1.385*	30.331*
4	1.361	29.397
3	1.276	26.704
2	1.017	20.543
Bottom 1	0.300	6.114
% AXIAL TILT	-15.213	-8.856
AVG BOT 8ft/12ft	1.1431	1.0922

Edit	Radial Power			
Zone	Avg.	Max.	IR	JR
19	0.346	0.590	55	40
20	0.297	0.310	57	42
21	0.417	0.742	53	42
22	0.687	0.903	13	14
23	1.267	1.458	35	22
24	1.387	1.489	37	22
25	1.077	1.272	47	44
26	1.386	1.462	31	28
27	1.289	1.498	25	24
28	1.147	1.428	43	20

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.498	27	25	24	1.788	0.805	26	39	20	7.28	0.754	33.9	24	31	30	4	8.22	0.776	39.6	23	31	18	4
1.494	27	23	26	1.790	0.804	26	41	22	7.29	0.753	31.9	23	31	16	4	8.18	0.772	39.6	23	17	32	4
1.489	24	37	22	1.844	0.792	23	39	26	7.28	0.752	31.9	23	45	30	4	8.15	0.766	39.1	23	31	16	4
1.487	24	39	24	1.849	0.790	23	25	22	7.24	0.748	31.8	23	31	18	4	8.11	0.763	39.1	23	15	32	4
1.484	24	37	24	1.852	0.788	24	35	26	7.24	0.748	31.8	23	43	30	4	7.86	0.757	41.6	23	39	26	4
1.480	24	29	30	1.833	0.786	26	35	20	7.15	0.741	34.2	24	37	22	4	7.84	0.755	41.6	23	25	40	4
1.478	24	39	22	1.863	0.784	24	37	20	7.15	0.741	34.2	24	39	24	4	7.85	0.753	41.2	24	31	32	5
1.462	26	31	28	1.839	0.783	26	35	28	7.12	0.738	34.2	24	37	24	4	7.78	0.750	41.7	24	39	24	4
1.461	26	39	20	1.842	0.782	26	27	26	7.08	0.734	33.5	23	31	14	4	7.75	0.748	41.8	24	23	40	4
1.460	26	41	22	1.867	0.782	24	41	38	7.08	0.733	34.0	24	39	22	4	7.77	0.744	41.1	23	47	34	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.11 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 3,500.0 MWd/MTU

```
Cycle:                          24                    Core Average Exposure: MWd/MTU    24526.7
Exposure: MWd/MTU (GWd)      3825.0 ( 481.22 )
Delta E: MWd/MTU, (GWd)       325.0 (  40.89 )              Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)          N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                    Top 24  0.178   4.201    19  0.343 0.586  55 40
Inlet Subcooling: Btu/lbm    -24.74                        23  0.324   9.648    20  0.293 0.306  57 42
Flow: Mlb/hr                  96.04 ( 98.00 %)             22  0.664  18.888    21  0.413 0.737  53 42
                                                           21  0.778  22.981    22  0.683 0.898  13 14
   2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             20  0.860  25.245    23  1.266 1.453  35 22
59                -- -- -- -- --             59            19  0.920  26.393    24  1.384 1.482  37 22
55          -- -- -- -- -- -- -- -- --       55            18  0.969  26.489    25  1.074 1.270  47 44
51       -- -- -- -- -- -- -- -- -- -- --    51            17  1.037  26.588    26  1.393 1.468  31 28
47    -- -- -- -- 24 -- -- -- 24 -- -- -- --  47           16  1.084  26.834    27  1.295 1.500  25 24
43    -- -- -- -- -- -- -- -- -- -- -- -- --  43           15  1.110  27.345    28  1.150 1.433  43 20
39 -- -- -- 24 -- -- --  8 -- -- -- 24 -- -- -- 39         14  1.117  28.002
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35         13  1.160  27.331
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31         12  1.145  27.741
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27         11  1.144  28.416
23 -- -- -- 24 -- -- -- 10 -- -- -- 24 -- -- -- 23         10  1.155  28.906
19    -- -- -- -- -- -- -- -- -- -- -- -- --  19            9  1.182  29.130
15    -- -- -- -- 24 -- -- -- 24 -- -- -- --  15            8  1.229  29.714
11       -- -- -- -- -- -- -- -- -- -- --    11             7  1.275  30.228
 7          -- -- -- -- -- -- -- -- --        7             6  1.341  30.505
 3                -- -- -- -- --              3             5  1.385* 30.760*
   2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              4  1.359  29.819
                                                            3  1.272  27.099
Control Rod Density: %          4.12                        2  1.012  20.857
                                                     Bottom 1  0.299   6.208
k-effective:                 0.99872
Void Fraction:                 0.381              % AXIAL TILT -14.993  -8.887
Core Delta-P: psia            21.738          AVG BOT 8ft/12ft  1.1419  1.0925
Core Plate Delta-P: psia      17.306
Coolant Temp: Deg-F            543.5
In Channel Flow: Mlb/hr        85.58   Active Channel Flow: Mlb/hr        81.77
Total Bypass Flow (%):          10.9   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00042
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.500	27	25	24	1.788	0.805	26	39	20	7.23	0.750	34.5	24	31	30	4	8.13	0.773	40.3	23	31	18	4
1.496	27	23	26	1.791	0.804	26	41	22	7.23	0.748	32.6	23	31	16	4	8.09	0.769	40.3	23	17	32	4
1.482	24	37	22	1.855	0.787	23	39	26	7.22	0.747	32.6	23	45	30	4	8.06	0.763	39.8	23	31	16	4
1.480	24	39	24	1.830	0.787	26	35	20	7.20	0.745	32.5	23	31	18	4	8.02	0.760	39.8	23	15	32	4
1.477	24	37	24	1.859	0.785	23	25	22	7.19	0.744	32.5	23	43	30	4	7.81	0.754	41.9	24	31	32	5
1.476	24	29	30	1.860	0.785	24	35	26	7.06	0.732	34.9	24	37	22	4	7.75	0.751	42.3	23	39	26	4
1.471	24	39	22	1.838	0.783	26	35	28	7.05	0.732	34.9	24	39	24	4	7.72	0.748	42.3	23	25	40	4
1.468	26	31	28	1.841	0.782	26	27	26	7.04	0.731	34.1	23	31	14	4	7.61	0.745	43.4	23	47	34	5
1.466	26	39	20	1.842	0.782	26	31	28	7.03	0.730	34.8	24	37	24	4	7.67	0.744	42.4	24	39	24	4
1.465	26	41	22	1.842	0.782	26	19	26	7.02	0.729	34.2	23	47	30	4	7.60	0.743	43.3	23	33	48	5

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.12 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 3,825.0 MWd/MTU

```
Cycle:                           24                      Core Average Exposure: MWd/MTU    24527.2
Exposure: MWd/MTU (GWd)      3826.0 ( 481.34 )
Delta E: MWd/MTU, (GWd)         1.0 (   0.13 )                Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)          N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                      Top 24  0.168   4.201   19  0.340 0.590  55 40
Inlet Subcooling: Btu/lbm    -24.74                          23  0.305   9.649   20  0.294 0.307   3 42
Flow: Mlb/hr                  96.04 ( 98.00 %)               22  0.624  18.888   21  0.414 0.749  53 42
                                                             21  0.729  22.981   22  0.680 0.910  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              20  0.803  25.246   23  1.311 1.476  21 32
59                -- -- -- -- -- --               59         19  0.846  26.394   24  1.308 1.493  23 32
55          -- -- -- -- -- -- -- -- -- --         55         18  0.882  26.490   25  1.086 1.311  47 44
51       -- -- -- -- -- -- -- -- -- -- -- --      51         17  0.937  26.589   26  1.370 1.466  27 36
47    -- -- -- -- -- -- 20 -- -- -- -- -- --      47         16  0.974  26.835   27  1.336 1.408  23 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --      43         15  0.994  27.346   28  1.157 1.449  17 42
39 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 39           14  1.000  28.003
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35           13  1.061  27.333
31 -- -- -- 20 -- -- --  8 -- -- -- 20 -- -- -- 31           12  1.094  27.743
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27           11  1.129  28.417
23 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 23           10  1.169  28.907
19    -- -- -- -- -- -- -- -- -- -- -- -- --      19          9  1.222  29.131
15    -- -- -- -- -- -- 20 -- -- -- -- -- --      15          8  1.289  29.715
11       -- -- -- -- -- -- -- -- -- -- -- --      11          7  1.356  30.229
 7          -- -- -- -- -- -- -- -- -- --          7          6  1.441  30.506
 3                -- -- -- -- -- --                3          5  1.510* 30.762*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               4  1.509  29.820
                                                              3  1.442  27.100
Control Rod Density: %         3.58                           2  1.167  20.858
                                                      Bottom  1  0.348   6.208
k-effective:                0.99864
Void Fraction:                0.395              % AXIAL TILT -22.299  -8.887
Core Delta-P: psia           21.909          AVG BOT 8ft/12ft  1.1691  1.0925
Core Plate Delta-P: psia     17.479
Coolant Temp: Deg-F           543.8
In Channel Flow: Mlb/hr       85.48   Active Channel Flow: Mlb/hr        81.62
Total Bypass Flow (%):         11.0   (of total core flow)
Total Water Rod Flow (%):       4.0   (of total core flow)
Source Convergence          0.00028
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.493	24	23	32	1.848	0.790	23	23	34	7.83	0.809	31.8	23	21	16	4	8.73	0.817	38.7	23	39	16	4
1.492	24	29	38	1.855	0.787	23	33	24	7.81	0.807	31.9	23	45	22	4	8.69	0.815	38.8	23	15	40	4
1.476	23	21	32	1.873	0.780	24	19	34	7.69	0.795	31.8	23	31	22	4	8.48	0.812	40.9	23	39	18	4
1.476	23	29	40	1.881	0.776	24	33	20	7.68	0.793	31.9	23	39	30	4	8.45	0.809	41.1	23	17	40	4
1.466	26	27	36	1.863	0.773	26	21	34	7.63	0.792	34.8	24	43	18	4	8.52	0.800	38.9	23	31	40	4
1.465	26	25	34	1.864	0.773	26	27	36	7.56	0.784	33.3	23	21	18	4	8.52	0.800	39.0	23	39	32	4
1.462	26	27	40	1.864	0.772	26	27	40	7.56	0.783	32.8	23	39	14	4	8.32	0.800	41.4	23	23	16	4
1.462	26	21	34	1.893	0.771	24	25	32	7.55	0.782	33.5	23	43	22	4	8.38	0.799	40.6	23	47	40	4
1.459	23	23	34	1.867	0.771	26	19	32	7.54	0.781	34.4	24	31	24	4	8.39	0.799	40.5	23	39	48	4
1.458	23	27	38	1.868	0.771	26	29	42	7.53	0.781	33.9	23	37	16	4	11.25	0.798	10.1	28	19	18	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.13 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 3,826.0 MWd/MTU

```
Cycle:                          24
Exposure: MWd/MTU (GWd)      4000.0 ( 503.23 )
Delta E: MWd/MTU, (GWd)       174.0 (  21.89 )
Power: MWt                   2957.0 (100.00 %)
Core Pressure: psia          1015.0
Inlet Subcooling: Btu/lbm    -24.74
Flow: Mlb/hr                  96.04 ( 98.00 %)

    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                -- -- -- -- --                59
55          -- -- -- -- -- -- -- -- --          55
51       -- -- -- -- -- -- -- -- -- -- --       51
47    -- -- -- -- -- -- 20 -- -- -- -- -- --    47
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43
39 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- 20 -- -- --  8 -- -- -- 20 -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 23
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19
15    -- -- -- -- -- -- 20 -- -- -- -- -- --    15
11       -- -- -- -- -- -- -- -- -- -- --       11
 7          -- -- -- -- -- -- -- -- --           7
 3                -- -- -- -- --                 3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58

Control Rod Density: %         3.58

k-effective:                0.99857
Void Fraction:                0.395
Core Delta-P: psia           21.909
Core Plate Delta-P: psia     17.478
Coolant Temp: Deg-F           543.8
In Channel Flow: Mlb/hr       85.48   Active Channel Flow: Mlb/hr        81.62
Total Bypass Flow (%):         11.0   (of total core flow)
Total Water Rod Flow (%):       4.0   (of total core flow)
Source Convergence          0.00028
```

Core Average Exposure: MWd/MTU   24701.7

N(PRA)	Axial Profile Power	Axial Profile Exposure
Top 24	0.168	4.227
23	0.307	9.706
22	0.627	19.007
21	0.731	23.120
20	0.804	25.399
19	0.847	26.556
18	0.884	26.658
17	0.939	26.761
16	0.976	27.010
15	0.996	27.524
14	1.001	28.182
13	1.061	27.514
12	1.093	27.929
11	1.128	28.610
10	1.168	29.106
9	1.220	29.337
8	1.288	29.928
7	1.354	30.454
6	1.441	30.745
5	1.509*	31.012*
4	1.508	30.070
3	1.439	27.339
2	1.164	21.051
Bottom 1	0.347	6.266
% AXIAL TILT	-22.166	-8.954
AVG BOT 8ft/12ft	1.1683	1.0928

Edit Zone	Radial Power Avg.	Radial Power Max.	IR	JR
19	0.338	0.587	55	40
20	0.292	0.306	3	42
21	0.412	0.747	53	42
22	0.677	0.907	13	48
23	1.311	1.474	21	32
24	1.307	1.490	23	32
25	1.084	1.309	47	44
26	1.375	1.469	27	36
27	1.339	1.410	23	36
28	1.159	1.451	17	42

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.490	24	23	32	1.851	0.789	23	23	34	7.79	0.806	32.2	23	21	16	4	8.67	0.815	39.1	23	39	16	4
1.489	24	29	38	1.858	0.786	23	33	24	7.77	0.804	32.3	23	45	22	4	8.63	0.812	39.2	23	15	40	4
1.474	23	21	32	1.873	0.779	24	19	34	7.64	0.790	32.2	23	31	22	4	8.44	0.811	41.4	23	39	18	4
1.473	23	29	40	1.881	0.776	24	27	42	7.61	0.790	35.2	24	43	18	4	8.40	0.808	41.5	23	17	40	4
1.469	26	27	36	1.860	0.774	26	21	34	7.63	0.789	32.3	23	39	30	4	8.47	0.798	39.4	23	31	40	4
1.468	26	25	34	1.862	0.773	26	27	40	7.53	0.781	33.7	23	21	18	4	8.46	0.798	39.4	23	39	32	4
1.466	26	27	40	1.862	0.773	26	27	36	7.53	0.781	33.2	23	39	14	4	8.33	0.798	41.0	23	47	40	4
1.466	26	21	34	1.865	0.772	26	19	32	7.52	0.780	33.8	23	43	22	4	8.34	0.798	40.9	23	39	48	4
1.456	23	23	34	1.865	0.772	26	29	42	7.49	0.778	34.3	23	37	16	4	8.27	0.798	41.8	23	23	16	4
1.456	23	27	38	1.868	0.771	26	35	34	7.51	0.778	33.3	23	47	22	4	11.23	0.797	10.7	28	19	18	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.14 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 4,000.0 MWd/MTU

```
Cycle:                          24                   Core Average Exposure: MWd/MTU    25201.7
Exposure: MWd/MTU (GWd)      4500.0 ( 566.14 )
Delta E: MWd/MTU, (GWd)       500.0 (  62.90 )                Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)          N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                    Top 24  0.170   4.302    19  0.334 0.581  55 40
Inlet Subcooling: Btu/lbm    -24.74                        23  0.310   9.872    20  0.288 0.301   3 42
Flow: Mlb/hr                  96.04 ( 98.00 %)             22  0.633  19.349    21  0.407 0.741  53 42
                                                           21  0.736  23.519    22  0.672 0.900  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            20  0.808  25.840    23  1.308 1.462  21 32
59                -- -- -- -- -- --               59       19  0.852  27.020    24  1.302 1.473  23 32
55          -- -- -- -- -- -- -- -- -- --         55       18  0.889  27.143    25  1.079 1.306  47 44
51       -- -- -- -- -- -- -- -- -- -- -- --      51       17  0.945  27.258    26  1.385 1.473  27 36
47    -- -- -- -- -- -- -- 20 -- -- -- -- -- --   47       16  0.983  27.514    27  1.348 1.414  23 36
43    -- -- -- -- -- -- -- -- -- -- -- -- -- --   43       15  1.003  28.039    28  1.164 1.459  17 42
39 -- -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 39     14  1.006  28.699
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35     13  1.064  28.034
31 -- -- -- 20 -- -- --  8 -- -- -- 20 -- -- -- -- 31     12  1.093  28.465
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27     11  1.126  29.162
23 -- -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 23     10  1.165  29.678
19    -- -- -- -- -- -- -- -- -- -- -- -- -- --   19        9  1.216  29.927
15    -- -- -- -- -- -- -- 20 -- -- -- -- -- --   15        8  1.284  30.541
11       -- -- -- -- -- -- -- -- -- -- -- --      11        7  1.351  31.098
 7          -- -- -- -- -- -- -- -- -- --          7        6  1.439  31.431
 3                -- -- -- -- -- --                3        5  1.507* 31.730*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             4  1.500  30.787
                                                            3  1.426  28.022
Control Rod Density: %         3.58                         2  1.150  21.603
                                                    Bottom  1  0.343   6.431
k-effective:                0.99851
Void Fraction:                0.393             % AXIAL TILT -21.664  -9.134
Core Delta-P: psia           21.904         AVG BOT 8ft/12ft  1.1660  1.0937
Core Plate Delta-P: psia     17.474
Coolant Temp: Deg-F           543.7
In Channel Flow: Mlb/hr       85.48   Active Channel Flow: Mlb/hr        81.63
Total Bypass Flow (%):         11.0   (of total core flow)
Total Water Rod Flow (%):       4.0   (of total core flow)
Source Convergence          0.00029
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR						LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.473	24	23	32	1.869	0.781	23	23	34	7.67	0.795	33.3	23	21	16	4	8.49	0.807	40.3	23	39	16	4
1.473	26	27	36	1.876	0.778	23	33	24	7.65	0.793	33.4	23	45	22	4	8.46	0.805	40.4	23	15	40	4
1.473	26	25	34	1.881	0.776	24	19	34	7.54	0.785	36.3	24	43	18	4	8.29	0.805	42.5	23	39	18	4
1.473	24	29	38	1.859	0.775	26	21	34	9.08	0.776	9.5	28	41	18	4	8.25	0.802	42.6	23	17	40	4
1.472	26	27	40	1.860	0.774	26	27	40	9.07	0.775	9.5	28	43	20	4	8.31	0.796	41.0	24	17	18	4
1.472	26	21	34	1.889	0.773	24	27	42	7.47	0.774	33.2	23	31	22	4	8.20	0.794	42.2	23	47	40	4
1.462	26	29	42	1.863	0.773	26	19	32	7.43	0.773	34.8	23	21	18	4	8.21	0.794	42.0	23	39	48	4
1.462	26	19	32	1.864	0.773	26	29	42	7.46	0.773	33.3	23	39	30	4	11.16	0.791	12.1	28	19	18	4
1.462	23	21	32	1.865	0.772	26	27	36	7.42	0.772	34.9	23	43	22	4	8.11	0.791	43.0	23	23	16	4
1.461	23	29	40	1.870	0.770	26	35	28	7.43	0.771	34.2	23	39	14	4	11.16	0.791	12.1	28	17	20	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.15 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 4,500.0 MWd/MTU

```
Cycle:                            24          Core Average Exposure: MWd/MTU    25701.2
Exposure: MWd/MTU (GWd)       5000.0 ( 629.04 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )
Power: MWt                    2957.0 (100.00 %)
Core Pressure: psia           1015.0
Inlet Subcooling: Btu/lbm     -24.74
Flow: Mlb/hr                   96.04 ( 98.00 %)

    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                -- -- -- -- --                59
55          -- -- -- -- -- -- -- -- --          55
51       -- -- -- -- -- -- -- -- -- -- --       51
47    -- -- -- -- -- -- 20 -- -- -- -- -- --    47
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43
39 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- 20 -- -- --  6 -- -- -- 20 -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- -- 23
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19
15    -- -- -- -- -- -- 20 -- -- -- -- -- --    15
11       -- -- -- -- -- -- -- -- -- -- --       11
 7          -- -- -- -- -- -- -- -- --           7
 3                -- -- -- -- --                 3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58

Control Rod Density: %          3.60

k-effective:                 0.99834
Void Fraction:                 0.392
Core Delta-P: psia            21.910
Core Plate Delta-P: psia      17.480
Coolant Temp: Deg-F            543.7
In Channel Flow: Mlb/hr        85.48   Active Channel Flow: Mlb/hr        81.63
Total Bypass Flow (%):          11.0   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00041
```

	Axial Profile	
N(PRA)	Power	Exposure
Top 24	0.171	4.378
23	0.311	10.039
22	0.633	19.693
21	0.734	23.920
20	0.807	26.281
19	0.852	27.486
18	0.890	27.630
17	0.948	27.756
16	0.988	28.020
15	1.008	28.556
14	1.010	29.217
13	1.064	28.555
12	1.092	29.000
11	1.123	29.714
10	1.161	30.248
9	1.213	30.516
8	1.282	31.153
7	1.351	31.742
6	1.442	32.116
5	1.510*	32.448*
4	1.500	31.502
3	1.424	28.701
2	1.145	22.151
Bottom 1	0.343	6.596
% AXIAL TILT	-21.542	-9.302
AVG BOT 8ft/12ft	1.1659	1.0945

Edit	Radial Power			
Zone	Avg.	Max.	IR	JR
19	0.330	0.577	55	40
20	0.284	0.297	3	42
21	0.402	0.737	53	42
22	0.668	0.895	13	48
23	1.304	1.447	21	32
24	1.293	1.453	23	32
25	1.077	1.305	13	44
26	1.394	1.475	27	40
27	1.359	1.424	15	36
28	1.172	1.469	17	42

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.475	26	27	40	1.865	0.772	26	19	32	7.59	0.789	34.4	23	21	16	4	8.37	0.805	41.4	23	39	16	4
1.475	26	21	34	1.866	0.772	26	29	42	7.58	0.787	34.5	23	45	22	4	8.18	0.803	43.7	23	39	18	4
1.471	26	27	36	1.894	0.771	23	23	34	7.52	0.784	37.3	24	43	18	4	8.34	0.802	41.6	23	15	40	4
1.471	26	25	34	1.894	0.771	24	19	34	9.14	0.781	10.6	28	41	18	4	8.15	0.801	43.8	23	17	40	4
1.469	28	17	42	1.869	0.770	26	21	34	9.14	0.781	10.6	28	43	20	4	8.27	0.800	42.1	24	17	18	4
1.469	26	19	32	1.871	0.770	26	27	40	9.09	0.777	9.9	26	41	16	5	8.13	0.796	43.3	23	47	40	4
1.469	26	29	42	1.901	0.768	23	33	24	9.09	0.777	9.9	26	45	20	5	8.14	0.796	43.1	23	39	48	4
1.468	28	19	44	1.902	0.767	24	27	42	7.38	0.769	35.8	23	21	18	4	11.17	0.792	13.4	28	19	18	4
1.453	24	23	32	1.880	0.766	26	27	36	7.38	0.768	35.3	23	39	14	4	11.16	0.791	13.3	28	17	20	4
1.452	24	29	38	1.885	0.764	26	35	34	7.36	0.767	35.9	23	43	22	4	8.01	0.790	44.1	23	23	16	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.16 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 5,000.0 MWd/MTU

```
Cycle:                            24
Exposure: MWd/MTU (GWd)       5500.0 ( 691.94 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )
Power: MWt                    2957.0 (100.00 %)
Core Pressure: psia           1015.0
Inlet Subcooling: Btu/lbm     -24.74
Flow: Mlb/hr                   96.04 ( 98.00 %)

    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                -- -- -- -- -- --                59
55          -- -- -- -- -- -- -- -- --          55
51       -- -- -- -- -- -- -- -- -- -- --       51
47    -- -- -- -- -- -- 20 -- -- -- -- -- --    47
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43
39 -- -- -- -- -- 10 -- -- --  8 -- -- -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- 20 -- -- --  0 -- -- -- 20 -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- -- --  8 -- -- -- 10 -- -- -- -- -- 23
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19
15    -- -- -- -- -- -- 20 -- -- -- -- -- --    15
11       -- -- -- -- -- -- -- -- -- -- --       11
 7          -- -- -- -- -- -- -- -- --          7
 3                -- -- -- -- -- --                3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58

Control Rod Density: %          3.72

k-effective:                 0.99792
Void Fraction:                 0.393
Core Delta-P: psia            21.938
Core Plate Delta-P: psia      17.507
Coolant Temp: Deg-F            543.7
In Channel Flow: Mlb/hr        85.47   Active Channel Flow: Mlb/hr        81.62
Total Bypass Flow (%):          11.0   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00040
```

Core Average Exposure: MWd/MTU   26201.2

N(PRA)	Axial Profile Power	Axial Profile Exposure
Top 24	0.166	4.453
23	0.303	10.206
22	0.618	20.037
21	0.717	24.320
20	0.789	26.723
19	0.839	27.952
18	0.881	28.117
17	0.943	28.256
16	0.986	28.529
15	1.009	29.075
14	1.012	29.737
13	1.065	29.077
12	1.092	29.535
11	1.124	30.264
10	1.162	30.818
9	1.215	31.104
8	1.286	31.764
7	1.359	32.386
6	1.455	32.803
5	1.526*	33.167*
4	1.515	32.216
3	1.437	29.380
2	1.155	22.697
Bottom 1	0.347	6.761
% AXIAL TILT	-22.265	-9.462
AVG BOT 8ft/12ft	1.1715	1.0954

Edit Zone	Radial Power Avg.	Radial Power Max.	IR	JR
19	0.328	0.573	55	40
20	0.281	0.294	3	42
21	0.399	0.735	7	42
22	0.666	0.893	13	48
23	1.300	1.431	21	32
24	1.278	1.431	23	32
25	1.076	1.309	13	44
26	1.401	1.478	21	34
27	1.371	1.440	15	36
28	1.183	1.483	17	42

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.483	28	17	42	1.873	0.769	26	19	32	9.30	0.794	11.0	26	19	16	5	8.30	0.812	43.3	24	17	18	4
1.482	28	19	44	1.874	0.768	26	29	42	9.29	0.794	11.1	26	45	42	5	8.29	0.810	43.2	23	39	16	4
1.478	26	21	34	1.912	0.763	24	19	34	9.29	0.794	11.8	28	19	18	4	8.15	0.809	44.8	23	39	18	4
1.478	26	27	40	1.887	0.763	26	21	34	9.28	0.793	11.8	28	43	20	4	8.25	0.808	43.3	23	15	40	4
1.475	26	19	32	1.887	0.763	26	27	40	7.59	0.791	35.4	23	21	16	4	8.12	0.807	44.9	23	17	40	4
1.475	26	29	42	1.920	0.760	24	27	42	7.56	0.790	38.4	24	43	18	4	8.14	0.805	44.4	23	47	40	4
1.468	26	27	36	1.923	0.759	23	23	34	7.57	0.788	35.6	23	45	22	4	8.15	0.805	44.3	23	39	48	4
1.467	26	25	34	1.897	0.759	26	27	36	9.11	0.779	11.2	27	17	16	4	11.27	0.799	14.8	28	19	18	4
1.446	26	17	38	1.903	0.757	26	35	28	9.11	0.779	11.3	27	45	18	4	11.27	0.799	14.8	28	17	20	4
1.446	26	23	44	1.929	0.757	23	33	24	7.41	0.773	36.3	23	21	14	4	7.98	0.795	45.2	23	23	16	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.17 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 5,500.0 MWd/MTU

```
Cycle:                            24                       Core Average Exposure: MWd/MTU    26701.2
Exposure: MWd/MTU (GWd)       6000.0 ( 754.85 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )                Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)          N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                    Top 24  0.165   4.527    19  0.324 0.570  55 40
Inlet Subcooling: Btu/lbm     -24.74                        23  0.300  10.369    20  0.277 0.290   3 42
Flow: Mlb/hr                   96.04 ( 98.00 %)             22  0.612  20.373    21  0.395 0.731   7 42
                                                            21  0.708  24.711    22  0.662 0.886  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             20  0.776  27.154    23  1.295 1.415  21 32
59                -- -- -- -- -- --                59       19  0.830  28.410    24  1.267 1.410  23 32
55          -- -- -- -- -- -- -- -- --             55       18  0.876  28.599    25  1.074 1.306  13 44
51       -- -- -- -- -- -- -- -- -- -- -- --       51       17  0.940  28.753    26  1.409 1.479  19 32
47    -- -- -- -- -- -- 20 -- -- -- -- -- --       47       16  0.986  29.037    27  1.383 1.451  15 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --       43       15  1.010  29.594    28  1.192 1.487  17 42
39 -- -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 39       14  1.013  30.258
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35       13  1.064  29.599
31 -- -- -- 20 -- -- --  0 -- -- -- 20 -- -- -- -- 31       12  1.090  30.070
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27       11  1.121  30.815
23 -- -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 23       10  1.160  31.387
19    -- -- -- -- -- -- -- -- -- -- -- -- --       19        9  1.214  31.693
15    -- -- -- -- -- -- 20 -- -- -- -- -- --       15        8  1.287  32.377
11       -- -- -- -- -- -- -- -- -- -- -- --       11        7  1.363  33.033
 7          -- -- -- -- -- -- -- -- --              7        6  1.465  33.496
 3                -- -- -- -- -- --                 3        5  1.539* 33.894*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              4  1.527  32.938
                                                             3  1.446  30.065
Control Rod Density: %          3.77                         2  1.160  23.247
                                                      Bottom 1  0.349   6.927
k-effective:                 0.99770
Void Fraction:                 0.393            % AXIAL TILT -22.659  -9.629
Core Delta-P: psia            21.958          AVG BOT 8ft/12ft  1.1745  1.0962
Core Plate Delta-P: psia      17.527
Coolant Temp: Deg-F            543.7
In Channel Flow: Mlb/hr        85.47   Active Channel Flow: Mlb/hr        81.61
Total Bypass Flow (%):          11.0   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00038
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.487	28	17	42	1.880	0.766	26	19	32	9.49	0.811	12.3	26	41	16	5	8.31	0.822	44.4	24	17	18	4
1.487	28	19	44	1.881	0.765	26	29	42	9.49	0.811	12.3	26	45	20	5	8.28	0.818	44.3	23	39	16	4
1.479	26	19	32	1.936	0.754	24	19	34	9.42	0.805	13.0	28	41	18	4	8.24	0.816	44.4	23	15	40	4
1.479	26	29	42	1.910	0.754	26	21	34	9.42	0.805	13.0	28	43	20	4	8.14	0.814	45.5	23	47	40	4
1.475	26	21	34	1.911	0.754	26	27	40	7.59	0.795	39.4	24	43	18	4	8.15	0.814	45.4	23	39	48	4
1.475	26	27	40	1.941	0.752	24	33	20	9.29	0.794	12.4	27	45	18	4	8.10	0.814	45.9	23	39	18	4
1.464	26	27	36	1.918	0.751	26	27	36	9.29	0.794	12.4	27	17	16	4	8.07	0.811	46.1	23	17	40	4
1.464	26	25	34	1.922	0.749	26	19	36	7.57	0.790	36.5	23	21	16	4	11.36	0.805	16.2	28	19	18	4
1.451	27	15	36	1.924	0.749	26	25	28	7.55	0.788	36.6	23	45	22	4	11.35	0.805	16.2	28	17	20	4
1.450	27	25	46	1.953	0.748	23	23	34	9.14	0.781	12.3	26	23	18	5	7.93	0.799	46.3	23	23	16	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.18 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 6,000.0 MWd/MTU

```
Cycle:                            24                    Core Average Exposure: MWd/MTU    27201.2
Exposure: MWd/MTU (GWd)       6500.0 ( 817.75 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )                Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                     Top 24  0.163   4.600    19  0.321 0.567  55 40
Inlet Subcooling: Btu/lbm     -24.74                         23  0.297  10.530    20  0.274 0.287  57 42
Flow: Mlb/hr                   96.04 ( 98.00 %)              22  0.605  20.706    21  0.391 0.728  53 42
                                                             21  0.698  25.097    22  0.659 0.882  47 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              20  0.769  27.578    23  1.290 1.400  21 30
59               -- -- -- -- --                59            19  0.825  28.864    24  1.256 1.402  19 28
55         -- -- -- -- -- -- -- -- --          55            18  0.872  29.078    25  1.072 1.307  47 44
51      -- -- -- -- -- -- -- -- -- -- --       51            17  0.939  29.249    26  1.417 1.485  19 30
47   -- -- -- -- -- -- 20 -- -- -- -- -- --    47            16  0.986  29.544    27  1.395 1.466  15 26
43   -- -- -- -- -- -- -- -- -- -- -- -- --    43            15  1.011  30.114    28  1.201 1.499  17 20
39 -- -- -- -- --  6 -- -- --  8 -- -- -- -- -- 39           14  1.012  30.779
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35           13  1.061  30.120
31 -- -- -- 20 -- -- --  0 -- -- -- 20 -- -- -- 31           12  1.084  30.604
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27           11  1.115  31.364
23 -- -- -- -- --  8 -- -- --  6 -- -- -- -- -- 23           10  1.154  31.955
19   -- -- -- -- -- -- -- -- -- -- -- -- --    19             9  1.209  32.281
15   -- -- -- -- -- -- 20 -- -- -- -- -- --    15             8  1.286  32.990
11      -- -- -- -- -- -- -- -- -- -- --       11             7  1.366  33.683
 7         -- -- -- -- -- -- -- -- --           7             6  1.475  34.194
 3               -- -- -- -- --                 3             5  1.554* 34.627*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               4  1.541  33.666
                                                              3  1.458  30.753
Control Rod Density: %          3.81                          2  1.168  23.800
                                                      Bottom  1  0.352   7.094
k-effective:                 0.99760
Void Fraction:                 0.393             % AXIAL TILT -23.024  -9.797
Core Delta-P: psia            21.976         AVG BOT 8ft/12ft  1.1770  1.0971
Core Plate Delta-P: psia      17.545
Coolant Temp: Deg-F            543.7
In Channel Flow: Mlb/hr        85.46   Active Channel Flow: Mlb/hr        81.60
Total Bypass Flow (%):          11.0   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00049
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.499	28	17	20	1.888	0.763	26	19	32	9.70	0.829	13.5	26	41	16	5	8.33	0.834	45.6	24	17	18	4
1.498	28	19	18	1.889	0.762	26	29	42	9.70	0.829	13.5	26	45	20	5	8.28	0.828	45.5	23	39	16	4
1.485	26	19	30	1.948	0.750	24	19	34	9.59	0.819	14.2	28	41	18	4	8.25	0.825	45.6	23	15	40	4
1.484	26	31	42	1.924	0.748	26	21	28	9.58	0.819	14.2	28	43	20	4	8.15	0.824	46.6	23	47	40	4
1.478	26	21	28	1.954	0.747	24	33	20	9.49	0.811	13.6	27	43	16	4	8.16	0.824	46.5	23	39	48	4
1.477	26	33	40	1.928	0.747	26	27	22	9.49	0.811	13.6	27	45	18	4	8.06	0.819	47.1	23	39	18	4
1.466	27	15	26	1.928	0.747	26	17	28	7.64	0.802	40.5	24	43	18	4	7.92	0.817	48.7	23	17	40	5
1.464	27	25	16	1.928	0.747	26	19	26	9.35	0.799	13.0	27	41	14	4	11.46	0.813	17.7	28	19	18	4
1.464	26	25	28	1.931	0.746	26	33	44	9.34	0.798	13.0	27	47	20	4	11.46	0.813	17.7	28	17	20	4
1.464	26	33	36	1.932	0.745	26	35	42	9.31	0.796	13.5	26	23	18	5	7.89	0.808	47.9	24	25	18	5

```
  * LHGR calculated with pin-power reconstruction
  * CPR calculated with pin-power reconstruction & CPR limit type 3
```

Figure A.19 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 6,500.0 MWd/MTU

```
Cycle:                            24                Core Average Exposure: MWd/MTU    27451.7
Exposure: MWd/MTU (GWd)       6750.0 ( 849.20 )
Delta E: MWd/MTU, (GWd)        250.0 (  31.45 )             Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)         N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                   Top 24  0.163   4.636    19  0.319 0.564  55 40
Inlet Subcooling: Btu/lbm     -24.74                       23  0.297  10.610    20  0.272 0.284  57 42
Flow: Mlb/hr                   96.04 ( 98.00 %)            22  0.605  20.871    21  0.388 0.725  53 42
                                                           21  0.698  25.287    22  0.657 0.879  47 48
   2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             20  0.768  27.788    23  1.288 1.393  21 30
59                 -- -- -- -- --             59           19  0.824  29.089    24  1.253 1.399  19 28
55           -- -- -- -- -- -- -- -- --       55           18  0.872  29.317    25  1.070 1.307  47 44
51        -- -- -- -- -- -- -- -- -- -- --    51           17  0.938  29.496    26  1.421 1.488  19 30
47     -- -- -- -- -- -- 20 -- -- -- -- -- -- 47           16  0.986  29.798    27  1.401 1.472  15 26
43     -- -- -- -- -- -- -- -- -- -- -- -- -- 43           15  1.010  30.374    28  1.205 1.504  17 20
39 -- -- -- -- -- --  6 -- -- --  8 -- -- -- -- -- 39      14  1.010  31.040
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35         13  1.058  30.380
31 -- -- -- 20 -- -- --  0 -- -- -- 20 -- -- -- 31         12  1.080  30.869
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27         11  1.110  31.637
23 -- -- -- -- -- --  8 -- -- --  6 -- -- -- -- -- 23      10  1.150  32.238
19     -- -- -- -- -- -- -- -- -- -- -- -- -- 19            9  1.206  32.573
15     -- -- -- -- -- -- 20 -- -- -- -- -- -- 15            8  1.283  33.296
11        -- -- -- -- -- -- -- -- -- -- --    11            7  1.366  34.008
 7           -- -- -- -- -- -- -- -- --        7            6  1.479  34.545
 3                 -- -- -- -- --              3            5  1.560* 34.998*
   2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              4  1.548  34.034
                                                            3  1.465  31.102
Control Rod Density: %          3.81                        2  1.172  24.079
                                                   Bottom  1  0.353   7.179
k-effective:                 0.99768
Void Fraction:                 0.393             % AXIAL TILT -23.099  -9.883
Core Delta-P: psia            21.983         AVG BOT 8ft/12ft  1.1772  1.0976
Core Plate Delta-P: psia      17.552
Coolant Temp: Deg-F            543.7
In Channel Flow: Mlb/hr        85.46   Active Channel Flow: Mlb/hr        81.60
Total Bypass Flow (%):          11.0   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00043
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.504	28	17	20	1.892	0.761	26	19	32	9.80	0.837	14.1	26	41	16	5	8.34	0.839	46.2	24	17	18	4
1.503	28	19	18	1.893	0.761	26	29	42	9.79	0.837	14.1	26	45	20	5	8.29	0.833	46.0	23	39	16	4
1.488	26	19	30	1.955	0.747	24	19	34	9.67	0.827	14.8	28	41	18	4	8.25	0.830	46.1	23	15	40	4
1.487	26	31	42	1.929	0.747	26	17	28	9.67	0.826	14.8	28	43	20	4	8.16	0.830	47.2	23	47	40	4
1.480	26	21	28	1.930	0.746	26	21	28	9.60	0.820	14.2	27	43	16	4	8.17	0.829	47.1	23	39	48	4
1.478	26	33	40	1.931	0.746	26	19	26	9.60	0.820	14.2	27	45	18	4	7.92	0.821	49.2	23	39	18	5
1.472	27	15	26	1.932	0.746	26	33	44	9.46	0.809	13.6	27	41	14	4	7.90	0.820	49.3	23	17	40	5
1.471	27	25	16	1.933	0.745	26	27	22	9.45	0.808	13.6	27	47	20	4	11.52	0.817	18.4	28	19	18	4
1.466	26	17	24	1.961	0.745	24	27	42	9.42	0.805	13.8	27	35	14	4	11.51	0.817	18.4	28	17	20	4
1.465	26	15	20	1.934	0.745	26	35	42	7.66	0.805	41.1	24	43	18	4	7.90	0.814	48.4	24	25	18	5

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.20 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 6,750.0 MWd/MTU

```
Cycle:                              24                      Core Average Exposure: MWd/MTU    27452.2
Exposure: MWd/MTU (GWd)      6751.0 ( 849.33 )
Delta E: MWd/MTU, (GWd)         1.0 (   0.13 )                     Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)                N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                          Top 24  0.191   4.636    19  0.318 0.553  55 40
Inlet Subcooling: Btu/lbm    -24.74                              23  0.346  10.610    20  0.267 0.279  57 42
Flow: Mlb/hr                  96.04 ( 98.00 %)                   22  0.705  20.872    21  0.383 0.704  53 42
                                                                 21  0.814  25.288    22  0.653 0.857  47 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                  20  0.891  27.789    23  1.241 1.417  35 40
59                -- -- -- -- --                 59              19  0.943  29.090    24  1.361 1.483  31 32
55          -- -- -- -- -- -- -- -- --           55              18  0.984  29.318    25  1.046 1.268  31 10
51       -- -- -- -- -- -- -- -- -- -- --        51              17  1.043  29.497    26  1.458 1.529  31 34
47    -- -- -- -- 20 -- -- -- 20 -- -- -- --     47              16  1.076  29.799    27  1.347 1.534  35 38
43    -- -- -- -- -- -- -- -- -- -- -- -- --     43              15  1.075  30.375    28  1.185 1.470  43 42
39 -- -- -- 20 -- -- -- 10 -- -- -- 20 -- -- -- 39              14  1.041  31.041
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35              13  1.062  30.381
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31              12  1.062  30.870
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27              11  1.074  31.638
23 -- -- -- 20 -- -- -- 10 -- -- -- 20 -- -- -- 23              10  1.099  32.239
19    -- -- -- -- -- -- -- -- -- -- -- -- --     19               9  1.143  32.575
15    -- -- -- -- 20 -- -- -- 20 -- -- -- --     15               8  1.209  33.297
11       -- -- -- -- -- -- -- -- -- -- --        11               7  1.283  34.010
 7          -- -- -- -- -- -- -- -- --            7               6  1.385  34.547
 3                -- -- -- -- --                  3               5  1.453* 34.999*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                   4  1.425  34.035
                                                                  3  1.330  31.103
Control Rod Density: %         4.47                               2  1.051  24.080
                                                           Bottom 1  0.314   7.179
k-effective:                0.99765
Void Fraction:                0.378             % AXIAL TILT -15.233  -9.883
Core Delta-P: psia           21.788         AVG BOT 8ft/12ft  1.1302  1.0976
Core Plate Delta-P: psia     17.355
Coolant Temp: Deg-F           543.4
In Channel Flow: Mlb/hr       85.57   Active Channel Flow: Mlb/hr        81.77
Total Bypass Flow (%):         10.9   (of total core flow)
Total Water Rod Flow (%):       4.0   (of total core flow)
Source Convergence          0.00037
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.534	27	35	38	1.811	0.795	26	29	34	9.36	0.800	13.2	26	31	12	5	8.06	0.807	45.7	23	47	32	4
1.531	27	37	36	1.814	0.794	26	21	42	9.34	0.799	13.2	26	49	32	5	8.05	0.806	45.6	23	31	48	4
1.529	26	31	34	1.817	0.793	26	19	40	9.32	0.797	13.8	26	33	16	5	8.14	0.802	44.1	23	31	16	4
1.525	26	33	32	1.825	0.789	26	27	44	9.32	0.796	14.3	26	43	28	5	7.73	0.798	48.7	23	47	34	5
1.518	26	35	42	1.832	0.786	26	35	28	9.32	0.796	14.3	26	33	18	5	8.09	0.798	44.1	23	15	32	4
1.514	26	41	36	1.833	0.786	26	27	36	9.32	0.796	13.8	26	45	28	5	7.72	0.796	48.6	23	33	48	5
1.508	26	39	42	1.833	0.786	26	33	30	9.26	0.791	13.5	26	33	12	5	7.73	0.794	48.2	23	31	18	5
1.508	26	41	40	1.833	0.786	26	35	20	9.24	0.790	13.5	26	49	34	5	7.70	0.791	48.2	23	17	32	5
1.503	26	27	44	1.836	0.784	26	43	34	9.10	0.778	14.1	26	31	28	5	7.74	0.788	47.4	24	31	32	5
1.499	26	43	34	1.863	0.784	24	29	32	9.09	0.777	14.1	26	33	30	5	10.92	0.775	15.9	26	49	32	5

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.21 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 6,751.0 MWd/MTU

```
Cycle:                            24                 Core Average Exposure: MWd/MTU    27701.2
 Exposure: MWd/MTU (GWd)       7000.0 ( 880.66 )
 Delta E: MWd/MTU, (GWd)        249.0 (  31.33 )
 Power: MWt                    2957.0 (100.00 %)
 Core Pressure: psia           1015.0
 Inlet Subcooling: Btu/lbm     -24.74
 Flow: Mlb/hr                   96.04 ( 98.00 %)
```

```
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                -- -- -- -- -- --             59
55          -- -- -- -- -- -- -- -- -- --          55
51       -- -- -- -- -- -- -- -- -- -- -- --       51
47    -- -- -- -- 20 -- -- -- 20 -- -- -- --    47
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43
39 -- -- -- 20 -- -- --  8 -- -- -- 20 -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- 20 -- -- --  8 -- -- -- 20 -- -- -- 23
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19
15    -- -- -- -- 20 -- -- -- 20 -- -- -- --    15
11       -- -- -- -- -- -- -- -- -- -- -- --       11
 7          -- -- -- -- -- -- -- -- -- --           7
 3                -- -- -- -- -- --              3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
```

Axial Profile

N(PRA)	Power	Exposure
Top 24	0.189	4.679
23	0.342	10.703
22	0.696	21.062
21	0.803	25.508
20	0.877	28.031
19	0.933	29.346
18	0.978	29.586
17	1.040	29.771
16	1.075	30.075
15	1.075	30.651
14	1.041	31.308
13	1.062	30.640
12	1.062	31.130
11	1.073	31.900
10	1.099	32.507
9	1.143	32.850
8	1.212	33.584
7	1.288	34.314
6	1.393	34.875
5	1.463*	35.344*
4	1.438	34.374
3	1.342	31.419
2	1.060	24.329
Bottom 1	0.317	7.254
% AXIAL TILT	-15.748	-9.896
AVG BOT 8ft/12ft	1.1339	1.0976

Edit Zone	Radial Power Avg.	Max.	IR	JR
19	0.318	0.553	55	40
20	0.266	0.278	57	42
21	0.382	0.704	53	42
22	0.653	0.857	13	48
23	1.238	1.405	39	36
24	1.351	1.467	31	32
25	1.048	1.270	51	32
26	1.459	1.520	31	28
27	1.354	1.528	37	36
28	1.192	1.477	43	42

```
 Control Rod Density: %         4.52

 k-effective:                0.99746
 Void Fraction:                0.379
 Core Delta-P: psia           21.806
 Core Plate Delta-P: psia     17.372
 Coolant Temp: Deg-F           543.5
 In Channel Flow: Mlb/hr       85.56   Active Channel Flow: Mlb/hr        81.76
 Total Bypass Flow (%):         10.9   (of total core flow)
 Total Water Rod Flow (%):       4.0   (of total core flow)
 Source Convergence          0.00036
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.528	27	37	36	1.819	0.791	26	19	40	9.48	0.810	13.8	26	31	12	5	7.85	0.811	48.8	23	47	32	5
1.527	27	35	38	1.821	0.791	26	21	42	9.46	0.809	13.8	26	49	32	5	7.84	0.810	48.8	23	31	48	5
1.520	26	31	28	1.836	0.784	26	43	34	9.42	0.805	14.4	26	33	16	5	8.12	0.805	44.7	23	31	16	4
1.519	26	33	32	1.839	0.783	26	27	44	9.41	0.805	14.4	26	45	28	5	7.74	0.804	49.2	23	47	34	5
1.517	26	41	36	1.850	0.779	26	19	36	9.40	0.803	14.9	26	33	18	5	7.73	0.802	49.2	23	33	48	5
1.515	26	25	42	1.853	0.777	26	35	20	9.40	0.803	14.9	26	43	28	5	8.07	0.801	44.7	23	15	32	4
1.511	26	41	40	1.857	0.776	26	31	28	9.39	0.803	14.1	26	33	12	5	7.73	0.798	48.7	23	31	18	5
1.510	26	39	42	1.858	0.775	26	33	30	9.37	0.801	14.1	26	49	34	5	7.70	0.795	48.7	23	17	32	5
1.505	26	43	34	1.862	0.774	26	27	36	9.15	0.782	14.7	26	31	28	5	7.70	0.788	47.9	24	31	32	5
1.502	26	27	44	1.862	0.773	26	35	28	9.14	0.781	14.6	26	33	30	5	11.01	0.781	16.6	26	49	32	5

 * LHGR calculated with pin-power reconstruction
 * CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.22 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 7,000.0 MWd/MTU

```
Cycle:                          24
Exposure: MWd/MTU (GWd)     7500.0 ( 943.56 )
Delta E: MWd/MTU, (GWd)      500.0 (  62.90 )
Power: MWt                  2957.0 (100.00 %)
Core Pressure: psia         1015.0
Inlet Subcooling: Btu/lbm   -24.74
Flow: Mlb/hr                 96.04 ( 98.00 %)

    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                -- -- -- -- --                59
55          -- -- -- -- -- -- -- -- --          55
51       -- -- -- -- -- -- -- -- -- -- --       51
47    -- -- -- -- 20 -- -- -- 20 -- -- -- --    47
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43
39 -- -- -- 20 -- -- --  8 -- -- -- 20 -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- 20 -- -- --  8 -- -- -- 20 -- -- -- 23
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19
15    -- -- -- -- 20 -- -- -- 20 -- -- -- --    15
11       -- -- -- -- -- -- -- -- -- -- --       11
 7          -- -- -- -- -- -- -- -- --           7
 3                -- -- -- -- --                 3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58

Control Rod Density: %        4.52

k-effective:               0.99759
Void Fraction:               0.379
Core Delta-P: psia          21.816
Core Plate Delta-P: psia    17.382
Coolant Temp: Deg-F          543.5
In Channel Flow: Mlb/hr      85.56   Active Channel Flow: Mlb/hr       81.75
Total Bypass Flow (%):        10.9   (of total core flow)
Total Water Rod Flow (%):      4.0   (of total core flow)
Source Convergence         0.00040
```

```
Core Average Exposure: MWd/MTU    28201.9

                Axial Profile     Edit    Radial Power
          N(PRA) Power Exposure   Zone  Avg.  Max.   IR JR
          Top 24  0.189   4.762    19  0.314 0.549  55 40
              23  0.343  10.887    20  0.262 0.274  57 42
              22  0.697  21.442    21  0.377 0.700  53 42
              21  0.803  25.946    22  0.649 0.852  13 48
              20  0.876  28.510    23  1.234 1.392  39 36
              19  0.933  29.856    24  1.341 1.451  29 30
              18  0.978  30.122    25  1.045 1.272   9 32
              17  1.040  30.320    26  1.466 1.520  41 36
              16  1.076  30.629    27  1.367 1.527  37 36
              15  1.076  31.205    28  1.201 1.486  43 42
              14  1.040  31.843
              13  1.057  31.159
              12  1.054  31.648
              11  1.065  32.424
              10  1.090  33.043
               9  1.136  33.402
               8  1.206  34.161
               7  1.285  34.927
               6  1.396  35.539
               5  1.470* 36.042*
               4  1.449  35.062
               3  1.354  32.061
               2  1.067  24.836
        Bottom 1  0.320   7.407

    % AXIAL TILT -15.777  -9.930
 AVG BOT 8ft/12ft  1.1338  1.0977
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.527	27	37	36	1.828	0.788	26	19	40	9.64	0.824	15.0	26	31	12	5	7.82	0.817	49.9	23	47	32	5
1.526	27	35	38	1.829	0.787	26	21	42	9.62	0.822	15.0	26	49	32	5	7.82	0.816	49.8	23	31	48	5
1.520	26	41	36	1.839	0.783	26	43	34	9.57	0.818	15.3	26	33	12	5	7.66	0.810	51.1	23	47	34	5
1.520	26	31	34	1.841	0.782	26	27	44	9.55	0.816	15.3	26	49	34	5	7.65	0.808	51.0	23	33	48	5
1.519	26	33	32	1.856	0.776	26	41	36	9.51	0.813	15.6	26	33	16	5	8.05	0.806	45.8	23	31	16	4
1.518	26	25	42	1.859	0.775	26	35	20	9.51	0.813	15.6	26	45	28	5	7.86	0.802	47.6	23	15	32	5
1.515	26	41	40	1.866	0.772	26	31	28	9.44	0.807	16.1	26	33	18	5	7.65	0.798	49.8	23	31	18	5
1.513	26	39	42	1.867	0.771	26	33	30	9.44	0.806	16.1	26	43	28	5	7.63	0.796	49.8	23	17	32	5
1.513	26	43	34	1.869	0.771	28	17	42	9.10	0.778	15.8	26	31	28	5	7.99	0.786	43.8	25	31	10	4
1.510	26	27	44	1.871	0.770	26	27	36	9.09	0.777	15.8	26	33	30	5	11.08	0.786	18.0	26	49	32	5

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.23 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 7,500.0 MWd/MTU

```
Cycle:                          24                     Core Average Exposure: MWd/MTU    28701.2
Exposure: MWd/MTU (GWd)     8000.0 (1006.50 )
Delta E: MWd/MTU, (GWd)      500.0 (  62.90 )              Axial Profile    Edit   Radial Power
Power: MWt                  2957.0 (100.00 %)         N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia         1015.0                   Top 24  0.187   4.846    19  0.312 0.547  55 40
Inlet Subcooling: Btu/lbm   -24.74                       23  0.339  11.071    20  0.260 0.272  57 42
Flow: Mlb/hr                 96.04 ( 98.00 %)            22  0.691  21.821    21  0.375 0.699  53 42
                                                         21  0.794  26.383    22  0.647 0.849  13 48
   2  6 10 14 18 22 26 30 34 38 42 46 50 54 58           20  0.871  28.989    23  1.228 1.385  45 32
59                -- -- -- -- --               59        19  0.930  30.366    24  1.327 1.428  31 32
55          -- -- -- -- -- -- -- -- --         55        18  0.979  30.657    25  1.045 1.277   9 32
51       -- -- -- -- -- -- -- -- -- -- --      51        17  1.044  30.868    26  1.470 1.523  41 36
47    -- -- -- -- 20 -- -- -- 20 -- -- -- --   47        16  1.081  31.182    27  1.381 1.524  37 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --   43        15  1.081  31.759    28  1.213 1.497  43 42
39 -- -- -- 20 -- -- --  6 -- -- -- 20 -- -- -- 39        14  1.044  32.379
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35        13  1.058  31.677
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31        12  1.053  32.165
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27        11  1.062  32.946
23 -- -- -- 20 -- -- --  6 -- -- -- 20 -- -- -- 23        10  1.087  33.577
19    -- -- -- -- -- -- -- -- -- -- -- -- --   19         9  1.132  33.953
15    -- -- -- -- 20 -- -- -- 20 -- -- -- --   15         8  1.203  34.735
11       -- -- -- -- -- -- -- -- -- -- --      11         7  1.283  35.540
 7          -- -- -- -- -- -- -- -- --          7         6  1.394  36.204
 3                -- -- -- -- --                3         5  1.470* 36.743*
   2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            4  1.456  35.752
                                                          3  1.364  32.707
Control Rod Density: %        4.57                        2  1.074  25.345
                                                   Bottom 1  0.321   7.560
k-effective:               0.99751
Void Fraction:               0.379               % AXIAL TILT -15.838  -9.964
Core Delta-P: psia          21.826            AVG BOT 8ft/12ft  1.1353  1.0979
Core Plate Delta-P: psia    17.393
Coolant Temp: Deg-F          543.4
In Channel Flow: Mlb/hr      85.56   Active Channel Flow: Mlb/hr        81.75
Total Bypass Flow (%):        10.9   (of total core flow)
Total Water Rod Flow (%):      4.0   (of total core flow)
Source Convergence         0.00038
```

```
                          Top Ten Thermal Limits Summary - Sorted by Margin

     Power                  MCPR                          APLHGR                              LHGR
Value FT  IR JR    Value Margin FT  IR JR    Value Margin Exp. FT  IR JR  K    Value Margin Exp. FT  IR JR  K
1.524  27 37 36    1.834  0.785  26 19 40    9.70  0.829 16.3  26 31 12  5    7.80  0.820 50.5  23 47 34  4
1.523  26 41 36    1.838  0.784  26 21 42    9.69  0.828 16.2  26 49 32  5    7.76  0.820 51.0  23 47 32  5
1.521  26 43 34    1.840  0.783  26 43 34    9.66  0.825 16.6  26 33 12  5    7.75  0.818 50.9  23 31 48  5
1.520  27 35 38    1.849  0.779  26 27 44    9.64  0.824 16.5  26 49 34  5    7.80  0.818 50.4  23 33 48  4
1.518  26 41 40    1.859  0.774  28 17 42    9.50  0.812 16.8  26 33 16  5    7.94  0.804 46.9  23 31 16  4
1.516  26 25 42    1.860  0.774  26 19 36    9.49  0.811 16.8  26 45 28  5    7.76  0.800 48.6  23 15 32  5
1.516  26 39 42    1.863  0.773  28 41 18    9.37  0.800 17.3  26 33 18  5    8.03  0.798 44.9  25 31 10  4
1.512  26 27 44    1.870  0.770  26 35 20    9.36  0.800 17.3  26 43 28  5    7.98  0.793 45.0  25  9 32  4
1.511  26 45 34    1.888  0.763  26 45 34    9.23  0.789 16.6  28 35 12  4    7.51  0.792 50.8  23 31 18  5
1.511  26 33 32    1.889  0.762  26 43 38    9.20  0.786 16.5  28 49 36  4    7.49  0.790 50.8  23 17 32  5

  * LHGR calculated with pin-power reconstruction
  * CPR calculated with pin-power reconstruction & CPR limit type 3
```

Figure A.24 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 8,000.0 MWd/MTU

```
Cycle:                          24                        Core Average Exposure: MWd/MTU    29201.2
Exposure: MWd/MTU (GWd)     8500.0 (1069.40 )
Delta E: MWd/MTU, (GWd)      500.0 (  62.90 )                Axial Profile    Edit   Radial Power
Power: MWt                  2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia         1015.0                     Top 24  0.186   4.929    19  0.310 0.544  55 40
Inlet Subcooling: Btu/lbm   -24.74                         23  0.337  11.254    20  0.257 0.269  57 42
Flow: Mlb/hr                 96.04 ( 98.00 %)              22  0.686  22.197    21  0.372 0.697  53 42
                                                           21  0.788  26.815    22  0.646 0.846  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            20  0.869  29.464    23  1.223 1.372  45 32
59                -- -- -- -- -- --                59      19  0.932  30.875    24  1.313 1.406  31 32
55          -- -- -- -- -- -- -- -- --             55      18  0.984  31.193    25  1.046 1.279   9 32
51       -- -- -- -- -- -- -- -- -- -- --          51      17  1.052  31.419    26  1.473 1.521  43 34
47    -- -- -- -- 20 -- -- -- 20 -- -- -- --       47      16  1.091  31.739    27  1.395 1.517  37 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --       43      15  1.092  32.315    28  1.225 1.507  43 42
39 -- -- -- 20 -- -- --  6 -- -- -- 20 -- -- -- 39         14  1.053  32.916
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35         13  1.063  32.196
31 -- -- -- -- --  6 -- -- --  6 -- -- -- -- -- 31         12  1.055  32.681
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27         11  1.062  33.466
23 -- -- -- 20 -- -- --  6 -- -- -- 20 -- -- -- 23         10  1.085  34.110
19    -- -- -- -- -- -- -- -- -- -- -- -- --       19       9  1.130  34.502
15    -- -- -- -- 20 -- -- -- 20 -- -- -- --       15       8  1.199  35.309
11       -- -- -- -- -- -- -- -- -- -- --          11       7  1.278  36.151
 7          -- -- -- -- -- -- -- -- --              7       6  1.385  36.868
 3                -- -- -- -- -- --                 3       5  1.460* 37.443*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             4  1.453  36.446
                                                            3  1.365  33.357
Control Rod Density: %        4.61                          2  1.073  25.856
                                                     Bottom 1  0.321   7.714
k-effective:               0.99755
Void Fraction:               0.378            % AXIAL TILT -15.553  -9.997
Core Delta-P: psia          21.828        AVG BOT 8ft/12ft  1.1353  1.0980
Core Plate Delta-P: psia    17.394
Coolant Temp: Deg-F          543.4
In Channel Flow: Mlb/hr      85.56   Active Channel Flow: Mlb/hr       81.76
Total Bypass Flow (%):        10.9   (of total core flow)
Total Water Rod Flow (%):      4.0   (of total core flow)
Source Convergence         0.00041
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.521	26	43	34	1.841	0.782	26	19	40	9.67	0.827	17.3	26	33	12	4	7.78	0.827	51.5	23	47	34	4
1.520	26	41	40	1.842	0.782	26	21	42	9.66	0.825	17.5	26	31	12	5	7.77	0.825	51.4	23	33	48	4
1.519	26	41	36	1.845	0.780	26	43	34	9.65	0.824	17.5	26	49	32	5	7.64	0.816	52.0	23	47	32	5
1.519	26	27	44	1.848	0.779	26	27	44	9.64	0.824	17.3	26	49	34	4	7.63	0.815	52.0	23	31	48	5
1.518	26	39	42	1.848	0.779	28	17	42	9.41	0.805	17.7	28	35	12	4	8.02	0.806	46.0	25	31	10	4
1.518	26	25	42	1.852	0.778	28	41	18	9.38	0.802	17.7	28	49	36	4	7.97	0.801	46.1	25	9	32	4
1.517	26	45	34	1.870	0.770	26	19	36	9.37	0.801	18.0	26	33	16	5	11.02	0.801	21.7	26	49	34	4
1.517	27	37	36	1.873	0.769	26	35	20	9.37	0.801	18.0	26	45	28	5	11.00	0.799	21.7	26	33	50	4
1.516	27	35	38	1.884	0.764	26	45	34	9.18	0.785	18.5	26	33	18	5	10.99	0.799	21.7	28	35	12	4
1.515	26	27	46	1.888	0.763	26	33	46	9.18	0.785	18.5	26	43	28	5	10.97	0.797	21.7	28	49	36	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.25 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 8,500.0 MWd/MTU

```
Cycle:                            24                       Core Average Exposure: MWd/MTU    29701.2
  Exposure: MWd/MTU (GWd)       9000.0 (1132.30 )
 Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )
 Power: MWt                    2957.0 (100.00 %)
 Core Pressure: psia           1015.0
 Inlet Subcooling: Btu/lbm     -24.74
 Flow: Mlb/hr                   96.04 ( 98.00 %)

    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                -- -- -- -- --                59
55          -- -- -- -- -- -- -- -- --          55
51       -- -- -- -- -- -- -- -- -- -- --       51
47    -- -- -- -- 18 -- -- -- 20 -- -- -- --    47
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43
39 -- -- -- 20 -- -- --  6 -- -- -- 20 -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- -- --  6 -- -- --  6 -- -- -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- 20 -- -- --  6 -- -- -- 20 -- -- -- 23
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19
15    -- -- -- -- 20 -- -- -- 18 -- -- -- --    15
11       -- -- -- -- -- -- -- -- -- -- --       11
 7          -- -- -- -- -- -- -- -- --           7
 3                -- -- -- -- --                 3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58

 Control Rod Density: %          4.66

 k-effective:                 0.99761
 Void Fraction:                 0.376
 Core Delta-P: psia            21.811
 Core Plate Delta-P: psia      17.378
 Coolant Temp: Deg-F            543.4
 In Channel Flow: Mlb/hr        85.57   Active Channel Flow: Mlb/hr        81.77
 Total Bypass Flow (%):          10.9   (of total core flow)
 Total Water Rod Flow (%):        4.0   (of total core flow)
 Source Convergence           0.00044
```

```
          Axial Profile     Edit   Radial Power
   N(PRA) Power Exposure    Zone  Avg.  Max.   IR JR
   Top 24  0.188    5.011    19  0.307 0.544   55 40
       23  0.342   11.435    20  0.255 0.268   57 42
       22  0.695   22.570    21  0.369 0.697   53 42
       21  0.798   27.245    22  0.643 0.846   47 48
       20  0.880   29.939    23  1.218 1.366   45 32
       19  0.944   31.384    24  1.304 1.393   31 32
       18  0.996   31.732    25  1.044 1.284   51 32
       17  1.064   31.973    26  1.476 1.528   43 34
       16  1.100   32.301    27  1.406 1.518   37 36
       15  1.096   32.877    28  1.235 1.519   43 42
       14  1.058   33.458
       13  1.067   32.717
       12  1.057   33.198
       11  1.063   33.987
       10  1.085   34.641
        9  1.128   35.049
        8  1.195   35.880
        7  1.269   36.760
        6  1.368   37.528
        5  1.437*  38.138*
        4  1.436   37.139
        3  1.353   34.007
        2  1.063   26.368
Bottom  1  0.318    7.868

    % AXIAL TILT -14.763 -10.024
AVG BOT 8ft/12ft  1.1308  1.0982
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.528	26	43	34	1.835	0.785	28	43	42	9.72	0.831	18.6	26	33	12	4	7.67	0.825	52.6	23	47	34	4
1.527	26	45	34	1.837	0.784	28	41	44	9.69	0.828	18.5	26	49	34	4	7.67	0.824	52.5	23	33	48	4
1.525	26	41	40	1.838	0.783	26	43	34	9.59	0.820	18.2	26	31	12	4	7.95	0.807	47.1	25	31	10	4
1.523	26	41	36	1.841	0.782	26	19	22	9.57	0.818	18.1	26	49	32	4	7.61	0.807	51.2	23	47	32	4
1.523	26	39	42	1.845	0.781	26	21	20	9.49	0.811	19.0	28	35	12	4	7.62	0.806	51.1	23	31	48	4
1.522	26	33	44	1.847	0.780	26	33	44	9.46	0.809	18.9	28	49	36	4	10.95	0.806	23.1	26	49	34	4
1.521	26	33	46	1.867	0.771	26	41	36	9.21	0.787	18.8	26	33	16	4	10.94	0.805	23.1	28	35	12	4
1.520	26	35	42	1.871	0.770	26	45	34	9.19	0.785	18.8	26	45	28	4	10.94	0.804	23.1	26	33	50	4
1.519	28	43	42	1.874	0.768	26	35	42	9.15	0.782	18.9	27	35	14	4	10.92	0.803	23.0	28	49	36	4
1.518	27	37	36	1.882	0.765	26	33	46	9.15	0.782	17.9	28	33	10	4	7.90	0.803	47.2	25	9	32	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.26 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 9,000.0 MWd/MTU

```
Cycle:                          24                          Core Average Exposure: MWd/MTU    30051.2
Exposure: MWd/MTU (GWd)      9350.0 (1176.30 )
Delta E: MWd/MTU, (GWd)       350.0 (  44.03 )                    Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)              N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                        Top 24  0.190   5.070    19  0.306 0.543  55 40
Inlet Subcooling: Btu/lbm    -24.74                            23  0.345  11.564    20  0.253 0.266  57 42
Flow: Mlb/hr                  96.04 ( 98.00 %)                 22  0.702  22.835    21  0.367 0.697  53 42
                                                               21  0.807  27.549    22  0.642 0.840  13 48
   2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                 20  0.889  30.276    23  1.214 1.362  45 32
59                -- -- -- -- --                59             19  0.954  31.746    24  1.299 1.386  31 32
55          -- -- -- -- -- -- -- -- --          55             18  1.007  32.113    25  1.044 1.289   9 32
51       -- -- -- -- -- -- -- -- -- -- --       51             17  1.074  32.366    26  1.479 1.534  45 34
47    -- -- -- -- 18 -- -- -- 18 -- -- -- --    47             16  1.108  32.697    27  1.413 1.518  37 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43             15  1.100  33.272    28  1.242 1.519  43 42
39 -- -- -- 20 -- -- --  6 -- -- -- 20 -- -- -- 39             14  1.063  33.840
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35             13  1.072  33.083
31 -- -- -- -- --  6 -- -- --  6 -- -- -- -- -- 31             12  1.061  33.560
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27             11  1.067  34.352
23 -- -- -- 20 -- -- --  6 -- -- -- 20 -- -- -- 23             10  1.088  35.014
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19              9  1.130  35.431
15    -- -- -- -- 18 -- -- -- 18 -- -- -- --    15              8  1.195  36.279
11       -- -- -- -- -- -- -- -- -- -- --       11              7  1.264  37.184
 7          -- -- -- -- -- -- -- -- --           7              6  1.355  37.984
 3                -- -- -- -- --                 3              5  1.416* 38.617*
   2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                  4  1.415  37.618
                                                                3  1.336  34.459
Control Rod Density: %          4.71                            2  1.049  26.722
                                                         Bottom 1  0.314   7.975
k-effective:                 0.99763
Void Fraction:                 0.374               % AXIAL TILT -14.082 -10.034
Core Delta-P: psia            21.795           AVG BOT 8ft/12ft  1.1270  1.0982
Core Plate Delta-P: psia      17.362
Coolant Temp: Deg-F            543.4
In Channel Flow: Mlb/hr        85.58   Active Channel Flow: Mlb/hr        81.79
Total Bypass Flow (%):          10.9   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00043
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.534	26	45	34	1.832	0.786	28	17	42	9.67	0.826	19.5	26	33	12	4	7.55	0.818	53.4	23	47	34	4
1.532	26	43	34	1.835	0.785	26	43	34	9.64	0.824	19.4	26	49	34	4	7.56	0.818	53.3	23	33	48	4
1.524	26	41	36	1.846	0.780	28	41	18	9.53	0.815	19.0	26	31	12	4	7.86	0.805	47.9	25	31	10	4
1.523	26	41	40	1.846	0.780	26	19	40	9.51	0.813	19.0	26	49	32	4	10.82	0.803	24.1	26	49	34	4
1.519	28	43	42	1.860	0.774	26	27	44	9.46	0.809	19.8	28	35	12	4	10.82	0.803	24.1	26	33	50	4
1.518	26	27	44	1.861	0.774	26	21	42	9.43	0.806	19.7	28	49	36	4	10.81	0.802	24.1	28	35	12	4
1.518	27	37	36	1.863	0.773	26	45	34	9.16	0.783	18.7	28	33	10	4	10.80	0.801	24.0	28	49	36	4
1.517	26	27	46	1.866	0.772	26	19	36	9.12	0.780	18.6	28	51	34	4	7.81	0.800	47.9	25	9	32	4
1.517	26	39	42	1.891	0.762	26	35	20	9.06	0.774	19.7	27	35	14	4	10.81	0.798	23.5	28	33	10	4
1.515	27	35	38	1.892	0.761	26	43	38	9.05	0.774	19.7	26	33	16	4	7.46	0.796	52.0	23	47	32	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.27 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 9,350.0 MWd/MTU

```
Cycle:                            24                          Core Average Exposure: MWd/MTU    30052.2
Exposure: MWd/MTU (GWd)      9351.0 (1176.40 )
Delta E: MWd/MTU, (GWd)         1.0 (   0.13 )                Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                      Top 24  0.176   5.070    19  0.304 0.549  55 40
Inlet Subcooling: Btu/lbm    -24.74                          23  0.319  11.564    20  0.257 0.269  57 42
Flow: Mlb/hr                  96.04 ( 98.00 %)               22  0.651  22.835    21  0.367 0.713  53 42
                                                             21  0.747  27.550    22  0.626 0.775  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              20  0.815  30.277    23  1.262 1.371  21 46
59                -- -- -- -- -- --                 59       19  0.857  31.747    24  1.226 1.374  17 44
55          -- -- -- -- -- -- -- -- -- --           55       18  0.891  32.114    25  1.053 1.289  47 44
51       -- -- -- -- -- -- -- -- -- -- -- --        51       17  0.953  32.367    26  1.454 1.527  15 42
47    -- -- 34 -- -- -- 12 -- -- -- 34 -- --        47       16  1.002  32.698    27  1.466 1.533  25 46
43    -- -- -- -- -- -- -- -- -- -- -- -- --        43       15  1.032  33.274    28  1.243 1.558  17 42
39 -- -- -- -- -- 10 -- -- -- 10 -- -- -- -- --     39       14  1.040  33.841
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- --     35       13  1.082  33.084
31 -- -- -- 12 -- -- --  0 -- -- -- 12 -- -- --     31       12  1.100  33.562
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- --     27       11  1.126  34.353
23 -- -- -- -- --  8 -- -- -- 10 -- -- -- -- --     23       10  1.161  35.015
19    -- -- -- -- -- -- -- -- -- -- -- -- --        19        9  1.210  35.433
15    -- -- 34 -- -- -- 12 -- -- -- 34 -- --        15        8  1.272  36.280
11       -- -- -- -- -- -- -- -- -- -- -- --        11        7  1.329  37.185
 7          -- -- -- -- -- -- -- -- -- --            7        6  1.413  37.985
 3                -- -- -- -- -- --                  3        5  1.476  38.618*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               4  1.482* 37.619
                                                              3  1.411  34.460
Control Rod Density: %          4.73                          2  1.118  26.723
                                                      Bottom  1  0.336   7.976
k-effective:                 0.99746
Void Fraction:                 0.383              % AXIAL TILT -20.285 -10.034
Core Delta-P: psia            21.937          AVG BOT 8ft/12ft  1.1619  1.0982
Core Plate Delta-P: psia      17.503
Coolant Temp: Deg-F            543.5
In Channel Flow: Mlb/hr        85.51   Active Channel Flow: Mlb/hr        81.68
Total Bypass Flow (%):          11.0   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00038
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.558	28	17	42	1.877	0.767	28	17	42	10.18	0.870	19.8	28	35	12	4	8.09	0.858	51.3	23	47	40	4
1.557	28	19	44	1.884	0.764	28	19	44	10.16	0.868	19.7	28	11	36	4	8.10	0.857	51.2	23	39	48	4
1.533	27	25	46	1.921	0.750	26	27	40	10.09	0.862	18.8	27	37	12	4	8.14	0.857	50.7	23	23	48	4
1.533	27	45	36	1.925	0.748	26	39	28	10.07	0.860	18.8	27	49	38	4	8.13	0.857	50.6	23	47	24	4
1.527	26	15	42	1.934	0.744	26	27	44	10.05	0.859	19.7	27	25	14	4	11.50	0.848	23.3	27	25	14	4
1.527	26	19	46	1.936	0.744	26	43	34	10.03	0.857	19.6	27	13	36	4	11.51	0.846	23.0	27	37	12	4
1.518	27	47	36	1.943	0.741	26	23	44	9.68	0.827	18.1	27	39	12	4	11.48	0.846	23.3	27	47	36	4
1.517	27	25	48	1.944	0.741	26	17	38	9.68	0.827	19.5	26	33	12	4	11.49	0.844	23.0	27	49	38	4
1.511	26	23	44	1.946	0.740	26	29	42	9.67	0.827	19.9	27	25	16	4	11.35	0.839	23.6	28	25	50	4
1.511	26	17	38	1.946	0.740	26	41	30	9.66	0.826	18.1	27	49	40	4	11.34	0.838	23.6	28	49	26	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.28 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 9,351.0 MWd/MTU

```
Cycle:                            24                      Core Average Exposure: MWd/MTU    30201.2
Exposure: MWd/MTU (GWd)       9500.0 (1195.20 )
Delta E: MWd/MTU, (GWd)        149.0 (  18.75 )                 Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                     Top 24  0.175   5.093    19  0.303 0.548   5 40
Inlet Subcooling: Btu/lbm     -24.74                         23  0.318  11.616    20  0.256 0.269   3 42
Flow: Mlb/hr                   96.04 ( 98.00 %)              22  0.648  22.941    21  0.367 0.713   7 42
                                                             21  0.744  27.671    22  0.626 0.776  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              20  0.811  30.410    23  1.260 1.370  15 40
59                -- -- -- -- -- --              59          19  0.856  31.886    24  1.221 1.375  17 44
55          -- -- -- -- -- -- -- -- -- --        55          18  0.892  32.260    25  1.054 1.290  13 44
51       -- -- -- -- -- -- -- -- -- -- -- --     51          17  0.956  32.517    26  1.454 1.532  15 42
47    -- -- 34 -- -- -- 12 -- -- -- 34 -- --     47          16  1.007  32.852    27  1.470 1.537  15 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --     43          15  1.038  33.432    28  1.247 1.561  17 42
39 -- -- -- -- -- 10 -- -- --  8 -- -- -- -- -- 39          14  1.046  34.000
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35          13  1.089  33.242
31 -- -- -- 12 -- -- --  0 -- -- -- 12 -- -- -- 31          12  1.106  33.722
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27          11  1.132  34.517
23 -- -- -- -- --  8 -- -- -- 10 -- -- -- -- -- 23          10  1.166  35.184
19    -- -- -- -- -- -- -- -- -- -- -- -- --     19           9  1.214  35.607
15    -- -- 34 -- -- -- 12 -- -- -- 34 -- --     15           8  1.275  36.461
11       -- -- -- -- -- -- -- -- -- -- -- --     11           7  1.329  37.374
 7          -- -- -- -- -- -- -- -- -- --         7           6  1.408  38.186
 3                -- -- -- -- -- --               3           5  1.467  38.828*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               4  1.472* 37.829
                                                              3  1.403  34.660
Control Rod Density: %          4.76                          2  1.113  26.882
                                                       Bottom 1  0.334   8.024
k-effective:                 0.99741
Void Fraction:                 0.383              % AXIAL TILT -20.155 -10.065
Core Delta-P: psia            21.934          AVG BOT 8ft/12ft  1.1624  1.0984
Core Plate Delta-P: psia      17.500
Coolant Temp: Deg-F            543.5
In Channel Flow: Mlb/hr        85.51   Active Channel Flow: Mlb/hr        81.68
Total Bypass Flow (%):          11.0   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00041
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.561	28	17	42	1.871	0.770	28	17	42	10.14	0.868	20.2	28	35	12	4	8.03	0.855	51.7	23	47	40	4
1.561	28	19	44	1.878	0.767	28	19	44	10.13	0.867	20.1	28	49	36	4	8.04	0.854	51.5	23	39	48	4
1.537	27	15	36	1.925	0.748	26	21	34	10.06	0.860	19.2	27	37	12	4	8.08	0.853	51.0	23	23	48	4
1.536	27	25	46	1.926	0.748	26	27	40	10.05	0.859	19.2	27	49	38	4	8.07	0.853	51.0	23	47	24	4
1.532	26	15	42	1.935	0.744	26	27	44	9.99	0.854	20.1	27	35	14	4	11.44	0.846	23.7	27	25	14	4
1.531	26	19	46	1.940	0.742	26	43	28	9.97	0.853	20.1	27	47	26	4	11.45	0.845	23.5	27	37	12	4
1.523	27	13	36	1.941	0.742	26	17	38	9.67	0.826	18.5	27	39	12	4	11.42	0.844	23.7	27	47	36	4
1.521	27	25	48	1.942	0.741	26	23	44	9.66	0.825	18.5	27	49	40	4	11.44	0.844	23.4	27	49	38	4
1.514	26	17	38	1.944	0.741	26	15	42	9.65	0.825	19.8	26	33	12	4	11.30	0.838	24.0	28	25	50	4
1.513	26	23	44	1.947	0.740	26	19	46	9.63	0.823	19.7	26	11	34	4	11.29	0.837	24.0	28	49	26	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.29 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 9,500.0 MWd/MTU

```
Cycle:                            24                          Core Average Exposure: MWd/MTU    30701.2
Exposure: MWd/MTU (GWd)      10000.0 (1258.10 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )               Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)          N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                     Top 24  0.177   5.171   19  0.303 0.548  55 40
Inlet Subcooling: Btu/lbm     -24.74                         23  0.321  11.786   20  0.255 0.267   3 42
Flow: Mlb/hr                   96.04 ( 98.00 %)              22  0.655  23.293   21  0.366 0.712   7 42
                                                             21  0.753  28.077   22  0.627 0.776  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              20  0.821  30.853   23  1.255 1.359  15 40
59                -- -- -- -- --                59           19  0.874  32.355   24  1.210 1.370  17 44
55          -- -- -- -- -- -- -- -- --          55           18  0.916  32.748   25  1.055 1.290  13 44
51       -- -- -- -- -- -- -- -- -- -- --       51           17  0.986  33.021   26  1.455 1.536  15 42
47    -- -- 34 -- -- -- 12 -- -- -- 34 -- --    47           16  1.041  33.371   27  1.478 1.540  15 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43           15  1.074  33.966   28  1.258 1.560  17 42
39 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 39           14  1.079  34.539
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35           13  1.118  33.775
31 -- -- -- 12 -- -- --  0 -- -- -- 12 -- -- -- 31           12  1.132  34.264
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27           11  1.153  35.072
23 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 23           10  1.184  35.756
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19            9  1.226  36.196
15    -- -- 34 -- -- -- 12 -- -- -- 34 -- --    15            8  1.278  37.068
11       -- -- -- -- -- -- -- -- -- -- --       11            7  1.318  38.007
 7          -- -- -- -- -- -- -- -- --           7            6  1.374  38.856
 3                -- -- -- -- --                 3            5  1.410* 39.527*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               4  1.403  38.531
                                                              3  1.333  35.329
Control Rod Density: %          4.80                          2  1.057  27.412
                                                       Bottom 1  0.317   8.184
k-effective:                 0.99746
Void Fraction:                 0.378             % AXIAL TILT -18.203 -10.166
Core Delta-P: psia            21.885         AVG BOT 8ft/12ft  1.1560  1.0990
Core Plate Delta-P: psia      17.451
Coolant Temp: Deg-F            543.4
In Channel Flow: Mlb/hr        85.54   Active Channel Flow: Mlb/hr        81.73
Total Bypass Flow (%):          10.9   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00041
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.560	28	17	42	1.857	0.775	28	17	42	9.76	0.846	21.5	28	35	12	4	7.66	0.824	52.6	23	39	48	4
1.559	28	19	44	1.865	0.772	28	19	44	9.76	0.845	21.4	28	11	36	4	7.65	0.824	52.8	23	47	40	4
1.540	27	15	36	1.918	0.751	26	15	42	9.71	0.834	20.5	27	37	12	4	7.69	0.822	52.1	23	23	48	4
1.538	27	25	46	1.929	0.746	26	19	46	9.71	0.833	20.4	27	11	38	4	7.68	0.821	52.1	23	47	24	4
1.536	26	15	42	1.930	0.746	27	15	26	9.52	0.824	21.4	27	25	14	4	10.95	0.819	24.9	27	23	12	4
1.534	26	19	46	1.933	0.745	26	27	44	9.52	0.824	21.3	27	13	36	4	10.95	0.818	24.8	27	11	38	4
1.531	27	13	36	1.935	0.744	26	17	34	9.31	0.804	21.1	26	33	12	4	10.90	0.817	25.2	27	25	14	4
1.528	27	25	48	1.936	0.744	27	25	16	9.30	0.802	21.0	26	11	34	4	10.90	0.816	25.1	27	13	26	4
1.509	26	17	38	1.947	0.740	27	25	48	9.37	0.801	19.7	27	21	12	4	10.77	0.812	25.8	28	35	12	4
1.508	26	23	44	1.948	0.739	26	17	38	9.37	0.801	19.7	27	11	40	4	10.77	0.811	25.7	28	11	36	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.30 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 10,000.0 MWd/MTU

```
Cycle:                           24                       Core Average Exposure: MWd/MTU    31201.2
Exposure: MWd/MTU (GWd)      10500.0 (1321.00 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )                 Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                     Top 24  0.176   5.250    19  0.302 0.548  55 40
Inlet Subcooling: Btu/lbm     -24.74                         23  0.319  11.959    20  0.254 0.266   3 42
Flow: Mlb/hr                   96.04 ( 98.00 %)              22  0.650  23.650    21  0.365 0.713   7 42
                                                             21  0.748  28.487    22  0.629 0.780  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              20  0.826  31.301    23  1.249 1.354  15 40
59                -- -- -- -- -- --              59          19  0.888  32.832    24  1.196 1.371  17 44
55          -- -- -- -- -- -- -- -- -- --        55          18  0.939  33.250    25  1.058 1.294  13 44
51       -- -- -- -- -- -- -- -- -- -- -- --     51          17  1.016  33.541    26  1.455 1.544  15 42
47    -- -- 34 -- -- -- 12 -- -- -- 34 -- --     47          16  1.077  33.907    27  1.486 1.545  15 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --     43          15  1.114  34.519    28  1.270 1.564  17 42
39 -- -- -- -- --  6 -- -- --  6 -- -- -- -- -- 39          14  1.119  35.095
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35          13  1.155  34.323
31 -- -- -- 12 -- -- --  0 -- -- -- 12 -- -- -- 31          12  1.166  34.819
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27          11  1.185  35.637
23 -- -- -- -- --  6 -- -- --  6 -- -- -- -- -- 23          10  1.211  36.336
19    -- -- -- -- -- -- -- -- -- -- -- -- --     19           9  1.247  36.790
15    -- -- 34 -- -- -- 12 -- -- -- 34 -- --     15           8  1.289  37.678
11       -- -- -- -- -- -- -- -- -- -- -- --     11           7  1.311  38.635
 7          -- -- -- -- -- -- -- -- -- --         7           6  1.340  39.511
 3                -- -- -- -- -- --               3           5  1.352* 40.198*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               4  1.327  39.199
                                                              3  1.254  35.965
Control Rod Density: %          4.90                          2  0.996  27.916
                                                       Bottom  1  0.298   8.336
k-effective:                 0.99743
Void Fraction:                 0.373             % AXIAL TILT -16.451 -10.232
Core Delta-P: psia            21.840         AVG BOT 8ft/12ft  1.1524  1.0994
Core Plate Delta-P: psia      17.407
Coolant Temp: Deg-F            543.3
In Channel Flow: Mlb/hr        85.56   Active Channel Flow: Mlb/hr        81.77
Total Bypass Flow (%):          10.9   (of total core flow)
Total Water Rod Flow (%):        4.0   (of total core flow)
Source Convergence           0.00048
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.564	28	17	42	1.833	0.785	28	17	42	9.30	0.815	22.7	28	25	50	4	7.14	0.789	55.3	23	47	40	5
1.563	28	19	44	1.841	0.782	28	19	44	9.29	0.814	22.6	28	11	36	4	7.14	0.789	55.1	23	39	48	5
1.545	27	15	36	1.883	0.765	26	15	42	9.27	0.804	21.7	27	23	50	4	7.16	0.785	54.5	23	23	48	5
1.544	26	15	42	1.894	0.760	26	19	46	9.26	0.804	21.6	27	11	38	4	10.37	0.785	26.3	27	23	12	4
1.543	27	25	46	1.899	0.758	27	15	26	9.00	0.788	22.5	27	25	48	4	10.37	0.785	26.2	27	11	38	4
1.542	26	19	46	1.905	0.756	27	25	16	8.99	0.787	22.5	27	13	36	4	7.15	0.785	54.5	23	47	24	5
1.540	27	13	36	1.911	0.754	26	17	34	8.88	0.775	22.2	26	27	12	4	10.33	0.782	26.3	28	25	50	4
1.537	27	25	48	1.912	0.753	27	25	48	8.88	0.775	22.2	26	11	34	4	10.24	0.781	27.1	28	11	36	4
1.516	27	13	42	1.916	0.751	26	27	44	8.98	0.774	20.9	27	21	12	4	10.29	0.781	26.6	27	25	14	4
1.516	27	15	44	1.917	0.751	27	47	26	8.98	0.774	20.9	27	11	40	4	10.29	0.780	26.5	27	13	26	4

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.31 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 10,500.0 MWd/MTU

```
Cycle:                            24                  Core Average Exposure: MWd/MTU    31701.2
Exposure: MWd/MTU (GWd)      11000.0 (1383.90 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )            Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)      N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                  Top 24  0.169   5.327   19  0.302 0.548  55 40
Inlet Subcooling: Btu/lbm     -24.74                      23  0.308  12.130   20  0.254 0.265   3 42
Flow: Mlb/hr                   96.04 ( 98.00 %)           22  0.639  24.003   21  0.365 0.715   7 42
                                                          21  0.748  28.895   22  0.631 0.784  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58           20  0.836  31.753   23  1.244 1.351  13 40
59                -- -- -- -- -- --              59       19  0.906  33.317   24  1.180 1.371  17 44
55          -- -- -- -- -- -- -- -- -- --        55       18  0.966  33.763   25  1.060 1.298  13 44
51       -- -- -- -- -- -- -- -- -- -- -- --     51       17  1.052  34.077   26  1.455 1.552  15 42
47    -- -- 34 -- -- -- 12 -- -- -- 34 -- --     47       16  1.120  34.461   27  1.494 1.551  15 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --     43       15  1.159  35.093   28  1.282 1.568  17 42
39 -- -- -- -- -- --  0 -- -- --  0 -- -- -- -- -- 39     14  1.164  35.671
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35     13  1.199  34.889
31 -- -- -- 12 -- -- --  0 -- -- -- 12 -- -- -- 31       12  1.206  35.390
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27     11  1.222  36.217
23 -- -- -- -- -- --  0 -- -- --  0 -- -- -- -- -- 23     10  1.243  36.929
19    -- -- -- -- -- -- -- -- -- -- -- -- --     19        9  1.271  37.394
15    -- -- 34 -- -- -- 12 -- -- -- 34 -- --     15        8  1.300  38.292
11       -- -- -- -- -- -- -- -- -- -- -- --     11        7  1.302  39.259
 7          -- -- -- -- -- -- -- -- -- --         7        6  1.302* 40.149
 3                -- -- -- -- -- --               3        5  1.285  40.842*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            4  1.241  39.832
                                                           3  1.162  36.562
Control Rod Density: %          5.18                       2  0.924  28.391
                                                   Bottom  1  0.277   8.479
k-effective:                 0.99740
Void Fraction:                 0.368             % AXIAL TILT -14.453 -10.268
Core Delta-P: psia            21.792          AVG BOT 8ft/12ft  1.1485  1.0998
Core Plate Delta-P: psia      17.357
Coolant Temp: Deg-F            543.2
In Channel Flow: Mlb/hr        85.59   Active Channel Flow: Mlb/hr        81.81
Total Bypass Flow (%):          10.9   (of total core flow)
Total Water Rod Flow (%):        3.9   (of total core flow)
Source Convergence           0.00045
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.568	28	17	42	1.807	0.797	28	17	42	8.75	0.775	23.9	28	25	50	4	6.80	0.759	56.1	23	39	48	5
1.567	28	19	44	1.814	0.794	28	19	44	8.74	0.775	23.8	28	11	36	4	6.79	0.759	56.2	23	47	40	5
1.552	26	15	42	1.846	0.780	26	15	42	8.72	0.766	22.8	27	23	50	4	6.81	0.755	55.5	23	23	48	5
1.551	26	19	46	1.857	0.776	26	19	46	8.72	0.765	22.8	27	11	38	4	6.80	0.754	55.5	23	47	24	5
1.551	27	15	36	1.867	0.771	27	15	26	8.40	0.744	23.7	27	25	48	4	9.71	0.744	27.6	28	25	50	4
1.549	27	13	36	1.872	0.769	27	25	16	8.40	0.743	23.7	27	13	36	4	9.63	0.743	28.4	28	11	36	4
1.549	27	25	46	1.875	0.768	27	25	48	8.48	0.738	22.0	27	21	50	4	9.70	0.743	27.5	27	11	38	4
1.546	27	25	48	1.880	0.766	27	47	26	8.48	0.738	22.0	27	11	40	4	9.70	0.743	27.6	27	23	12	4
1.528	27	13	42	1.883	0.765	26	17	34	8.37	0.738	23.3	26	11	34	4	6.79	0.740	53.8	23	25	10	5
1.527	27	15	44	1.884	0.764	27	15	18	8.36	0.738	23.3	26	27	50	4	6.79	0.739	53.8	23	9	26	5

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.32 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 11,000.0 MWd/MTU

```
Cycle:                           24                         Core Average Exposure: MWd/MTU    32201.2
Exposure: MWd/MTU (GWd)      11500.0 (1446.80 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )                   Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)             N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                      Top 24  0.172   5.402    19  0.301 0.548  55 40
Inlet Subcooling: Btu/lbm     -24.74                          23  0.314  12.296    20  0.252 0.265   3 42
Flow: Mlb/hr                   96.04 ( 98.00 %)               22  0.652  24.351    21  0.364 0.716   7 42
                                                              21  0.765  29.302    22  0.631 0.788  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               20  0.855  32.209    23  1.238 1.352  13 40
59                -- -- -- -- -- --                59         19  0.929  33.813    24  1.175 1.373  17 44
55          -- -- -- -- -- -- -- -- -- --          55         18  0.997  34.292    25  1.061 1.302  13 44
51       -- -- -- -- -- -- -- -- -- -- -- --       51         17  1.091  34.631    26  1.456 1.562  15 42
47    -- -- 34 -- -- -- 10 -- -- -- 34 -- --    47            16  1.164  35.038    27  1.500 1.559  13 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43            15  1.207  35.689    28  1.291 1.574  17 42
39 -- -- -- -- -- --  0 -- -- --  0 -- -- -- -- -- 39         14  1.210  36.270
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35            13  1.242  35.477
31 -- -- -- 12 -- -- --  0 -- -- -- 12 -- -- -- 31            12  1.245  35.981
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27            11  1.256  36.816
23 -- -- -- -- -- --  0 -- -- --  0 -- -- -- -- -- 23         10  1.271  37.538
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19             9  1.290  38.010
15    -- -- 34 -- -- -- 10 -- -- -- 34 -- --    15             8  1.304* 38.911
11       -- -- -- -- -- -- -- -- -- -- -- --       11          7  1.282  39.880
 7          -- -- -- -- -- -- -- -- -- --           7          6  1.252  40.769
 3                -- -- -- -- -- --                 3          5  1.208  41.454*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                4  1.143  40.423
                                                               3  1.057  37.116
Control Rod Density: %          5.23                           2  0.841  28.831
                                                        Bottom 1  0.252   8.611
k-effective:                 0.99759
Void Fraction:                 0.361            % AXIAL TILT -11.685 -10.272
Core Delta-P: psia            21.723        AVG BOT 8ft/12ft  1.1391  1.1001
Core Plate Delta-P: psia      17.288
Coolant Temp: Deg-F            543.0
In Channel Flow: Mlb/hr        85.64   Active Channel Flow: Mlb/hr        81.87
Total Bypass Flow (%):          10.8   (of total core flow)
Total Water Rod Flow (%):        3.9   (of total core flow)
Source Convergence           0.00047
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.574	28	17	42	1.780	0.809	28	17	42	8.22	0.733	24.5	28	25	50	5	6.53	0.733	56.6	23	47	40	6
1.572	28	19	44	1.789	0.805	28	19	44	8.21	0.732	24.5	28	11	36	5	6.84	0.733	52.3	24	17	18	9
1.562	26	15	42	1.808	0.796	26	15	42	8.19	0.724	23.5	27	23	50	5	6.53	0.732	56.5	23	39	48	6
1.559	27	13	36	1.823	0.790	26	19	46	8.19	0.723	23.5	27	11	38	5	6.58	0.729	55.4	23	17	40	10
1.559	27	15	36	1.831	0.786	27	15	26	7.94	0.708	24.4	27	25	48	5	6.54	0.727	55.7	23	23	48	6
1.558	26	19	46	1.840	0.782	27	47	26	7.94	0.707	24.4	27	13	36	5	6.56	0.726	55.3	23	39	18	10
1.546	27	25	46	1.843	0.781	27	15	18	7.86	0.703	24.8	26	27	12	5	6.53	0.726	55.7	23	47	24	6
1.545	27	25	48	1.844	0.781	27	47	42	7.86	0.702	24.7	26	11	34	5	6.65	0.724	53.9	23	15	40	10
1.539	27	13	42	1.849	0.779	27	17	16	8.00	0.702	22.8	27	21	50	5	6.79	0.723	51.8	24	15	16	9
1.538	27	15	44	1.850	0.778	26	17	34	8.00	0.702	22.8	27	11	40	5	6.62	0.721	53.8	23	39	16	10

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.33 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 11,500.0 MWd/MTU

```
Cycle:                           24                  Core Average Exposure: MWd/MTU    32651.2
Exposure: MWd/MTU (GWd)     11950.0 (1503.40 )
Delta E: MWd/MTU, (GWd)       450.0 (  56.61 )             Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)        N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                  Top 24  0.175   5.471    19  0.301 0.546  55 40
Inlet Subcooling: Btu/lbm    -24.74                      23  0.319  12.449    20  0.251 0.263   3 42
Flow: Mlb/hr                  96.04 ( 98.00 %)           22  0.662  24.670    21  0.363 0.715   7 42
                                                         21  0.779  29.677    22  0.632 0.792  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58          20  0.872  32.630    23  1.231 1.348  13 40
59                -- -- -- -- -- --                59    19  0.947  34.270    24  1.169 1.374  17 44
55          -- -- -- -- -- -- -- -- -- --          55    18  1.023  34.783    25  1.062 1.304  13 44
51       -- -- -- -- -- -- -- -- -- -- -- --       51    17  1.124  35.149    26  1.456 1.568  15 42
47    -- -- 34 -- -- -- 10 -- -- -- 34 -- --       47    16  1.203  35.577    27  1.506 1.558  13 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --       43    15  1.248  36.249    28  1.300 1.578  17 42
39 -- -- -- -- -- --  0 -- -- --  0 -- -- -- -- -- 39    14  1.251  36.831
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35    13  1.281  36.024
31 -- -- -- 10 -- -- --  0 -- -- -- 10 -- -- -- -- 31    12  1.281  36.530
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27    11  1.288  37.370
23 -- -- -- -- -- --  0 -- -- --  0 -- -- -- -- -- 23    10  1.296  38.099
19    -- -- -- -- -- -- -- -- -- -- -- -- --       19     9  1.306* 38.572
15    -- -- 34 -- -- -- 10 -- -- -- 34 -- --       15     8  1.306  39.471
11       -- -- -- -- -- -- -- -- -- -- -- --       11     7  1.262  40.430
 7          -- -- -- -- -- -- -- -- -- --           7     6  1.207  41.306
 3                -- -- -- -- -- --                 3     5  1.140  41.972*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58           4  1.059  40.913
                                                          3  0.969  37.569
Control Rod Density: %         5.27                       2  0.772  29.192
                                                   Bottom  1  0.231   8.720
k-effective:                0.99760
Void Fraction:                0.355            % AXIAL TILT  -9.306 -10.237
Core Delta-P: psia           21.666        AVG BOT 8ft/12ft  1.1312  1.1002
Core Plate Delta-P: psia     17.231
Coolant Temp: Deg-F           542.9
In Channel Flow: Mlb/hr       85.67   Active Channel Flow: Mlb/hr        81.92
Total Bypass Flow (%):         10.8   (of total core flow)
Total Water Rod Flow (%):       3.9   (of total core flow)
Source Convergence          0.00044
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.578	28	17	42	1.761	0.818	28	17	42	8.09	0.715	19.0	28	17	42	15	6.91	0.747	53.1	24	17	18	9
1.577	28	19	44	1.768	0.814	28	19	44	8.09	0.715	18.9	28	19	44	15	6.68	0.746	56.2	23	17	40	10
1.568	26	15	42	1.780	0.809	26	15	42	7.99	0.706	18.9	26	27	36	15	6.66	0.744	56.1	23	39	18	10
1.566	26	19	46	1.790	0.804	26	19	46	7.99	0.706	18.9	26	25	34	15	6.75	0.741	54.6	23	47	40	9
1.558	27	13	36	1.813	0.794	27	15	18	7.82	0.704	25.5	28	25	50	5	6.73	0.740	54.7	23	15	40	10
1.555	27	25	48	1.815	0.793	27	47	42	7.82	0.704	25.5	28	11	36	5	6.74	0.740	54.5	23	39	48	9
1.555	27	15	36	1.815	0.793	27	17	16	7.97	0.698	17.9	27	15	44	14	6.72	0.738	54.6	23	39	16	10
1.553	27	25	46	1.818	0.792	27	41	48	7.96	0.697	17.9	27	17	46	14	6.98	0.735	50.7	24	15	46	10
1.546	27	15	44	1.823	0.790	27	15	26	7.90	0.696	18.6	26	21	34	15	6.60	0.728	54.9	23	23	48	9
1.545	27	13	42	1.825	0.789	27	25	48	7.90	0.696	18.6	26	27	40	15	6.60	0.728	54.9	23	47	24	9

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.34 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 11,950.0 MWd/MTU

```
Cycle:                            24                    Core Average Exposure: MWd/MTU    32652.2
Exposure: MWd/MTU (GWd)      11951.0 (1503.50 )
Delta E: MWd/MTU, (GWd)          1.0 (   0.13 )             Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)        N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                  Top 24  0.213   5.471    19  0.298 0.529  55 40
Inlet Subcooling: Btu/lbm     -24.74                      23  0.386  12.449    20  0.242 0.253  57 42
Flow: Mlb/hr                   96.04 ( 98.00 %)           22  0.790  24.671    21  0.355 0.682  53 42
                                                          21  0.916  29.678    22  0.636 0.824  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58           20  1.005  32.631    23  1.186 1.360  47 32
59                -- -- -- -- --               59         19  1.073  34.271    24  1.280 1.397  29 30
55          -- -- -- -- -- -- -- -- -- --      55         18  1.123  34.784    25  1.034 1.297  51 32
51       -- -- -- -- -- -- -- -- -- -- -- --   51         17  1.178  35.150    26  1.500 1.554  45 34
47    -- -- -- -- 14 -- -- -- 14 -- -- -- --   47         16  1.218  35.578    27  1.439 1.534  23 26
43    -- -- -- -- -- -- -- -- -- -- -- -- --   43         15  1.232* 36.250    28  1.281 1.520  43 20
39 -- -- -- 14 -- -- --  6 -- -- -- 14 -- -- -- 39         14  1.209  36.832
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35         13  1.218  36.026
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31         12  1.203  36.531
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27         11  1.199  37.371
23 -- -- -- 14 -- -- --  8 -- -- -- 14 -- -- -- 23         10  1.203  38.100
19    -- -- -- -- -- -- -- -- -- -- -- -- --   19          9  1.215  38.573
15    -- -- -- -- 14 -- -- -- 14 -- -- -- --   15          8  1.226  39.472
11       -- -- -- -- -- -- -- -- -- -- -- --   11          7  1.205  40.431
 7          -- -- -- -- -- -- -- -- -- --       7          6  1.164  41.307
 3                -- -- -- -- --                3          5  1.104  41.973*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            4  1.025  40.914
                                                           3  0.934  37.570
Control Rod Density: %          5.11                       2  0.741  29.193
                                                   Bottom  1  0.221   8.720
k-effective:                 0.99761
Void Fraction:                 0.345             % AXIAL TILT  -3.652 -10.237
Core Delta-P: psia            21.525         AVG BOT 8ft/12ft  1.0822  1.1002
Core Plate Delta-P: psia      17.091
Coolant Temp: Deg-F            542.8
In Channel Flow: Mlb/hr        85.75   Active Channel Flow: Mlb/hr        82.04
Total Bypass Flow (%):          10.7   (of total core flow)
Total Water Rod Flow (%):        3.9   (of total core flow)
Source Convergence           0.00043
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.554	26	45	34	1.768	0.814	26	43	28	8.35	0.739	19.1	27	23	26	15	7.33	0.765	49.8	24	31	32	15
1.552	26	33	16	1.769	0.814	26	33	18	8.35	0.739	19.1	27	25	24	15	7.13	0.748	50.2	24	23	24	15
1.550	26	43	28	1.790	0.805	26	45	34	8.38	0.738	18.4	26	29	28	15	6.97	0.747	52.3	24	39	24	14
1.548	26	33	18	1.793	0.803	26	33	16	8.38	0.737	18.4	26	27	30	15	6.95	0.745	52.4	24	23	40	14
1.534	27	23	26	1.794	0.803	26	41	26	8.17	0.722	18.9	26	27	26	15	6.91	0.728	50.7	23	39	26	14
1.534	27	25	24	1.796	0.802	26	35	20	8.17	0.722	18.9	26	25	28	15	6.97	0.728	49.9	24	39	40	15
1.534	26	31	28	1.798	0.801	28	41	18	8.11	0.715	18.6	26	19	26	15	6.93	0.726	50.2	24	33	34	15
1.533	26	41	26	1.801	0.800	28	17	20	8.10	0.714	18.6	26	25	20	15	6.89	0.726	50.7	23	25	40	14
1.532	26	35	20	1.804	0.798	26	31	28	8.00	0.703	18.3	26	17	28	15	7.07	0.725	47.9	24	25	26	15
1.532	26	33	30	1.805	0.798	26	33	32	7.99	0.703	18.3	26	19	22	15	6.72	0.711	51.0	24	31	26	15

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.35 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 11,951.0 MWd/MTU

```
Cycle:                            24
Exposure: MWd/MTU (GWd)      12000.0 (1509.70 )
Delta E: MWd/MTU, (GWd)         49.0 (   6.16 )
Power: MWt                    2957.0 (100.00 %)
Core Pressure: psia           1015.0
Inlet Subcooling: Btu/lbm     -24.74
Flow: Mlb/hr                   96.04 ( 98.00 %)

    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                -- -- -- -- -- --             59
55          -- -- -- -- -- -- -- -- -- --       55
51       -- -- -- -- -- -- -- -- -- -- -- --    51
47    -- -- -- -- 14 -- -- -- 14 -- -- -- --    47
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43
39 -- -- -- 14 -- -- --  6 -- -- -- 14 -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- 14 -- -- --  6 -- -- -- 14 -- -- -- 23
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19
15    -- -- -- -- 14 -- -- -- 14 -- -- -- --    15
11       -- -- -- -- -- -- -- -- -- -- -- --    11
 7          -- -- -- -- -- -- -- -- -- --        7
 3                -- -- -- -- -- --              3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58

Control Rod Density: %          5.13

k-effective:                 0.99757
Void Fraction:                 0.344
Core Delta-P: psia            21.522
Core Plate Delta-P: psia      17.088
Coolant Temp: Deg-F            542.8
In Channel Flow: Mlb/hr        85.75   Active Channel Flow: Mlb/hr        82.04
Total Bypass Flow (%):          10.7   (of total core flow)
Total Water Rod Flow (%):        3.9   (of total core flow)
Source Convergence           0.00039
```

```
Core Average Exposure: MWd/MTU    32701.2

          Axial Profile    Edit   Radial Power
    N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
    Top 24  0.213   5.480   19  0.298 0.530  55 40
        23  0.385  12.469   20  0.242 0.253  57 42
        22  0.787  24.713   21  0.355 0.683  53 42
        21  0.913  29.727   22  0.636 0.826  13 48
        20  1.005  32.685   23  1.186 1.362  47 32
        19  1.075  34.328   24  1.277 1.392  31 32
        18  1.127  34.844   25  1.035 1.299  51 32
        17  1.183  35.211   26  1.500 1.557  45 34
        16  1.224  35.640   27  1.440 1.531  37 36
        15  1.238* 36.312   28  1.283 1.522  43 42
        14  1.214  36.893
        13  1.223  36.084
        12  1.207  36.589
        11  1.202  37.429
        10  1.206  38.158
         9  1.217  38.631
         8  1.226  39.529
         7  1.203  40.487
         6  1.159  41.362
         5  1.097  42.024*
         4  1.017  40.962
         3  0.926  37.614
         2  0.734  29.227
 Bottom  1  0.220   8.731

      % AXIAL TILT  -3.450 -10.221
  AVG BOT 8ft/12ft  1.0820  1.1002
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.557	26	45	34	1.764	0.817	26	43	34	8.37	0.741	19.1	27	37	36	15	7.34	0.767	49.9	24	31	32	15
1.552	26	43	34	1.774	0.812	26	33	18	8.36	0.740	19.1	27	35	38	15	6.98	0.749	52.4	24	39	24	14
1.549	26	33	46	1.784	0.807	26	45	34	8.40	0.740	18.5	26	31	34	15	7.13	0.749	50.3	24	23	24	15
1.542	26	33	44	1.791	0.804	26	41	36	8.40	0.739	18.5	26	33	32	15	6.97	0.748	52.5	24	23	40	14
1.534	26	41	36	1.794	0.803	26	33	46	8.18	0.724	19.0	26	25	34	15	7.00	0.732	49.9	24	39	40	15
1.531	27	37	36	1.796	0.802	28	17	42	8.18	0.724	19.0	26	27	26	15	6.92	0.730	50.8	23	39	26	14
1.528	26	33	32	1.800	0.800	28	41	18	8.15	0.719	18.7	26	41	36	15	6.91	0.729	50.8	23	25	40	14
1.527	27	35	38	1.801	0.800	26	25	42	8.12	0.717	18.7	26	25	42	15	6.94	0.729	50.3	24	33	34	15
1.527	26	31	34	1.806	0.797	26	33	32	8.05	0.708	18.4	26	43	34	15	7.07	0.725	48.0	24	25	26	15
1.526	26	35	42	1.807	0.797	26	31	34	8.03	0.707	18.4	26	41	40	15	6.77	0.714	50.8	23	15	32	14

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.36 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 12,000.0 MWd/MTU

```
Cycle:                            24                        Core Average Exposure: MWd/MTU    33201.9
Exposure: MWd/MTU (GWd)      12500.0 (1572.60 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )                  Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)            N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                       Top 24  0.217   5.576    19  0.298 0.531  55 40
Inlet Subcooling: Btu/lbm     -24.74                           23  0.393  12.678    20  0.242 0.253  57 42
Flow: Mlb/hr                   96.04 ( 98.00 %)                22  0.805  25.146    21  0.355 0.685  53 42
                                                               21  0.937  30.231    22  0.638 0.826  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                20  1.034  33.242    23  1.182 1.355  47 32
59                -- -- -- -- -- --                 59         19  1.109  34.926    24  1.263 1.366  31 32
55          -- -- -- -- -- -- -- -- -- --           55         18  1.164  35.471    25  1.038 1.304  51 32
51       -- -- -- -- -- -- -- -- -- -- -- --        51         17  1.223  35.845    26  1.498 1.559  45 34
47    -- -- -- -- 14 -- -- -- 14 -- -- -- --        47         16  1.265  36.280    27  1.448 1.525  45 36
43    -- -- -- -- -- -- -- -- -- -- -- -- --        43         15  1.278* 36.960    28  1.294 1.525  43 42
39 -- -- -- 14 -- -- --  6 -- -- -- 14 -- -- -- 39             14  1.252  37.528
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35             13  1.258  36.692
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31             12  1.239  37.188
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27             11  1.229  38.025
23 -- -- -- 14 -- -- --  6 -- -- -- 14 -- -- -- 23             10  1.226  38.754
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19              9  1.226  39.223
15    -- -- -- -- 14 -- -- -- 14 -- -- -- --    15              8  1.219  40.112
11       -- -- -- -- -- -- -- -- -- -- -- --       11           7  1.172  41.053
 7          -- -- -- -- -- -- -- -- -- --           7           6  1.104  41.901
 3                -- -- -- -- -- --                 3           5  1.022  42.529*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                 4  0.929  41.426
                                                                3  0.837  38.034
Control Rod Density: %          5.13                            2  0.665  29.561
                                                         Bottom 1  0.199   8.831
k-effective:                 0.99758
Void Fraction:                 0.337              % AXIAL TILT  -0.549 -10.036
Core Delta-P: psia            21.456          AVG BOT 8ft/12ft  1.0699  1.0994
Core Plate Delta-P: psia      17.022
Coolant Temp: Deg-F            542.6
In Channel Flow: Mlb/hr        85.79   Active Channel Flow: Mlb/hr        82.10
Total Bypass Flow (%):          10.7   (of total core flow)
Total Water Rod Flow (%):        3.9   (of total core flow)
Source Convergence           0.00036
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.559	26	45	34	1.755	0.820	26	43	34	8.53	0.758	19.6	26	31	34	15	7.37	0.778	50.9	24	31	32	15
1.551	26	33	16	1.763	0.817	26	45	34	8.52	0.758	19.6	26	33	32	15	7.11	0.761	52.2	24	39	24	15
1.550	26	43	34	1.765	0.816	26	33	18	8.46	0.756	20.2	27	37	36	15	7.17	0.760	51.3	24	23	24	15
1.541	26	33	18	1.772	0.812	26	33	16	8.45	0.755	20.2	27	35	38	15	7.10	0.760	52.2	24	23	40	15
1.534	26	49	32	1.783	0.808	28	17	42	8.35	0.744	19.8	26	41	36	15	7.09	0.749	50.9	24	39	40	15
1.532	26	49	34	1.785	0.807	28	41	18	8.32	0.741	19.7	26	35	42	15	6.97	0.743	51.8	23	39	26	14
1.527	26	31	12	1.786	0.806	26	41	36	8.30	0.741	20.1	26	35	34	15	6.96	0.741	51.8	23	25	40	14
1.527	26	41	36	1.795	0.802	26	35	20	8.29	0.740	20.1	26	33	26	15	6.98	0.740	51.3	24	33	34	15
1.526	26	33	12	1.800	0.800	26	41	40	8.31	0.738	19.4	26	43	34	15	7.11	0.736	49.0	24	25	26	15
1.525	27	45	36	1.806	0.797	26	39	42	8.27	0.734	19.4	26	33	44	15	6.90	0.735	51.7	23	15	32	14

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.37 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 12,500.0 MWd/MTU

```
Cycle:                            24                        Core Average Exposure: MWd/MTU    33701.2
Exposure: MWd/MTU (GWd)      13000.0 (1635.50 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )                  Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)            N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                       Top 24  0.226   5.672   19  0.298 0.533  39  6
Inlet Subcooling: Btu/lbm     -24.74                           23  0.409  12.890   20  0.241 0.252  57 42
Flow: Mlb/hr                   96.04 ( 98.00 %)                22  0.841  25.583   21  0.354 0.691  41  8
                                                               21  0.984  30.742   22  0.639 0.829  47 14
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                20  1.082  33.807   23  1.181 1.357  33 14
59                -- -- -- -- -- --                59          19  1.158  35.532   24  1.251 1.343  31 30
55          -- -- -- -- -- -- -- -- -- --          55          18  1.213  36.108   25  1.040 1.313  31 10
51       -- -- -- -- -- -- -- -- -- -- -- --       51          17  1.271  36.490   26  1.495 1.569  33 16
47    -- -- -- -- 14 -- -- -- 14 -- -- -- --    47          16  1.306  36.932   27  1.453 1.544  35 14
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43          15  1.313* 37.618   28  1.304 1.537  41 18
39 -- -- -- 14 -- -- --  8 -- -- -- 14 -- -- -- 39          14  1.282  38.173
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35          13  1.285  37.309
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31          12  1.260  37.795
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27          11  1.244  38.627
23 -- -- -- 14 -- -- --  8 -- -- -- 14 -- -- -- 23          10  1.231  39.354
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19           9  1.220  39.817
15    -- -- -- -- 14 -- -- -- 16 -- -- -- --    15           8  1.196  40.692
11       -- -- -- -- -- -- -- -- -- -- -- --       11           7  1.129  41.612
 7          -- -- -- -- -- -- -- -- -- --           7           6  1.041  42.427
 3                -- -- -- -- -- --                 3           5  0.942  43.016*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                 4  0.841  41.869
                                                                3  0.750  38.433
Control Rod Density: %          5.06                            2  0.596  29.877
                                                         Bottom  1  0.178   8.926
k-effective:                 0.99753
Void Fraction:                 0.329              % AXIAL TILT   3.098  -9.834
Core Delta-P: psia            21.374          AVG BOT 8ft/12ft  1.0510  1.0985
Core Plate Delta-P: psia      16.940
Coolant Temp: Deg-F            542.4
In Channel Flow: Mlb/hr        85.84   Active Channel Flow: Mlb/hr        82.17
Total Bypass Flow (%):          10.6   (of total core flow)
Total Water Rod Flow (%):        3.8   (of total core flow)
Source Convergence           0.00028
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.569	26	33	16	1.722	0.836	26	33	16	8.58	0.770	20.7	26	35	20	15	7.33	0.782	51.9	24	31	32	15
1.557	26	45	28	1.737	0.829	26	33	18	8.60	0.770	20.4	26	33	18	15	7.35	0.774	50.7	23	39	18	17
1.553	26	33	18	1.746	0.825	28	41	18	8.52	0.768	21.3	27	35	24	15	7.24	0.774	52.1	24	39	24	16
1.552	26	33	12	1.751	0.822	26	45	28	8.55	0.768	20.7	26	31	28	15	7.14	0.765	52.3	24	23	24	15
1.548	26	31	12	1.753	0.821	26	43	28	8.55	0.767	20.7	26	33	30	15	7.06	0.764	53.2	24	23	40	15
1.544	27	35	14	1.754	0.821	27	35	14	8.50	0.767	21.3	27	37	26	15	7.28	0.762	50.0	23	39	16	18
1.543	26	43	28	1.762	0.817	28	43	20	8.53	0.764	20.4	26	39	20	15	7.07	0.760	52.5	23	31	18	15
1.542	27	35	16	1.764	0.816	26	39	20	8.48	0.762	20.8	26	41	26	15	7.11	0.758	51.9	24	39	40	15
1.541	26	49	32	1.767	0.815	27	35	16	8.54	0.760	19.8	26	33	16	15	7.04	0.756	52.6	23	31	16	14
1.540	26	49	34	1.767	0.815	26	35	20	8.48	0.760	20.5	26	43	28	15	7.09	0.753	51.5	23	39	26	15

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.38 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 13,000.0 MWd/MTU

```
Cycle:                          24                     Core Average Exposure: MWd/MTU    34201.2
Exposure: MWd/MTU (GWd)    13500.0 (1698.40 )
Delta E: MWd/MTU, (GWd)      500.0 (  62.90 )                Axial Profile    Edit   Radial Power
Power: MWt                  2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia         1015.0                     Top 24  0.234   5.772    19  0.299 0.535  55 40
Inlet Subcooling: Btu/lbm   -24.74                         23  0.423  13.110    20  0.242 0.254   3 42
Flow: Mlb/hr                 96.04 ( 98.00 %)              22  0.871  26.041    21  0.356 0.693   7 42
                                                           21  1.024  31.277    22  0.642 0.830  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            20  1.130  34.398    23  1.181 1.352  47 28
59                -- -- -- -- --              59           19  1.212  36.165    24  1.231 1.332  43 18
55          -- -- -- -- -- -- -- -- -- --     55           18  1.273  36.772    25  1.046 1.320  51 32
51       -- -- -- -- -- -- -- -- -- -- -- --  51           17  1.339  37.160    26  1.485 1.561  45 28
47    -- -- -- -- 16 -- -- -- 14 -- -- -- --  47           16  1.358* 37.604    27  1.462 1.544  47 26
43    -- -- -- -- -- -- -- -- -- -- -- -- --  43           15  1.353  38.294    28  1.318 1.543  11 36
39 -- -- -- 16 -- -- --  8 -- -- -- 16 -- -- -- 39        14  1.312  38.833
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35        13  1.311  37.939
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31        12  1.279  38.413
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27        11  1.254  39.237
23 -- -- -- 16 -- -- --  8 -- -- -- 16 -- -- -- 23        10  1.232  39.958
19    -- -- -- -- -- -- -- -- -- -- -- -- --  19            9  1.207  40.408
15    -- -- -- -- 14 -- -- -- 16 -- -- -- --  15            8  1.166  41.262
11       -- -- -- -- -- -- -- -- -- -- -- --  11            7  1.079  42.149
 7          -- -- -- -- -- -- -- -- -- --      7            6  0.974  42.922
 3                -- -- -- -- --               3            5  0.862  43.464*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             4  0.754  42.269
                                                            3  0.666  38.791
Control Rod Density: %        4.94                          2  0.529  30.161
                                                    Bottom  1  0.158   9.012
k-effective:               0.99750
Void Fraction:               0.320             % AXIAL TILT   6.995  -9.585
Core Delta-P: psia          21.291         AVG BOT 8ft/12ft  1.0309  1.0975
Core Plate Delta-P: psia    16.857
Coolant Temp: Deg-F          542.2
In Channel Flow: Mlb/hr      85.89   Active Channel Flow: Mlb/hr        82.25
Total Bypass Flow (%):        10.6   (of total core flow)
Total Water Rod Flow (%):      3.8   (of total core flow)
Source Convergence         0.00045
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.561	26	45	28	1.719	0.855	26	45	28	8.68	0.785	21.5	26	43	28	15	7.51	0.801	51.8	23	17	40	17
1.561	26	11	34	1.730	0.850	26	33	16	8.69	0.782	20.9	26	45	28	15	7.50	0.798	51.7	23	39	18	17
1.558	26	49	32	1.737	0.846	26	43	28	8.63	0.780	21.5	26	33	18	15	7.39	0.793	52.5	24	39	24	17
1.554	26	33	16	1.739	0.845	27	47	26	8.67	0.779	20.8	28	17	42	17	7.39	0.793	52.5	24	23	40	17
1.553	26	33	12	1.741	0.844	26	49	34	8.58	0.779	21.9	26	41	26	15	7.45	0.788	51.0	23	15	40	18
1.548	26	31	12	1.742	0.844	28	17	42	8.67	0.778	20.7	28	19	44	17	7.47	0.787	50.7	23	15	24	18
1.544	27	47	26	1.742	0.844	28	41	18	8.60	0.778	21.6	26	41	22	15	7.41	0.783	51.0	23	39	16	18
1.543	28	11	36	1.746	0.842	26	27	44	8.59	0.777	21.5	26	39	20	15	7.37	0.779	51.0	24	39	40	17
1.538	27	35	14	1.750	0.840	26	49	32	8.63	0.776	20.9	26	33	16	15	7.27	0.777	52.0	23	23	46	16
1.538	26	43	28	1.750	0.840	27	25	48	8.56	0.776	21.9	26	35	20	15	7.19	0.776	52.9	24	31	32	15

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.39 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 13,500.0 MWd/MTU

```
Cycle:                          24                 Core Average Exposure: MWd/MTU    34301.2
Exposure: MWd/MTU (GWd)    13600.0 (1711.00 )
Delta E: MWd/MTU, (GWd)      100.0 (  12.58 )              Axial Profile    Edit   Radial Power
Power: MWt                  2957.0 (100.00 %)         N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia         1015.0                   Top 24  0.235   5.793    19  0.300 0.534  55 40
Inlet Subcooling: Btu/lbm   -24.74                       23  0.426  13.155    20  0.242 0.254   3 42
Flow: Mlb/hr                 96.04 ( 98.00 %)            22  0.878  26.135    21  0.356 0.692   7 42
                                                         21  1.033  31.389    22  0.642 0.828  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58          20  1.141  34.522    23  1.181 1.349  13 28
59                -- -- -- -- --                59       19  1.224  36.298    24  1.227 1.330  17 18
55          -- -- -- -- -- -- -- -- --          55       18  1.286  36.911    25  1.047 1.320   9 32
51       -- -- -- -- -- -- -- -- -- -- --       51       17  1.354  37.301    26  1.483 1.562  11 34
47    -- -- -- -- 16 -- -- -- 14 -- -- -- --    47       16  1.370* 37.744    27  1.463 1.543  47 26
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43       15  1.361  38.433    28  1.321 1.543  11 36
39 -- -- -- 16 -- -- --  8 -- -- -- 16 -- -- -- 39       14  1.318  38.968
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35       13  1.316  38.067
31 -- -- -- -- --  8 -- -- --  8 -- -- -- -- -- 31       12  1.283  38.539
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27       11  1.256  39.360
23 -- -- -- 16 -- -- --  8 -- -- -- 16 -- -- -- 23       10  1.231  40.079
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19        9  1.203  40.525
15    -- -- -- -- 16 -- -- -- 16 -- -- -- --    15        8  1.158  41.373
11       -- -- -- -- -- -- -- -- -- -- --       11        7  1.068  42.252
 7          -- -- -- -- -- -- -- -- --           7        6  0.959  43.015
 3                -- -- -- -- --                 3        5  0.845  43.546*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58           4  0.737  42.341
                                                          3  0.649  38.854
Control Rod Density: %        4.92                        2  0.515  30.212
                                                   Bottom 1  0.154   9.027
k-effective:               0.99753
Void Fraction:               0.319             % AXIAL TILT   7.847  -9.525
Core Delta-P: psia          21.273         AVG BOT 8ft/12ft  1.0265  1.0972
Core Plate Delta-P: psia    16.839
Coolant Temp: Deg-F          542.2
In Channel Flow: Mlb/hr      85.90   Active Channel Flow: Mlb/hr        82.26
Total Bypass Flow (%):        10.6   (of total core flow)
Total Water Rod Flow (%):      3.8   (of total core flow)
Source Convergence         0.00050
```

```
                     Top Ten Thermal Limits Summary - Sorted by Margin

     Power               MCPR                      APLHGR                            LHGR
Value FT   IR JR   Value Margin FT   IR JR   Value Margin Exp. FT   IR JR  K   Value Margin Exp. FT   IR JR  K
1.562  26 11 34    1.722  0.854  26 45 28    8.68  0.786 21.8  26 43 28 15    7.52  0.803 52.0  23 17 40 17
1.559  26 33 12    1.723  0.853  26 33 16    8.68  0.786 21.7  26 33 18 15    7.52  0.802 51.9  23 39 18 17
1.559  26 11 32    1.739  0.846  26 49 34    8.70  0.784 21.2  26 45 28 15    7.55  0.797 50.9  23 15 24 18
1.558  26 45 28    1.739  0.845  26 17 28    8.70  0.784 21.1  26 33 16 15    7.54  0.796 50.9  23 23 16 18
1.557  26 33 16    1.740  0.845  27 47 26    8.71  0.784 21.0  28 17 20 17    7.39  0.795 52.7  24 39 24 17
1.556  26 31 12    1.741  0.845  26 27 18    8.71  0.783 20.9  28 19 18 17    7.38  0.794 52.7  24 23 40 17
1.543  28 11 36    1.743  0.843  28 41 18    8.64  0.780 21.3  26 19 22 16    7.47  0.791 51.3  23 15 40 18
1.543  27 47 26    1.745  0.842  28 17 42    8.57  0.779 22.1  26 41 26 15    7.44  0.788 51.2  23 39 16 18
1.542  27 35 14    1.745  0.842  27 15 26    8.64  0.779 21.3  26 21 20 16    7.53  0.782 49.3  24 17 26 17
1.541  28 35 12    1.747  0.842  26 49 32    8.57  0.779 22.1  26 35 20 15    7.52  0.780 49.3  24 25 18 17

  * LHGR calculated with pin-power reconstruction
  * CPR calculated with pin-power reconstruction & CPR limit type 3
```

Figure A.40 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 13,600.0 MWd/MTU

```
Cycle:                           24                      Core Average Exposure: MWd/MTU    34302.2
Exposure: MWd/MTU (GWd)     13601.0 (1711.10 )
Delta E: MWd/MTU, (GWd)         1.0 (   0.13 )                 Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                     Top 24  0.209   5.793    19  0.297 0.540  55 40
Inlet Subcooling: Btu/lbm    -24.74                         23  0.377  13.156    20  0.246 0.257   3 42
Flow: Mlb/hr                  96.04 ( 98.00 %)              22  0.774  26.136    21  0.360 0.710  53 42
                                                            21  0.905  31.390    22  0.641 0.855  13 48
     2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            20  1.009  34.523    23  1.208 1.350  47 40
59                -- -- -- -- -- --             59          19  1.090  36.299    24  1.170 1.372  17 18
55          -- -- -- -- -- -- -- -- --          55          18  1.151  36.913    25  1.064 1.314  47 44
51       -- -- -- -- -- -- -- -- -- -- --       51          17  1.240  37.303    26  1.447 1.583  15 42
47    -- -- -- -- -- --  6 -- -- -- -- -- --    47          16  1.304  37.745    27  1.517 1.570  15 44
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43          15  1.332  38.434    28  1.329 1.580  17 42
39 -- -- -- -- -- 12 -- -- -- 12 -- -- -- -- -- 39          14  1.320  38.969
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35          13  1.341* 38.068
31 -- -- --  6 -- -- --  6 -- -- --  6 -- -- -- 31          12  1.325  38.540
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27          11  1.312  39.361
23 -- -- -- -- -- 12 -- -- -- 12 -- -- -- -- -- 23          10  1.295  40.080
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19           9  1.271  40.526
15    -- -- -- -- -- --  6 -- -- -- -- -- --    15           8  1.231  41.374
11       -- -- -- -- -- -- -- -- -- -- --       11           7  1.147  42.253
 7          -- -- -- -- -- -- -- -- --           7           6  1.045  43.016
 3                -- -- -- -- -- --              3           5  0.940  43.547*
     2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             4  0.839  42.342
                                                             3  0.754  38.855
Control Rod Density: %         4.17                          2  0.606  30.212
                                                     Bottom  1  0.183   9.027
k-effective:                0.99751
Void Fraction:                0.333             % AXIAL TILT   0.428  -9.524
Core Delta-P: psia           21.437         AVG BOT 8ft/12ft  1.0778  1.0972
Core Plate Delta-P: psia     17.005
Coolant Temp: Deg-F           542.5
In Channel Flow: Mlb/hr       85.80   Active Channel Flow: Mlb/hr        82.12
Total Bypass Flow (%):         10.7   (of total core flow)
Total Water Rod Flow (%):       3.8   (of total core flow)
Source Convergence          0.00046
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.583	26	15	42	1.719	0.855	26	15	42	8.47	0.771	22.3	28	43	42	15	6.80	0.763	56.6	23	15	40	11
1.582	26	19	46	1.724	0.853	26	41	16	8.47	0.771	22.3	28	19	44	15	6.80	0.763	56.5	23	39	16	11
1.580	28	17	42	1.729	0.850	27	15	18	8.43	0.760	21.1	26	45	42	15	7.05	0.762	53.2	23	31	40	15
1.579	28	19	44	1.730	0.850	27	17	16	8.43	0.759	21.0	26	19	46	15	7.05	0.762	53.2	23	39	32	15
1.570	27	15	44	1.734	0.848	28	17	42	8.33	0.750	21.1	27	45	44	15	6.80	0.759	56.1	23	17	40	13
1.569	27	17	46	1.738	0.846	28	41	18	8.33	0.750	21.0	27	43	46	15	6.82	0.759	55.8	23	47	40	11
1.565	27	13	42	1.746	0.842	27	47	42	8.25	0.748	21.7	27	45	36	15	6.66	0.759	57.9	23	39	18	11
1.564	27	19	48	1.748	0.841	27	41	48	8.25	0.747	21.7	27	25	46	15	6.81	0.757	55.7	23	39	48	11
1.555	27	13	36	1.777	0.827	27	25	48	8.18	0.742	21.9	26	23	44	15	6.76	0.755	56.1	23	47	24	11
1.554	27	25	48	1.778	0.827	27	47	26	8.17	0.742	21.9	26	43	38	15	6.76	0.755	56.1	23	23	48	11

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.41 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 13,601.0 MWd/MTU

```
Cycle:                              24
Exposure: MWd/MTU (GWd)        14000.0 (1761.30 )
Delta E: MWd/MTU, (GWd)          399.0 (  50.20 )
Power: MWt                      2957.0 (100.00 %)
Core Pressure: psia             1015.0
Inlet Subcooling: Btu/lbm       -24.74
Flow: Mlb/hr                     96.04 ( 98.00 %)

    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58
59                  -- -- -- -- -- --               59
55            -- -- -- -- -- -- -- -- -- --         55
51         -- -- -- -- -- -- -- -- -- -- -- --      51
47      -- -- -- -- -- -- 10 -- -- -- -- -- -- --   47
43      -- -- -- -- -- -- -- -- -- -- -- -- -- --   43
39 -- -- -- -- -- 12 -- -- -- 14 -- -- -- -- -- 39
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35
31 -- -- -- 10 -- -- --  6 -- -- -- 10 -- -- -- 31
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27
23 -- -- -- -- -- 14 -- -- -- 14 -- -- -- -- -- 23
19      -- -- -- -- -- -- -- -- -- -- -- -- -- --   19
15      -- -- -- -- -- -- 10 -- -- -- -- -- -- --   15
11         -- -- -- -- -- -- -- -- -- -- -- --      11
 7            -- -- -- -- -- -- -- -- -- --          7
 3                  -- -- -- -- -- --                3
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58

Control Rod Density: %            3.91

k-effective:                   0.99748
Void Fraction:                   0.324
Core Delta-P: psia              21.334
Core Plate Delta-P: psia        16.903
Coolant Temp: Deg-F              542.3
In Channel Flow: Mlb/hr          85.86   Active Channel Flow: Mlb/hr        82.21
Total Bypass Flow (%):            10.6   (of total core flow)
Total Water Rod Flow (%):          3.8   (of total core flow)
Source Convergence             0.00046
```

Core Average Exposure: MWd/MTU 34701.2

	Axial Profile N(PRA)	Power	Exposure
Top	24	0.226	5.867
	23	0.409	13.318
	22	0.840	26.472
	21	0.990	31.784
	20	1.097	34.963
	19	1.166	36.774
	18	1.219	37.415
	17	1.289	37.825
	16	1.335	38.281
	15	1.350*	38.982
	14	1.328	39.511
	13	1.344	38.593
	12	1.322	39.058
	11	1.300	39.874
	10	1.274	40.586
	9	1.240	41.018
	8	1.187	41.843
	7	1.090	42.690
	6	0.978	43.413
	5	0.865	43.904*
	4	0.761	42.661
	3	0.678	39.142
	2	0.545	30.443
Bottom	1	0.165	9.097
% AXIAL TILT		4.945	-9.363
AVG BOT 8ft/12ft		1.0478	1.0967

Edit Zone	Radial Power Avg.	Max.	IR	JR
19	0.297	0.540	55	40
20	0.244	0.256	57	42
21	0.359	0.708	53	42
22	0.640	0.846	47	48
23	1.213	1.342	47	40
24	1.167	1.362	43	44
25	1.061	1.304	47	44
26	1.447	1.573	45	42
27	1.513	1.562	47	36
28	1.333	1.569	43	42

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.573	26	45	42	1.721	0.854	26	41	16	8.55	0.785	23.1	28	43	20	15	7.97	0.804	46.4	24	37	38	19
1.572	26	19	16	1.727	0.851	27	17	16	8.55	0.784	23.1	28	41	18	15	7.87	0.800	47.2	24	39	38	19
1.569	28	43	42	1.728	0.851	27	15	18	8.54	0.775	21.9	26	45	42	15	7.86	0.799	47.2	24	37	40	19
1.568	28	19	18	1.728	0.851	26	45	42	8.54	0.774	21.9	26	41	16	15	7.87	0.791	46.1	24	39	40	19
1.562	27	47	36	1.736	0.847	28	41	18	8.42	0.764	21.9	27	45	44	15	7.16	0.773	53.0	23	39	32	17
1.561	27	25	14	1.741	0.844	28	43	42	8.42	0.764	21.8	27	43	16	15	7.12	0.768	53.0	23	31	40	17
1.557	27	45	44	1.744	0.843	27	47	42	8.36	0.763	22.6	27	45	36	15	6.98	0.767	54.6	23	39	18	15
1.556	27	47	42	1.747	0.842	27	41	48	8.36	0.763	22.5	27	35	16	15	7.07	0.766	53.3	24	31	24	17
1.556	27	17	16	1.757	0.837	27	47	26	8.39	0.760	21.7	27	47	36	15	7.01	0.765	54.0	23	47	40	14
1.555	27	19	14	1.765	0.833	27	49	24	8.39	0.760	21.7	27	35	14	15	7.00	0.763	53.9	23	39	48	14

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.42 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 14,000.0 MWd/MTU

```
Cycle:                              24                    Core Average Exposure: MWd/MTU    35201.2
Exposure: MWd/MTU (GWd)      14500.0 (1824.20 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )                 Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                     Top 24  0.216   5.967    19  0.289 0.527  55 40
Inlet Subcooling: Btu/lbm     -24.74                         23  0.391  13.538    20  0.237 0.248  57 42
Flow: Mlb/hr                   96.04 ( 98.00 %)              22  0.805  26.929    21  0.349 0.693  53 42
                                                             21  0.950  32.323    22  0.626 0.829  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              20  1.052  35.563    23  1.203 1.320  47 40
59                -- -- -- -- -- --                59        19  1.134  37.412    24  1.236 1.353  43 44
55          -- -- -- -- -- -- -- -- --             55        18  1.204  38.083    25  1.041 1.283  47 44
51       -- -- -- -- -- -- -- -- -- -- --          51        17  1.294  38.504    26  1.455 1.562  45 42
47    -- -- -- -- -- --  8 -- -- -- -- -- --       47        16  1.353  38.969    27  1.506 1.540  45 44
43    -- -- -- -- -- -- -- -- -- -- -- -- --       43        15  1.377  39.677    28  1.318 1.575  43 42
39 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 39           14  1.363  40.195
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35           13  1.389* 39.252
31 -- -- --  8 -- -- --  8 -- -- --  8 -- -- -- 31           12  1.368  39.706
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27           11  1.342  40.511
23 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 23           10  1.307  41.210
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19            9  1.263  41.619
15    -- -- -- -- -- --  6 -- -- -- -- -- --    15            8  1.199  42.409
11       -- -- -- -- -- -- -- -- -- -- --       11            7  1.091  43.209
 7          -- -- -- -- -- -- -- -- --           7            6  0.968  43.879
 3                -- -- -- -- -- --              3            5  0.847  44.316*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               4  0.739  43.024
                                                              3  0.657  39.465
Control Rod Density: %          2.38                          2  0.531  30.702
                                                       Bottom 1  0.162   9.176
k-effective:                 0.99746
Void Fraction:                 0.323              % AXIAL TILT   4.389  -9.101
Core Delta-P: psia            21.358          AVG BOT 8ft/12ft  1.0596  1.0956
Core Plate Delta-P: psia      16.931
Coolant Temp: Deg-F            542.2
In Channel Flow: Mlb/hr        85.84   Active Channel Flow: Mlb/hr        82.18
Total Bypass Flow (%):          10.6   (of total core flow)
Total Water Rod Flow (%):        3.8   (of total core flow)
Source Convergence           0.00050
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.575	28	43	42	1.721	0.854	28	17	42	8.71	0.807	24.3	28	43	42	15	7.14	0.802	56.6	23	17	40	14
1.575	28	41	44	1.725	0.852	28	41	18	8.70	0.807	24.3	28	41	44	15	7.14	0.801	56.6	23	39	18	14
1.562	26	45	42	1.732	0.849	26	15	42	8.61	0.794	23.7	26	41	40	15	7.09	0.799	57.0	24	23	40	14
1.562	26	41	46	1.737	0.846	26	41	16	8.61	0.794	23.6	26	39	42	15	7.09	0.798	56.9	24	39	24	14
1.540	27	45	44	1.745	0.842	27	15	18	8.57	0.786	23.0	26	45	42	15	7.19	0.790	54.6	24	39	40	15
1.539	27	43	46	1.746	0.842	27	17	16	8.57	0.785	23.0	26	41	46	15	6.99	0.789	57.2	23	39	16	12
1.537	27	47	42	1.767	0.832	27	47	42	8.49	0.784	23.9	26	43	38	15	6.98	0.789	57.2	23	15	40	12
1.536	27	41	48	1.768	0.832	27	41	48	8.48	0.784	24.0	26	35	42	15	7.21	0.786	53.9	24	41	42	14
1.536	27	47	36	1.781	0.825	27	25	48	8.48	0.784	24.0	26	41	36	15	7.01	0.778	55.6	23	31	40	15
1.535	27	35	48	1.781	0.825	26	41	40	8.50	0.784	23.8	26	37	44	15	7.05	0.778	55.0	24	23	24	15

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.43 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 14,500.0 MWd/MTU

```
Cycle:                            24                      Core Average Exposure: MWd/MTU    35701.2
Exposure: MWd/MTU (GWd)      15000.0 (1887.10 )
Delta E: MWd/MTU, (GWd)        500.0 (  62.90 )                Axial Profile    Edit   Radial Power
Power: MWt                    2957.0 (100.00 %)           N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                     Top 24  0.239   6.063    19  0.290 0.526  55 40
Inlet Subcooling: Btu/lbm     -24.74                         23  0.432  13.748    20  0.236 0.246  57 42
Flow: Mlb/hr                   96.04 ( 98.00 %)              22  0.892  27.367    21  0.347 0.689  53 42
                                                             21  1.061  32.840    22  0.627 0.812  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58              20  1.183  36.138    23  1.219 1.305  47 38
59                -- -- -- -- --                59           19  1.264  38.032    24  1.218 1.325  43 44
55          -- -- -- -- -- -- -- -- --          55           18  1.318  38.742    25  1.038 1.261  47 44
51       -- -- -- -- -- -- -- -- -- -- --       51           17  1.383  39.187    26  1.454 1.534  45 20
47    -- -- -- -- -- -- 16 -- -- -- -- -- --    47           16  1.405* 39.665    27  1.496 1.548  47 26
43    -- -- -- -- -- -- -- -- -- -- -- -- --    43           15  1.399  40.385    28  1.325 1.540  43 20
39 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 39           14  1.364  40.897
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35           13  1.378  39.932
31 -- -- -- 16 -- -- -- 12 -- -- -- 16 -- -- -- 31           12  1.344  40.376
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27           11  1.305  41.168
23 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 23           10  1.260  41.851
19    -- -- -- -- -- -- -- -- -- -- -- -- --    19            9  1.204  42.231
15    -- -- -- -- -- -- 16 -- -- -- -- -- --    15            8  1.127  42.980
11       -- -- -- -- -- -- -- -- -- -- --       11            7  1.007  43.729
 7          -- -- -- -- -- -- -- -- --           7            6  0.876  44.340
 3                -- -- -- -- --                 3            5  0.751  44.720*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               4  0.644  43.376
                                                              3  0.567  39.778
Control Rod Density: %          1.93                          2  0.458  30.956
                                                      Bottom  1  0.139   9.254
k-effective:                 0.99736
Void Fraction:                 0.310              % AXIAL TILT  10.987  -8.855
Core Delta-P: psia            21.216          AVG BOT 8ft/12ft  1.0142  1.0947
Core Plate Delta-P: psia      16.790
Coolant Temp: Deg-F            541.9
In Channel Flow: Mlb/hr        85.92   Active Channel Flow: Mlb/hr        82.30
Total Bypass Flow (%):          10.5   (of total core flow)
Total Water Rod Flow (%):        3.8   (of total core flow)
Source Convergence           0.00044
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.548	27	47	26	1.725	0.852	27	47	26	8.65	0.810	25.4	28	43	20	15	7.58	0.819	53.1	23	31	16	18
1.548	27	25	14	1.727	0.851	27	25	48	8.65	0.809	25.3	28	41	18	15	7.56	0.818	53.2	23	15	32	18
1.540	28	43	20	1.736	0.847	27	25	16	8.63	0.796	23.8	27	35	14	15	7.64	0.812	51.5	23	47	32	18
1.540	28	41	18	1.738	0.846	27	15	26	8.56	0.796	24.7	27	35	16	15	7.63	0.810	51.5	23	31	48	18
1.540	28	49	36	1.741	0.845	28	49	26	8.61	0.795	23.9	27	47	36	15	7.15	0.803	56.6	23	39	18	15
1.539	28	25	12	1.742	0.844	28	25	50	8.55	0.795	24.7	27	45	26	15	7.33	0.803	54.4	23	39	32	17
1.534	26	45	20	1.743	0.843	28	41	18	8.54	0.795	24.7	26	41	22	15	7.33	0.803	54.3	23	31	40	17
1.533	26	41	16	1.744	0.843	28	17	42	8.54	0.795	24.7	26	39	20	15	7.14	0.802	56.7	23	17	40	15
1.528	27	25	16	1.755	0.838	26	41	16	8.51	0.793	24.9	26	37	18	15	7.26	0.798	54.6	23	31	18	17
1.528	27	45	26	1.757	0.837	26	15	42	8.51	0.793	24.9	26	43	24	15	8.15	0.797	43.2	24	31	32	20

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.44 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 15,000.0 MWd/MTU

```
Cycle:                           24                       Core Average Exposure: MWd/MTU    35951.2
Exposure: MWd/MTU (GWd)     15250.0 (1918.60 )
Delta E: MWd/MTU, (GWd)       250.0 (  31.45 )                   Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)             N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                          Top 24  0.224   6.116   19  0.289 0.522  55 40
Inlet Subcooling: Btu/lbm    -24.74                              23  0.404  13.864   20  0.232 0.242  57 42
Flow: Mlb/hr                  96.04 ( 98.00 %)                   22  0.834  27.609   21  0.342 0.679  53 42
                                                                 21  0.993  33.129   22  0.624 0.793  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                  20  1.108  36.461   23  1.260 1.329  33 14
59                -- -- -- -- -- --              59              19  1.191  38.377   24  1.179 1.295  43 44
55          -- -- -- -- -- -- -- -- -- --        55              18  1.254  39.102   25  1.031 1.296  51 32
51       -- -- -- -- -- -- -- -- -- -- -- --     51              17  1.336  39.551   26  1.455 1.563  33 12
47    -- -- -- -- -- -- -- -- -- -- -- -- -- --  47              16  1.386  40.027   27  1.487 1.579  25 14
43    -- -- -- -- -- -- -- -- -- -- -- -- -- --  43              15  1.401  40.746   28  1.331 1.567  25 12
39 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 39              14  1.382  41.248
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35              13  1.410* 40.270
31 -- -- -- -- -- -- -- --  6 -- -- -- -- -- -- 31              12  1.384  40.705
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27              11  1.347  41.488
23 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 23              10  1.300  42.160
19    -- -- -- -- -- -- -- -- -- -- -- -- -- --  19               9  1.241  42.523
15    -- -- -- -- -- -- -- -- -- -- -- -- -- --  15               8  1.162  43.249
11       -- -- -- -- -- -- -- -- -- -- -- --     11               7  1.039  43.969
 7          -- -- -- -- -- -- -- -- -- --         7               6  0.907  44.549
 3                -- -- -- -- -- --               3               5  0.782  44.898*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                   4  0.676  43.530
                                                                  3  0.601  39.913
Control Rod Density: %         0.49                               2  0.489  31.065
                                                          Bottom  1  0.150   9.287
k-effective:                0.99768
Void Fraction:                0.316              % AXIAL TILT   7.684  -8.688
Core Delta-P: psia           21.293          AVG BOT 8ft/12ft  1.0410  1.0940
Core Plate Delta-P: psia     16.871
Coolant Temp: Deg-F           542.0
In Channel Flow: Mlb/hr       85.87   Active Channel Flow: Mlb/hr        82.22
Total Bypass Flow (%):         10.6   (of total core flow)
Total Water Rod Flow (%):       3.8   (of total core flow)
Source Convergence          0.00038
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.579	27	25	14	1.701	0.864	27	47	26	8.78	0.814	24.4	27	35	14	15	7.36	0.823	56.3	23	47	32	13
1.579	27	47	26	1.702	0.864	27	25	48	8.76	0.813	24.4	27	47	26	15	7.36	0.823	56.2	23	31	48	13
1.567	28	25	12	1.709	0.860	28	49	26	8.66	0.809	25.2	27	35	16	15	7.31	0.811	55.5	23	31	16	14
1.567	28	49	36	1.710	0.860	28	25	50	8.65	0.809	25.3	27	45	26	15	7.31	0.811	55.5	23	47	34	14
1.563	26	33	12	1.719	0.855	26	49	34	8.69	0.807	24.5	26	33	16	15	7.31	0.810	55.4	23	33	48	14
1.563	26	49	34	1.720	0.855	26	33	50	8.68	0.807	24.6	26	45	28	15	7.28	0.808	55.6	23	15	32	14
1.552	27	35	16	1.729	0.850	27	25	16	8.74	0.802	23.2	28	35	12	15	7.09	0.799	56.9	23	47	24	14
1.552	27	45	26	1.733	0.848	27	15	26	8.79	0.802	22.4	26	33	12	15	7.10	0.798	56.8	23	23	48	14
1.547	26	49	32	1.737	0.846	26	33	16	8.72	0.801	23.3	28	49	36	15	7.07	0.794	56.6	23	23	16	14
1.547	26	31	12	1.742	0.844	26	15	34	8.56	0.801	25.2	26	33	18	15	7.05	0.792	56.6	23	15	24	14

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.45 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 15,250.0 MWd/MTU

```
Cycle:                           24                         Core Average Exposure: MWd/MTU    36106.4
Exposure: MWd/MTU (GWd)     15405.2 (1938.10 )
Delta E: MWd/MTU, (GWd)       155.2 (  19.53 )                   Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)             N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                        Top 24  0.226   6.147    19  0.284 0.514  55 40
Inlet Subcooling: Btu/lbm    -24.21                            23  0.410  13.932    20  0.228 0.238  57 42
Flow: Mlb/hr                  98.00 (100.00 %)                 22  0.847  27.750    21  0.336 0.670  53 42
                                                               21  1.010  33.297    22  0.615 0.781  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                20  1.126  36.649    23  1.253 1.316  33 48
59                -- -- -- -- -- --                  59        19  1.207  38.579    24  1.217 1.284  35 44
55          -- -- -- -- -- -- -- -- -- --            55        18  1.268  39.315    25  1.020 1.285  51 32
51       -- -- -- -- -- -- -- -- -- -- -- --         51        17  1.347  39.770    26  1.470 1.549  33 12
47    -- -- -- -- -- -- -- -- -- -- -- -- -- --      47        16  1.393  40.248    27  1.477 1.564  35 14
43    -- -- -- -- -- -- -- -- -- -- -- -- -- --      43        15  1.405  40.969    28  1.319 1.553  25 12
39 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 39             14  1.385  41.469
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35             13  1.412* 40.484
31 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 31             12  1.384  40.916
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27             11  1.346  41.693
23 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 23             10  1.297  42.357
19    -- -- -- -- -- -- -- -- -- -- -- -- -- --      19         9  1.235  42.709
15    -- -- -- -- -- -- -- -- -- -- -- -- -- --      15         8  1.153  43.420
11       -- -- -- -- -- -- -- -- -- -- -- --         11         7  1.028  44.123
 7          -- -- -- -- -- -- -- -- -- --             7         6  0.893  44.683
 3                -- -- -- -- -- --                   3         5  0.766  45.014*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                 4  0.659  43.630
                                                                3  0.584  40.002
Control Rod Density: %         0.00                             2  0.476  31.137
                                                       Bottom  1  0.146   9.309
k-effective:                0.99807
Void Fraction:                0.312            % AXIAL TILT   8.623  -8.600
Core Delta-P: psia           21.927        AVG BOT 8ft/12ft  1.0349  1.0936
Core Plate Delta-P: psia     17.507
Coolant Temp: Deg-F           542.0
In Channel Flow: Mlb/hr       87.66   Active Channel Flow: Mlb/hr       83.96
Total Bypass Flow (%):         10.5   (of total core flow)
Total Water Rod Flow (%):       3.8   (of total core flow)
Source Convergence          0.00035
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power				MCPR					APLHGR							LHGR						
Value	FT	IR	JR	Value	Margin	FT	IR	JR	Value	Margin	Exp.	FT	IR	JR	K	Value	Margin	Exp.	FT	IR	JR	K
1.564	27	35	14	1.732	0.849	27	47	26	8.71	0.811	24.7	27	35	14	15	7.29	0.818	56.6	23	47	32	13
1.563	27	47	26	1.732	0.849	27	25	48	8.70	0.810	24.8	27	47	26	15	7.30	0.818	56.5	23	31	48	13
1.553	28	25	12	1.738	0.846	28	49	26	8.60	0.807	25.6	27	35	16	15	7.26	0.807	55.8	23	47	34	14
1.553	28	49	36	1.739	0.845	28	25	50	8.59	0.806	25.6	27	45	26	15	7.26	0.807	55.7	23	33	48	14
1.549	26	33	12	1.748	0.841	26	49	34	8.64	0.805	24.9	26	33	16	15	7.25	0.807	55.8	23	31	16	14
1.549	26	49	34	1.749	0.841	26	33	50	8.63	0.804	24.9	26	45	28	15	7.22	0.804	55.9	23	15	32	14
1.537	27	35	16	1.761	0.835	27	25	16	8.53	0.800	25.6	26	33	18	15	7.04	0.795	57.2	23	47	24	14
1.537	27	45	26	1.765	0.833	27	15	26	8.52	0.800	25.6	26	43	34	15	7.04	0.795	57.1	23	23	48	14
1.533	26	31	12	1.768	0.831	26	33	16	8.69	0.800	23.5	28	35	12	15	7.01	0.790	56.9	23	23	16	14
1.533	26	49	32	1.773	0.829	26	15	34	8.74	0.800	22.8	26	33	12	15	6.99	0.788	56.9	23	15	24	14

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.46 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 15,405.2 MWd/MTU

```
Cycle:                            24                  Core Average Exposure: MWd/MTU    36186.7
Exposure: MWd/MTU (GWd)     15484.8 (1948.10 )
Delta E: MWd/MTU, (GWd)        79.6 (  10.01 )              Axial Profile    Edit   Radial Power
Power: MWt                   2957.0 (100.00 %)        N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                  Top 24  0.229   6.163    19  0.284 0.514  55 40
Inlet Subcooling: Btu/lbm    -23.70                      23  0.414  13.967    20  0.228 0.238  57 42
Flow: Mlb/hr                  99.96 (102.00 %)           22  0.857  27.824    21  0.336 0.670  53 42
                                                         21  1.021  33.385    22  0.616 0.781  13 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58          20  1.138  36.747    23  1.252 1.316  33 48
59                -- -- -- -- --               59        19  1.218  38.685    24  1.215 1.283  35 44
55          -- -- -- -- -- -- -- -- --         55        18  1.277  39.426    25  1.021 1.287  29 10
51       -- -- -- -- -- -- -- -- -- -- -- --   51        17  1.353  39.883    26  1.469 1.552  33 12
47    -- -- -- -- -- -- -- -- -- -- -- -- -- --  47      16  1.397  40.363    27  1.478 1.565  35 14
43    -- -- -- -- -- -- -- -- -- -- -- -- -- --  43      15  1.408  41.085    28  1.322 1.556  25 12
39 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 39       14  1.387  41.582
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35       13  1.413* 40.594
31 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 31       12  1.385  41.024
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27       11  1.345  41.798
23 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 23       10  1.295  42.458
19    -- -- -- -- -- -- -- -- -- -- -- -- -- --  19       9  1.231  42.804
15    -- -- -- -- -- -- -- -- -- -- -- -- -- --  15       8  1.146  43.508
11       -- -- -- -- -- -- -- -- -- -- -- --   11         7  1.019  44.200
 7          -- -- -- -- -- -- -- -- --          7         6  0.882  44.750
 3                -- -- -- -- --                3         5  0.754  45.072*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58           4  0.648  43.679
                                                          3  0.573  40.046
Control Rod Density: %         0.00                       2  0.467  31.173
                                                   Bottom 1  0.143   9.321
k-effective:                0.99794
Void Fraction:                0.308             % AXIAL TILT   9.264  -8.552
Core Delta-P: psia           22.574         AVG BOT 8ft/12ft  1.0308  1.0934
Core Plate Delta-P: psia     18.153
Coolant Temp: Deg-F           542.1
In Channel Flow: Mlb/hr       89.45   Active Channel Flow: Mlb/hr        85.69
Total Bypass Flow (%):         10.5   (of total core flow)
Total Water Rod Flow (%):       3.8   (of total core flow)
Source Convergence          0.00044
```

Top Ten Thermal Limits Summary - Sorted by Margin

Power Value	FT	IR	JR	MCPR Value	Margin	FT	IR	JR	APLHGR Value	Margin	Exp.	FT	IR	JR	K	LHGR Value	Margin	Exp.	FT	IR	JR	K
1.565	27	35	14	1.741	0.844	27	47	26	8.74	0.814	24.9	27	35	14	15	7.30	0.820	56.7	23	47	32	13
1.564	27	47	26	1.741	0.844	27	25	48	8.72	0.813	24.9	27	47	26	15	7.30	0.820	56.7	23	31	48	13
1.556	28	25	12	1.746	0.842	28	49	26	8.62	0.809	25.7	27	35	16	15	7.28	0.811	56.0	23	47	34	14
1.556	28	49	36	1.747	0.842	28	25	50	8.61	0.809	25.8	27	45	26	15	7.28	0.811	55.9	23	33	48	14
1.552	26	33	12	1.756	0.837	26	49	34	8.65	0.808	25.0	26	33	16	15	7.26	0.809	55.9	23	31	16	14
1.552	26	49	34	1.756	0.837	26	33	50	8.64	0.807	25.1	26	45	28	15	7.22	0.805	56.0	23	15	32	14
1.537	27	35	16	1.771	0.830	27	25	16	8.78	0.804	23.0	26	33	12	15	7.05	0.798	57.3	23	47	24	14
1.537	27	45	26	1.776	0.828	27	15	26	8.72	0.804	23.7	28	35	12	15	7.05	0.797	57.3	23	23	48	14
1.536	26	31	12	1.780	0.826	26	33	16	8.76	0.803	23.0	26	49	34	15	7.02	0.792	57.1	23	23	16	14
1.536	26	49	32	1.782	0.825	26	49	32	8.70	0.803	23.8	28	49	36	15	7.00	0.790	57.1	23	15	24	14

* LHGR calculated with pin-power reconstruction
* CPR calculated with pin-power reconstruction & CPR limit type 3

Figure A.47 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 15,484.8 MWd/MTU

```
Cycle:                            24                    Core Average Exposure: MWd/MTU    36664.4
Exposure: MWd/MTU (GWd)     15962.5 (2008.20 )
Delta E: MWd/MTU, (GWd)       477.7 (  60.10 )                Axial Profile    Edit   Radial Power
Power: MWt                   2735.9 ( 92.52 %)          N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia          1015.0                    Top 24  0.242   6.262    19  0.285 0.517  55 40
Inlet Subcooling: Btu/lbm    -22.15                        23  0.442  14.187    20  0.228 0.238  57 42
Flow: Mlb/hr                  99.96 (102.00 %)             22  0.916  28.284    21  0.337 0.674  53 42
                                                           21  1.089  33.934    22  0.619 0.782  47 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58            20  1.205  37.359    23  1.247 1.320  33 48
59                -- -- -- -- -- --              59        19  1.275  39.335    24  1.199 1.283  43 44
55          -- -- -- -- -- -- -- -- -- --        55        18  1.321  40.105    25  1.029 1.305  31 10
51       -- -- -- -- -- -- -- -- -- -- -- --     51        17  1.386  40.573    26  1.460 1.570  33 12
47    -- -- -- -- -- -- -- -- -- -- -- -- -- --  47        16  1.419  41.055    27  1.483 1.572  35 14
43    -- -- -- -- -- -- -- -- -- -- -- -- -- --  43        15  1.420* 41.780    28  1.340 1.573  25 12
39 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 39        14  1.391  42.265
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35        13  1.415  41.256
31 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 31        12  1.381  41.671
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27        11  1.336  42.426
23 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 23        10  1.277  43.060
19    -- -- -- -- -- -- -- -- -- -- -- -- -- --  19         9  1.203  43.368
15    -- -- -- -- -- -- -- -- -- -- -- -- -- --  15         8  1.108  44.021
11       -- -- -- -- -- -- -- -- -- -- -- --     11         7  0.970  44.653
 7          -- -- -- -- -- -- -- -- -- --         7         6  0.828  45.139
 3                -- -- -- -- -- --               3         5  0.698  45.402*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58             4  0.594  43.962
                                                            3  0.524  40.296
Control Rod Density: %          0.00                        2  0.428  31.376
                                                     Bottom 1  0.132   9.384
k-effective:                 0.99731
Void Fraction:                 0.289             % AXIAL TILT  12.666  -8.243
Core Delta-P: psia            22.189         AVG BOT 8ft/12ft  1.0078  1.0921
Core Plate Delta-P: psia      17.775
Coolant Temp: Deg-F            542.2
In Channel Flow: Mlb/hr        89.70  Active Channel Flow: Mlb/hr       86.04
Total Bypass Flow (%):          10.3  (of total core flow)
Total Water Rod Flow (%):        3.7  (of total core flow)
Source Convergence           0.00039
```

```
                        Top Ten Thermal Limits Summary - Sorted by Margin

     Power                 MCPR                        APLHGR                             LHGR
Value FT  IR JR     Value Margin FT  IR JR     Value Margin Exp. FT  IR JR  K     Value Margin Exp. FT  IR JR  K
1.573  28 25 12     1.862  0.789  28 49 26     8.19  0.771 26.0  27 35 14 15     6.76  0.768 57.7  23 31 48 13
1.573  28 49 26     1.862  0.789  28 25 50     8.17  0.770 26.0  27 47 26 15     6.89  0.768 56.0  23 33 48 15
1.572  27 35 14     1.864  0.789  27 25 48     8.28  0.766 24.1  26 33 12 15     6.88  0.768 56.0  23 47 34 15
1.571  27 47 26     1.864  0.789  27 47 26     8.22  0.766 24.8  28 35 12 15     6.81  0.768 56.9  23 47 32 14
1.570  26 33 12     1.868  0.787  26 33 50     8.26  0.765 24.1  26 49 34 15     6.64  0.757 58.1  23 31 16 14
1.569  26 49 34     1.869  0.787  26 49 34     8.20  0.764 24.8  28 49 26 15     6.60  0.754 58.1  23 15 32 14
1.552  26 31 12     1.898  0.775  26 31 50     8.03  0.762 26.8  27 35 16 15     6.55  0.750 58.3  23 47 24 14
1.551  26 49 32     1.898  0.774  26 49 32     8.02  0.761 26.8  27 45 26 15     6.56  0.749 58.2  23 23 48 14
1.536  27 35 16     1.899  0.774  27 25 16     8.24  0.761 23.8  26 31 12 15     6.51  0.742 58.0  23 23 16 14
1.535  27 45 26     1.903  0.772  27 15 26     8.06  0.759 26.1  26 33 16 15     6.66  0.741 55.8  24 25 18 15

  * LHGR calculated with pin-power reconstruction
  * CPR calculated with pin-power reconstruction & CPR limit type 3
```

Figure A.48 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 15,962.5 MWd/MTU

```
Cycle:                            24                        Core Average Exposure: MWd/MTU    37064.4
Exposure: MWd/MTU (GWd)      16362.5 (2058.50 )
Delta E: MWd/MTU, (GWd)        400.0 (  50.32 )                  Axial Profile    Edit   Radial Power
Power: MWt                    2513.4 ( 85.00 %)            N(PRA) Power Exposure  Zone  Avg.  Max.  IR JR
Core Pressure: psia           1015.0                      Top 24  0.253   6.350    19  0.284 0.517  55 40
Inlet Subcooling: Btu/lbm     -20.61                          23  0.466  14.383    20  0.227 0.237  57 42
Flow: Mlb/hr                   99.96 (102.00 %)               22  0.969  28.695    21  0.336 0.678  55 34
                                                              21  1.149  34.422    22  0.621 0.781  47 48
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58               20  1.263  37.898    23  1.243 1.324  33 14
59                 -- -- -- -- -- --                 59       19  1.323  39.904    24  1.185 1.285  43 18
55           -- -- -- -- -- -- -- -- -- --           55       18  1.358  40.692    25  1.036 1.322  31 10
51        -- -- -- -- -- -- -- -- -- -- -- --        51       17  1.414  41.164    26  1.454 1.587  33 12
47     -- -- -- -- -- -- -- -- -- -- -- -- -- --     47       16  1.437* 41.643    27  1.489 1.580  35 14
43     -- -- -- -- -- -- -- -- -- -- -- -- -- --     43       15  1.429  42.367    28  1.356 1.590  35 12
39 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 39       14  1.395  42.839
35 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 35       13  1.417  41.811
31 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 31       12  1.380  42.213
27 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 27       11  1.329  42.948
23 -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- -- 23       10  1.262  43.558
19     -- -- -- -- -- -- -- -- -- -- -- -- -- --     19        9  1.179  43.830
15     -- -- -- -- -- -- -- -- -- -- -- -- -- --     15        8  1.073  44.436
11        -- -- -- -- -- -- -- -- -- -- -- --        11        7  0.927  45.015
 7           -- -- -- -- -- -- -- -- -- --            7        6  0.780  45.445
 3                 -- -- -- -- -- --                  3        5  0.650  45.659*
    2  6 10 14 18 22 26 30 34 38 42 46 50 54 58                4  0.548  44.180
                                                               3  0.483  40.488
Control Rod Density: %          0.00                           2  0.394  31.533
                                                       Bottom  1  0.122   9.432
k-effective:                 0.99733
Void Fraction:                 0.270                % AXIAL TILT  15.614  -7.957
Core Delta-P: psia            21.825            AVG BOT 8ft/12ft  0.9877  1.0908
Core Plate Delta-P: psia      17.418
Coolant Temp: Deg-F            542.3
In Channel Flow: Mlb/hr        89.93   Active Channel Flow: Mlb/hr        86.39
Total Bypass Flow (%):          10.0   (of total core flow)
Total Water Rod Flow (%):        3.6   (of total core flow)
Source Convergence           0.00049


                         Top Ten Thermal Limits Summary - Sorted by Margin

     Power                  MCPR                        APLHGR                              LHGR
Value FT   IR JR      Value Margin FT   IR JR     Value Margin Exp. FT   IR JR  K     Value Margin Exp. FT   IR JR  K
1.590  28 35 12      1.996  0.737  27 47 26      7.61  0.723 26.9  27 35 14 15      6.41  0.721 56.8  23 33 48 15
1.589  28 49 26      1.996  0.736  28 49 26      7.74  0.722 25.0  26 33 12 15      6.41  0.721 56.9  23 47 34 15
1.587  26 33 12      1.997  0.736  27 25 48      7.68  0.722 25.7  28 35 12 15      6.36  0.716 56.8  23 31 48 14
1.585  26 49 34      1.998  0.736  28 25 50      7.60  0.721 26.9  27 47 26 15      6.36  0.716 56.9  23 47 32 14
1.580  27 35 14      2.007  0.732  26 33 50      7.72  0.720 25.1  26 49 34 15      6.11  0.703 58.8  23 31 16 14
1.579  27 47 26      2.008  0.732  26 49 34      7.66  0.720 25.7  28 49 26 15      6.08  0.700 58.9  23 15 32 14
1.568  26 31 12      2.039  0.721  27 25 16      7.70  0.717 24.8  26 31 12 15      6.05  0.698 59.1  23 47 24 14
1.566  26 49 32      2.040  0.720  26 31 50      7.67  0.715 24.8  26 49 32 15      6.06  0.698 59.0  23 23 48 14
1.548  27 23 12      2.042  0.720  26 49 32      7.44  0.711 27.7  27 35 16 15      3.21  0.697 71.9  22 33  6 11
1.548  27 49 24      2.045  0.719  27 15 26      7.42  0.710 27.7  27 45 26 15      6.12  0.691 57.3  23 39 18 17

  * LHGR calculated with pin-power reconstruction
  * CPR calculated with pin-power reconstruction & CPR limit type 3
```

Figure A.49 Quad Cities Unit 2 Representative Cycle 24 Control Rod Pattern and Axial Distributions at 16,362.5 MWd/MTU

Appendix B Elevation Views of the Quad Cities Unit 2 Representative Cycle 24 Fresh Reload Batch Fuel Assemblies

[

]

Figure B.1 Elevation View for the Quad Cities Unit 2 Representative Cycle 24 Fresh Fuel ATRIUM 10XM [ ] Fuel Assembly Design

[

]

Figure B.2 Elevation View for the Quad Cities Unit 2 Representative Cycle 24 Fresh Fuel ATRIUM 10XM [ ] Fuel Assembly Design

[

]

Figure B.3 Elevation View for the Quad Cities Unit 2 Representative Cycle 24 Fresh Fuel ATRIUM 10XM [ ] Fuel Assembly Design

Appendix C Quad Cities Unit 2 Representative Cycle 24 Fresh Fuel Locations

Table C.1 Quad Cities Unit 2 Representative Cycle 24 Reload Fuel Identification and Locations (Core Coordinates)

Assembly Type: ATRIUM 10XM QCI2-24
Bundle Description: [ ]
Number Loaded: 96

Fuel ID	Core Coord.	Fuel ID	Core Coord.	Fuel ID	Core Coord.	Fuel ID	Core Coord.
XRA001	27-32	XRA025	19-32	XRA049	15-34	XRA073	11-32
XRA002	29-34	XRA026	29-42	XRA050	27-46	XRA074	29-50
XRA003	31-34	XRA027	31-42	XRA051	33-46	XRA075	31-50
XRA004	33-32	XRA028	41-32	XRA052	45-34	XRA076	49-32
XRA005	33-30	XRA029	41-30	XRA053	45-28	XRA077	49-30
XRA006	31-28	XRA030	31-20	XRA054	33-16	XRA078	31-12
XRA007	29-28	XRA031	29-20	XRA055	27-16	XRA079	29-12
XRA008	27-30	XRA032	19-30	XRA056	15-28	XRA080	11-30
XRA009	25-34	XRA033	19-36	XRA057	19-40	XRA081	11-34
XRA010	27-36	XRA034	25-42	XRA058	21-42	XRA082	27-50
XRA011	33-36	XRA035	35-42	XRA059	39-42	XRA083	33-50
XRA012	35-34	XRA036	41-36	XRA060	41-40	XRA084	49-34
XRA013	35-28	XRA037	41-26	XRA061	41-22	XRA085	49-28
XRA014	33-26	XRA038	35-20	XRA062	39-20	XRA086	33-12
XRA015	27-26	XRA039	25-20	XRA063	21-20	XRA087	27-12
XRA016	25-28	XRA040	19-26	XRA064	19-22	XRA088	11-28
XRA017	21-34	XRA041	17-34	XRA065	17-38	XRA089	15-42
XRA018	27-40	XRA042	27-44	XRA066	23-44	XRA090	19-46
XRA019	33-40	XRA043	33-44	XRA067	37-44	XRA091	41-46
XRA020	39-34	XRA044	43-34	XRA068	43-38	XRA092	45-42
XRA021	39-28	XRA045	43-28	XRA069	43-24	XRA093	45-20
XRA022	33-22	XRA046	33-18	XRA070	37-18	XRA094	41-16
XRA023	27-22	XRA047	27-18	XRA071	23-18	XRA095	19-16
XRA024	21-28	XRA048	17-28	XRA072	17-24	XRA096	15-20

Table C.2 Quad Cities Unit 2 Representative Cycle 24 Reload Fuel Identification and Locations (Core Coordinates)

Assembly Type: ATRIUM 10XM QCI2-24
Bundle Description: [ ]
Number Loaded: 64

Fuel ID	Core Coord.	Fuel ID	Core Coord.	Fuel ID	Core Coord.	Fuel ID	Core Coord.
XRA097	23-36	XRA113	13-36	XRA129	15-44	XRA145	11-40
XRA098	25-38	XRA114	25-48	XRA130	17-46	XRA146	21-50
XRA099	35-38	XRA115	35-48	XRA131	43-46	XRA147	39-50
XRA100	37-36	XRA116	47-36	XRA132	45-44	XRA148	49-40
XRA101	37-26	XRA117	47-26	XRA133	45-18	XRA149	49-22
XRA102	35-24	XRA118	35-14	XRA134	43-16	XRA150	39-12
XRA103	25-24	XRA119	25-14	XRA135	17-16	XRA151	21-12
XRA104	23-26	XRA120	13-26	XRA136	15-18	XRA152	11-22
XRA105	15-36	XRA121	11-38	XRA137	13-42	XRA153	11-42
XRA106	25-46	XRA122	23-50	XRA138	19-48	XRA154	19-50
XRA107	35-46	XRA123	37-50	XRA139	41-48	XRA155	41-50
XRA108	45-36	XRA124	49-38	XRA140	47-42	XRA156	49-42
XRA109	45-26	XRA125	49-24	XRA141	47-20	XRA157	49-20
XRA110	35-16	XRA126	37-12	XRA142	41-14	XRA158	41-12
XRA111	25-16	XRA127	23-12	XRA143	19-14	XRA159	19-12
XRA112	15-26	XRA128	11-24	XRA144	13-20	XRA160	11-20

Table C.3 Quad Cities Unit 2 Representative Cycle 24 Reload Fuel Identification and Locations (Core Coordinates)

Assembly Type: ATRIUM 10XM QCI2-24
Bundle Description: [ ]
Number Loaded: 88

Fuel ID	Core Coord.	Fuel ID	Core Coord.	Fuel ID	Core Coord.	Fuel ID	Core Coord.
XRA161	17-42	XRA183	27-10	XRA205	51-24	XRA227	45-48
XRA162	19-44	XRA184	9-28	XRA206	37-10	XRA228	47-46
XRA163	41-44	XRA185	7-32	XRA207	23-10	XRA229	47-16
XRA164	43-42	XRA186	29-54	XRA208	9-24	XRA230	45-14
XRA165	43-20	XRA187	31-54	XRA209	7-36	XRA231	15-14
XRA166	41-18	XRA188	53-32	XRA210	25-54	XRA232	13-16
XRA167	19-18	XRA189	53-30	XRA211	35-54	XRA233	11-44
XRA168	17-20	XRA190	31- 8	XRA212	53-36	XRA234	17-50
XRA169	11-36	XRA191	29- 8	XRA213	53-26	XRA235	43-50
XRA170	25-50	XRA192	7-30	XRA214	35- 8	XRA236	49-44
XRA171	35-50	XRA193	7-34	XRA215	25- 8	XRA237	49-18
XRA172	49-36	XRA194	27-54	XRA216	7-26	XRA238	43-12
XRA173	49-26	XRA195	33-54	XRA217	7-38	XRA239	17-12
XRA174	35-12	XRA196	53-34	XRA218	23-54	XRA240	11-18
XRA175	25-12	XRA197	53-28	XRA219	37-54	XRA241	9-42
XRA176	11-26	XRA198	33- 8	XRA220	53-38	XRA242	19-52
XRA177	9-34	XRA199	27- 8	XRA221	53-24	XRA243	41-52
XRA178	27-52	XRA200	7-28	XRA222	37- 8	XRA244	51-42
XRA179	33-52	XRA201	9-38	XRA223	23- 8	XRA245	51-20
XRA180	51-34	XRA202	23-52	XRA224	7-24	XRA246	41-10
XRA181	51-28	XRA203	37-52	XRA225	13-46	XRA247	19-10
XRA182	33-10	XRA204	51-38	XRA226	15-48	XRA248	9-20

Appendix D Quad Cities Unit 2 Representative Cycle 24 Radial Exposure and Power Distributions

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29
60											44.462	41.789	43.535	43.430	43.693
58										41.583	43.173	42.753	42.791	41.728	43.825
56							44.934	44.200	44.099	42.128	41.847	41.770	40.811	40.673	40.099
54						44.579	43.418	42.578	19.457	39.207	18.618	0.000	0.000	0.000	0.000
52					44.586	43.642	41.600	36.066	19.469	0.000	20.339	0.000	22.175	0.000	19.949
50				44.511	43.637	43.066	36.590	17.483	0.000	0.000	0.000	0.000	0.000	0.000	0.000
48			44.926	43.054	41.661	36.513	35.708	0.000	20.006	0.000	22.434	23.610	0.000	22.941	22.898
46			44.124	42.508	36.175	17.506	0.000	20.936	0.000	0.000	22.975	23.238	0.000	0.000	22.850
44			44.063	19.476	19.466	0.000	20.028	0.000	21.579	0.000	22.573	0.000	21.313	0.000	22.036
42		34.318	42.020	39.125	0.000	0.000	0.000	0.000	0.000	21.579	0.000	22.537	0.000	19.976	0.000
40	44.396	43.066	41.846	18.602	20.333	0.000	22.463	23.026	22.631	0.000	23.309	23.618	21.942	0.000	22.902
38	40.606	42.671	41.472	0.000	0.000	0.000	23.615	23.232	0.000	22.569	23.638	23.353	0.000	20.788	23.506
36	43.724	42.718	39.793	0.000	22.192	0.000	0.000	0.000	21.339	0.000	21.954	0.000	20.980	0.000	22.911
34	43.347	42.763	40.717	0.000	0.000	0.000	22.976	0.000	0.000	19.982	0.000	20.781	0.000	22.054	0.000
32	43.679	43.815	40.040	0.000	19.973	0.000	22.925	22.855	22.047	0.000	22.935	23.506	22.954	0.000	22.088
30	43.672	43.777	40.029	0.000	19.955	0.000	22.897	22.867	22.056	0.000	22.938	23.486	22.942	0.000	22.092
28	43.325	42.749	40.757	0.000	0.000	0.000	22.957	0.000	0.000	19.958	0.000	20.759	0.000	22.045	0.000
26	43.747	42.676	40.607	0.000	22.195	0.000	0.000	0.000	21.316	0.000	21.924	0.000	20.989	0.000	22.922
24	41.654	42.695	41.490	0.000	0.000	0.000	23.607	23.221	0.000	22.548	23.626	23.335	0.000	20.811	23.523
22	44.411	43.110	41.869	18.627	20.317	0.000	22.456	23.023	22.615	0.000	23.337	23.635	21.971	0.000	22.928
20		34.411	42.063	39.123	0.000	0.000	0.000	0.000	0.000	21.606	0.000	22.563	0.000	19.999	0.000
18			44.062	19.458	19.485	0.000	20.022	0.000	21.548	0.000	22.591	0.000	21.341	0.000	22.057
16			44.158	42.525	36.134	17.494	0.000	20.923	0.000	0.000	23.003	23.246	0.000	0.000	22.869
14			44.916	43.042	41.763	36.524	35.759	0.000	20.031	0.000	22.457	23.618	0.000	22.970	22.929
12				44.536	43.666	43.048	36.620	17.487	0.000	0.000	0.000	0.000	0.000	0.000	0.000
10					44.563	43.702	41.723	36.108	19.447	0.000	20.354	0.000	22.188	0.000	19.973
8						44.567	43.375	42.660	19.481	39.249	18.601	0.000	0.000	0.000	0.000
6							44.928	44.150	44.079	42.129	41.964	41.598	39.819	40.823	40.158
4										41.523	43.148	42.799	42.778	41.726	43.795
2											44.462	40.704	43.493	43.433	43.707

Figure D.1 Quad Cities Unit 2 Representative Cycle 24 BOC Exposure Distribution (0.0 GWd/MTU)

	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
60	43.698	43.295	43.502	41.791	44.453										
58	43.746	41.757	42.775	42.769	43.124	41.550									
56	40.158	40.689	40.832	41.616	41.941	42.159	44.039	44.124	44.886						
54	0.000	0.000	0.000	0.000	18.597	39.244	19.463	42.691	43.375	44.588					
52	19.955	0.000	22.164	0.000	20.333	0.000	19.443	36.114	41.720	43.715	44.434				
50	0.000	0.000	0.000	0.000	0.000	0.000	0.000	17.470	36.637	43.011	43.679	44.553			
48	22.904	22.944	0.000	23.591	22.436	0.000	20.009	0.000	35.748	36.547	41.500	43.004	44.893		
46	22.812	0.000	0.000	23.228	22.970	0.000	0.000	20.913	0.000	17.485	36.146	42.561	44.133		
44	22.003	0.000	21.320	0.000	22.556	0.000	21.546	0.000	20.015	0.000	19.468	19.457	44.027		
42	0.000	19.980	0.000	22.541	0.000	21.585	0.000	0.000	0.000	0.000	0.000	39.165	42.085	34.432	
40	22.885	0.000	21.947	23.598	23.308	0.000	22.594	22.999	22.449	0.000	20.308	18.608	41.864	43.119	44.414
38	23.503	20.789	0.000	23.319	23.593	22.544	0.000	23.208	23.580	0.000	0.000	0.000	39.837	42.706	41.669
36	22.883	0.000	20.968	0.000	21.921	0.000	21.313	0.000	0.000	0.000	22.162	0.000	40.765	42.682	43.725
34	0.000	22.034	0.000	20.754	0.000	19.954	0.000	0.000	22.948	0.000	0.000	0.000	40.617	42.739	43.329
32	22.075	0.000	22.911	23.470	22.903	0.000	22.002	22.811	22.890	0.000	19.954	0.000	40.052	43.742	43.668
30	22.102	0.000	22.936	23.508	22.914	0.000	22.016	22.830	22.927	0.000	19.969	0.000	40.056	43.786	43.670
28	0.000	22.070	0.000	20.800	0.000	19.992	0.000	0.000	22.968	0.000	0.000	0.000	40.729	42.795	43.348
26	22.925	0.000	20.995	0.000	21.965	0.000	21.332	0.000	0.000	0.000	22.167	0.000	40.627	42.727	43.762
24	23.526	20.811	0.000	23.338	23.624	22.557	0.000	23.241	23.609	0.000	0.000	0.000	41.635	42.691	41.677
22	22.914	0.000	21.967	23.625	23.338	0.000	22.612	23.003	22.455	0.000	20.352	18.620	41.872	43.116	44.426
20	0.000	19.997	0.000	22.555	0.000	21.604	0.000	0.000	0.000	0.000	0.000	39.170	42.039	34.355	
18	22.027	0.000	21.329	0.000	22.602	0.000	21.593	0.000	20.022	0.000	19.486	19.479	44.056		
16	22.835	0.000	0.000	23.253	23.002	0.000	0.000	20.958	0.000	17.529	36.167	42.545	44.142		
14	22.926	22.967	0.000	23.615	22.455	0.000	20.030	0.000	35.751	36.548	41.510	43.029	44.914		
12	0.000	0.000	0.000	0.000	0.000	0.000	0.000	17.494	36.647	43.060	43.639	44.561			
10	19.971	0.000	22.178	0.000	20.356	0.000	19.482	36.104	41.780	43.735	44.590				
8	0.000	0.000	0.000	0.000	18.622	39.263	19.474	42.686	43.386	44.632					
6	40.169	40.844	39.817	41.720	41.963	42.156	44.072	44.192	44.945						
4	43.801	41.761	42.769	42.800	43.188	41.582									
2	43.706	43.443	43.486	40.719	44.505										

Figure D.1 Quad Cities Unit 2 Representative Cycle 24 BOC Exposure Distribution (0.0 GWd/MTU) *(Continued)*

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29
60											47.093	44.895	46.845	46.860	47.133
58										45.855	48.386	48.703	49.165	48.370	50.371
56							47.846	48.466	50.009	49.884	51.058	51.805	51.349	51.398	50.951
54						47.874	48.711	50.301	31.816	51.042	34.515	17.028	17.553	17.758	17.852
52					48.037	49.062	49.882	47.599	35.884	18.647	39.482	20.634	41.847	21.133	40.264
50				47.814	49.060	50.914	47.989	34.374	19.774	20.907	21.577	22.058	22.592	22.184	21.839
48			47.841	48.315	49.943	47.939	49.734	19.562	40.957	22.351	42.485	43.716	23.112	44.336	42.323
46			48.411	50.253	47.715	34.419	19.586	41.266	23.016	23.004	43.451	43.737	23.469	23.011	42.525
44			50.025	31.874	35.915	19.803	41.000	23.034	43.896	23.973	44.562	23.302	43.695	23.466	43.858
42		38.984	49.866	51.019	18.697	20.945	22.380	23.028	23.989	43.948	23.172	44.361	23.502	42.468	23.082
40	47.100	48.385	51.126	34.564	39.528	21.621	42.543	43.518	44.627	23.179	42.987	43.289	43.791	23.119	42.727
38	43.777	48.701	51.607	17.113	20.702	22.108	43.756	43.757	23.314	44.393	43.303	43.033	23.309	42.793	43.207
36	47.070	49.173	50.548	17.676	41.932	22.646	23.149	23.493	43.730	23.505	43.803	23.313	43.249	23.322	44.526
34	46.814	49.400	51.612	17.870	21.211	22.237	44.399	23.037	23.481	42.483	23.127	42.789	23.318	44.029	22.878
32	47.157	50.429	50.975	17.942	40.343	21.885	42.379	42.555	43.881	23.092	42.765	43.214	44.550	22.856	41.561
30	47.142	50.376	50.943	17.920	40.317	21.878	42.352	42.566	43.886	23.092	42.767	43.192	44.539	22.855	41.567
28	46.771	49.340	51.585	17.801	21.173	22.216	44.372	23.031	23.478	42.460	23.126	42.768	23.312	44.020	22.876
26	47.057	49.076	51.117	17.585	41.889	22.620	23.136	23.488	43.706	23.504	43.777	23.313	43.250	23.319	44.535
24	44.784	48.673	51.561	17.041	20.657	22.084	43.737	43.741	23.314	44.375	43.292	43.012	23.306	42.810	43.228
22	47.094	48.391	51.108	34.547	39.487	21.601	42.526	43.509	44.608	23.175	43.008	43.293	43.805	23.115	42.753
20		39.051	49.881	50.995	18.673	20.928	22.368	23.020	23.982	43.963	23.164	44.373	23.487	42.477	23.075
18			50.014	31.846	35.917	19.790	40.986	23.027	43.862	23.966	44.573	23.295	43.706	23.454	43.863
16			48.434	50.260	47.672	34.399	19.578	41.248	23.012	23.000	43.472	43.742	23.464	23.003	42.529
14			47.828	48.299	50.032	47.944	49.778	19.557	40.978	22.350	42.506	43.727	23.115	44.358	42.337
12				47.831	49.084	50.903	48.025	34.379	19.775	20.911	21.584	22.070	22.607	22.197	21.850
10					48.017	49.125	50.012	47.649	35.871	18.655	39.507	20.658	41.887	21.170	40.314
8						47.867	48.676	50.388	31.846	51.090	34.512	17.062	17.628	17.829	17.911
6							47.842	48.419	49.991	49.886	51.170	51.672	50.535	51.673	51.053
4										45.804	48.372	48.780	49.213	48.429	50.384
2											47.103	43.843	46.840	46.894	47.171

Figure D.2 Quad Cities Unit 2 Representative Cycle 24 EOC Exposure Distribution (16.4 GWd/MTU)

	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
60	47.139	46.730	46.811	44.893	47.079										
58	50.297	48.403	49.145	48.713	48.327	45.822									
56	51.005	51.409	51.365	51.641	51.123	49.905	49.947	48.390	47.799						
54	17.851	17.756	17.544	17.012	34.482	51.071	31.819	50.410	48.675	47.891					
52	40.269	21.130	41.835	20.629	39.474	18.647	35.864	47.653	50.012	49.143	47.899				
50	21.839	22.181	22.590	22.056	21.577	20.909	19.777	34.369	48.050	50.876	49.110	47.858			
48	42.325	44.334	23.106	43.700	42.486	22.352	40.965	19.567	49.776	47.983	49.817	48.279	47.815		
46	42.493	23.006	23.460	43.723	43.445	23.004	23.018	41.251	19.595	34.412	47.700	50.310	48.422		
44	43.829	23.460	43.691	23.292	44.543	23.969	43.868	23.040	40.999	19.815	35.926	31.864	49.995		
42	23.082	42.463	23.486	44.350	23.158	43.944	23.988	23.036	22.392	20.958	18.708	51.066	49.936	39.098	
40	42.713	23.114	43.780	43.245	42.966	23.171	44.596	43.504	42.541	21.636	39.521	34.593	51.176	48.451	47.120
38	43.203	42.790	23.294	42.978	43.245	44.366	23.317	43.740	43.735	22.117	20.708	17.131	50.115	48.749	44.825
36	44.501	23.317	43.228	23.299	43.766	23.501	43.706	23.495	23.151	22.646	41.898	17.640	51.388	49.119	47.055
34	22.879	44.008	23.308	42.759	23.122	42.454	23.481	23.036	44.372	22.229	21.190	17.828	51.405	49.347	46.782
32	41.549	22.854	44.508	43.166	42.727	23.092	43.843	42.520	42.348	21.880	40.313	17.909	50.961	50.341	47.135
30	41.571	22.850	44.527	43.194	42.731	23.087	43.852	42.531	42.376	21.872	40.318	17.901	50.954	50.378	47.132
28	22.875	44.036	23.302	42.787	23.111	42.480	23.472	23.025	44.376	22.208	21.161	17.786	51.544	49.373	46.786
26	44.535	23.317	43.245	23.290	43.794	23.494	43.715	23.480	23.129	22.610	41.856	17.568	51.121	49.114	47.067
24	43.232	42.804	23.289	42.997	43.276	44.369	23.304	43.752	43.733	22.073	20.642	17.023	51.677	48.661	44.802
22	42.740	23.110	43.789	43.270	42.994	23.158	44.597	43.487	42.520	21.589	39.501	34.524	51.102	48.390	47.105
20	23.074	42.474	23.480	44.355	23.148	43.951	23.972	23.011	22.359	20.916	18.657	51.025	49.852	38.995	
18	43.839	23.454	43.698	23.291	44.578	23.960	43.893	23.014	40.974	19.779	35.907	31.852	50.004		
16	42.498	23.002	23.466	43.759	43.479	22.998	23.000	41.268	19.567	34.425	47.695	50.274	48.419		
14	42.335	44.357	23.120	43.735	42.515	22.348	40.967	19.540	49.756	47.959	49.806	48.289	47.826		
12	21.851	22.201	22.612	22.076	21.587	20.906	19.761	34.359	48.020	50.898	49.056	47.856			
10	40.311	21.175	41.886	20.664	39.509	18.651	35.888	47.616	50.022	49.135	48.034				
8	17.912	17.833	17.637	17.078	34.539	51.098	31.824	50.391	48.661	47.921					
6	51.063	51.691	50.538	51.802	51.185	49.913	49.980	48.450	47.847						
4	50.394	48.459	49.207	48.787	48.419	45.858									
2	47.169	46.901	46.833	43.860	47.146										

Figure D.2 Quad Cities Unit 2 Representative Cycle 24 EOC Exposure Distribution (16.4 GWd/MTU) *(Continued)*

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29
60											0.197	0.224	0.235	0.240	0.239
58										0.319	0.379	0.420	0.436	0.448	0.437
56							0.230	0.323	0.435	0.557	0.636	0.665	0.673	0.667	0.676
54						0.262	0.409	0.565	0.866	0.797	1.045	1.012	1.004	0.997	1.002
52					0.275	0.419	0.610	0.805	1.091	1.131	1.209	1.185	1.162	1.152	1.190
50				0.262	0.419	0.575	0.789	1.113	1.183	1.201	1.242	1.247	1.247	1.151	1.123
48			0.230	0.406	0.608	0.790	0.864	1.095	1.304	1.285	1.342	1.335	1.281	1.247	1.045
46			0.324	0.566	0.805	1.115	1.096	1.175	1.294	1.300	1.392	1.380	1.322	1.236	1.101
44			0.439	0.869	1.094	1.184	1.305	1.295	1.364	1.393	1.395	1.335	1.384	1.307	1.371
42		0.348	0.564	0.801	1.134	1.203	1.286	1.301	1.394	1.373	1.280	1.346	1.334	1.439	1.378
40	0.203	0.387	0.641	1.050	1.212	1.244	1.343	1.392	1.396	1.280	1.014	1.036	1.383	1.406	1.501
38	0.229	0.427	0.667	1.017	1.189	1.249	1.337	1.381	1.335	1.345	1.036	1.045	1.378	1.504	1.548
36	0.238	0.443	0.685	1.011	1.167	1.250	1.282	1.323	1.384	1.334	1.383	1.378	1.458	1.428	1.516
34	0.243	0.450	0.686	1.004	1.157	1.153	1.248	1.237	1.307	1.439	1.406	1.504	1.427	1.449	1.347
32	0.242	0.442	0.683	1.008	1.194	1.125	1.046	1.102	1.371	1.378	1.500	1.548	1.514	1.346	1.099
30	0.241	0.441	0.681	1.006	1.193	1.124	1.046	1.102	1.371	1.378	1.500	1.547	1.514	1.346	1.099
28	0.241	0.445	0.680	0.999	1.154	1.152	1.247	1.237	1.307	1.439	1.406	1.504	1.427	1.448	1.347
26	0.235	0.438	0.665	1.005	1.163	1.248	1.281	1.322	1.384	1.334	1.382	1.377	1.457	1.427	1.515
24	0.227	0.422	0.662	1.012	1.186	1.247	1.335	1.380	1.334	1.345	1.035	1.044	1.377	1.503	1.546
22	0.201	0.384	0.638	1.047	1.210	1.242	1.342	1.392	1.395	1.279	1.013	1.035	1.381	1.405	1.499
20		0.346	0.561	0.800	1.132	1.201	1.285	1.300	1.393	1.372	1.279	1.344	1.333	1.438	1.376
18			0.438	0.868	1.092	1.183	1.304	1.295	1.363	1.393	1.394	1.334	1.383	1.306	1.369
16			0.323	0.565	0.805	1.114	1.095	1.174	1.294	1.300	1.391	1.379	1.322	1.236	1.100
14			0.230	0.405	0.608	0.790	0.862	1.094	1.304	1.285	1.342	1.335	1.281	1.246	1.044
12				0.262	0.419	0.576	0.790	1.114	1.183	1.201	1.242	1.247	1.248	1.152	1.124
10					0.276	0.419	0.616	0.806	1.092	1.132	1.209	1.187	1.165	1.155	1.193
8						0.262	0.410	0.565	0.867	0.797	1.046	1.014	1.009	1.002	1.007
6							0.230	0.323	0.435	0.557	0.635	0.663	0.684	0.682	0.681
4										0.319	0.379	0.422	0.441	0.453	0.441
2											0.198	0.227	0.238	0.242	0.241

Figure D.3 Quad Cities Unit 2 Representative Cycle 24 Radial Power Distribution at 0.0 MWd/MTU

	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
60	0.239	0.240	0.235	0.224	0.196										
58	0.437	0.448	0.436	0.419	0.377	0.318									
56	0.676	0.666	0.672	0.659	0.633	0.556	0.434	0.323	0.230						
54	1.002	0.996	1.003	1.010	1.043	0.796	0.866	0.564	0.410	0.262					
52	1.190	1.152	1.162	1.184	1.208	1.130	1.091	0.805	0.615	0.419	0.276				
50	1.122	1.150	1.246	1.246	1.241	1.200	1.182	1.114	0.790	0.576	0.420	0.262			
48	1.044	1.246	1.280	1.334	1.341	1.284	1.304	1.095	0.863	0.791	0.612	0.407	0.230		
46	1.101	1.235	1.321	1.378	1.391	1.299	1.294	1.175	1.096	1.115	0.806	0.566	0.324		
44	1.371	1.306	1.382	1.333	1.393	1.392	1.363	1.295	1.305	1.184	1.094	0.870	0.439		
42	1.377	1.438	1.332	1.342	1.276	1.370	1.392	1.300	1.286	1.203	1.134	0.801	0.564	0.348	
40	1.501	1.405	1.380	1.032	1.009	1.277	1.394	1.392	1.343	1.244	1.213	1.051	0.643	0.388	0.203
38	1.547	1.503	1.375	1.041	1.032	1.343	1.333	1.379	1.336	1.249	1.189	1.018	0.679	0.427	0.228
36	1.516	1.427	1.456	1.375	1.379	1.332	1.383	1.322	1.281	1.248	1.166	1.008	0.678	0.441	0.237
34	1.347	1.448	1.426	1.503	1.404	1.438	1.306	1.236	1.247	1.152	1.154	1.000	0.671	0.447	0.242
32	1.099	1.345	1.514	1.546	1.499	1.376	1.370	1.101	1.045	1.124	1.192	1.005	0.680	0.441	0.241
30	1.098	1.345	1.513	1.546	1.499	1.376	1.370	1.101	1.045	1.123	1.192	1.005	0.680	0.440	0.241
28	1.346	1.447	1.425	1.501	1.404	1.437	1.305	1.235	1.246	1.151	1.153	0.998	0.679	0.444	0.240
26	1.515	1.426	1.455	1.374	1.378	1.331	1.382	1.321	1.280	1.246	1.162	1.003	0.663	0.437	0.235
24	1.546	1.501	1.374	1.039	1.031	1.342	1.332	1.378	1.334	1.246	1.184	1.011	0.660	0.421	0.226
22	1.499	1.404	1.378	1.030	1.008	1.276	1.392	1.390	1.340	1.241	1.208	1.045	0.636	0.383	0.201
20	1.376	1.436	1.331	1.341	1.275	1.369	1.391	1.299	1.284	1.200	1.131	0.798	0.561	0.346	
18	1.369	1.305	1.381	1.332	1.392	1.390	1.361	1.293	1.303	1.182	1.091	0.867	0.438		
16	1.100	1.235	1.321	1.378	1.389	1.298	1.292	1.172	1.094	1.113	0.804	0.565	0.323		
14	1.044	1.246	1.280	1.334	1.340	1.283	1.302	1.092	0.860	0.789	0.610	0.405	0.230		
12	1.124	1.152	1.248	1.247	1.241	1.199	1.181	1.110	0.786	0.573	0.418	0.261			
10	1.193	1.155	1.165	1.187	1.209	1.130	1.090	0.802	0.605	0.417	0.274				
8	1.007	1.002	1.009	1.015	1.046	0.796	0.865	0.563	0.407	0.261					
6	0.680	0.682	0.684	0.669	0.637	0.557	0.434	0.322	0.229						
4	0.441	0.452	0.441	0.423	0.380	0.319									
2	0.241	0.242	0.238	0.227	0.198										

Figure D.3 Quad Cities Unit 2 Representative Cycle 24 Radial Power Distribution at 0.0 MWd/MTU *(Continued)*

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29
60											0.135	0.165	0.179	0.189	0.191
58										0.218	0.275	0.326	0.358	0.378	0.376
56							0.134	0.208	0.301	0.409	0.511	0.591	0.644	0.661	0.675
54						0.148	0.246	0.388	0.667	0.669	0.933	1.156	1.222	1.248	1.251
52					0.153	0.248	0.404	0.611	0.932	1.219	1.164	1.408	1.238	1.474	1.286
50				0.148	0.248	0.378	0.592	0.935	1.270	1.401	1.468	1.516	1.555	1.551	1.535
48			0.134	0.245	0.403	0.593	0.781	1.238	1.222	1.475	1.274	1.289	1.564	1.316	1.304
46			0.209	0.388	0.610	0.935	1.238	1.175	1.462	1.489	1.252	1.267	1.536	1.527	1.268
44			0.303	0.668	0.932	1.270	1.222	1.462	1.279	1.489	1.241	1.467	1.282	1.497	1.245
42		0.237	0.413	0.669	1.220	1.401	1.474	1.489	1.489	1.258	1.441	1.215	1.447	1.261	1.452
40	0.138	0.279	0.512	0.934	1.164	1.467	1.272	1.251	1.240	1.440	1.206	1.189	1.196	1.415	1.202
38	0.168	0.329	0.594	1.159	1.409	1.515	1.287	1.265	1.465	1.213	1.187	1.187	1.396	1.190	1.162
36	0.180	0.362	0.656	1.226	1.239	1.554	1.562	1.534	1.280	1.445	1.194	1.396	1.201	1.391	1.161
34	0.190	0.382	0.672	1.252	1.474	1.550	1.314	1.525	1.494	1.259	1.413	1.189	1.391	1.199	1.415
32	0.193	0.378	0.677	1.254	1.286	1.533	1.302	1.266	1.243	1.450	1.200	1.161	1.159	1.413	1.250
30	0.192	0.378	0.676	1.253	1.286	1.533	1.302	1.266	1.242	1.450	1.200	1.161	1.159	1.413	1.250
28	0.189	0.380	0.669	1.249	1.473	1.550	1.314	1.525	1.495	1.259	1.413	1.189	1.390	1.199	1.415
26	0.179	0.359	0.638	1.223	1.237	1.554	1.562	1.535	1.281	1.446	1.195	1.396	1.201	1.391	1.161
24	0.166	0.327	0.591	1.157	1.408	1.515	1.288	1.266	1.466	1.214	1.188	1.187	1.396	1.189	1.161
22	0.137	0.278	0.511	0.933	1.164	1.468	1.273	1.251	1.240	1.441	1.206	1.188	1.195	1.415	1.201
20		0.236	0.412	0.669	1.220	1.401	1.475	1.489	1.490	1.258	1.441	1.214	1.447	1.260	1.452
18			0.303	0.668	0.932	1.270	1.223	1.463	1.281	1.490	1.241	1.467	1.282	1.497	1.245
16			0.208	0.389	0.611	0.936	1.239	1.176	1.463	1.490	1.252	1.268	1.537	1.527	1.269
14			0.134	0.246	0.403	0.593	0.780	1.238	1.222	1.476	1.274	1.289	1.565	1.316	1.305
12				0.148	0.248	0.378	0.593	0.935	1.271	1.402	1.469	1.517	1.556	1.552	1.536
10					0.153	0.249	0.406	0.611	0.933	1.220	1.165	1.410	1.240	1.476	1.287
8						0.148	0.247	0.389	0.668	0.668	0.934	1.159	1.226	1.252	1.254
6							0.134	0.208	0.301	0.410	0.510	0.593	0.656	0.671	0.677
4										0.219	0.276	0.328	0.362	0.382	0.378
2											0.135	0.167	0.181	0.190	0.192

Figure D.4 Quad Cities Unit 2 Representative Cycle 24 Radial Power Distribution at 15,484.8 MWd/MTU (EOFP)

	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
60	0.191	0.189	0.179	0.165	0.135										
58	0.376	0.379	0.359	0.326	0.275	0.218									
56	0.674	0.661	0.645	0.590	0.509	0.409	0.301	0.208	0.134						
54	1.251	1.249	1.222	1.157	0.933	0.668	0.667	0.388	0.246	0.148					
52	1.286	1.474	1.239	1.409	1.165	1.220	0.933	0.611	0.405	0.249	0.153				
50	1.535	1.551	1.556	1.517	1.469	1.402	1.271	0.936	0.593	0.379	0.249	0.148			
48	1.305	1.316	1.565	1.290	1.275	1.476	1.223	1.239	0.781	0.593	0.406	0.246	0.134		
46	1.269	1.527	1.537	1.268	1.253	1.490	1.463	1.176	1.239	0.937	0.611	0.389	0.209		
44	1.245	1.498	1.283	1.468	1.242	1.491	1.281	1.463	1.223	1.271	0.933	0.669	0.303		
42	1.453	1.262	1.448	1.216	1.443	1.259	1.491	1.490	1.476	1.403	1.221	0.670	0.413	0.238	
40	1.203	1.416	1.197	1.190	1.209	1.443	1.242	1.253	1.274	1.469	1.166	0.936	0.514	0.280	0.138
38	1.163	1.191	1.398	1.189	1.190	1.216	1.467	1.268	1.289	1.517	1.411	1.161	0.607	0.330	0.167
36	1.162	1.393	1.203	1.398	1.197	1.448	1.283	1.536	1.564	1.556	1.240	1.226	0.648	0.362	0.180
34	1.416	1.200	1.392	1.191	1.416	1.262	1.497	1.527	1.316	1.552	1.475	1.252	0.663	0.382	0.190
32	1.251	1.415	1.161	1.163	1.203	1.453	1.245	1.269	1.304	1.536	1.287	1.254	0.677	0.378	0.192
30	1.250	1.414	1.161	1.163	1.202	1.453	1.245	1.268	1.304	1.535	1.287	1.253	0.677	0.378	0.192
28	1.415	1.199	1.392	1.190	1.416	1.261	1.497	1.527	1.316	1.551	1.475	1.250	0.669	0.380	0.189
26	1.161	1.392	1.202	1.398	1.196	1.448	1.282	1.537	1.564	1.556	1.239	1.223	0.638	0.360	0.179
24	1.162	1.190	1.398	1.188	1.189	1.216	1.468	1.268	1.289	1.517	1.409	1.157	0.590	0.327	0.166
22	1.201	1.415	1.196	1.189	1.207	1.443	1.242	1.253	1.274	1.469	1.165	0.934	0.510	0.278	0.137
20	1.452	1.261	1.448	1.215	1.443	1.259	1.491	1.491	1.476	1.402	1.220	0.669	0.412	0.237	
18	1.245	1.498	1.283	1.468	1.242	1.491	1.281	1.464	1.224	1.272	0.933	0.668	0.303		
16	1.269	1.528	1.537	1.268	1.253	1.491	1.463	1.176	1.240	0.937	0.611	0.389	0.209		
14	1.305	1.316	1.565	1.289	1.274	1.476	1.223	1.239	0.781	0.594	0.405	0.246	0.134		
12	1.536	1.552	1.556	1.517	1.469	1.402	1.271	0.935	0.592	0.378	0.249	0.148			
10	1.287	1.476	1.240	1.410	1.165	1.220	0.933	0.611	0.403	0.248	0.153				
8	1.254	1.252	1.226	1.159	0.935	0.669	0.667	0.388	0.246	0.148					
6	0.677	0.671	0.656	0.595	0.511	0.410	0.301	0.208	0.134						
4	0.378	0.381	0.362	0.328	0.276	0.218									
2	0.192	0.190	0.181	0.167	0.135										

Figure D.4 Quad Cities Unit 2 Representative Cycle 24 Radial Power Distribution at 15,484.8 MWd/MTU (EOFP) *(Continued)*

ATTACHMENT 22

Reload Safety Analysis Report
(Non-Proprietary)

ANP-3361NP
Revision 0

Quad Cities Unit 2 Cycle 24 Representative Cycle Design Reload Safety Analysis

December 2014

AREVA Inc.

ANP-3361NP
Revision 0

Quad Cities Unit 2 Cycle 24 Representative Cycle Design Reload Safety Analysis

AREVA Inc.

ANP-3361NP
Revision 0

Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial issue

Contents

Tables

Figures

Nomenclature

AOO	anticipated operational occurrence
ARO	all control rods out
ASME	American Society of Mechanical Engineers
AST	alternative source term
ATWS	anticipated transient without scram
ATWS-RPT	anticipated transient without scram recirculation pump trip
BOC	beginning-of-cycle
BPWS	banked position withdrawal sequence
BSP	backup stability protection
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CFR	Code of Federal Regulations
COLR	core operating limits report
CPR	critical power ratio
CRDA	control rod drop accident
CRWE	control rod withdrawal error
ECCS	emergency core cooling system
EFPD	effective full-power days
EFPH	effective full-power hours
EOCLB	end-of-cycle licensing basis
EOFP	end of full power
EOOS	equipment out-of-service
FHA	fuel handling accident
FHOOS	feedwater heaters out-of-service
FWCF	feedwater controller failure
FWTR	feedwater temperature reduction
HCOM	hot channel oscillation magnitude
IHPCIS	inadvertent high pressure coolant injection startup
ISS	intermediate scram speed
LFWH	loss of feedwater heating
LHGR	linear heat generation rate
$LHGRFAC_f$	flow-dependent linear heat generation rate multipliers
$LHGRFAC_p$	power-dependent linear heat generation rate multipliers
LOCA	loss-of-coolant accident
LOSC	loss of stator cooling
LPRM	local power range monitor
LRNB	generator load rejection with no bypass

Nomenclature *(Continued)*

MAPLHGR	maximum average planar linear heat generation rate
MCFL	maximum combined flow limit
MCPR	minimum critical power ratio
$MCPR_f$	flow-dependent minimum critical power ratio
$MCPR_p$	power-dependent minimum critical power ratio
MSIV	main steam isolation valve
MSIVOOS	main steam isolation valve out-of-service
NEOC	near end-of-cycle
NSS	nominal scram speed
NRC	Nuclear Regulatory Commission, U.S.
OLMCPR	operating limit minimum critical power ratio
OPRM	oscillation power range monitor
P_{bypass}	power below which direct scram on TSV/TCV closure is bypassed
PCOOS	pressure controller out-of-service
PCT	peak cladding temperature
PLU	power load unbalance
PLUOOS	power load unbalance out-of-service
PRFO	pressure regulator failure open
RBM	rod block monitor
RHR	residual heat removal
RPT	recirculation pump trip
SLC	standby liquid control
SLMCPR	safety limit minimum critical power ratio
SLO	single-loop operation
SRV	safety/relief valve
SRVOOS	safety/relief valve out-of-service
TBVOOS	turbine bypass valves out-of-service
TCV	turbine control valve
TIP	traversing incore probe
TLO	two-loop operation
TSSS	technical specifications scram speed
TSV	turbine stop valve
TTNB	turbine trip with no bypass
UFSAR	updated final safety analysis report
ΔCPR	change in critical power ratio

1.0 Introduction

Reload licensing analyses results generated by AREVA Inc. (AREVA) are presented in support of the Quad Cities Unit 2 Cycle 24 representative cycle design. The analyses reported in this document were performed using methodologies previously approved for generic application to boiling water reactors with some exceptions which are explicitly described in Reference 2. The NRC technical limitations associated with the application of the approved methodologies have been satisfied by these analyses.

The Cycle 24 representative core design consists of a total of 724 fuel assemblies, including 248 fresh ATRIUM™ 10XM* assemblies and 476 irradiated OPTIMA2 assemblies. The licensing analysis supports the core design presented in Reference 1.

The reload licensing analyses for the Cycle 24 representative core design were performed for the potentially limiting events and analyses that were identified in the disposition of events. A summary of the disposition of events is presented in Section 2. The results of the analyses are used to establish the Technical Specifications/COLR limits and ensure that the design and licensing criteria are met. The design and safety analyses are based on the design and operational assumptions and plant parameters provided by the utility. The results of the reload licensing analysis support operation for the power/flow map presented in Figure 1.1 and also support operation with the equipment out-of-service (EOOS) scenarios presented in Table 1.1.

---

* ATRIUM is a trademark of AREVA Inc.

Table 1.1 EOOS
Operating Conditions

Single-loop operation (SLO)*
Turbine bypass valves out-of-service (TBVOOS)
Feedwater heaters out-of-service (FHOOS)
One safety relief valve out-of-service (SRVOOS)
One main steam isolation valve out-of-service (MSIVOOS)†
Power load unbalance out-of-service (PLUOOS)
1 stuck closed TCV or TSV‡
TCV slow closure
Pressure controller out-of-service (PCOOS)
Up to 40% of the TIP channels out-of-service
Up to 50% of the LPRMs out-of-service

* SLO is only supported at power levels less than 50% of rated and core flows less than 51% of rated.
† Operation with One MSIVOOS is only supported at power levels less than 75% of rated.
‡ Operation with 1 stuck closed TCV or TSV is only supported at power levels less than 75% of rated.

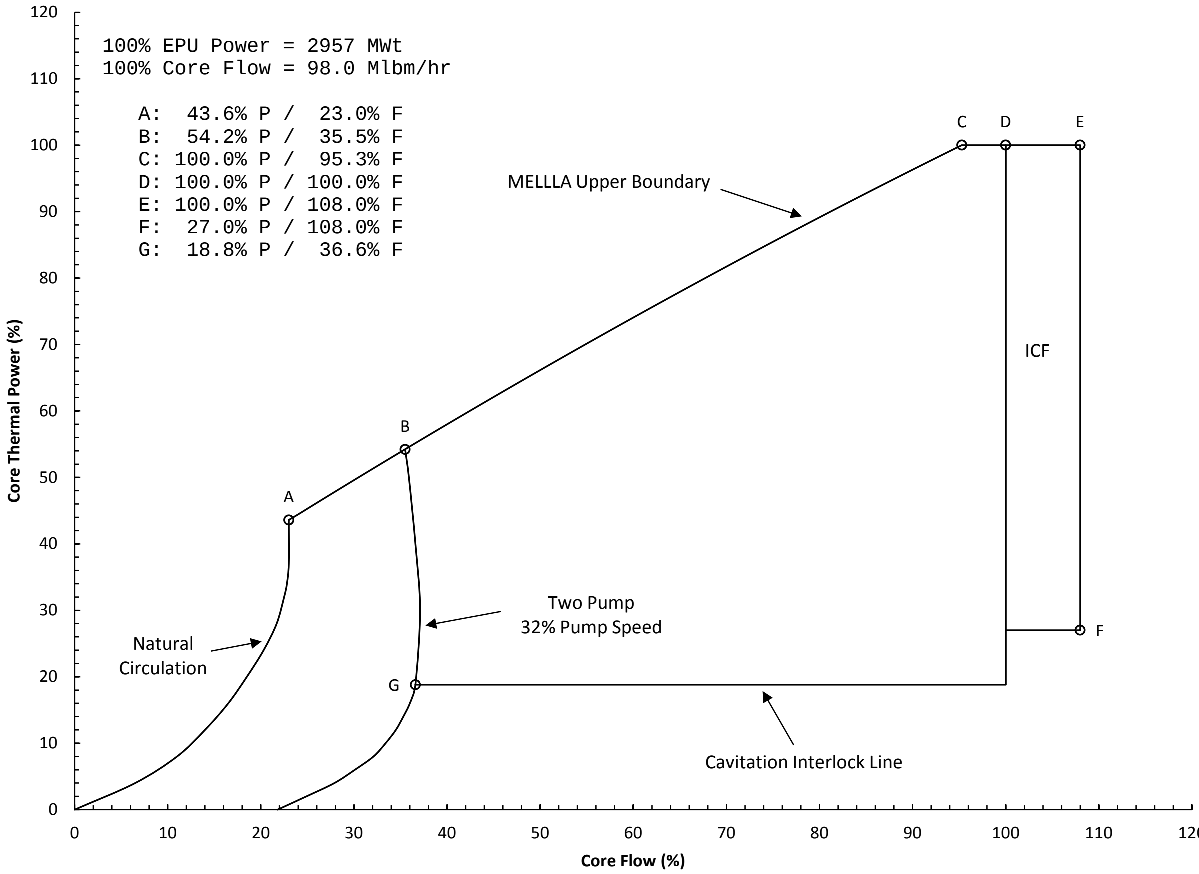


Figure 1.1 Quad Cities Unit 2
Power/Flow Map

2.0 Disposition of Events for ATRIUM 10XM Fuel Introduction

The objective of the disposition of events is to identify the limiting events which must be analyzed to support operation at the Quad Cities Nuclear Power Station with the introduction of ATRIUM 10XM fuel. Events and analyses identified as potentially limiting are either evaluated generically for the introduction of ATRIUM 10XM fuel or on a cycle-specific basis.

The first step in the disposition of events is to identify the licensing basis of the plant. Included in the licensing basis are descriptions of the postulated events/analyses and the associated criteria. Fuel-related system design criteria which must be met to ensure regulatory compliance and safe operation are also included. The Quad Cities licensing basis is contained in the Updated Final Safety Analysis Report (UFSAR), the Technical Specifications, Core Operating Limits Reports (COLR), and other reload analysis reports.

AREVA reviewed all the fuel-related design criteria, events, and analyses identified in the licensing basis. In many cases, when the operating limits are established to ensure acceptable consequences of an anticipated operational occurrence or accident, the fuel-related aspects of the system design criteria are met. All the fuel-related events were reviewed and dispositioned into one of the following categories:

- ***No further analysis required.*** This classification may result from one of the following:
  - The consequences of the event are bound by consequences of a different event.
  - The consequences of the event are benign, i.e., the event causes no significant change in margins to the operating limits.
  - The event or safety design basis is not affected by the introduction of a new fuel design and/or the current analysis of record remains applicable.
- ***Address event each reload.*** The consequences of the event are potentially limiting and need to be addressed each reload.
- ***Address for initial reload.*** This classification may result from one of the following:
  - The analysis is performed using conservative bounding assumptions and inputs such that the initial reload results will remain applicable for future reloads of the same fuel design.
  - Results from the first reload will be used to quantitatively demonstrate that the results remain applicable for future reloads of the same fuel design because the consequences are benign or bound by those of another event.

A summary of the disposition of events results is presented in Tables 2.1, 2.2 and 2.3. Table 2.1 presents a list of the events and analyses, the corresponding UFSAR section, the disposition status, and any applicable comments. Table 2.2 presents a summary of the disposition of fuel-related design criteria evaluations. Table 2.3 presents a summary of the disposition of events for the EOOS scenarios.

Table 2.1 Disposition of Events Summary for ATRIUM 10XM Fuel Introduction at Quad Cities

UFSAR Section	Event	Disposition Status	Comments
15.1.1	Loss of Feedwater Heater (LFWH)	Address each reload	Potentially limiting AOO.
15.1.2	Increase in Feedwater Flow (FWCF) - Maximum Demand	Address each reload	Potentially limiting AOO.
15.1.3	Increase in Steam Flow (PRFO)	No further analysis required	Consequences bound by the TTNB.
15.2.1	Steam Pressure Regulator Malfunction	No further analysis required	Benign event.
15.2.2	Load Rejection with bypass and with TBVs fail to open (LRNB)	Address each reload	Potentially limiting AOO with bypass inoperable. At 50% power and below, the effects of the PLU device need to be addressed.
15.2.3	Turbine Trip with bypass and with TBVs fail to open (TTNB)	Address each reload	Potentially limiting AOO with bypass inoperable.
15.2.4	Inadvertent Closure of Main Steam Isolation Valves (MSIV) - 4 valve closure - single valve closure	No further analyses required	Consequences bound by the LRNB and TTNB for both single and 4 valve closures.
15.2.5	Loss of Condenser Vacuum	No further analyses required	Consequences bound by the TTNB.
15.2.6	Loss of AC Power	No further analyses required	Consequences bound by LRNB/TTNB.
15.2.7	Loss of Normal Feedwater Flow	No further analysis required	Benign event.
15.3.1	Single and Multiple Recirculation Pump Trips	No further analyses required	Benign event.
15.3.2	Recirculation Flow Controller Malfunctions	No further analyses required	Benign event and bound by pump trips.
15.3.3	Recirculation Pump Shaft Seizure	No further analysis required	Non-limiting event in two loop operation. See Table 2.3 for SLO disposition.
15.3.4	Recirculation Pump Shaft Break	No further analysis required	Bound by consequences of the pump seizure event.
15.3.5	Jet Pump Malfunction	No further analysis required	Bound by other flow reduction events.
15.3.6	Transients During Single Loop Operation	Discussed in Table 2.3	
15.4.1	Control Rod Withdrawal Error During Refueling	No further analyses required	Benign event.

UFSAR Section	Event	Disposition Status	Comments
15.4.2	Rod Withdrawal Error - at Power	Address each reload	Potentially limiting AOO.
15.4.3	Control Rod Maloperation	No further analyses required	Not analyzed for Quad Cities.
15.4.4	Startup of Idle Recirculation Loop at Incorrect Temperature	No further analyses required	Non-limiting event.
15.4.5	Recirculation Loop Flow Controller Failure with Increasing Flow	No further analyses required	Non-limiting event. Flow-dependent limits are bounding.
15.4.6	Chemical and Volume Control System Malfunction	No further analyses required	Event is not applicable to Quad Cities.
15.4.7	Mislocated Fuel Assembly Accident	Address each reload	Infrequent event criteria per AREVA methodology.
15.4.8	Misoriented Fuel Assembly Accident	Address each reload	Infrequent event criteria per AREVA methodology.
15.4.9	Control Rod Ejection Event	No further analyses required	Event is not applicable at Quad Cities.
15.4.10	Control Rod Drop Accident	Address each reload	Potentially limiting event.
15.4.11	Thermal Hydraulic Instability Transient	Address each reload	Potentially limiting event.
15.5.1	Inadvertent Initiation of HPCI During Power Operation	Address for initial reload	At low power, level control may not keep water level from reaching high level trip.
15.6.1	Inadvertent Opening of a Safety Valve, Relief Valve, or Safety Relief Valve	No further analysis required	Benign event.
15.6.2	Break in Reactor Coolant Pressure Boundary Instrument Line Outside Containment	No further analysis required	Benign event relative to acceptance criteria.
15.6.3	Steam Generator Tube Failure	No further analysis required	Event is not applicable to Quad Cities.
15.6.4	Steam System Line Break Outside Containment	No additional analysis required	Fuel-related consequences bound by limiting LOCA evaluation. Current radiological release evaluation remains applicable.
15.6.5	Loss of Coolant Accident Resulting from Piping Breaks Inside Containment	Address for initial reload	Fuel response and ECCS performance addressed in UFSAR 6.3.3. Applicability of the AST evaluation to ATRIUM 10XM fuel is needed. Exelon will perform the assessment based on source term inputs provided by AREVA.
15.7.1	Postulated Liquid Releases Due to Liquid Tank Failures	No further analysis required	No impact on current evaluation.

UFSAR Section	Event	Disposition Status	Comments
15.7.2	Design Basis Fuel Handling Accidents Inside Containment and Spent Fuel Storage Buildings	Address for initial reload	Potentially limiting accident. Dose consequences need to meet AST evaluation. Applicability of the AST evaluation to ATRIUM 10XM fuel is needed. Exelon will perform the assessment based on source term inputs provided by AREVA.
15.7.3	Spent Fuel Cask Drop Accident	No further analysis required	No longer considered a credible design basis accident.
15.7.4	Radioactive Gas Waste System Leak or Failure	No further analysis required	Current evaluation remains applicable as it is not affected by fuel design.
15.8	Anticipated Transient Without Scram	Address each reload	Peak pressure evaluation needs to be addressed each reload. Pressure margin to lifting the SLC system relief valve needs to be evaluated. Evaluations needed to show current long term ATWS evaluations for suppression pool temperature and containment remain applicable. PCT and oxidation are bound by LOCA. The current ATWS with core instability analysis remains applicable for the introduction of ATRIUM 10XM fuel.
	Backup Stability Protection (BSP)	Address each reload	Required to establish exclusion regions to support operation when the OPRM system is inoperable.
	Safety Limit MCPR	Address each reload	Analyses performed to establish the SLMCPR for the cycle.
	Slow Flow Run-up	Address each reload	Analysis results form the basis of the flow-dependent operating limits.
	Loss of Stator Cooling (LOSC)	Address at least for the initial reload	Potentially limiting AOO at certain power levels.
	Station Blackout	No further analyses required	Bound by other pressurization events.

Table 2.2 Disposition of Licensing and Safety Design Criteria Evaluations for Quad Cities

UFSAR Section	Design Criteria	Disposition Status	Comments
3.9	Mechanical Systems and Components	Address for initial reload	ATRIUM 10XM seismic evaluation is needed.
4.2	Fuel System Design	Address for initial reload	The analysis scope is described in References 5 and 6. Analyses for initial reload are applicable for follow-on reloads of the same fuel design.
4.3	Nuclear Design	Address each reload	The cycle-specific analysis scope for nuclear design is generally described in References 5, 19 and 25.
4.4	Thermal and Hydraulic Design	Address for initial reload	The thermal hydraulic compatibility analysis is addressed for the initial reload. Other thermal hydraulic design criteria discussed in Reference 5 are addressed by establishing operating limits to ensure AOO, accident and stability criteria are met.
4.6	Functional Design of Reactivity Control Systems	Address each reload	Scram speed dependent operating limits are established on a cycle-specific basis. The CRDA event is addressed in UFSAR Section 15.4.10.
5.2.2	Overpressurization Protection	Address each reload	ASME overpressurization analysis will be performed each reload. Potentially limiting analyses include MSIV, TCV, and TSV closures; including the FWCF with TBVOOS.
5.3	Reactor Vessels	No further analyses required	Current evaluation remains applicable.
5.4.1	Reactor Recirculation System	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
5.4.6	Reactor Core Isolation Cooling System	No further analyses required	Reactor core isolation cooling system safety design basis is independent of fuel design.
5.4.7	Residual Heat Removal (RHR) System	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
5.4.9	Main Steam Line and Feedwater Piping	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
5.4.13	Safety/Relief Valves	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
6.2.1	Primary Containment Functional Design	No additional analysis required	Current evaluation remains applicable.
6.2.2	Containment Heat Removal	No further analyses required	The containment cooling subsystem of the RHR system is not impacted by fuel design.

UFSAR Section	Design Criteria	Disposition Status	Comments
6.2.3	Secondary Containment System	No further analyses required	Current evaluation remains applicable.
6.2.5	Combustible Gas Control in Containment	No further analysis required	Current evaluation remains applicable.
6.3	Emergency Core Cooling Systems (ECCS)	Address each reload	Limiting break characteristics are identified for the initial ATRIUM 10XM reload. Heatup analysis for reload fuel is expected for follow-on reloads. Also see the disposition for LOCA addressed in UFSAR 15.6.5.
6.4	Habitability System	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
7.2	Reactor Protection System	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
7.3.2	Primary Containment Isolation	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
7.6.1.	Nuclear Instrumentation	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
7.6.2	Reactor vessel instrumentation	No further analyses required	The safety design basis for the reactor vessel instrumentation is independent of the fuel design.
7.7.1	Reactor Control Rod Control Systems	No additional analysis required	The reactor control rod system is independent of fuel design.
7.7.2	Rod Worth Minimizer System	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events, particularly the CRDA event.
7.7.3.1	Recirculation Flow Control System	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
7.7.4	Pressure Regulator and Turbine Generator Control System	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
7.7.5	Feedwater Level Control System	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.
7.8	ATWS Mitigation System	No additional analysis required	Fuel-related design basis items that may need to be evaluated are addressed in other portions of the disposition of events.

UFSAR Section	Design Criteria	Disposition Status	Comments
9.1.1	New Fuel Storage	Address for initial reload	Exelon has selected a third party vendor to perform this evaluation. Any validation of the criticality requirements will be Exelon's responsibility.
9.1.2	Spent Fuel Storage	Address for initial reload	Criticality – to be addressed by a third party. Thermal Hydraulic evaluation – no further analysis required. Mechanical – seismic evaluation is needed.
9.1.3	Spent Fuel Pool Cooling and Cleanup System	No further analyses required	The safety design basis for the spent fuel storage pool cooling and cleanup system is unaffected by the introduction of ATRIUM 10XM fuel.
9.3.4	Standby Liquid Control (SLC) System	Address each reload	Analysis performed each reload to verify adequate SLC system shutdown capability.
9.5.1	Fire Protection Systems	No further analyses required	Current analysis remains applicable.
11.0	Radioactive Waste Management	No further analyses required	The radionuclide source terms are generic and are unaffected by the introduction of ATRIUM 10XM fuel.
12.3	Radiation Protection Design Features	No additional analysis required	Introduction of ATRIUM 10XM fuel has no impact on the evaluations.

Table 2.3 Disposition of Equipment Out of Service Flexibility Options on Limiting Events

Option	Affected Limiting Event/Analysis	Comments
MSIV Out of Service	Slow flow run up	The impact of the increase in steam line pressure drop on the slow flow run-up analysis will be evaluated each reload.
SRV Out of Service		This condition will be included as part of the base case analyses where applicable.
FWTR/Feedwater Heater Out of Service	FWCF LRNB TTNB Option III Stability Solution IHPCIS CRWE Backup Stability Protection (BSP)	This scenario should be examined each reload for the FWCF and stability analyses. For the LRNB and TTNB events, reduced feedwater temperature impact at high power should be examined for the initial reload. The IHPCIS event needs to be reviewed at least for the initial reload to ensure it is non-limiting. The CRWE event will be addressed for the initial reload.
Single Loop Operation	LOCA SLMCPR Recirculation Pump Seizure BSP	The impact of SLO will be addressed for the initial cycle in the break spectrum analyses. SLO SLMCPR will be evaluated each reload, including the impact on uncertainties associated with LPRMs out of service, TIP channels out of service and the LPRM calibration interval. SLO pump seizure event is a potentially limiting event relative to the AOO acceptance criteria and will be evaluated each reload. SLO impact on BSP will be addressed for the initial reload.
Turbine Bypass Valves Out of Service	FWCF LFWH CRWE LOSC Slow flow run up IHPCIS PRFO	The FWCF with TBVOOS will be evaluated each reload. The other events may be impacted if the base case analyses indicate the steam flow increases above the capacity of the turbine control valves. PRFO and IHPCIS events will be evaluated if a high level trip occurs in the base case analyses.
PLUOOS	LRNB	Condition is bound by TCV slow closure condition.

Option	Affected Limiting Event/Analysis	Comments
1 Stuck Closed TCV/TSV	FWCF Slow flow run up	FWCF analyses will be performed for the initial reload to determine the maximum power level supported for this condition. Slow flow run up will be evaluated if the steam flow in the base case analysis increases above the MCFL limit for this condition. While not expected, analyses using steady state methods are potentially impacted if, in the base case analyses, the steam flow increases above the MCFL for this condition.
TCV Slow Closure	LRNB	LRNB analyses to consider various combinations of fast and slow closing TCVs. Limiting combinations will be evaluated each reload.
Pressure Controller Out of Service	NA	Events with PCOOS are bound by the TCV slow closure events for the closure event and bound by TTNB for the opening event. No additional analysis required.
TIPs Out of Service LPRMS Out of Service Extended LPRM calibration interval	SLMCPR	Base analysis to include uncertainties associated with these conditions.
Combined EOOS Conditions		
FWTR/FHOOS and MSIVOOS	NA	No additional analyses required.
FWTR/FHOOS and TBVOOS	FWCF IHPCIS	FWCF will be addressed each reload. The IHPCIS event needs to be reviewed to ensure it is non-limiting at least for the initial reload.
FWTR/FHOOS with PLUOOS	NA	No additional analyses required.
FWTR/FHOOS with 1 Stuck Closed TCV/TSV	NA	No additional analyses required.
FWTR/FHOOS with TCV Slow Closure	LRNB	Analyses will be performed for the initial reload to determine if the decrease in steam flow due to lower feedwater temperature offsets the further closed TCV position.
FWTR/FHOOS with PCOOS	NA	No additional analyses required.
SLO with 1 MSIVOOS	NA	Results for TLO with 1 MSIVOOS are bounding or applicable. No additional analyses required.
SLO with TBVOOS	Recirculation pump seizure	If the pump seizure evaluation results in a high level trip, then an evaluation with TBVOOS will be needed.
SLO with PLUOOS	NA	Results for TLO with PLUOOS are bounding or applicable. No additional analyses required.

Option	Affected Limiting Event/Analysis	Comments
SLO with 1 Stuck Closed TCV/TSV	NA	Results for TLO with 1 stuck closed TCV/TSV are bounding or applicable. No additional analyses required.
SLO with TCV Slow Closure	NA	Results for TLO with TCV slow closure are bounding or applicable. No additional analyses required.
SLO with PCOOS	NA	Results for TLO with PCOOS are bounding or applicable. No additional analyses required.
PCOOS and 1 Stuck Closed TCV/TSV	NA	Equivalent to or bound by the PCOOS condition by itself or the 1 stuck closed TCV/TSV condition by itself. No additional analyses required.
PCOOS and PLUOOS	NA	Equivalent to or bound by the PCOOS condition by itself or the PLUOOS condition by itself. No additional analyses required.
PLUOOS and 1 Stuck Closed TCV/TSV	NA	Equivalent to or bound by the PLUOOS condition by itself or the 1 stuck closed TCV/TSV condition by itself. No additional analyses required.

3.0 Mechanical Design Analysis

The mechanical design analyses for ATRIUM 10XM fuel are presented in the applicable mechanical and thermal-mechanical design reports (References 3 and 4). The maximum exposure limits for the ATRIUM 10XM fuel are:

54.0 GWd/MTU average assembly exposure
62.0 GWd/MTU rod average exposure (full-length fuel rods)

The ATRIUM 10XM LHGR limits are presented in Section 8.0. The OPTIMA2 LHGR multipliers presented in Section 8.0 ensure that the thermal-mechanical design criteria for OPTIMA2 fuel are satisfied. The fuel cycle design analyses (Reference 1) have verified that all ATRIUM 10XM and OPTIMA2 fuel assemblies remain within licensed burnup limits.

4.0 Thermal-Hydraulic Design Analysis

4.1 *Thermal-Hydraulic Design and Compatibility*

The ATRIUM 10XM fuel is analyzed and monitored with the ACE critical power correlation (Reference 9). The OPTIMA2 fuel is analyzed and monitored with the SPCB critical power correlation (Reference 10) **[ ]** The SPCB additive constants and additive constant uncertainty for the OPTIMA2 fuel were developed using the indirect approach described in Reference 11.

The results of the thermal-hydraulic characterization and compatibility analyses are presented in the thermal-hydraulic design report (Reference 7). The analysis results demonstrate that the thermal-hydraulic design and compatibility criteria are satisfied for the Quad Cities transition cores consisting of ATRIUM 10XM and OPTIMA2 fuel.

4.2 *Safety Limit MCPR Analysis*

The safety limit MCPR (SLMCPR) is determined such that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition during normal operation or an anticipated operational occurrence (AOO). The SLMCPR for all fuel in the Quad Cities Unit 2 Cycle 24 representative core design was determined using the methodology described in Reference 8 **[ ]** The SLMCPR analysis explicitly includes the effects of channel bow and assumes that no fuel channels are used for more than one fuel bundle lifetime.

The analysis was performed with a power distribution that conservatively represents expected reactor operation through the cycle. The fuel- and plant-related uncertainties used in the SLMCPR analysis are presented in Table 4.1. The radial and nodal power uncertainties used in the analysis include the effects of up to 40% of the TIP channels out-of-service, up to 50% of the LPRMs out-of-service, and a 2500 EFPH LPRM calibration interval. The radial and nodal power uncertainties allow for startup with up to 40% of the TIP channels out-of-service.

The analysis results support a two-loop operation (TLO) SLMCPR of 1.10 and a single-loop operation (SLO) SLMCPR of 1.11. Table 4.2 presents a summary of the analysis results including the SLMCPR and the percentage of rods expected to experience boiling transition. The values currently in the Quad Cities Technical Specifications are 1.12 for TLO and 1.14 for

SLO. The limits presented in this report are based on the current TLO SLMCPR of 1.12 and SLO SLMCPR of 1.14.

4.3 *Core Hydrodynamic Stability*

Quad Cities has implemented BWROG Long Term Stability Solution Option III (Oscillation Power Range Monitor-OPRM). Reload validation has been performed in accordance with Reference 12. The stability based Operating Limit MCPR (OLMCPR) is provided for two conditions as a function of OPRM amplitude setpoint in Table 4.3. The two conditions evaluated are for a postulated oscillation at 45% core flow steady state operation (SS) and following a two recirculation pump trip (2PT) from the limiting full power operation state point. The power- and flow-dependent limits for the Unit 2 Cycle 24 representative core design provide adequate protection against violation of the SLMCPR for postulated reactor instability as long as the operating limit is greater than or equal to the specified value for the selected OPRM setpoint. The results in Table 4.3 are valid for normal and reduced feedwater temperature operation and allow for the dome pressure variation specified by Exelon. The limits in this table are based upon the current Technical Specification TLO SLMCPR value of 1.12.

AREVA has performed calculations for the relative change in CPR as a function of the calculated hot channel oscillation magnitude (HCOM). These calculations were performed with the RAMONA5-FA code in accordance with Reference 13. This code is a coupled neutronic-thermal-hydraulic three-dimensional transient model for the purpose of determining the relationship between the relative change in ΔCPR and the HCOM on a plant specific basis. The stability-based OLMCPRs are calculated using the most limiting of the calculated change in relative ΔCPR for a given oscillation magnitude or the generic value provided in Reference 12. The generic value was determined to be limiting for the Unit 2 Cycle 24 representative core design.

In cases where the OPRM system is declared inoperable, Backup Stability Protection (BSP) is provided in accordance with Reference 14. BSP curves have been evaluated using STAIF (Reference 15) to determine endpoints that meet decay ratio criteria for the BSP Base Minimal Region I (scram region) and Base Minimal Region II (controlled entry region). Stability boundaries based on these endpoints are then determined using the modified shape function provided by Exelon. Analyses have been performed to support operation with both nominal and reduced feedwater temperature conditions. The stability regions have been verified to remain applicable for single loop operation (SLO) and allow for the dome pressure variation specified by Exelon. The core has been determined to remain stable at the lowest achievable core flow rate at 25% of rated core thermal power, which demonstrates that the current OPRM trip-enabled region remains applicable. The endpoints for the BSP regions are provided in Table 4.4.

Table 4.1 Fuel- and Plant-Related Uncertainties for Safety Limit MCPR Analyses

Parameter	Uncertainty
Fuel-Related Uncertainties	
[	
	]
Plant-Related Uncertainties	
Feedwater flow rate	0.229 Mlbm/hr*
Feedwater temperature	8.50°F*
Core pressure	22.56 psig*
Total core flow rate TLO SLO	 2.5% 6%

* [ ]

Table 4.2 Results Summary for Safety Limit MCPR Analyses

SLMCPR	Percentage of Rods in Boiling Transition
TLO – 1.10	0.0513
SLO – 1.11	0.0293

Table 4.3 OPRM Setpoints

OPRM Setpoint	OLMCPR (SS)	OLMCPR (2PT)
1.05	1.24	1.27
1.06	1.27	1.30
1.07	1.30	1.33
1.08	1.32	1.35
1.09	1.35	1.38
1.10	1.38	1.42
1.11	1.42	1.45
1.12	1.45	1.48
1.13	1.48	1.52
1.14	1.52	1.55
1.15	1.56	1.59
Acceptance Criteria	Less than or equal to the Off-Rated OLMCPR at 45% Flow	Less than or equal to the Rated Power OLMCPR as described in Section 8.0

**Table 4.4 BSP Endpoints for
Quad Cities Unit 2 Cycle 24
Representative Core Design**

Feedwater Temperature Operation Mode	Region	End Point Designation	Power (% rated)	Flow (% rated)
Nominal	Scram	IA	61.05	43.70
Nominal	Scram	IB	24.90	20.80
Nominal	Controlled entry	IIA	67.40	51.55
Nominal	Controlled entry	IIB	23.90	20.60
FHOOS	Scram	IA	67.20	51.40
FHOOS	Scram	IB	26.20	21.00
FHOOS	Controlled entry	IIA	68.80	53.30
FHOOS	Controlled entry	IIB	23.90	20.60

5.0 Anticipated Operational Occurrences

This section describes the analyses performed to determine the power- and flow-dependent MCPR operating limits for base case operation for the Quad Cities Unit 2 Cycle 24 representative core design.

COTRANSA2 (Reference 16), XCOBRA-T (Reference 17), XCOBRA (Reference 18), and CASMO-4/MICROBURN-B2 (Reference 19) are the major codes used in the thermal limits analyses as described in the AREVA THERMEX methodology report (Reference 18) and neutronics methodology report (Reference 19). COTRANSA2 is a system transient simulation code, which includes an axial one-dimensional neutronics model that captures the effects of axial power shifts associated with the system transients. XCOBRA-T is a transient thermal-hydraulics code used in the analysis of thermal margins for the limiting fuel assembly. XCOBRA is used in steady-state analyses.

Fuel pellet-to-cladding gap conductance values are based on RODEX2 (Reference 20) calculations.

The ACE/ATRIUM 10XM critical power correlation (Reference 9) is used to evaluate the thermal margin for the ATRIUM 10XM fuel. The SPCB critical power correlation (Reference 10 **[** **]**) is used in the thermal margin evaluations for the OPTIMA2 fuel. The application of the SPCB correlation to OPTIMA2 fuel follows the indirect process described in Reference 11 **[** **]**.

5.1 *System Transients*

The reactor plant parameters for the system transient analyses were provided by the utility. Analyses have been performed to determine power-dependent MCPR limits and power-dependent LHGR multipliers that protect operation throughout the power/flow domain shown in Figure 1.1.

At Quad Cities, direct scram on turbine stop valve (TSV) position and turbine control valve (TCV) fast closure are bypassed at power levels less than or equal to 38.5% of rated (P_{bypass}). Scram will occur if and when the high pressure or high neutron flux scram setpoint is reached.

Reference 21 indicates that MCPR and LHGR limits only need to be monitored at power levels greater than or equal to 25% of rated, which is the lowest power analyzed for this report.

The limiting exposure for rated power pressurization transients is typically at end of full power (EOFP) when the control rods are fully withdrawn. To provide additional margin to the operating limits earlier in the cycle, analyses were also performed to establish operating limits at a near end-of-cycle (NEOC) exposure of 14,000 MWd/MTU. Analyses were performed at cycle exposures prior to NEOC to ensure that the operating limits provide the necessary protection. The end-of-cycle licensing basis (EOCLB) analysis was performed at EOFP + 25 EFPD (16,072 MWd/MTU). Table 5.1 presents the licensing basis exposures used to develop the neutronics inputs to the transient analyses for the Quad Cities Unit 2 Cycle 24 representative core design. The same methods described in this report will be used to establish operating limits to support final feedwater temperature reduction and/or power coastdown operation to extend cycle operation.

All pressurization transients assumed that the single most beneficial relief, safety or safety/relief valve is out of service. Since the Target Rock safety relief valve has the highest flow capacity and provides the most pressure relief, it was assumed out of service in the pressurization transient analyses. This basis supports operation with 1 SRV out-of-service.

The Quad Cities turbine bypass system includes 9 bypass valves. However, for base case analyses in which credit is taken for turbine bypass operation, only 8 of the turbine bypass valves are assumed operable.

Variations in feedwater temperature of +10/-30°F from the nominal feedwater temperature are considered base case operation and not an EOOS condition. The nominal feedwater temperature is provided as a function of power as

$$FWT = -98\ FRP^2 + 242.78\ FRP + 210.82$$

where

FWT = nominal feedwater temperature (°F)
FRP = fraction of rated power

The base case operating limits also support operation for a range of dome pressures. The minimum dome pressure is defined as a function of power as

$$MDP = 116.96\ FRP^2 - 9.69\ FRP + 897$$

where

MDP = minimum dome pressure (psia)

and the maximum dome pressure is 1019.7 psia for all power levels. Analyses were performed to support operation in the allowable ranges.

The thermal limits are applicable for single and three element feedwater control systems and for dome pressure and throttle pressure regulator control systems.

The 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%, 20%, 50%, and 90% 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. The Quad Cities Technical Specifications (Reference 21) allow operation with up to 12 “slow” and 1 stuck control rod. One additional control rod is assumed to fail to scram. Conservative adjustments to the NSS, ISS, and TSSS scram speeds were made to the analysis inputs to appropriately account for these effects on scram reactivity. For cases below 38.5% power, the results are relatively insensitive to scram speed, and only TSSS analyses are performed. At 38.5% power (P_{bypass}), some analyses were performed both with and without bypass of the direct scram function which can result in a step change in the operating limits.

5.1.1 Load Rejection No Bypass (LRNB)

The load rejection causes a fast closure of the turbine control valves. The resulting compression wave travels through the steam lines into the vessel and creates a rapid pressurization. The increase in pressure causes a decrease in core voids, which in turn causes a rapid increase in power. The fast closure of the turbine control valves also causes a reactor scram. Turbine bypass system operation, which also mitigates the consequences of the event, is not credited. The excursion of the core power due to the void collapse is terminated primarily by the reactor scram and revoiding of the core.

For power levels less than or equal to 50% of rated, the LRNB analyses assume that the power load unbalance (PLU) is inoperable. With the PLU inoperable, the LRNB sequence of events is different than the standard event. Instead of a fast closure, the TCVs close in servo mode and there is no direct scram on TCV closure. A turbine trip (i.e., TSV closure) signal is assumed to occur 0.625 second after the load rejection signal due to crediting the 86 device. The turbine trip signal results in closure of the TSVs and a reactor scram. Below P_{bypass}, there is no direct scram on the TSV closure. As a result, the event continues until the high pressure scram set point is reached.

LRNB analyses were performed for a range of power/flow conditions to support generation of the thermal limits. Tables 5.3 and 5.4 present the base case limiting LRNB transient analysis results used to generate the NEOC and EOCLB operating limits for TSSS, ISS and NSS insertion times. Figures 5.1 – 5.2 show the responses of various reactor and plant parameters during the LRNB event initiated at 100% of rated power and 108% of rated core flow with TSSS insertion times.

5.1.2 Turbine Trip No Bypass (TTNB)

The turbine trip causes a closure of the turbine stop valves. The resulting compression wave travels through the steam lines into the vessel and creates a rapid pressurization. The increase in pressure causes a decrease in core voids, which in turn causes a rapid increase in power. The closure of the turbine stop valves also causes a reactor scram. Turbine bypass system operation, which also mitigates the consequences of the event, is not credited. The excursion of the core power due to the void collapse is terminated primarily by the reactor scram and revoiding of the core. Below P_{bypass}, there is no direct scram on the TSV closure. As a result, the event will continue until the high pressure scram set point is reached.

Tables 5.5 and 5.6 present the base case TTNB transient analysis results for TSSS, ISS, and NSS insertion times for the Quad Cities representative cycle.

5.1.3 Feedwater Controller Failure (FWCF)

The increase in feedwater flow due to a failure of the feedwater control system to maximum demand results in an increase in the water level and a decrease in the coolant temperature at the core inlet. The increase in core inlet subcooling causes an increase in core power. As the feedwater flow continues at maximum demand, the water level continues to rise and eventually reaches the high water level trip setpoint. The initial water level is conservatively assumed to be at the low-level normal operating range to delay the high-level trip and maximize the core inlet subcooling resulting from the FWCF. The high water level trip causes the turbine stop valves to close in order to prevent damage to the turbine from excessive liquid inventory in the steam line. The feedwater pumps also trip on high level. The valve closures create a compression wave that travels to the core causing a void collapse and subsequent rapid power excursion. The closure of the turbine stop valves also initiates a reactor scram. Eight of the nine installed turbine bypass valves are assumed operable and provide pressure relief. The core power excursion is mitigated in part by the pressure relief, but the primary mechanism for termination of the event is reactor scram.

FWCF analyses were performed for a range of power/flow conditions to support generation of the thermal limits. Tables 5.7 and 5.8 present the base case limiting FWCF transient analysis results used to generate the NEOC and EOCLB operating limits for TSSS, ISS, and NSS insertion times. Figures 5.3 – 5.4 show the responses of various reactor and plant parameters during the FWCF event initiated at 100% of rated power and 108% of rated core flow with TSSS insertion times.

5.1.4 Inadvertent HPCI Startup (IHPCIS)

The inadvertent HPCI startup results in cold water being injected in the feedwater line sparger. This causes an increase in core inlet subcooling and a subsequent increase in the core power. The water injection also causes an increase in the water level. In an effort to maintain the water level, the feedwater/level control system responds to the water level increase by decreasing the feedwater flow. At most power levels, the decrease in feedwater flow is sufficient to offset the HPCI injection flow and avoid reaching the high level trip. At lower power levels, the decrease in feedwater flow may not be enough to offset the HPCI injection so the water level may increase until a high level turbine trip occurs. The turbine trip results in a pressurization event similar to the TTNB discussion above. For power levels at which the high level set point is not reached, the power will increase some due to the increase in subcooling and the core will reach a new pseudo steady state condition at a higher power.

IHPCIS analyses were performed for a range of power/flow conditions to support generation of the thermal limits. The results showed that above P_{bypass}, the feedwater/level control system decreases the feedwater flow such that a high level trip does not occur. Below P_{bypass}, avoiding the high level trip is not assured so the analyses were performed forcing the high level trip to occur. Table 5.9 presents the base case limiting IHPCIS transient analysis results.

5.1.5 Loss of Stator Cooling (LOSC)

A loss of stator cooling event results in a turbine runback or decrease in turbine steam flow. The analysis assumes the turbine steam flow decreases or runs back at a rate of 40% of rated steam flow per minute (maximum value). The runback continues until the steam flow reaches 23% of rated steam flow (minimum value). Since the runback rate is relatively slow, the turbine bypass valves open to maintain the target pressure. If the decrease in steam flow is less than the capacity of the bypass valves, there is no system pressurization and the reactor remains at the initial state but with some of the steam flow passing through the bypass valves instead of the turbine. At power levels with steam flows greater than the sum of the bypass capacity and 23% of rated steam flow, the system starts to slowly pressurize when the turbine bypass valves reach the maximum relief capacity. The pressure slowly increases until the high pressure scram set point is reached.

Table 5.10 presents the base case LOSC transient analysis results for the Quad Cities representative cycle.

5.1.6 Loss of Feedwater Heating (LFWH)

The loss of feedwater heating event analysis supports a 120°F decrease in the feedwater temperature. The temperature is assumed to decrease linearly over 58.6 seconds. The result is an increase in core inlet subcooling, which reduces voids, thereby increasing core power and shifting the axial power distribution toward the bottom of the core. As a result of the axial power shift and increased core power, voids begin to build up in the bottom region of the core, providing negative feedback to the increased subcooling effect. The negative feedback moderates the core power increase. Although there is a substantial increase in core thermal power during the event, the increase in steam flow is much less because a large part of the added power is used to overcome the increase in inlet subcooling. The increase in steam flow is accommodated by the pressure control system via the TCVs and/or the turbine bypass valves, so no pressurization occurs. The COTRANSA2/XCOBRA/XCOBRA-T code package was used to analyze this event due to the time over which the feedwater temperature decrease occurs. The LFWH results are presented in Table 5.11. The base case results also show that at power levels less than 99% of rated there is sufficient capacity in the turbine control valves to accommodate the increase in steam flow during the event and no turbine bypass valves open. The LFWH results are used to establish the $MCPR_p$ limits and $LHGRFAC_p$ multipliers.

5.1.7 Control Rod Withdrawal Error (CRWE)

The control rod withdrawal error transient is an inadvertent reactor operator initiated withdrawal of a control rod. This withdrawal increases local power and core thermal power, lowering the core MCPR. Quad Cities Unit 2 is a partial-ARTS plant so the CRWE analysis is performed assuming the reactor is above 75% power when the event occurs and the rated power conservative error rod patterns are used to determine the change in MCPR over the event. The CRWE transient for Quad Cities is analyzed without credit taken for the rod block monitor (RBM) system and is analyzed assuming no xenon. The analysis further assumes that the plant could be operating in either an A or B sequence control rod pattern. The limiting ΔCPR for the base case CRWE event is 0.32 for both nominal and reduced feedwater temperature conditions and allows for one TBVOOS. Analyses have been performed for TBVOOS and include the effects of pressurization when steam flow exceeds the TCV capacity. The limiting ΔCPR results for this EOOS case is 0.35. Operation with 1 Stuck Closed TCV/TSV is limited to 75% of rated core thermal power which eliminates the possibility of system pressurization during the CRWE event which results in this EOOS condition being bound by the base case results.

Analyses demonstrate that the 1% strain and centerline melt criteria are met for both ATRIUM 10XM without any LHGR set down for the base case and all EOOS conditions. The results show that the OPTIMA2 fuel also meets the thermal-mechanical design criteria (i.e., strain and centerline melt) with the base case LHGR multipliers presented in Section 8.2.

5.2 *Slow Flow Runup Analysis*

Flow-dependent MCPR and LHGR limits are established to support operation at off-rated core flow conditions. The limits are based on the CPR and heat flux changes experienced by the fuel during slow flow excursions. The slow flow excursion event assumes a failure of the recirculation flow control system such that the core flow increases slowly to the maximum flow physically permitted by the equipment (110% of rated core flow). An uncontrolled increase in flow creates the potential for a significant increase in core power and heat flux. A conservatively steep flow runup path was used in the analyses. Slow flow runup analyses were performed to support operation in all the EOOS scenarios – including One MSIVOOS, TBVOOS, and 1 stuck closed TCV/TSV. Operation with One MSIVOOS causes a larger increase in pressure and power during the flow excursion which results in a steeper flow runup path. For TBVOOS and 1 stuck closed TCV/TSV, the steam flow increases beyond the capacity of the available TCVs and turbine bypass valves resulting in a pressurization of the system. This system pressurization is accounted for in the analyses.

$MCPR_f$ limits are determined for all fuel types in the core. XCOBRA is used to calculate the change in critical power ratio during a two-loop flow runup to the maximum flow rate. The $MCPR_f$ limit is set such that the increase in core power, resulting from the maximum increase in core flow, assures that the TLO safety limit MCPR is not violated. Calculations were performed for a range of initial flow rates to determine the corresponding MCPR values that put the limiting assembly on the safety limit MCPR at the high flow condition at the end of the flow excursion.

$MCPR_f$ limits that provide the required protection are presented in Table 8.16. The $MCPR_f$ limits are applicable for all exposures in the Unit 2 Cycle 24 representative core design.

Flow runup analyses were performed with CASMO-4/MICROBURN-B2 to determine flow-dependent LHGR multipliers ($LHGRFAC_f$) for ATRIUM 10XM and OPTIMA2 fuel. The analysis assumes that the recirculation flow increases slowly along the limiting rod line to the maximum flow physically permitted by the equipment. A series of flow excursion analyses were performed at several exposures throughout the cycle starting from different initial power/flow conditions. Xenon is assumed to remain constant during the event. The $LHGRFAC_f$ multipliers are established to provide protection against fuel centerline melt and overstraining of the cladding during a flow runup. Analyses were performed to support operation in all the EOOS scenarios – including One MSIVOOS, TBVOOS, and 1 stuck closed TSV/TCV. As described above, these 4 EOOS conditions can impact the results of the analysis due to the increase in pressure during the flow runup. This system pressurization is accounted for in the analyses.

The Cycle 24 $LHGRFAC_f$ multipliers are presented in Table 8.25 for ATRIUM 10XM fuel. A process consistent with the Westinghouse thermal-mechanical methodology was used to determine flow-dependent LHGR multipliers for OPTIMA2 fuel. The OPTIMA2 $LHGRFAC_f$ multipliers, presented in Table 8.26, provide protection against fuel centerline melt and overstraining of the cladding for OPTIMA2 fuel during operation at off-rated core flow conditions. The same set of $LHGRFAC_f$ multipliers supports base case operation and all the EOOS conditions for all Cycle 24 exposures.

The maximum flow during a flow excursion in single-loop operation is much less than the maximum flow during two-loop operation. Therefore, the flow-dependent MCPR limits and LHGR multipliers for two-loop operation are applicable for SLO.

5.3 *Equipment Out-of-Service Scenarios*

The following equipment out-of-service (EOOS) scenarios are supported for Quad Cities:

- One safety/relief valve out-of-service (One SRVOOS)
- Feedwater heater out-of-service (FHOOS)
- Turbine bypass valves out-of-service (TBVOOS)
- Combined FHOOS and TBVOOS
- Turbine control valve slow closure
- Power load unbalance out-of-service (PLUOOS)
- Pressure controller out-of-service (PCOOS)
- One stuck closed TCV or TSV
- One main steam isolation valve out-of-service (One MSIVOOS)
- Single-loop operation (SLO)

Operating limits for the EOOS scenario are established using the base case results for those events not affected by the specific scenario and analyses performed explicitly modeling the EOOS condition.

5.3.1 One SRVOOS

As noted earlier, the base case pressurization transient analyses were performed with the most beneficial relief, safety or safety/relief valve out-of-service. Therefore, the base case operating limits support operation with one relief, safety or safety/relief valve out-of-service. The EOOS operating limits also support operation with one relief, safety or safety/relief valve out-of-service.

5.3.2 FHOOS

The FHOOS scenario assumes a feedwater temperature reduction of 120°F at rated power and steam flow. At lower power levels the feedwater temperature reduction is less. Analyses were performed to support operation with reduced feedwater temperature at rated and reduced power operation. The effect of the reduced feedwater temperature is an increase in the core inlet subcooling which can change the axial power shape and core void fraction. In addition, the steam flow for a given power level decreases since more power is required to increase the enthalpy of the coolant to saturated conditions. The consequences of the FWCF event are potentially more severe as a result of the increase in core inlet subcooling during the overcooling phase of the event. While the decrease in steam flow tends to make the LRNB event less severe, the TCV initial position is further closed which tends to make the event more severe, especially at higher power levels.

FWCF events were analyzed to ensure that appropriate FHOOS operating limits are established. Results of high power LRNB and TTNB analyses show that the event with FHOOS is bound by the base case analysis. Analysis results also show that the IHPCIS with FHOOS is bound by the base case IHPCIS event. CRWE analyses were also evaluated for FHOOS and the base case result reported in Section 5.1.7 remains applicable.

5.3.3 TBVOOS

For this EOOS scenario, operation with TBVOOS means that the opening of the of the turbine bypass valves cannot be assured, thereby reducing the pressure relief capacity during anticipated operation occurrences. While the base case LRNB and TTNB events are analyzed assuming the turbine bypass valves are inoperable, operation with TBVOOS has a potential adverse effect on other events including: FWCF, LFWH, IHPCIS, LOSC, and CRWE. Analyses with the TBVOOS were performed for high power LFWH and CRWE events since the base case analyses showed that the steam flow increase during the event was greater than the available overcapacity of the turbine control valves. Analyses of the LOSC event with TBVOOS were performed because the decreased relief capacity with the TBVOOS results in system pressurization at a lower power level than the base case event. The low power IHPCIS events were analyzed with TBVOOS since they have the potential to reach the high level trip set point which results in a turbine trip. Analyses of the FWCF event with TBVOOS were also performed. The results of these EOOS analyses formed the basis of establishing the power-dependent MCPR limits and LHGR multipliers for the TBVOOS scenario.

Slow flow runup analyses were performed assuming pressurization occurs at or near the end of the flow run up excursion when the steam flow reaches the capacity of the turbine control valves. The results of the slow flow runup analysis with TBVOOS are used to establish flow-dependent MCPR limits and LHGR multipliers for the TBVOOS scenario as described in Section 5.2.

5.3.4 Combined FHOOS and TBVOOS

FWCF and IHPCIS analyses with both FHOOS and TBVOOS were performed to establish operating limits for the combined FHOOS and TBVOOS scenario. The results of the IHPCIS analysis show that the event with FHOOS and TBVOOS is bound by the IHPCIS event with TBVOOS.

5.3.5 Turbine Control Valve Slow Closure

LRNB analyses were performed to evaluate the impact of a TCV slow closure. The analyses included evaluating different combinations of valves closing slowly with a full stroke closure time of 5.0 seconds with the other valve(s) closing in fast closure mode with a full stroke closure time of 0.150 second. Combinations of 3 fast/ 1 slow, 2 fast/ 2 slow, 1 fast/ 3 slow and 4 slow were analyzed. No direct scram on TCV fast closure was credited. The event progresses towards the high flux scram and high pressure scram set points. The analyses assumed a turbine stop valve closure occurs when either the high flux or high pressure scram set point is reached. The reactor scram in the event occurs on the later of either TSV position or the earliest scram set point signal (high flux or high pressure). At power levels above 68% rated, the high flux scram set point was always attained for all the slow closure combinations. Below 68% power, the high flux scram was not credited in the analysis to ensure that all valve closure combinations are supported. The TCV slow closure results were used to establish power-dependent MCPR limits and LHGR multipliers. Analysis results for TCV slow closure with FHOOS show they are bound by the TCV slow closure results.

5.3.6 PLUOOS

The response to a load rejection event with the PLUOOS is different than the standard load rejection described in Section 5.1.1. With the PLUOOS, the TCVs close in the slow or servo mode and there is no direct scram on TCV fast closure. The sequence of events is similar to the PLU inoperable condition in which all the TCVs also close slowly. However, the turbine trip is not assumed to occur 0.625 seconds after the event initiation. Rather, the scenario is modeled assuming a TSV closure occurs when either the high flux or high pressure scram set point is reached, similar to the TCV slow closure scenario. The reactor scram in the event occurs on the later of either TSV position or the earliest scram set point signal (high flux or high pressure). In each case, the scram occurs on either a high flux or high pressure scram signal as the delays are always longer than the scram delay on TSV closure. The PLUOOS is a subset of the TCV slow closure condition; therefore, the TCV slow closure operating limits are applied for operation with PLUOOS. Operation with PLUOOS and FHOOS is supported by the TCV slow closure with FHOOS limits.

5.3.7 PCOOS

The PCOOS scenario assumes one pressure controller is out-of-service. In most situations, the backup controller will take over when the pressure mismatch occurs. However, if the pressure controller fails with one controller out of service, there is no backup and the pressure controller will open or close the TCVs resulting in a pressure regulator failure open (PRFO) or pressure regulator failure closure event. The consequences of the PRFO event are bound by the TTNB event as indicated in Section 2 and the pressure regulator failure closure event with one PCOOS results in the same sequence of events as LRNB with PLUOOS. Therefore, the TCV slow closure operating limits are applied for operation with PCOOS. The TCV slow closure limits also support operation with both PCOOS and PLUOOS. Operation with PCOOS and FHOOS is supported by the TCV slow closure with FHOOS limits.

5.3.8 One Stuck Closed TCV/TSV

With one of the turbine control valves assumed stuck closed, the other three TCVs will be further opened when operating at a given power level. In addition, the highest attainable power is decreased because of the decreased total steam flow capacity of the TCVs. At the same initial power with the valves further open, TCV closure events such as the LRNB are less severe because the pressurization occurs over a longer period of time. In addition, the steam line pressure drop would be slightly larger which also makes the pressurization less severe. While the FWCF event is not affected during the turbine stop valve closure portion of the event it may be impacted during the overcooling phase. At a certain power level, the TCVs will be in the full open position with no ability to accommodate any increase steam flow during the overcooling phase. The result would be an increase in pressure prior to the high level trip and a more severe event. Analyses show that below 75% of rated power, there would be no system pressurization during the overcooling phase of the FWCF at all feedwater temperature conditions (including the base case feedwater temperature range of +10/-30°F from nominal and FHOOS). Therefore, operation with 1 stuck closed TCV is supported with a maximum power of 75% of rated with the base case $MCPR_p$ limits and $LHGRFAC_p$ multipliers. The FHOOS $MCPR_p$ limits and $LHGRFAC_p$ multipliers support operation with the combined 1 stuck closed TCV and FHOOS condition. The TCV slow closure $MCPR_p$ limits and $LHGRFAC_p$ multipliers support operation with the following combined EOOS conditions: PCOOS with 1 stuck closed TCV; and PLUOOS with 1 stuck closed TCV.

As noted earlier, slow flow runup analyses were performed to establish flow-dependent limits for operation with 1 stuck closed TCV/TSV.

The discussion presented above for 1 stuck closed TCV is also applicable to the 1 stuck closed turbine stop valve condition.

5.3.9 One MSIVOOS

Operation with One MSIVOOS is supported for operation less than 75% of rated power. At these reduced power levels, the flow through any one steam line will not be greater than the flow at rated power when all MSIVs are available. Since all four turbine control valves are available, adequate pressure control can be maintained. The main difference in operation with One MSIVOOS is that the steam line pressure drop between the steam dome and the turbine valves is higher than if all MSIVs are available. Since low steam line pressure drop is limiting for pressurization transients, the results of the pressurization events with all MSIVs in service bound the results with One MSIVOOS. In addition, operation with One MSIVOOS has no impact on the other non-pressurization events evaluated to establish power-dependent operating limits. Therefore, the power-dependent operating limits applicable to base case operation with all MSIVs in service remain applicable for operation with One MSIVOOS for power levels less than or equal to 75% of rated. As noted earlier, slow flow runup analyses were performed to support operation with One MSIVOOS. The FHOOS power-dependent operating limits are applicable for operating with One MSIVOOS and FHOOS. The One MSIVOOS flow-dependent operating limits are applicable for operation in the combined One MSIVOOS and FHOOS condition.

5.3.10 Single-Loop Operation

In SLO, the two-loop operation ΔCPRs and LHGRFAC multipliers from the anticipated operation occurrences remain applicable.

While the SLO pump seizure is classified as a limiting fault, the event is evaluated to protect the AOO acceptance criteria. A SLO pump seizure analysis was performed at 50% rated power and a core flow of 50 Mlbm/hr. After the initial significant core flow decrease, the core flow recovers some and the system experiences a level swell to the point where the high level trip set point is reached. A nominal level setpoint is used in the analysis to ensure that a turbine trip would occur should the level increase enough to cause a turbine trip. The subsequent turbine trip causes a pressurization event with the associated power and heat flux excursion. The analysis also assumed that the turbine bypass valves are out of service. The limiting CPR occurs near the time of the lowest core flow during the event, well before the turbine trip occurs. As a result, the results for the SLO pump seizure base case event remain limiting whether or not the turbine trip occurs or whether or not the turbine bypass valves are out of service. The EOOS scenarios do not adversely impact the consequences of the SLO pump seizure event. The ΔCPRs for the analysis are 1.01 for the OPTIMA2 fuel and 0.93 for the ATRIUM 10XM fuel. The event is a power decrease event so there is no threat to the transient thermal-mechanical limits during this event. The results for the state point analyzed are limiting for lower power levels and core flows.

Figures 5.5 and 5.6 show the response of various reactor and plant parameters during the SLO pump seizure event. Since the SLO pump seizure ΔCPR results are limiting for single loop operation and are used to establish the SLO MCPR operating limits, SLO is only supported for power levels ≤ 50% of rated and core flows ≤ 50 Mlbm/hr (51% of rated).

In addition to the inclusion of the SLO pump seizure analysis results, the other impacts on the MCPR and MAPLHGR limits for SLO are an increase of 0.02 in the SLMCPR as discussed in Section 4.2, and the application of an SLO MAPLHGR multiplier discussed in Section 8.3. The SLO MCPR results are presented in Section 8.1 for base SLO and for SLO with the EOOS scenarios. The TLO EOOS $LHGRFAC_p$ multipliers remain applicable in SLO.

5.4 *Licensing Power Shape*

The licensing axial power profile used by AREVA for the plant transient analyses bounds the projected end of full power axial power profile. The conservative licensing axial power profile generated at the EOCLB core average exposure of 36,774 MWd/MTU is given in Table 5.12. Cycle 24 operation is considered to be in compliance when:

- The normalized power generated in the bottom 7 nodes from the projected EOFP solution at the state conditions provided in Table 5.12 is greater than the normalized power generated in the bottom 7 nodes in the licensing basis axial power profile and the individual normalized power from the EOFP solution is greater than the corresponding normalized power from the licensing basis axial power profile for at least 6 of the 7 bottom nodes.

- The projected EOFP condition occurs at a core average exposure less than or equal to EOCLB.

If the criteria cannot be fully met, the licensing basis may nevertheless remain valid but further assessment will be required.

The licensing basis power profile in Table 5.12 was calculated using the MICROBURN-B2 code. Compliance analyses must also be performed using MICROBURN-B2. Note that the power profile comparison should be done without incorporating instrument updates to the axial profile because the updated power is not used in the core monitoring system to accumulate assembly burnups.

Table 5.1 Exposure Basis for Quad Cities Unit 2 Cycle 24 Representative Cycle Transient Analysis

Cycle Exposure at End of Interval (MWd/MTU)	Core Average Exposure (MWd/MTU)*	Comments
0	20,702	Beginning of cycle
14,000	34,702	Break point for exposure-dependent $MCPR_p$ limits (NEOC)
16,072	36,774	Design basis rod patterns to EOFP + 25 EFPD (EOCLB)

* Note that the thermal limits presented in Section 8.0 are based on core average exposure.

**Table 5.2 Scram Speed
Insertion Times***

Control Rod Position (% Insertion)	TSSS Time (sec)	ISS Time (sec)	NSS Time (sec)
0 (*full-out*)	0.00	0.00	0.00
0	0.20	0.20	0.20
5	0.48	0.36	0.324
20	0.89	0.72	0.694
50	1.98	1.58	1.510
90	3.44	2.80	2.670

* Adjustments to the NSS, ISS, and TSSS scram speeds inputs were made in the analyses to account for the effects of slow and stuck rods.

Table 5.3 NEOC Base Case LRNB Transient Results

Power (% rated)	OPTIMA2 ΔCPR	ATRIUM 10XM ΔCPR
TSSS Insertion Times		
100	0.34	0.31
90	0.34	0.31
80	0.35	0.32
70	0.38	0.36
60	0.38	0.34
50	0.39	0.32
50 PLU inoperable	0.46	0.36
38.5 PLU inoperable	0.47	0.35
38.5 at > 60%F below P_{bypass}	1.20	1.23
38.5 at ≤ 60%F below P_{bypass}	0.93	1.03
25 at > 60%F below P_{bypass}	1.58	1.66
25 at ≤ 60%F below P_{bypass}	1.25	1.42
NSS Insertion Times		
100	0.29	0.26
90	0.30	0.27
80	0.31	0.28
70	0.31	0.33
60	0.30	0.32
50	0.31	0.30
50 PLU inoperable	0.39	0.33
38.5 PLU inoperable	0.41	0.31
ISS Insertion Times		
100	0.30	0.27
90	0.31	0.28
80	0.32	0.29
70	0.31	0.33
60	0.30	0.33
50	0.32	0.31
50 PLU inoperable	0.41	0.33
38.5 PLU inoperable	0.42	0.32

Table 5.4 EOCLB Base Case LRNB Transient Results

Power (% rated)	OPTIMA2 ΔCPR	ATRIUM 10XM ΔCPR
TSSS Insertion Times		
100	0.37	0.33
90	0.38	0.33
80	0.38	0.33
70	0.42	0.36
60	0.44	0.35
50	0.45	0.34
50 PLU inoperable	0.51	0.37
38.5 PLU inoperable	0.50	0.36
38.5 at > 60%F below P_{bypass}	1.20	1.23
38.5 at ≤ 60%F below P_{bypass}	0.93	1.03
25 at > 60%F below P_{bypass}	1.58	1.66
25 at ≤ 60%F below P_{bypass}	1.25	1.42
NSS Insertion Times		
100	0.34	0.30
90	0.35	0.31
80	0.36	0.32
70	0.35	0.35
60	0.34	0.33
50	0.36	0.31
50 PLU inoperable	0.44	0.34
38.5 PLU inoperable	0.44	0.32
ISS Insertion Times		
100	0.35	0.31
90	0.36	0.32
80	0.36	0.32
70	0.37	0.35
60	0.36	0.34
50	0.37	0.32
50 PLU inoperable	0.45	0.34
38.5 PLU inoperable	0.45	0.33

Table 5.5 NEOC Base Case TTNB Transient Results

Power (% rated)	OPTIMA2 ΔCPR	ATRIUM 10XM ΔCPR
TSSS Insertion Times		
100	0.34	0.31
90	0.34	0.31
80	0.34	0.31
70	0.36	0.31
60	0.35	0.30
50	0.37	0.28
38.5	0.36	0.26
38.5 at > 60%F below P_{bypass}	1.20	1.23
38.5 at ≤ 60%F below P_{bypass}	0.93	1.03
25 at > 60%F below P_{bypass}	1.58	1.66
25 at ≤ 60%F below P_{bypass}	1.25	1.42
NSS Insertion Times		
100	0.29	0.26
90	0.29	0.27
80	0.30	0.27
70	0.30	0.28
60	0.29	0.27
50	0.28	0.25
38.5	0.28	0.22
ISS Insertion Times		
100	0.30	0.27
90	0.30	0.27
80	0.31	0.28
70	0.30	0.28
60	0.29	0.28
50	0.30	0.26
38.5	0.29	0.22

Table 5.6 EOCLB Base Case TTNB Transient Results

Power (% rated)	OPTIMA2 ΔCPR	ATRIUM 10XM ΔCPR
TSSS Insertion Times		
100	0.37	0.33
90	0.38	0.33
80	0.38	0.33
70	0.41	0.32
60	0.42	0.30
50	0.43	0.30
38.5	0.40	0.28
38.5 at > 60%F below P_{bypass}	1.20	1.23
38.5 at ≤ 60%F below P_{bypass}	0.93	1.03
25 at > 60%F below P_{bypass}	1.58	1.66
25 at ≤ 60%F below P_{bypass}	1.25	1.42
NSS Insertion Times		
100	0.34	0.31
90	0.35	0.31
80	0.35	0.31
70	0.34	0.30
60	0.33	0.29
50	0.35	0.27
38.5	0.31	0.23
ISS Insertion Times		
100	0.35	0.31
90	0.35	0.31
80	0.36	0.32
70	0.35	0.30
60	0.34	0.29
50	0.36	0.27
38.5	0.32	0.24

Table 5.7 NEOC Base Case FWCF Transient Results

Power (% rated)	OPTIMA2 ΔCPR	ATRIUM 10XM ΔCPR
TSSS Insertion Times		
100	0.31	0.29
90	0.35	0.33
80	0.39	0.38
70	0.45	0.43
60	0.52	0.50
50	0.62	0.59
38.5	0.79	0.76
38.5 at > 60%F below P_{bypass}	1.17	1.24
38.5 at ≤ 60%F below P_{bypass}	1.10	1.24
25 at > 60%F below P_{bypass}	1.40	1.42
25 at ≤ 60%F below P_{bypass}	1.33	1.42
NSS Insertion Times		
100	0.28	0.26
90	0.31	0.30
80	0.36	0.35
70	0.42	0.40
60	0.49	0.48
50	0.60	0.57
38.5	0.77	0.74
ISS Insertion Times		
100	0.28	0.27
90	0.32	0.30
80	0.37	0.35
70	0.42	0.41
60	0.50	0.48
50	0.60	0.58
38.5	0.78	0.74

Table 5.8 EOCLB Base Case FWCF Transient Results

Power (% rated)	OPTIMA2 ΔCPR	ATRIUM 10XM ΔCPR
TSSS Insertion Times		
100	0.34	0.31
90	0.37	0.34
80	0.41	0.38
70	0.46	0.43
60	0.52	0.50
50	0.62	0.59
38.5	0.79	0.76
38.5 at > 60%F below P_{bypass}	1.17	1.24
38.5 at ≤ 60%F below P_{bypass}	1.10	1.24
25 at > 60%F below P_{bypass}	1.40	1.42
25 at ≤ 60%F below P_{bypass}	1.33	1.42
NSS Insertion Times		
100	0.32	0.29
90	0.35	0.32
80	0.39	0.37
70	0.44	0.41
60	0.50	0.48
50	0.60	0.57
38.5	0.77	0.74
ISS Insertion Times		
100	0.32	0.29
90	0.35	0.33
80	0.40	0.37
70	0.45	0.41
60	0.51	0.48
50	0.60	0.58
38.5	0.78	0.74

Table 5.9 EOCLB Base Case IHPCIS Transient Results*

Power (% rated)	OPTIMA2 ΔCPR	ATRIUM 10XM ΔCPR
All Insertion Times		
80	0.19	0.22
50	0.58	0.53
38.5	0.79	0.73
38.5 at > 60%F below P_{bypass}	0.98	1.08
38.5 at ≤ 60%F below P_{bypass}	0.94	1.05
25 at > 60%F below P_{bypass}	0.98	0.96
25 at ≤ 60%F below P_{bypass}	1.08	0.98

* 100% power analyses were not performed due to the significant available margin to limits.

Table 5.10 EOCLB Base Case LOSC Transient Results

Power (% rated)	OPTIMA2 ΔCPR	ATRIUM 10XM ΔCPR
TSSS Insertion Times		
100	0.19	0.19
90	0.25	0.26
80	0.33	0.33
70	0.41	0.42
65	0.44	0.46
NSS and ISS Insertion Times		
100	0.18	0.19
90	0.25	0.26
80	0.33	0.33
70	0.40	0.42
65	0.44	0.46

Table 5.11 Limiting Loss of Feedwater Heating Transient Analysis Results, All Exposures

Power (% rated)	OPTIMA2 ΔCPR	ATRIUM 10XM ΔCPR
All Scram Insertion Times		
100	0.22	0.21
95	0.23	0.22
90	0.24	0.21
80	0.22	0.25
75	0.26	0.25
70	0.27	0.26
65	0.28	0.26
60	0.29	0.27
55	0.49	0.44
50	0.58	0.53
38.5	0.72	0.67
25	0.41	0.37

Table 5.12 Licensing Basis Core Average Axial Power Profile

State Conditions for Power Shape Evaluation	
Power, MWt	2957.0
MICROBURN-B2 pressure, psia	1015.0
Inlet subcooling, Btu/lbm	22.3
Flow, Mlb/hr	105.84
Control state	ARO
Core average exposure (EOCLB), MWd/MTU	36,774

Licensing Axial Power Profile (Normalized)

	Node	Power
Top	24	0.241
	23	0.440
	22	0.908
	21	1.082
	20	1.208
	19	1.293
	18	1.354
	17	1.434
	16	1.478
	15	1.485
	14	1.458
	13	1.482
	12	1.448
	11	1.402
	10	1.340
	9	1.254
	8	1.134
	7	0.959
	6	0.773
	5	0.602
	4	0.466
	3	0.377
	2	0.294
Bottom	1	0.090

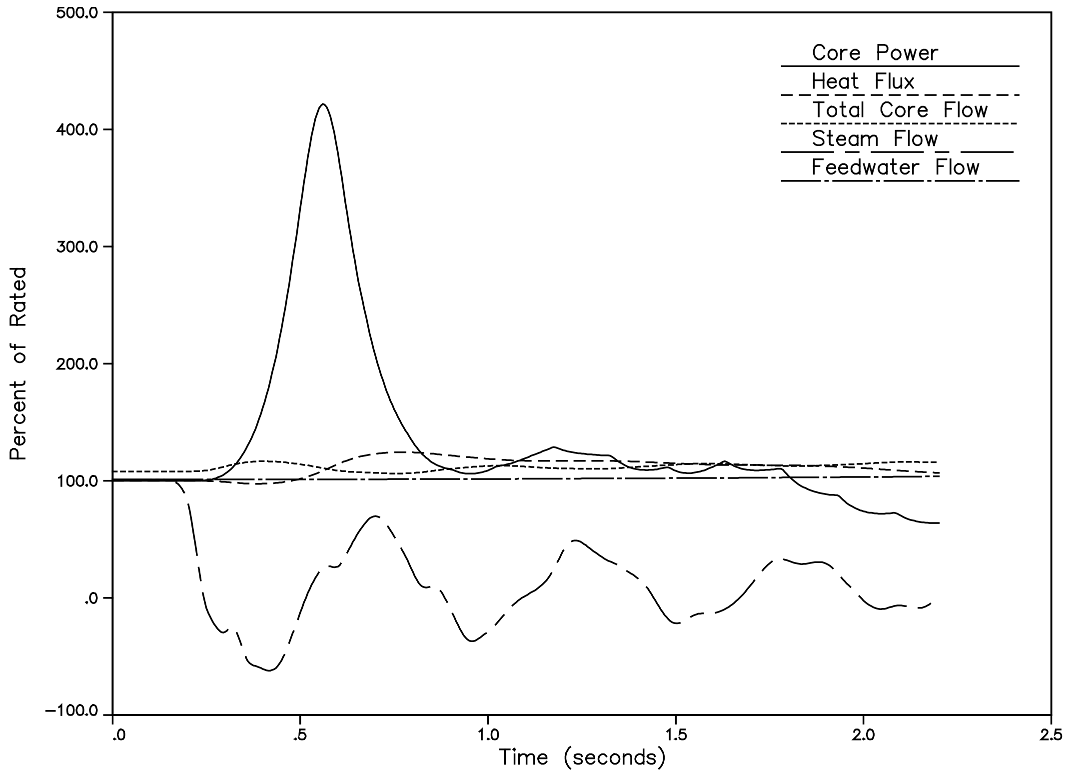


Figure 5.1 EOCLB LRNB at 100P/108F – TSSS Key Parameters

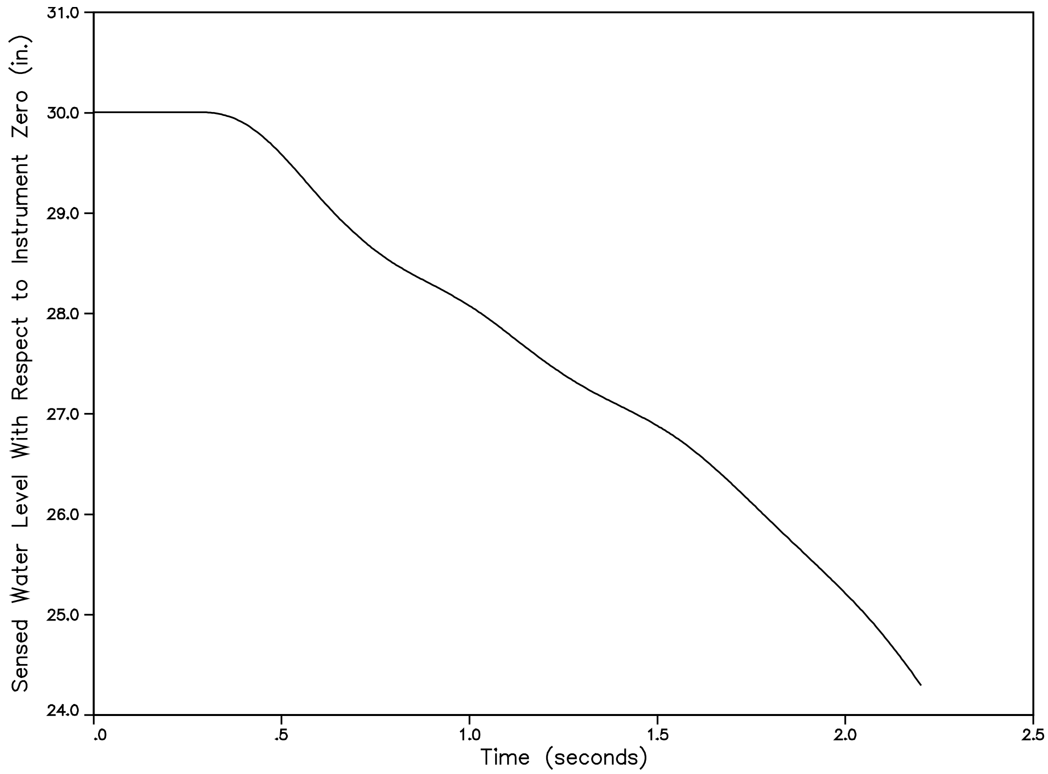


Figure 5.2 EOCLB LRNB at 100P/108F – TSSS Sensed Water Level

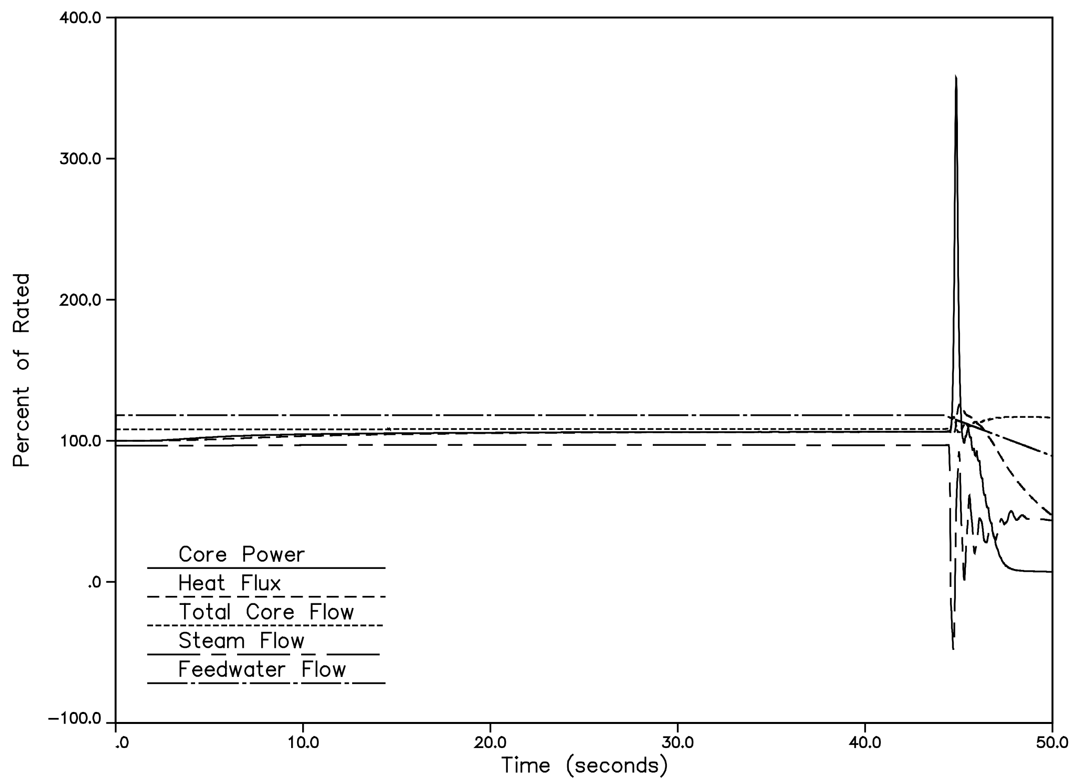


Figure 5.3 EOCLB FWCF at 100P/108F – TSSS
Key Parameters

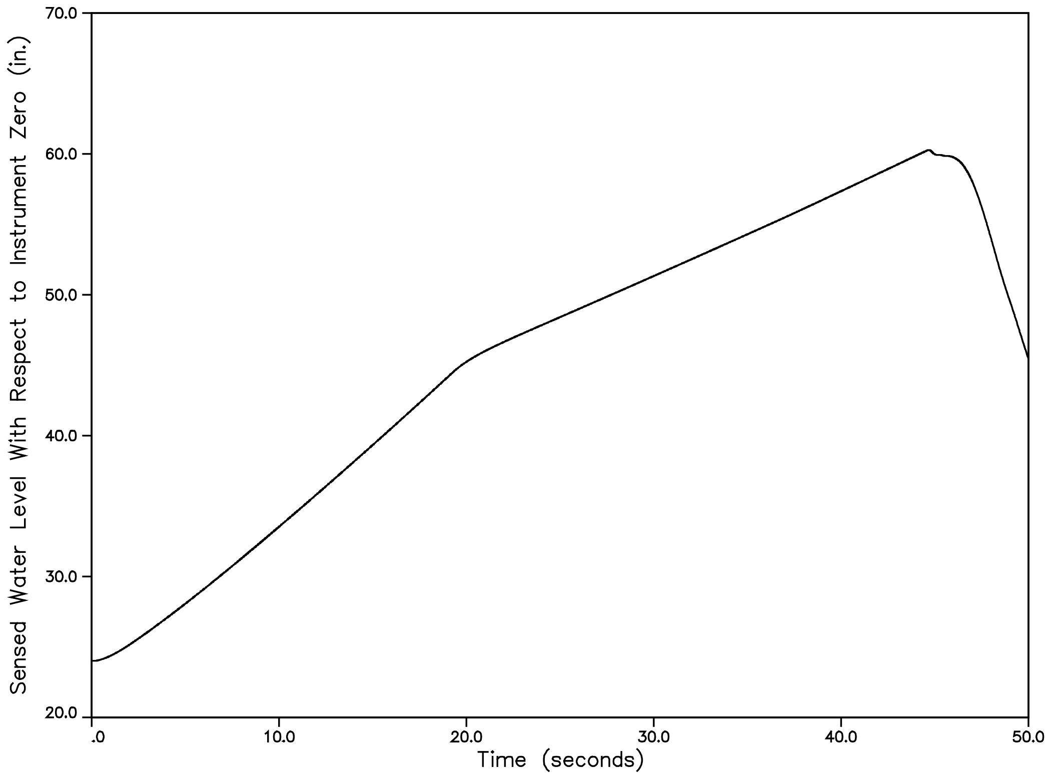


**Figure 5.4 EOCLB FWCF at 100P/108F – TSSS
Sensed Water Level**

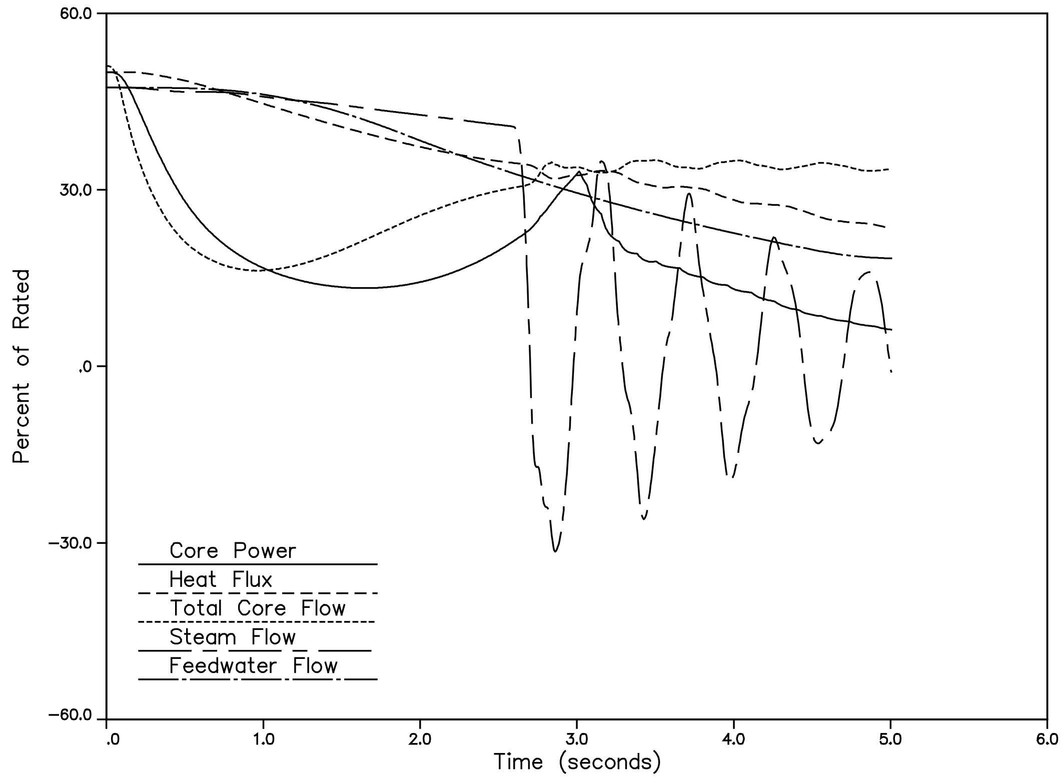


Figure 5.5 SLO Pump Seizure at 50P/51F – TSSS Key Parameters

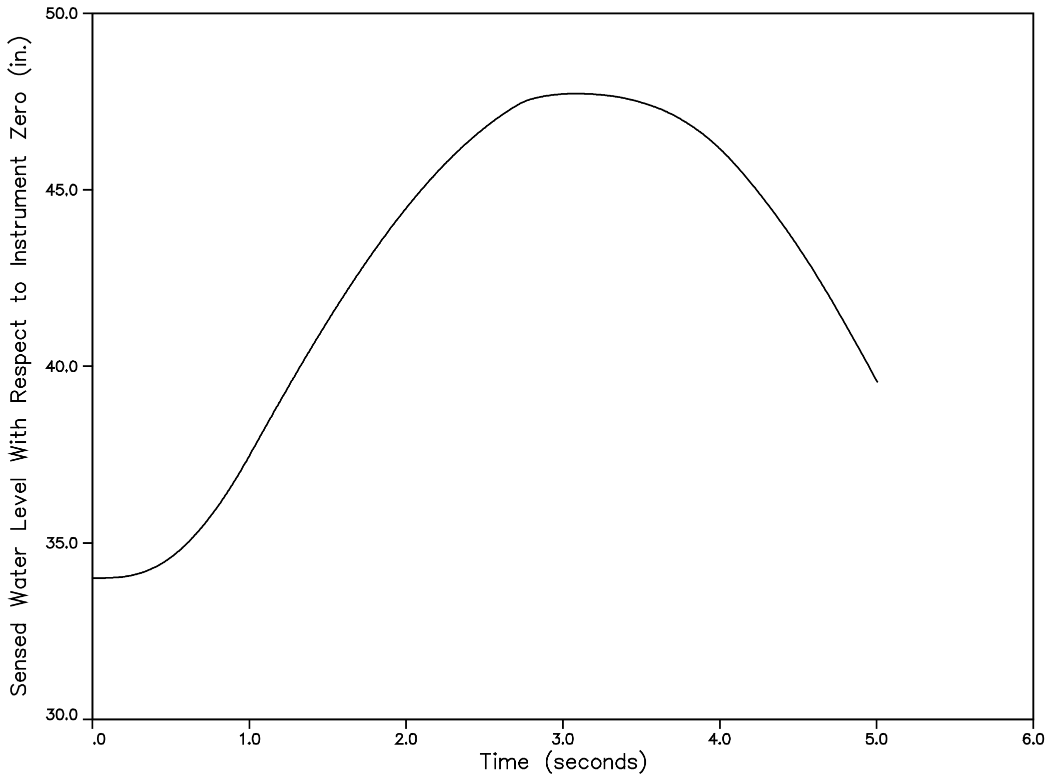


Figure 5.6 SLO Pump Seizure at 50P/51F – TSSS Sensed Water Level

6.0 Postulated Accidents

6.1 *Loss-of-Coolant Accident (LOCA)*

The results of the ATRIUM 10XM LOCA analysis are presented in References 22 and 23. The ATRIUM 10XM PCT is 2138°F. The peak local metal water reaction is 4.05% and the core wide metal water reaction is < 1%. The SLO MAPLHGR multiplier is 0.80.

The LOCA analysis for OPTIMA2 fuel is presented in Reference 24 (UFSAR).

6.2 *Control Rod Drop Accident (CRDA)*

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 in Reference 29. Reference 25 describes the approved AREVA generic CRDA methodology. Subsequent calculations have shown that the methodology is applicable to fuel modeled with the CASMO4/MICROBURN-B2 code system. The CRDA analysis was performed with the approved methodology described in Reference 25.

The CRDA analysis results demonstrate that the maximum deposited fuel rod enthalpy is less than the NRC licensing threshold of 280 cal/g and that the estimated number of fuel rods that exceed the fuel damage threshold of 170 cal/g is less than the number of failed rods supported by the Quad Cities licensing basis. Quad Cities UFSAR Section 15.4.10 identifies a value of 850 failed rods for 8x8 and 9x9 fuel designs which was later increased to 1153 rods for the GE14 (10x10) fuel design at uprated power. The number of fuel rods estimated to exceed the fuel damage threshold is well below the more limiting value of 850 for all fuel designs.

Maximum dropped control rod worth, mk	11.13
Core average Doppler coefficient, $\Delta k/k/^{o}F$	-10.17×10^{-6}
Effective delayed neutron fraction	0.0052
Four-bundle local peaking factor	1.389
Maximum deposited fuel rod enthalpy, cal/g	190.6
Maximum number of rods exceeding 170 cal/g	546

6.3 *Fuel and Equipment Handling Accident*

AREVA has performed a fuel handling accident analysis for the ATRIUM 10XM fuel design and determined that a maximum of 162 fuel rods fail. This number of failed rods is less than the 176 failed rods in the GE14 10x10 FHA analysis, and is bounded by dose from the licensing basis failure of 111 fuel rods in the GE 7x7 fuel. Therefore, the current fuel handling accident AST analysis remains applicable for the introduction of the ATRIUM 10XM fuel design.

6.4 *Fuel Loading Error (Infrequent Event)*

There are two types of fuel loading errors possible in a BWR: the mislocation of a fuel assembly in a core position prescribed to be loaded with another fuel assembly, and the misorientation of a fuel assembly with respect to the control blade. As described in Reference 26, the fuel loading error is characterized as an infrequent event. The acceptance criteria are that the offsite dose consequences due to the event shall not exceed a small fraction of the 10 CFR 50.67 limits.

6.4.1 Mislocated Fuel Bundle

AREVA has performed a cycle specific fuel mislocation error analysis for the Quad Cities Unit 2 Cycle 24 representative core design which covers both the ATRIUM 10XM and OPTIMA2 fuel types. This analysis evaluated the impact of a mislocated assembly against potential fuel rod failure mechanisms due to increased LHGR and reduced CPR. Based on this analysis, the offsite dose criteria (a small fraction of 10 CFR 50.67) is conservatively satisfied. A dose consequence evaluation is not necessary since no rod approached the fuel centerline melt or 1% strain limits, and less than 0.1% of the fuel rods are expected to experience boiling transition which could result in a dryout induced failure.

6.4.2 Misoriented Fuel Bundle

AREVA has performed cycle specific fuel assembly misorientation analyses for the ATRIUM 10XM and OPTIMA2 fuel assemblies in the Quad Cities Unit 2 Cycle 24 representative core design. The analysis was performed assuming that the limiting assembly was loaded with either a 90° or 180° misorientation and depleted through the cycle without operator interaction. These analyses demonstrate that the small fraction of 10 CFR 50.67 offsite dose criteria is conservatively satisfied. A dose consequence evaluation is not necessary since less than 0.1% of fuel rods approached the fuel centerline melt or 1% strain limits and less than 0.1% of the fuel rods are expected to experience boiling transition.

7.0 Special Analyses

7.1 *ASME Overpressurization Analysis*

This section describes the maximum overpressurization analyses performed to demonstrate compliance with the ASME Boiler and Pressure Vessel Code. The analysis shows that the safety and safety/relief valves at Quad Cities Unit 2 have sufficient capacity and performance to prevent the reactor vessel pressure from reaching the safety limit of 110% of the design pressure.

Analyses were performed with the AREVA plant simulator code COTRANSA2 (Reference 16) for 102% power and both 95.3% and 108% core flow at the highest Cycle 24 exposure where rated power operation can be attained. The following events were evaluated: MSIV closure, TCV closure, TSV closure and the FWCF event with TBVOOS. The limiting overpressurization event was the FWCF with TBVOOS. The base FWCF event is described in Section 5.1.3. The main differences in the FWCF overpressurization analysis are that no credit is taken for the direct scram on TSV position or pressure relief from the turbine bypass valves. Sensitivity analyses showed that crediting the ATWS-RPT resulted in higher peak vessel and peak dome pressures. The following 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.
- TSSS insertion times were used.
- A nominal ATWS-RPT set point of 1170 psig was used.
- The initial dome pressure was set at the maximum allowed by the Technical Specifications, 1019.7 psia (1005 psig).

Results of the limiting overpressurization analyses are presented in Table 7.1. Figures 7.1 – 7.4 show the response of various reactor plant parameters during the limiting FWCF with TBVOOS event. 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 and dome pressure limit of 1345 psig are not exceeded. The ASME analysis results are applicable to all EOOS conditions presented in Table 1.1.

7.2 ***ATWS Event Evaluation***

7.2.1 ATWS Overpressurization Analysis

This section describes the analyses performed to demonstrate that the peak vessel pressure for the limiting ATWS event is less than the ASME Service Level C limit of 120% of the design pressure (1500 psig). The ATWS overpressurization analyses were performed at 100% power at 95.3% and 108% flow. The MSIV closure and pressure regulator failure open (PRFO) events were evaluated. Failure of the pressure regulator in the open position causes the turbine control and turbine bypass valves to open such that steam flow increases until the maximum flow through all TCVs and all 9 turbine bypass valves is attained. The system pressure decreases until the low pressure setpoint is reached, resulting in the closure of the MSIVs. The resulting pressurization wave causes a decrease in core voids and an increase in core pressure thereby increasing the core power.

The following assumptions were made in the analyses:

- The analytical limit ATWS-RPT setpoint (1200 psig) and function were assumed.
- All relief, safety and safety/relief valves were assumed in service.
- All scram functions were disabled.
- The initial dome pressure was set to 1004.3 psia.
- The MSIV closure is based on a nominal closure time of 4.0 seconds for both events.

Results of ATWS overpressurization analyses are presented in Table 7.2. Figures 7.5 – 7.8 show the response of various reactor plant parameters during the limiting PRFO event, the event which results in the maximum vessel pressure. The maximum lower plenum pressure is 1489 psig and the maximum dome pressure is 1473 psig. The 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 ATWS maximum vessel pressure limit of 1500 psig is not exceeded.

The analyses also showed that 57 seconds after event initiation, the maximum pressure in the reactor vessel at the SLC system injection elevation is 1346 psig.

7.2.2 Long-Term Evaluation

Fuel design differences may impact the power and pressure excursion experienced during the ATWS event. This in turn may impact the amount of steam discharged to the suppression pool and containment. **[**

]

Relative to the 10 CFR 50.46 acceptance criteria (i.e., PCT and cladding oxidation), the consequences of an ATWS event are bound by those of the limiting LOCA event.

7.3 *Standby Liquid Control System*

In the event that the control rod scram function becomes incapable of rendering the core in a shutdown state, the standby liquid control (SLC) system 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 Quad Cities Unit 2 SLC system is required to be able to inject 918 ppm natural boron equivalent at 68°F into the reactor coolant (including a 25% allowance for imperfect mixing, leakage, and volume of other piping connected to the reactor). AREVA has performed an analysis that demonstrates that the SLC system meets the required shutdown capability for the Cycle 24 representative core design. The analysis was performed to support a coolant temperature of 358.3°F with a boron concentration equivalent to 918 ppm at 68°F. The temperature of 358.3°F corresponds to the low pressure permissive for the RHR shutdown cooling suction valves, and represents the maximum reactivity condition with soluble boron in the coolant. The analysis shows the core to be subcritical throughout the cycle by at least 6.54% Δk/k.

Table 7.1 ASME Overpressurization Analysis Results*

Event	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel Pressure Lower-Plenum (psig)	Maximum Dome Pressure (psig)
MSIV Closure (102P/108F)	282	127	1360	1340
MSIV Closure (102P/95.3F)	268	126	1359	1340
TCV Closure (102P/108F)	454	128	1360	1340
TCV Closure (102P/95.3F)	398	127	1358	1339
TSV Closure (102P/108F)	453	128	1360	1340
TSV Closure (102P/95.3F)	397	127	1358	1339
FWCF with TBVOOS (102P/108F)	477	133	1362	1342
FWCF with TBVOOS (102P/95.3F)	420	131	1360	1341

* The peak pressure results include adjustments to address the NRC concerns discussed in Section 7.1.

Table 7.2 ATWS Overpressurization Analysis Results*

Event	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel Pressure Lower-Plenum (psig)	Maximum Dome Pressure (psig)
MSIV closure (100P/108F)	237	150	1468	1452
MSIV closure (100P/95.3F)	230	147	1479	1463
PRFO (100P/108F)	251	158	1480	1464
PRFO (100P/95.3F)	240	154	1489	1473

* The peak pressure results include adjustments to address the NRC concerns discussed in Section 7.2.

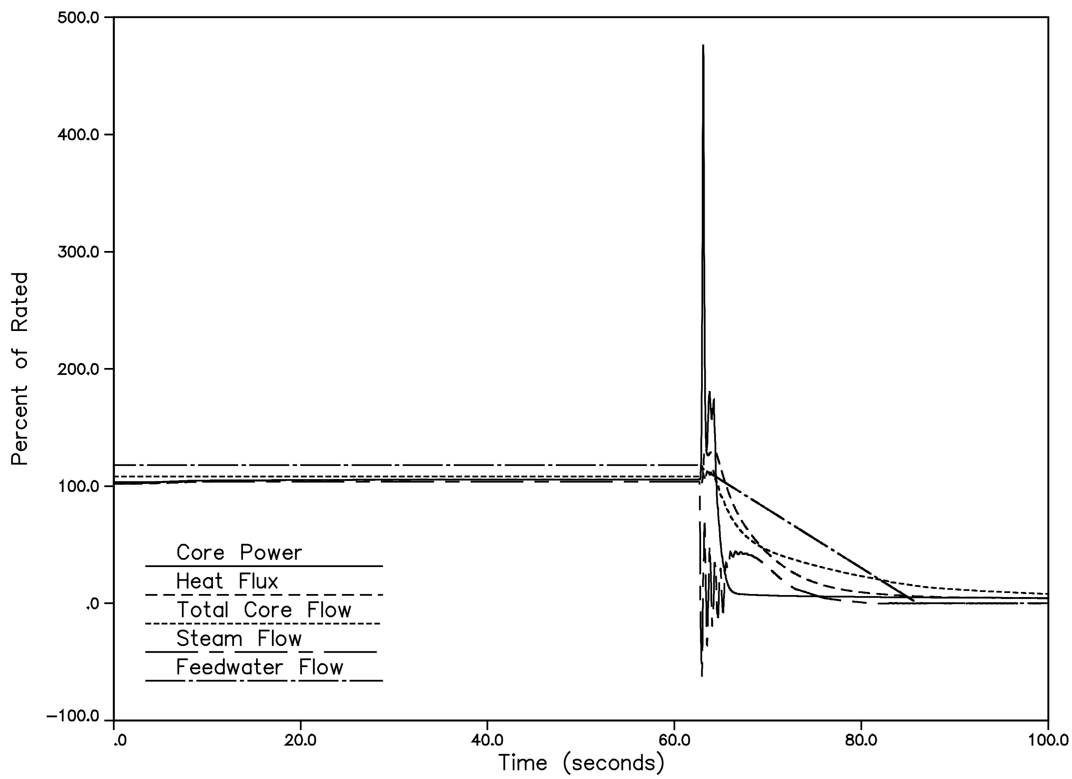


Figure 7.1 FWCF with TBVOOS ASME Overpressurization Event at 102P/108F – Key Parameters

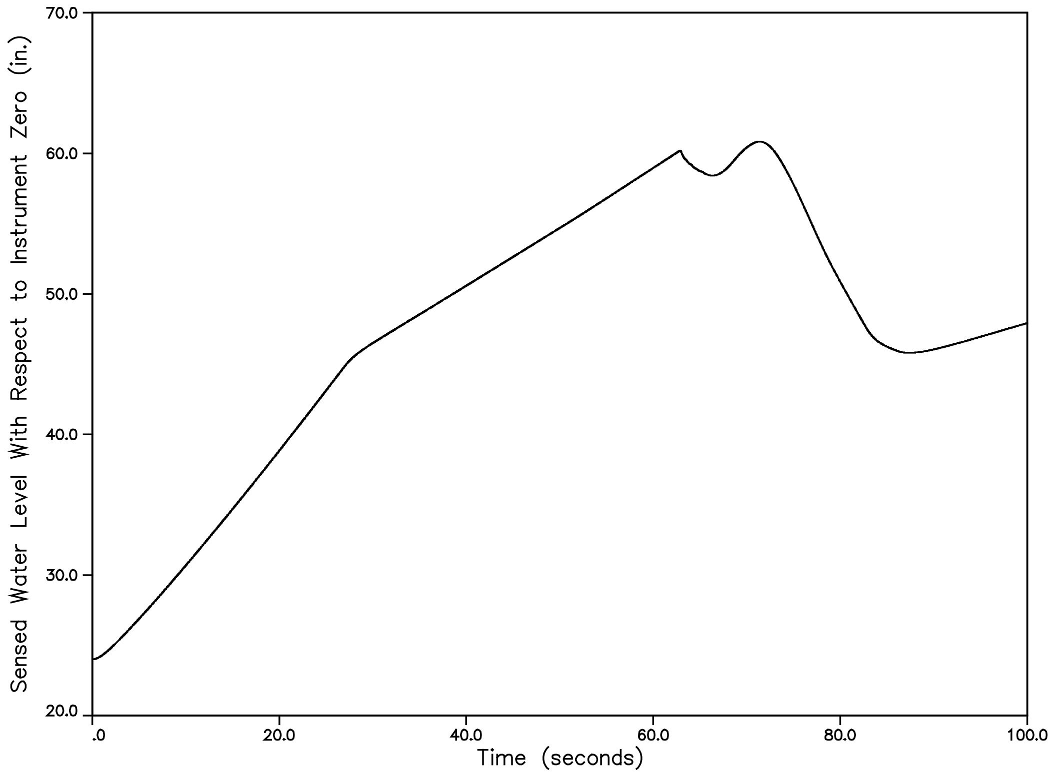


Figure 7.2 FWCF with TBVOOS ASME Overpressurization Event at 102P/108F – Sensed Water Level

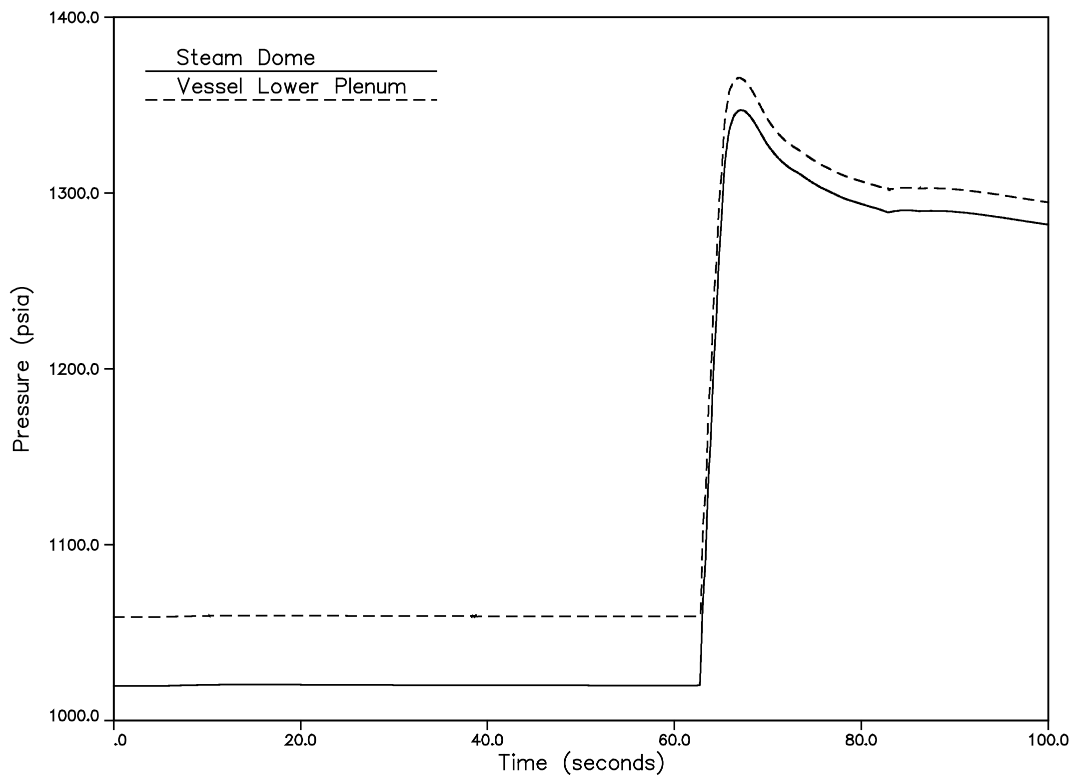


Figure 7.3 FWCF with TBVOOS ASME Overpressurization Event at 102P/108F – Vessel Pressures*

* The pressures presented in this figure do not include the adjustments associated with the NRC concerns discussed in Section 7.1.

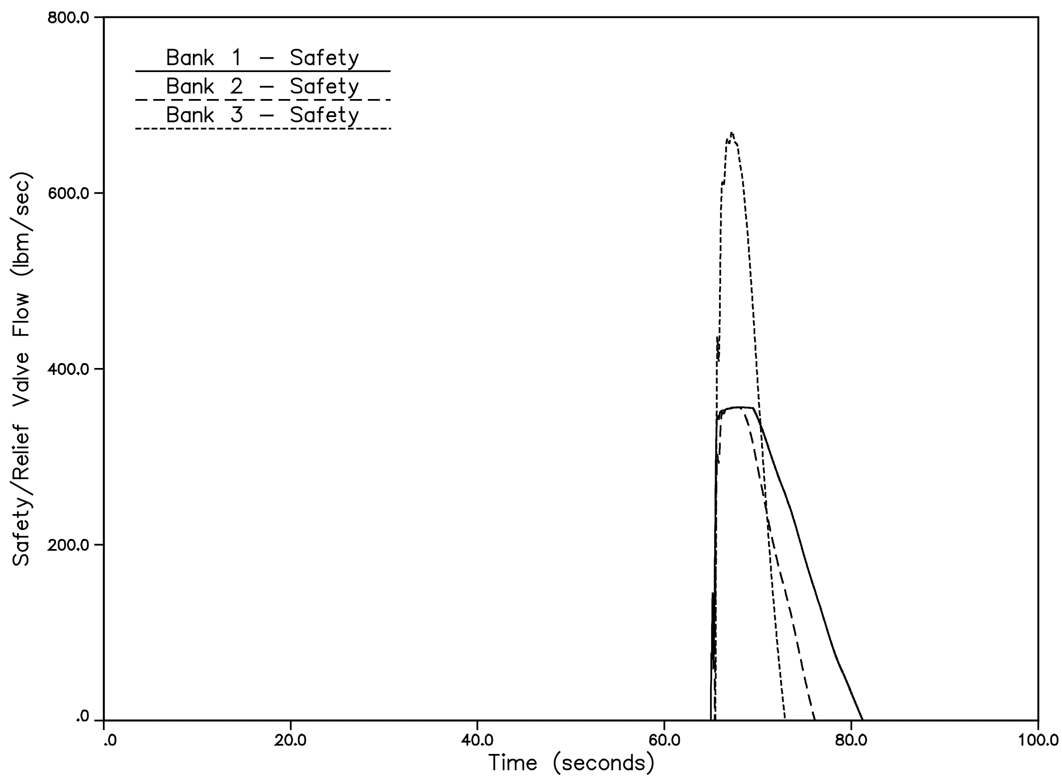


Figure 7.4 FWCF with TBVOOS ASME Overpressurization Event at 102P/108F – Safety/Relief Valve Flow Rates

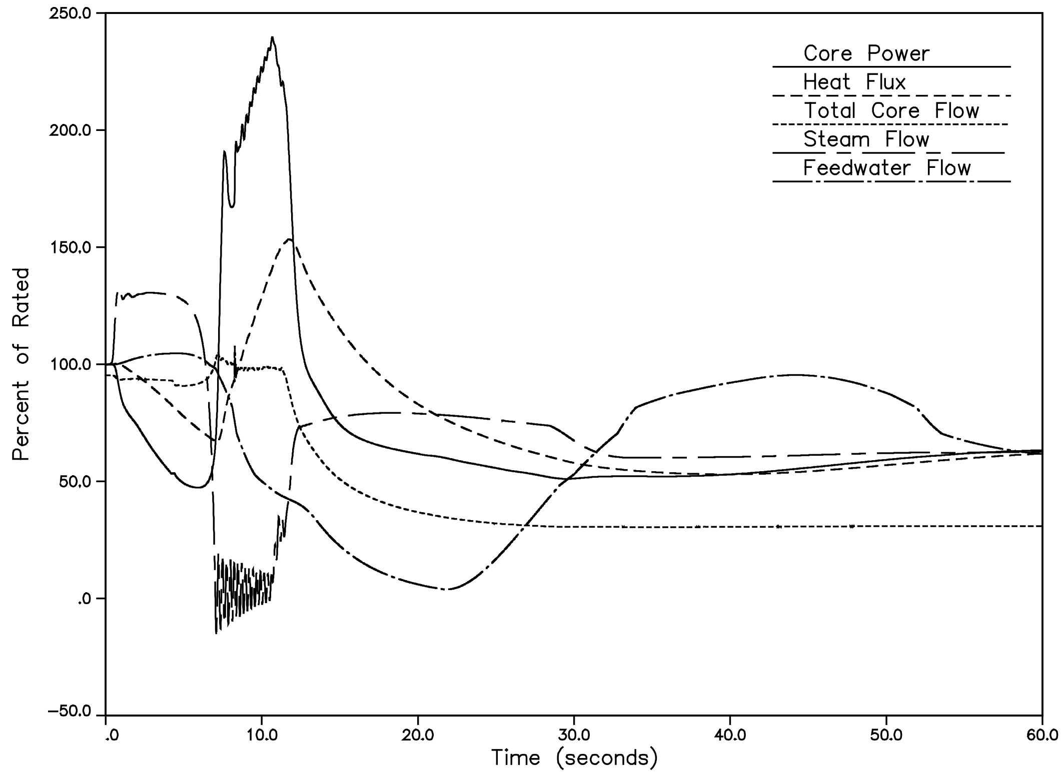


Figure 7.5 PRFO ATWS Overpressurization Event at 100P/95.3F – Key Parameters

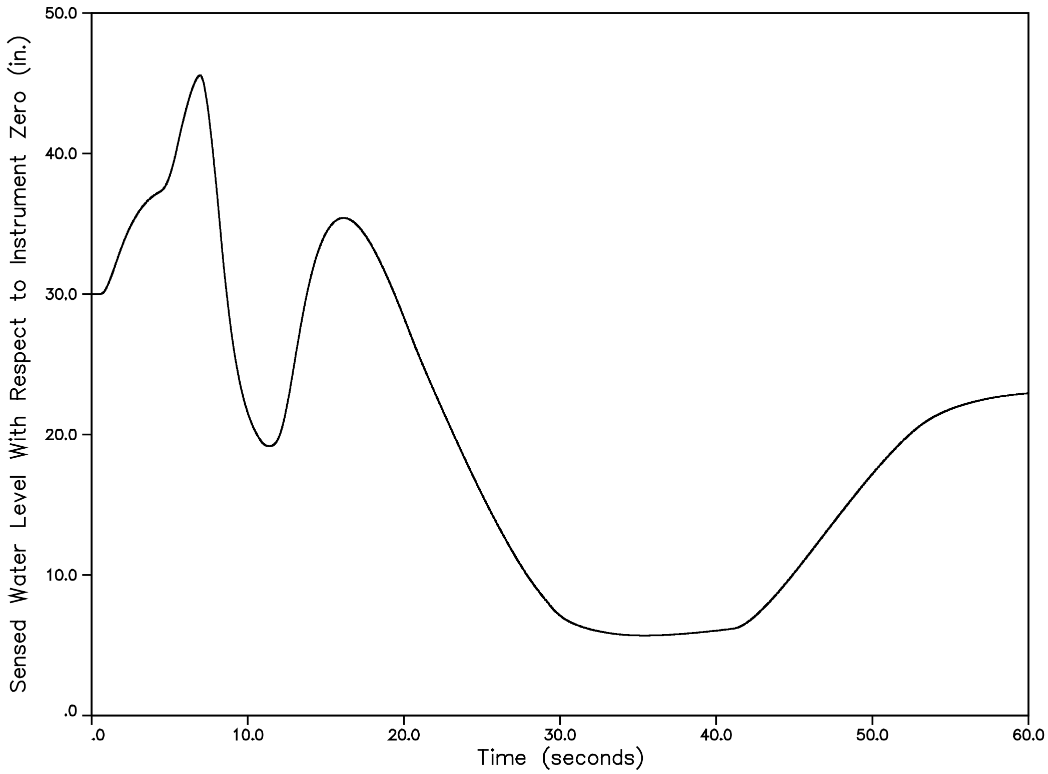


Figure 7.6 PRFO ATWS Overpressurization Event at 100P/95.3F – Sensed Water Level

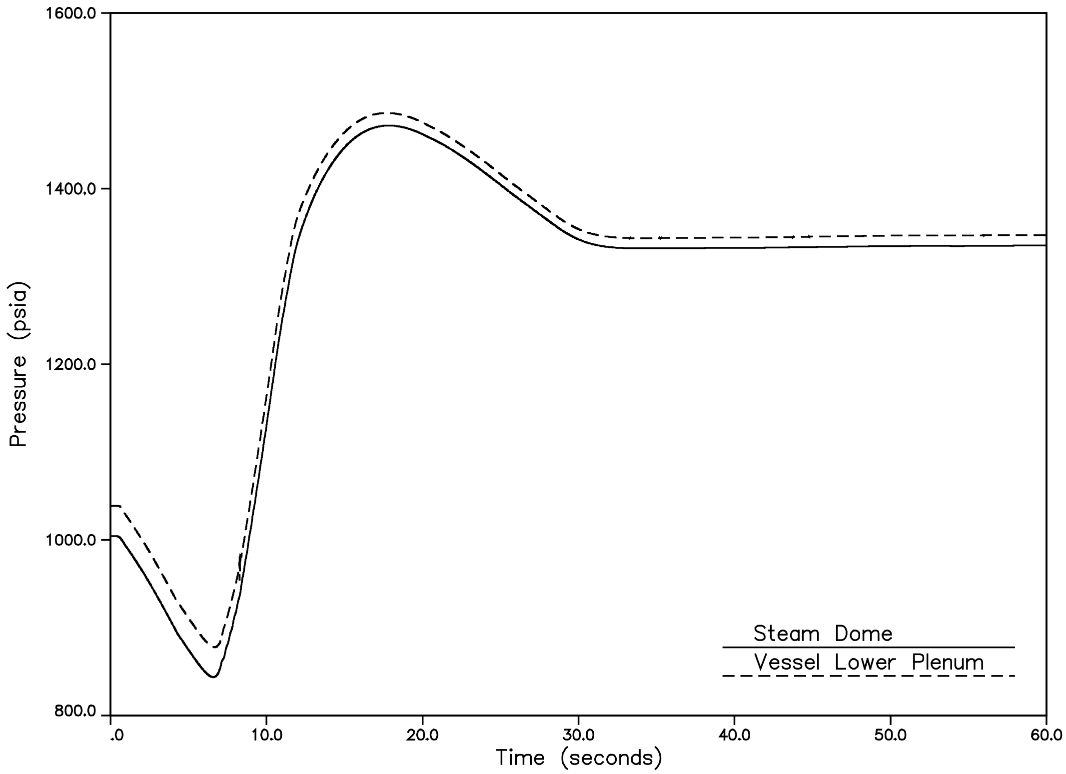


Figure 7.7 PRFO ATWS Overpressurization Event at 100P/95.3F – Vessel Pressures*

* The pressures presented in this figure do not include the adjustments associated with the NRC concerns discussed in Section 7.2.

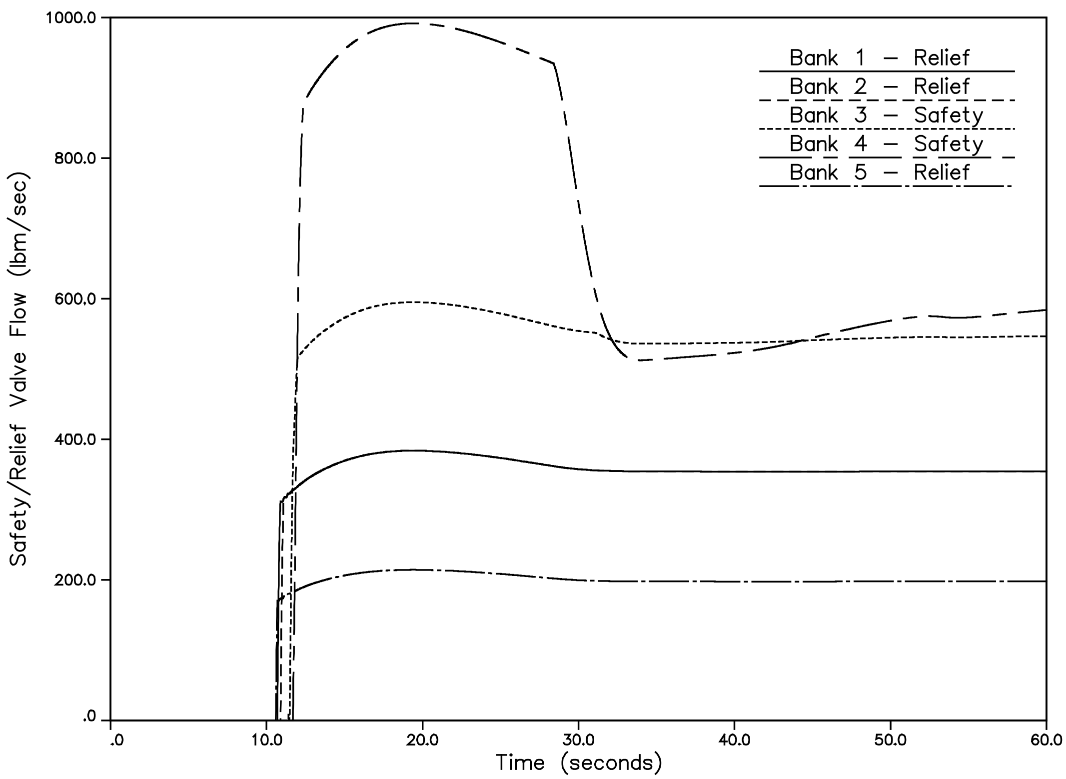


Figure 7.8 PRFO ATWS Overpressurization Event at 100P/95.3F – Safety/Relief Valve Flow Rates

8.0 Operating Limits and COLR Input

The thermal limits for the Quad Cities Unit 2 Cycle 24 representative core design are presented in this section. Limits are provided to support base case two loop operation, base case single loop operation and several EOOS scenarios. Several of the operating limits are applicable for multiple EOOS conditions. Table 8.1 identifies the appropriate operating limits and multipliers (both MCPR and LHGRFAC) that should be applied for the various EOOS conditions. Table 8.1 also presents any applicable power and core flow restrictions for the EOOS scenarios. The base case and EOOS limits and multipliers support operation with 8 of the 9 turbine bypass operational (i.e., 1 bypass valve out-of-service) with the exception of the TBVOOS condition in which all bypass valves are inoperable.

8.1 *MCPR Limits*

The determination of the MCPR limits for the Quad Cities Unit 2 Cycle 24 representative core design is based on the analyses of the limiting anticipated operational occurrences (AOOs). The MCPR operating limits are established so that less than 0.1% 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 Technical Specifications TLO SLMCPR of 1.12 and SLO SLMCPR of 1.14. Exposure-dependent MCPR limits were established to support operation from BOC to NEOC (core average exposure of 34,702 MWd/MTU) and NEOC to EOCLB (core average exposure of 36,774 MWd/MTU) as defined by the core average exposures listed in Table 5.1. The limits for the later exposure range can be used earlier in the cycle as they are the same or more conservative.

Two-loop operation $MCPR_p$ limits for ATRIUM 10XM and OPTIMA2 fuel are presented in Tables 8.2 – 8.13 for base case operation and the EOOS conditions. Limits are presented for NSS, ISS and TSSS insertion times for the exposure ranges considered. Tables 8.2 – 8.7 present the exposure-dependent $MCPR_p$ limits for the ATRIUM 10XM fuel for all scram insertion times. Tables 8.8 – 8.13 present the exposure-dependent $MCPR_p$ limits for the OPTIMA2 fuel for all scram insertion times. All the MCPR limits support operation with any combination of 1 SRVOOS, up to 40% of the TIP channels out-of-service, and up to 50% of the LPRMS out-of-service.

$MCPR_p$ limits for SLO are presented in Tables 8.14 and 8.15. The SLO $MCPR_p$ limits are established using the TLO AOO results, a SLMCPR of 1.14 and the results of the SLO pump seizure analysis. In most cases, the pump seizure analysis results are limiting for power levels between 38.5 and 50% of rated. Single loop operation is limited to operation less than 50% rated power and less than 51% core flow (50 Mlbm/hr). The core monitoring system may require inputs below P_{bypass} for core flows above and below 60% of rated even though SLO is limited to core flows below 51% of rated. The below P_{bypass} SLO $MCPR_p$ limits presented in Tables 8.14 and 8.15 can be used for the high and low flow values in the core monitoring system input.

$MCPR_f$ limits that protect against fuel failures during a postulated slow flow excursion are presented in Table 8.16 and are applicable for all Cycle 24 exposures.

The $MCPR_p$ and $MCPR_f$ limits presented in Tables 8.2 through 8.16 are operating limit MCPR values. The margin to the OLMCPR is determined using the limiting or highest MCPR from the applicable $MCPR_p$ or $MCPR_f$ limits for the given power/flow statepoint.

If there is a need to input $MCPR_p$ limits in the core monitoring system for power levels above 100% of rated, the rated power $MCPR_p$ limit can be used as it would be bounding.

8.2 *LHGR Limits*

The LHGR limits for ATRIUM 10XM fuel are presented in Table 8.17. The LHGR limits for OPTIMA2 fuel presented in Reference 28 remain applicable for transition cores. The power- and flow-dependent multipliers ($LHGRFAC_p$ and $LHGRFAC_f$) 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 (Reference 27). 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 EOOS operation from BOC to NEOC (core average exposure of 34,702 MWd/MTU) and from NEOC to EOCLB (core average exposure of 36,774 MWd/MTU) for NSS, ISS and TSSS insertion times. The ATRIUM 10XM and OPTIMA2 $LHGRFAC_p$ multipliers are presented in Tables 8.18 through 8.24. The LHGR multipliers for the later exposure range can be used earlier in the cycle as they are the same or more conservative.

$LHGRFAC_f$ 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 in Tables 8.25 and 8.26 respectively. They are applicable for all Cycle 24 exposures and the EOOS conditions identified in Table 8.1.

The $LHGRFAC_p$ and $LHGRFAC_f$ multipliers presented in Tables 8.18 through 8.26 are applied to the ATRIUM 10XM and OPTIMA2 LHGR limits. 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 statepoint 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% of rated, the rated power $LHGRFAC_p$ multiplier can be used.

8.3 *MAPLHGR Limits*

The ATRIUM 10XM TLO MAPLHGR limits are presented in Tables 8.27 and 8.28. For operation in SLO, a multiplier of 0.8 must be applied to the TLO MAPLHGR limits.

The OPTIMA2 MAPLHGR limits (including the SLO MAPLHGR multiplier) are presented in Reference 28 and remain applicable for transition cores.

Table 8.1 Applicable Operating Limits and Multipliers for EOOS Conditions*

EOOS Condition	Applicable $MCPR_p$ Limits and $LHGRFAC_p$ Multipliers	Applicable $MCPR_f$ Limits	Power/Flow restrictions
Base TLO	Base TLO	Base	
Base SLO	Base SLO	Base	<50%P, <51%F
FHOOS	FHOOS	Base	
PLUOOS TLO	TCV Slow TLO	Base	
PLUOOS SLO	TCV Slow SLO	Base	<50%P, <51%F
PLUOOS TLO FHOOS	TCV Slow TLO and FHOOS	Base	
PLUOOS and 1 stuck closed TCV/TSV TLO	TCV Slow TLO	1 stuck closed TCV/TSV	<75%P
TBVOOS TLO	TBVOOS TLO	TBVOOS	
TBVOOS SLO	TBVOOS SLO	TBVOOS	<50%P, <51%F
TBVOOS TLO FHOOS	TBVOOS and FHOOS	TBVOOS	
TCV Slow TLO	TCV Slow TLO	Base	
TCV Slow SLO	TCV Slow SLO	Base	<50%P, <51%F
TCV Slow TLO FHOOS	TCV Slow TLO and FHOOS	Base	
1 stuck closed TCV/TSV TLO	Base TLO	1 stuck closed TCV/TSV	<75%P
1 stuck closed TCV/TSV SLO	Base SLO	1 stuck closed TCV/TSV	<50%P, <51%F
1 stuck closed TCV/TSV TLO FHOOS	FHOOS	1 stuck closed TCV/TSV	<75%P
PCOOS TLO	TCV Slow TLO	Base	
PCOOS SLO	TCV Slow SLO	Base	<50%P, <51%F
PCOOS TLO FHOOS	TCV Slow TLO and FHOOS	Base	
PCOOS and PLUOOS TLO	TCV Slow TLO	Base	
PCOOS and 1 stuck closed TCV/TSV TLO	TCV Slow TLO	1 stuck closed TCV/TSV	<75%P
One MSIVOOS TLO	Base TLO	One MSIVOOS	<75%P
One MSIVOOS SLO	Base SLO	One MSIVOOS	<50%P, <51%F
One MSIVOOS TLO and FHOOS	FHOOS	One MSIVOOS	<75%P

* $LHGRFAC_f$ multipliers presented in Tables 8.25 and 8.26 for ATRIUM 10XM and OPTIMA2 fuel are applicable for all base case and EOOS conditions.

Table 8.2 ATRIUM 10XM TLO $MCPR_p$ Limits for NSS Insertion Times BOC to 34,702 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.54	2.54	2.36	1.86		1.44
	> 60	2.78	2.78	2.36			
TBVOOS	≤ 60	3.41	3.41	2.54	1.94		1.47
	> 60	3.51	3.51	2.73			
TCV Slow Closure	≤ 60	2.54	2.54	2.36	2.33	1.98	1.44
	> 60	2.78	2.78	2.36			
FHOOS	≤ 60	2.75	2.75	2.43	1.97		1.44
	> 60	2.78	2.78	2.43			
TBVOOS and FHOOS	≤ 60	3.55	3.55	2.65	2.01		1.47
	> 60	3.71	3.71	2.81			
TCV Slow Closure and FHOOS	≤ 60	2.75	2.75	2.43	2.33	1.98	1.44
	> 60	2.78	2.78	2.43			

Table 8.3 ATRIUM 10XM TLO $MCPR_p$ Limits for ISS Insertion Times BOC to 34,702 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.54	2.54	2.36	1.86		1.44
	> 60	2.78	2.78	2.36			
TBVOOS	≤ 60	3.41	3.41	2.54	1.94		1.47
	> 60	3.51	3.51	2.73			
TCV Slow Closure	≤ 60	2.54	2.54	2.36	2.34	1.98	1.44
	> 60	2.78	2.78	2.36			
FHOOS	≤ 60	2.75	2.75	2.43	1.97		1.44
	> 60	2.78	2.78	2.43			
TBVOOS and FHOOS	≤ 60	3.55	3.55	2.65	2.02		1.47
	> 60	3.71	3.71	2.81			
TCV Slow Closure and FHOOS	≤ 60	2.75	2.75	2.43	2.34	1.98	1.44
	> 60	2.78	2.78	2.43			

Table 8.4 ATRIUM 10XM TLO MCPR$_p$ Limits for TSSS Insertion Times BOC to 34,702 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.54	2.54	2.36	1.88		1.44
	> 60	2.78	2.78	2.36			
TBVOOS	≤ 60	3.41	3.41	2.54	1.94		1.47
	> 60	3.51	3.51	2.73			
TCV Slow Closure	≤ 60	2.54	2.54	2.36	2.35	2.00	1.47
	> 60	2.78	2.78	2.36			
FHOOS	≤ 60	2.75	2.75	2.43	2.02		1.44
	> 60	2.78	2.78	2.43			
TBVOOS and FHOOS	≤ 60	3.55	3.55	2.65	2.04		1.48
	> 60	3.71	3.71	2.81			
TCV Slow Closure and FHOOS	≤ 60	2.75	2.75	2.43	2.35	2.00	1.47
	> 60	2.78	2.78	2.43			

Table 8.5 ATRIUM 10XM TLO MCPR$_p$ Limits for NSS Insertion Times 34,702 to 36,774 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.54	2.54	2.36	1.86		1.44
	> 60	2.78	2.78	2.36			
TBVOOS	≤ 60	3.41	3.41	2.54	1.94		1.47
	> 60	3.51	3.51	2.73			
TCV Slow Closure	≤ 60	2.54	2.54	2.36	2.33	1.98	1.46
	> 60	2.78	2.78	2.36			
FHOOS	≤ 60	2.75	2.75	2.43	1.97		1.44
	> 60	2.78	2.78	2.43			
TBVOOS and FHOOS	≤ 60	3.55	3.55	2.65	2.01		1.47
	> 60	3.71	3.71	2.81			
TCV Slow Closure and FHOOS	≤ 60	2.75	2.75	2.43	2.33	1.98	1.46
	> 60	2.78	2.78	2.43			

Table 8.6 ATRIUM 10XM TLO $MCPR_p$ Limits for ISS Insertion Times 34,702 to 36,774 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.54	2.54	2.36	1.86		1.44
	> 60	2.78	2.78	2.36			
TBVOOS	≤ 60	3.41	3.41	2.54	1.94		1.47
	> 60	3.51	3.51	2.73			
TCV Slow Closure	≤ 60	2.54	2.54	2.36	2.34	1.98	1.46
	> 60	2.78	2.78	2.36			
FHOOS	≤ 60	2.75	2.75	2.43	1.97		1.44
	> 60	2.78	2.78	2.43			
TBVOOS and FHOOS	≤ 60	3.55	3.55	2.65	2.02		1.48
	> 60	3.71	3.71	2.81			
TCV Slow Closure and FHOOS	≤ 60	2.75	2.75	2.43	2.34	1.98	1.46
	> 60	2.78	2.78	2.43			

Table 8.7 ATRIUM 10XM TLO $MCPR_p$ Limits for TSSS Insertion Times 34,702 to 36,774 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.54	2.54	2.36	1.88		1.45
	> 60	2.78	2.78	2.36			
TBVOOS	≤ 60	3.41	3.41	2.54	1.94		1.49
	> 60	3.51	3.51	2.73			
TCV Slow Closure	≤ 60	2.54	2.54	2.36	2.35	2.00	1.49
	> 60	2.78	2.78	2.36			
FHOOS	≤ 60	2.75	2.75	2.43	2.02		1.45
	> 60	2.78	2.78	2.43			
TBVOOS and FHOOS	≤ 60	3.55	3.55	2.65	2.04		1.50
	> 60	3.71	3.71	2.81			
TCV Slow Closure and FHOOS	≤ 60	2.75	2.75	2.43	2.35	2.00	1.49
	> 60	2.78	2.78	2.43			

Table 8.8 OPTIMA2 TLO MCPR$_p$ Limits for NSS Insertion Times
BOC to 34,702 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.45	2.45	2.22	1.91		1.44
	> 60	2.70	2.70	2.32			
TBVOOS	≤ 60	3.28	3.28	2.45	1.97		1.47
	> 60	3.59	3.59	2.77			
TCV Slow Closure	≤ 60	2.45	2.45	2.30	2.30	1.97	1.46
	> 60	2.70	2.70	2.32			
FHOOS	≤ 60	2.65	2.65	2.32	2.03		1.44
	> 60	2.73	2.73	2.40			
TBVOOS and FHOOS	≤ 60	3.41	3.41	2.52	2.07		1.47
	> 60	3.70	3.70	2.87			
TCV Slow Closure and FHOOS	≤ 60	2.65	2.65	2.32	2.30	1.97	1.46
	> 60	2.73	2.73	2.40			

Table 8.9 OPTIMA2 TLO MCPR$_p$ Limits for ISS Insertion Times
BOC to 34,702 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.45	2.45	2.22	1.91		1.44
	> 60	2.70	2.70	2.32			
TBVOOS	≤ 60	3.28	3.28	2.45	1.97		1.47
	> 60	3.59	3.59	2.77			
TCV Slow Closure	≤ 60	2.45	2.45	2.31	2.31	1.98	1.47
	> 60	2.70	2.70	2.32			
FHOOS	≤ 60	2.65	2.65	2.32	2.03		1.44
	> 60	2.73	2.73	2.40			
TBVOOS and FHOOS	≤ 60	3.41	3.41	2.52	2.08		1.47
	> 60	3.70	3.70	2.87			
TCV Slow Closure and FHOOS	≤ 60	2.65	2.65	2.32	2.31	1.98	1.47
	> 60	2.73	2.73	2.40			

Table 8.10 OPTIMA2 TLO MCPR$_p$ Limits for TSSS Insertion Times BOC to 34,702 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.45	2.45	2.22	1.91		1.46
	> 60	2.70	2.70	2.32			
TBVOOS	≤ 60	3.28	3.28	2.45	2.00		1.50
	> 60	3.59	3.59	2.77			
TCV Slow Closure	≤ 60	2.45	2.45	2.32	2.32	2.01	1.50
	> 60	2.70	2.70	2.32			
FHOOS	≤ 60	2.65	2.65	2.32	2.06		1.46
	> 60	2.73	2.73	2.40			
TBVOOS and FHOOS	≤ 60	3.41	3.41	2.52	2.12		1.51
	> 60	3.70	3.70	2.87			
TCV Slow Closure and FHOOS	≤ 60	2.65	2.65	2.32	2.32	2.01	1.50
	> 60	2.73	2.73	2.40			

Table 8.11 OPTIMA2 TLO MCPR$_p$ Limits for NSS Insertion Times 34,702 to 36,774 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.45	2.45	2.22	1.91		1.46
	> 60	2.70	2.70	2.32			
TBVOOS	≤ 60	3.28	3.28	2.45	1.97		1.50
	> 60	3.59	3.59	2.77			
TCV Slow Closure	≤ 60	2.45	2.45	2.30	2.30	1.98	1.50
	> 60	2.70	2.70	2.32			
FHOOS	≤ 60	2.65	2.65	2.32	2.03		1.46
	> 60	2.73	2.73	2.40			
TBVOOS and FHOOS	≤ 60	3.41	3.41	2.52	2.07		1.51
	> 60	3.70	3.70	2.87			
TCV Slow Closure and FHOOS	≤ 60	2.65	2.65	2.32	2.30	1.98	1.50
	> 60	2.73	2.73	2.40			

Table 8.12 OPTIMA2 TLO MCPR$_p$ Limits for ISS Insertion Times
34,702 to 36,774 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.45	2.45	2.22	1.91		1.47
	> 60	2.70	2.70	2.32			
TBVOOS	≤ 60	3.28	3.28	2.45	1.97		1.50
	> 60	3.59	3.59	2.77			
TCV Slow Closure	≤ 60	2.45	2.45	2.31	2.31	1.99	1.50
	> 60	2.70	2.70	2.32			
FHOOS	≤ 60	2.65	2.65	2.32	2.03		1.47
	> 60	2.73	2.73	2.40			
TBVOOS and FHOOS	≤ 60	3.41	3.41	2.52	2.08		1.51
	> 60	3.70	3.70	2.87			
TCV Slow Closure and FHOOS	≤ 60	2.65	2.65	2.32	2.31	1.99	1.50
	> 60	2.73	2.73	2.40			

Table 8.13 OPTIMA2 TLO MCPR$_p$ Limits for TSSS Insertion Times
34,702 to 36,774 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)					
		0	25	≤ 38.5	>38.5	70	100
Base	≤ 60	2.45	2.45	2.22	1.91		1.49
	> 60	2.70	2.70	2.32			
TBVOOS	≤ 60	3.28	3.28	2.45	2.00		1.52
	> 60	3.59	3.59	2.77			
TCV Slow Closure	≤ 60	2.45	2.45	2.32	2.32	2.03	1.52
	> 60	2.70	2.70	2.32			
FHOOS	≤ 60	2.65	2.65	2.32	2.06		1.49
	> 60	2.73	2.73	2.40			
TBVOOS and FHOOS	≤ 60	3.41	3.41	2.52	2.12		1.53
	> 60	3.70	3.70	2.87			
TCV Slow Closure and FHOOS	≤ 60	2.65	2.65	2.32	2.32	2.03	1.52
	> 60	2.73	2.73	2.40			

Table 8.14 ATRIUM 10XM SLO MCPR$_p$ Limits All Exposures

EOOS Condition	Scram Speed	Core Power (% rated)				
		0	25	≤ 38.5	>38.5	50
SLO Base	all	2.56	2.56	2.38	2.07	2.07
SLO and TBVOOS	all	3.43	3.43	2.56	2.07	2.07
SLO and TCV Slow Closure	NSS	2.56	2.56	2.38	2.35	2.23
	ISS	2.56	2.56	2.38	2.36	2.23
	TSS	2.56	2.56	2.38	2.37	2.25

Table 8.15 OPTIMA2 SLO MCPR$_p$ Limits All Exposures

EOOS Condition	Scram Speed	Core Power (% rated)				
		0	25	≤ 38.5	>38.5	50
SLO Base	all	2.47	2.47	2.24	2.15	2.15
SLO and TBVOOS	all	3.30	3.30	2.47	2.15	2.15
SLO and TCV Slow Closure	NSS	2.47	2.47	2.32	2.32	2.21
	ISS	2.47	2.47	2.33	2.33	2.22
	TSSS	2.47	2.47	2.34	2.34	2.24

Table 8.16 Flow-Dependent MCPR Limits
ATRIUM 10XM and OPTIMA2 Fuel

Core Flow (% of rated)	ATRIUM 10XM	OPTIMA2
Supports base case, FHOOS, PCOOS, PLUOOS, TCV Slow Closure, PLUOOS and PCOOS in TLO and SLO		
0	1.70	1.70
35	1.70	1.70
108	1.19	1.19
Supports any scenario with One MSIVOOS		
0	1.81	1.81
35	1.81	1.81
108	1.19	1.19
Supports any scenario with TBVOOS		
0	1.91	1.91
35	1.91	1.91
108	1.35	1.35
Supports any scenario with 1 stuck closed TCV/TSV		
0	2.85	2.85
35	2.85	2.85
108	1.54	1.54

Table 8.17 Steady-State LHGR Limits

Peak Pellet Exposure (GWd/MTU)	ATRIUM 10XM LHGR (kW/ft)
0.0	14.1
18.9	14.1
74.4	7.4

Table 8.18 ATRIUM 10XM LHGRFAC$_p$ Multipliers for All Scram Insertion Times, All Exposures

<table>
<tr><th rowspan="2">EOOS Condition</th><th rowspan="2">Core Flow (% rated)</th><th colspan="7">Core Power (% rated)</th></tr>
<tr><th>0</th><th>25</th><th>≤ 38.5</th><th>>38.5</th><th>≤ 60</th><th>>60</th><th>100</th></tr>
<tr><td rowspan="2">Base</td><td>≤ 60</td><td>0.44</td><td>0.44</td><td>0.51</td><td rowspan="2">0.53</td><td rowspan="2">0.73</td><td rowspan="2">0.85</td><td rowspan="2">0.98</td></tr>
<tr><td>> 60</td><td>0.44</td><td>0.44</td><td>0.51</td></tr>
<tr><td rowspan="2">TBVOOS</td><td>≤ 60</td><td>0.37</td><td>0.37</td><td>0.48</td><td rowspan="2">0.53</td><td rowspan="2">0.73</td><td rowspan="2">0.85</td><td rowspan="2">0.97</td></tr>
<tr><td>> 60</td><td>0.37</td><td>0.37</td><td>0.48</td></tr>
<tr><td rowspan="2">TCV Slow Closure</td><td>≤ 60</td><td>0.44</td><td>0.44</td><td>0.51</td><td rowspan="2">0.53</td><td rowspan="2">0.73</td><td rowspan="2">0.85</td><td rowspan="2">0.98</td></tr>
<tr><td>> 60</td><td>0.44</td><td>0.44</td><td>0.51</td></tr>
<tr><td rowspan="2">FHOOS</td><td>≤ 60</td><td>0.39</td><td>0.39</td><td>0.48</td><td rowspan="2">0.53</td><td rowspan="2">0.73</td><td rowspan="2">0.83</td><td rowspan="2">0.98</td></tr>
<tr><td>> 60</td><td>0.39</td><td>0.39</td><td>0.48</td></tr>
<tr><td rowspan="2">TBVOOS and FHOOS</td><td>≤ 60</td><td>0.34</td><td>0.34</td><td>0.45</td><td rowspan="2">0.53</td><td rowspan="2">0.73</td><td rowspan="2">0.82</td><td rowspan="2">0.97</td></tr>
<tr><td>> 60</td><td>0.34</td><td>0.34</td><td>0.45</td></tr>
<tr><td rowspan="2">TCV Slow Closure and FHOOS</td><td>≤ 60</td><td>0.39</td><td>0.39</td><td>0.48</td><td rowspan="2">0.53</td><td rowspan="2">0.73</td><td rowspan="2">0.83</td><td rowspan="2">0.98</td></tr>
<tr><td>> 60</td><td>0.39</td><td>0.39</td><td>0.48</td></tr>
</table>

Table 8.19 OPTIMA2 $LHGRFAC_p$ Multipliers for NSS Insertion Times
BOC to 34,702 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)									
		0	25	≤38.5	>38.5	50	70	80	<90	≥90	100
Base	≤ 60	0.45	0.45	0.50	0.54	0.63	0.78	0.85	0.85	0.93	0.95
	> 60	0.44	0.44	0.50							
TBVOOS	≤ 60	0.35	0.35	0.47	0.54	0.63	0.76	0.80	0.83	0.83	0.84
	> 60	0.35	0.35	0.43							
TCV Slow Closure	≤ 60	0.45	0.45	0.50	0.54	0.62	0.66	0.85	0.85	0.93	0.95
	> 60	0.44	0.44	0.50							
FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.63	0.71	0.80	0.82	0.82	0.85
	> 60	0.40	0.40	0.48							
TBVOOS and FHOOS	≤ 60	0.33	0.33	0.44	0.54	0.63	0.71	0.78	0.80	0.80	0.83
	> 60	0.33	0.33	0.42							
TCV Slow Closure and FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.62	0.66	0.80	0.82	0.82	0.85
	> 60	0.40	0.40	0.48							

Table 8.20 OPTIMA2 $LHGRFAC_p$ Multipliers for ISS Insertion Times
BOC to 34,702 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)									
		0	25	≤38.5	>38.5	50	70	80	<90	≥90	100
Base	≤ 60	0.45	0.45	0.50	0.54	0.63	0.77	0.85	0.85	0.93	0.95
	> 60	0.44	0.44	0.50							
TBVOOS	≤ 60	0.35	0.35	0.47	0.54	0.63	0.76	0.80	0.82	0.82	0.83
	> 60	0.35	0.35	0.43							
TCV Slow Closure	≤ 60	0.45	0.45	0.50	0.54	0.62	0.66	0.85	0.85	0.93	0.95
	> 60	0.44	0.44	0.50							
FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.63	0.70	0.80	0.82	0.82	0.85
	> 60	0.40	0.40	0.48							
TBVOOS and FHOOS	≤ 60	0.33	0.33	0.44	0.54	0.63	0.70	0.78	0.80	0.80	0.83
	> 60	0.33	0.33	0.42							
TCV Slow Closure and FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.62	0.66	0.80	0.82	0.82	0.85
	> 60	0.40	0.40	0.48							

Table 8.21 OPTIMA2 LHGRFAC$_p$ Multipliers for TSSS Insertion Times
BOC to 34,702 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)									
		0	25	≤38.5	>38.5	50	70	80	<90	≥90	100
Base	≤ 60	0.45	0.45	0.50	0.54	0.63	0.74	0.83	0.83	0.93	0.95
	> 60	0.44	0.44	0.50							
TBVOOS	≤ 60	0.35	0.35	0.47	0.54	0.63	0.73	0.80	0.81	0.81	0.83
	> 60	0.35	0.35	0.43							
TCV Slow Closure	≤ 60	0.45	0.45	0.50	0.54	0.62	0.65	0.83	0.83	0.93	0.94
	> 60	0.44	0.44	0.50							
FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.62	0.67	0.79	0.82	0.82	0.85
	> 60	0.40	0.40	0.48							
TBVOOS and FHOOS	≤ 60	0.33	0.33	0.44	0.54	0.62	0.67	0.76	0.79	0.79	0.83
	> 60	0.33	0.33	0.42							
TCV Slow Closure and FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.62	0.65	0.79	0.82	0.82	0.85
	> 60	0.40	0.40	0.48							

Table 8.22 OPTIMA2 LHGRFAC$_p$ Multipliers for NSS Insertion Times
34,702 to 36,774 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)										
		0	25	≤38.5	>38.5	50	70	80	90	<92	≥92	100
Base	≤ 60	0.45	0.45	0.50	0.54	0.63	0.77	0.82	0.85	0.85	0.93	0.94
	> 60	0.44	0.44	0.50								
TBVOOS	≤ 60	0.35	0.35	0.47	0.54	0.63	0.74	0.79	0.81			0.84
	> 60	0.35	0.35	0.43								
TCV Slow Closure	≤ 60	0.45	0.45	0.50	0.54	0.62	0.66	0.82	0.85	0.85	0.93	0.93
	> 60	0.44	0.44	0.50								
FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.63	0.70	0.80	0.82			0.85
	> 60	0.40	0.40	0.48								
TBVOOS and FHOOS	≤ 60	0.33	0.33	0.44	0.54	0.63	0.70	0.77	0.80			0.83
	> 60	0.33	0.33	0.42								
TCV Slow Closure and FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.62	0.66	0.80	0.82			0.85
	> 60	0.40	0.40	0.48								

Table 8.23 OPTIMA2 LHGRFAC$_p$ Multipliers for ISS Insertion Times 34,702 to 36,774 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)										
		0	25	≤38.5	>38.5	50	70	80	90	<93	≥93	100
Base	≤ 60	0.45	0.45	0.50	0.54	0.63	0.77	0.82	0.85	0.85	0.92	0.93
	> 60	0.44	0.44	0.50								
TBVOOS	≤ 60	0.35	0.35	0.47	0.54	0.63	0.74	0.79	0.81			0.83
	> 60	0.35	0.35	0.43								
TCV Slow Closure	≤ 60	0.45	0.45	0.50	0.54	0.62	0.66	0.82	0.85	0.85	0.92	0.93
	> 60	0.44	0.44	0.50								
FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.63	0.70	0.79	0.82			0.85
	> 60	0.40	0.40	0.48								
TBVOOS and FHOOS	≤ 60	0.33	0.33	0.44	0.54	0.63	0.70	0.77	0.80			0.83
	> 60	0.33	0.33	0.42								
TCV Slow Closure and FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.62	0.66	0.79	0.82			0.85
	> 60	0.40	0.40	0.48								

Table 8.24 OPTIMA2 LHGRFAC$_p$ Multipliers for TSSS Insertion Times 34,702 to 36,774 MWd/MTU Core Average Exposure

EOOS Condition	Core Flow (% rated)	Core Power (% rated)										
		0	25	≤38.5	>38.5	50	70	80	90	<97	≥97	100
Base	≤ 60	0.45	0.45	0.50	0.54	0.63	0.74	0.81	0.83	0.85	0.92	0.92
	> 60	0.44	0.44	0.50								
TBVOOS	≤ 60	0.35	0.35	0.47	0.54	0.63	0.73	0.78	0.80			0.82
	> 60	0.35	0.35	0.43								
TCV Slow Closure	≤ 60	0.45	0.45	0.50	0.54	0.62	0.64	0.81	0.83	0.85	0.92	0.92
	> 60	0.44	0.44	0.50								
FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.62	0.67	0.79	0.82			0.85
	> 60	0.40	0.40	0.48								
TBVOOS and FHOOS	≤ 60	0.33	0.33	0.44	0.54	0.62	0.67	0.76	0.79			0.82
	> 60	0.33	0.33	0.42								
TCV Slow Closure and FHOOS	≤ 60	0.40	0.40	0.48	0.54	0.62	0.64	0.79	0.82			0.85
	> 60	0.40	0.40	0.48								

Table 8.25 ATRIUM 10XM
$LHGRFAC_f$ Multipliers
All Cycle 24 Exposures, All EOOS

Core Flow (% of rated)	$LHGRFAC_f$
0.0	0.60
35.0	0.60
75.0	1.00
108.0	1.00

Table 8.26 OPTIMA2
$LHGRFAC_f$ Multipliers
All Cycle 24 Exposures, All EOOS

Core Flow (% of rated)	$LHGRFAC_f$
0.0	0.45
35.0	0.45
90.0	1.00
108.0	1.00

Table 8.27 AREVA Fuel
MAPLHGR Limits – Bottom Lattices

Average Planar Exposure (GWd/MTU)	ATRIUM 10XM MAPLHGR (kW/ft)
0.0	11.7
20.0	11.7
67.0	6.6
Limits for the following Cycle 24 lattices:	

Table 8.28 AREVA Fuel
MAPLHGR Limits – Top Lattices

Average Planar Exposure (GWd/MTU)	ATRIUM 10XM MAPLHGR (kW/ft)
0.0	11.7
15.0	11.7
67.0	6.6
Limits for the following Cycle 24 lattices:	

9.0 References

1. ANP-3293P Revision 0, *Quad Cities Unit 2 Cycle 24 Representative Fuel Cycle Design,* AREVA, October 2014.

2. ANP-3338P Revision 0, *Applicability of AREVA BWR Methods to the Dresden and Quad Cities Reactors Operating at Extended Power Uprate*, AREVA, November 2014.

3. ANP-3305P Revision 0, *Mechanical Design Report for Quad Cities and Dresden ATRIUM 10XM Fuel Assemblies*, AREVA, June 2014.

4. ANP-3324P Revision 0, *ATRIUM 10XM Fuel Rod Thermal-Mechanical Design for Quad Cities Unit 2 Cycle 24 Representative Fuel Cycle Design*, AREVA, October 2014.

5. ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs,* Advanced Nuclear Fuels Corporation, May 1995.

6. EMF-93-177(P)(A) Revision 1, *Mechanical Design for BWR Fuel Channels*, Framatome ANP, August 2005.

7. ANP-3287P Revision 1, *Quad Cities Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM™ 10XM Fuel Assemblies,* AREVA, November 2014.

8. ANP-10307PA Revision 0, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,* AREVA NP, June 2011.

9. ANP-10298P-A Revision 1, *ACE/ATRIUM 10XM Critical Power Correlation*, AREVA, March 2014.

10. EMF-2209(P)(A) Revision 3, *SPCB Critical Power Correlation*, AREVA NP, September 2009.

11. EMF-2245(P)(A) Revision 0, *Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel*, Siemens Power Corporation, August 2000.

12. NEDO-32465-A, *Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications,* GE Nuclear Energy, August 1996.

13. BAW-10255PA Revision 2, *Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code*, AREVA NP, May 2008.

14. OG02-0119-260, "Backup Stability Protection (BSP) for Inoperable Option III Solution," GE Nuclear Energy, July 17, 2002.

15. EMF-CC-074(P)(A) Volume 4 Revision 0, *BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2,* Siemens Power Corporation, August 2000.

16. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses*, Advanced Nuclear Fuels Corporation, August 1990.

17. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis,* Exxon Nuclear Company, February 1987.

18. 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, January 1987.

19. 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, October 1999.

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

21. *Technical Specifications,* Quad Cities Nuclear Power Station Unit 1 and 2.

22. ANP-3328P Revision 0, *Quad Cities Units 1 and 2 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel,* AREVA, December 2014.

23. ANP-3356P Revision 0, *Quad Cities Units 1 and 2 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel*, AREVA, December 2014.

24. *Updated Final Safety Analysis Report,* Quad Cities Nuclear Power Station Unit 1 and 2.

25. 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, March 1983.

26. XN-NF-80-19(P)(A) Volume 4 Revision 1, *Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads*, Exxon Nuclear Company, June 1986.

27. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, AREVA NP, February 2008.

28. Letter, S Darin (Exelon) to US Nuclear Regulatory Commission, “Core Operating Limits Report for Quad Cities Unit 2 Cycle 23,” SVP-14-034, May 2, 2014 (NRC ascension number ML14126A600).

29. Exelon TODI NF140642, “AREVA Rod Drop Accident Sequence Information,” October 2014 (38-2201261-000).

ATTACHMENT 23

LOCA Break Spectrum Analysis Report
(Non-Proprietary)

ANP-3328NP
Revision 0

Quad Cities Units 1 and 2 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel

December 2014

AREVA Inc.

ANP-3328NP
Revision 0

Quad Cities Units 1 and 2
LOCA Break Spectrum Analysis
for ATRIUM 10XM Fuel

AREVA Inc.

ANP-3328NP
Revision 0

Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial issue.

Contents

Tables

Figures

Nomenclature

ADS	automatic depressurization system
ANS	American Nuclear Society
APF	axial peaking factor
BL	broken loop
BOL	beginning of life
BWR	boiling-water reactor
CFR	Code of Federal Regulations
DEG	double-ended guillotine
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOB	end of blowdown
EPU	extended power uprate
HPCI	high-pressure coolant injection
ID	inside diameter
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MSIV	main steam isolation valve
MWR	metal-water reaction
NRC	Nuclear Regulatory Commission, U.S.
OD	outside diameter
PCT	peak cladding temperature
PD	pump discharge
PS	pump suction
RDIV	recirculation discharge isolation valve
RPF	radial peaking factor
SF-ADS	single failure of ADS valve
SF-DGEN	single failure of diesel generator
SF-HPCI	single failure of the HPCI system
SF-LPCI	single failure of an LPCI injection valve
SF-LSL	single failure of the loop selection logic
SLO	single-loop operation
TCV	turbine control valve
TLO	two-loop operation
TSV	turbine step valve
UFSAR	updated safety analysis report
VDC	volts direct current

1.0 Introduction

The results of a loss-of-coolant accident (LOCA) break spectrum analysis at extended power uprate (EPU) conditions for Quad Cities are documented in this report. The purpose of the break spectrum analysis is to identify the parameters that result in the highest calculated peak cladding temperature (PCT) during a postulated LOCA. The LOCA parameters addressed in this report include the following:

- Break location
- Break type (double-ended guillotine (DEG) or split)
- Break size
- Limiting emergency core cooling system (ECCS) single failure
- Axial power shape (top- or mid-peaked)

The analyses documented in this report were performed with LOCA Evaluation Models developed by AREVA Inc. (AREVA) and approved for reactor licensing analyses by the U.S. Nuclear Regulatory Commission (NRC). The models and computer codes used by AREVA for LOCA analyses are collectively referred to as the EXEM BWR-2000 Evaluation Model (References 1 – 4). The EXEM BWR-2000 Evaluation Model and NRC approval are documented in Reference 1. A summary description of the LOCA analysis methodology is provided in Section 4.0. The calculations described in this report were performed in conformance with 10 CFR 50 Appendix K requirements and satisfy the event acceptance criteria identified in 10 CFR 50.46. **[**

]

The break spectrum analyses documented in this report 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% of 2957 MWt, providing a licensing basis power of 3016.14 MWt. The 2.0% increase reflects the maximum uncertainty in monitoring reactor power, as per NRC requirements. The limiting assembly in the core was assumed to be at a maximum average planar linear heat generation rate (MAPLHGR) limit of 11.7 kW/ft. Other initial conditions used in the analyses are described in Section 4.0.

This report identifies the limiting LOCA break characteristics (location, type, size, single failure and axial power shape) that will be used in future analyses to determine the MAPLHGR limit

---

* ATRIUM is a trademark of AREVA Inc.

versus exposure for ATRIUM 10XM fuel contained in Quad Cities. Even though the limiting break will not change with exposure or nuclear fuel design, the value of PCT calculated for any given set of break characteristics is dependent on exposure and local power peaking. Therefore, heatup analyses are performed to determine the PCT versus exposure for each nuclear design in the core. The heatup analyses are performed each cycle using the limiting boundary conditions determined in the break spectrum analysis. The maximum PCT versus exposure from the heatup analyses are documented in the MAPLHGR report.

[

] Limiting reactor power and core flow conditions were selected with consideration for the EPU range of operating conditions. This report also addresses long-term coolability.

The impact on LOCA of operation with equipment out-of-service and their combinations as per the COLR have been considered. The LOCA analyses include the effects of 1 relief valve out-of-service. Operation with an ADS valve out-of-service is not currently allowed by the Quad Cities Technical Specification. Since the consequences of a LOCA would be more severe during operation with one of the recirculation lines out-of-service, this report presents results for single-loop operation (SLO). Operation with other allowed equipment out-of-service conditions are supported (turbine bypass valves, feedwater heaters, one MSIV, power load unbalance, pressure controller, TCV slow closure, and one stuck closed TCV or TSV).

2.0 Summary of Results

Based on analyses presented in this report, the limiting break characteristics are identified below.

Limiting LOCA Break Characteristics	
Location	Recirculation discharge pipe
Type / size	Split break / 0.13 ft^2
Single failure	HPCI
Axial power shape	Top-peaked
Initial state	102% power / [ ]

A more detailed discussion of results is provided in Sections 6.0 – 7.0.

[

] The break characteristics identified in this report can be used in subsequent fuel type specific LOCA heatup analyses to determine the MAPLHGR limit appropriate for the fuel type.

The SLO LOCA analyses support operation with an ATRIUM 10XM MAPLHGR multiplier of 0.80 applied to the normal two-loop operation MAPLHGR limit.

The long-term coolability evaluation confirms that the ECCS capacity is sufficient to maintain adequate cooling in an ATRIUM 10XM core for an extended period after a LOCA.

[

]

While the fuel rod temperatures in the limiting plane of the hot channel during a LOCA are dependent on exposure, the factors that determine the limiting break characteristics are primarily associated with the reactor system and are not dependent on fuel-exposure characteristics. Fuel parameters that are dependent on exposure (e.g., stored energy, local peaking) have an insignificant effect on the reactor system response during a LOCA. The limiting break characteristics are determined using BOL fuel conditions for a representative ATRIUM 10XM lattice design and conservative stored energy. These limiting break conditions are applicable for exposed fuel. Fuel exposure effects are addressed in heatup analyses performed to determine or verify MAPLHGR limits versus exposure for each fuel design.

The break spectrum analysis was performed using the NRC approved AREVA EXEM BWR-2000 LOCA methodology. A modified application approach to **[** **]** is presented in Section 4.4. This modification is conservative relative to the application approach for the approved methodology utilized in Reference 1. The modified application approach was communicated to the NRC in Reference 8. The NRC acknowledged the modified approach in Reference 9.

Differences between the break spectrum results for the AREVA fuel and the co-resident legacy fuel will primarily be the result of differences between the fuel vendor LOCA methodologies.

3.0 LOCA Description

3.1 *Accident Description*

The LOCA is described in the Code of Federal Regulations 10 CFR 50.46 as a hypothetical accident that results in a loss of reactor coolant from breaks in reactor coolant pressure boundary piping up to and including a break equivalent in size to a double-ended rupture of the largest pipe in the reactor coolant system. There is not a specifically identified cause that results in the pipe break. However, for the purpose of identifying a design basis accident, the pipe break is postulated to occur inside the primary containment before the first isolation valve.

For a boiling water reactor (BWR), a LOCA may occur over a wide spectrum of break locations and sizes. Responses to the break vary significantly over the break spectrum. The largest possible break is a double-ended rupture of a recirculation pipe; however, this is not necessarily the most severe challenge to the emergency core cooling system (ECCS). A double-ended rupture of a main steam line causes the most rapid primary system depressurization, but because of other phenomena, steam line breaks are seldom limiting with respect to the event acceptance criteria (10 CFR 50.46). In addition to break location dependence, different break sizes in the same pipe produce quite different event responses, and the largest break area is not necessarily the most severe challenge to the event acceptance criteria. Because of these complexities, an analysis covering the full range of break sizes and locations is performed to identify the limiting break characteristics.

Regardless of the initiating break characteristics, the event response is conveniently separated into three phases: the blowdown phase, the refill phase, and the reflood phase. The relative duration of each phase is strongly dependent upon the break size and location. The last two phases are often combined and will be discussed together in this report.

During the blowdown phase of a LOCA, there is a net loss of coolant inventory, an increase in fuel cladding temperature due to core flow degradation, and for the larger breaks, the core becomes fully or partially uncovered. There is a rapid decrease in pressure during the blowdown phase. During the early phase of the depressurization, the exiting coolant provides core cooling. Later in the blowdown, core cooling is provided by lower plenum flashing as the system continues to depressurize and the injection of ECCS flows. The blowdown phase is defined to end when rated LPCS flow is attained.

In the refill phase of a LOCA, the ECCS is functioning and there is a net increase of coolant inventory. During this phase the core sprays provide core cooling and, along with low-pressure and high-pressure coolant injection (LPCI and HPCI), supply liquid to refill the lower portion of the reactor vessel. In general, the core heat transfer to the coolant is less than the fuel decay heat rate and the fuel cladding temperature continues to increase during the refill phase.

In the reflood phase, the coolant inventory has increased to the point where the mixture level reenters the core region. During the core reflood phase, cooling is provided above the mixture level by entrained reflood liquid and below the mixture level by pool boiling. Sufficient coolant eventually reaches the core hot node and the fuel cladding temperature decreases.

3.2 *Acceptance Criteria*

A LOCA is a potentially limiting event that may place constraints on fuel design, local power peaking, and in some cases, acceptable core power level. During a LOCA, the normal transfer of heat from the fuel to the coolant is disrupted. As the liquid inventory in the reactor decreases, the decay heat and stored energy of the fuel cause a heatup of the undercooled fuel assembly. In order to limit the amount of heat that can contribute to the heatup of the fuel assembly during a LOCA, an operating limit on the MAPLHGR is applied to each fuel assembly in the core.

The Code of Federal Regulations prescribes specific acceptance criteria (10 CFR 50.46) for a LOCA event as well as specific requirements and acceptable features for Evaluation Models (10 CFR 50 Appendix K). The conformance of the EXEM BWR-2000 LOCA Evaluation Models to Appendix K is described in Reference 1. The ECCS must be designed such that the plant response to a LOCA meets the following acceptance criteria specified in 10 CFR 50.46:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- The calculated local oxidation of the cladding shall nowhere exceed 0.17 times the local cladding thickness.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume, were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria are commonly referred to as the peak cladding temperature (PCT) criterion, the local oxidation criterion, the hydrogen generation criterion, the coolable geometry criterion, and the long-term cooling criterion. A MAPLHGR limit versus fuel exposure is established to ensure that these criteria are met. For jet pump BWRs, the most challenging criterion is that PCT must not exceed 2200°F. LOCA PCT results are provided in Sections 6.0 - 7.0 to determine the limiting LOCA event.

LOCA analysis results demonstrating that the PCT, local oxidation, and hydrogen generation criteria are met are provided in follow-on MAPLHGR report and cycle specific heatup analyses performed to determine MAPLHGR limits versus exposure for each fuel design. Cycle-specific heatup analyses are performed to demonstrate that the MAPLHGR limit versus exposure for the ATRIUM 10XM fuel remains applicable for cycle-specific nuclear designs. Compliance with these three criteria ensures that a coolable geometry is maintained. Long-term coolability criterion is discussed in Section 8.0.

4.0 LOCA Analysis Description

The Evaluation Model used for the break spectrum analysis is the EXEM BWR-2000 LOCA analysis methodology described in Reference 1. The EXEM BWR-2000 methodology employs three major computer codes to evaluate the system and fuel response during all phases of a LOCA. These are the RELAX, HUXY, and RODEX2 computer codes. RELAX is used to calculate the system and hot channel response during the blowdown, refill and reflood phases of the LOCA. The HUXY code is used to perform heatup calculations for the entire LOCA, and calculates the PCT and local clad oxidation at the axial plane of interest. RODEX2 is used to determine fuel parameters (such as stored energy) for input to the other LOCA codes. The code interfaces for the LOCA methodology are illustrated in Figure 4.1.

A complete analysis for a given break size starts with the specification of fuel parameters using RODEX2 (Reference 4). RODEX2 is used to determine the initial stored energy for both the blowdown analysis (RELAX hot channel) and the heatup analysis (HUXY). This is accomplished by ensuring that the initial stored energy in RELAX and HUXY is the same or higher than that calculated by RODEX2 for the power, exposure, and fuel design being considered.

4.1 *Blowdown Analysis*

The RELAX code (Reference 1) is used to calculate the system thermal-hydraulic response during the blowdown phase of the LOCA. For the system blowdown analysis, the core is represented by an average core channel. The reactor core is modeled with heat generation rates determined from reactor kinetics equations with reactivity feedback and with decay heating as required by Appendix K of 10 CFR 50. The reactor vessel nodalization for the system analysis is shown in Figure 4.3. This nodalization is consistent with that used in the topical report submitted to the NRC (Reference 1).

The RELAX blowdown analysis is performed from the time of the break initiation through the end of blowdown (EOB). The system blowdown calculation provides the upper and lower plenum transient boundary conditions for the hot channel analysis.

Following the system blowdown calculation, another RELAX analysis is performed to analyze the maximum power assembly (hot channel) of the core. The RELAX hot channel blowdown calculation determines hot channel fuel, cladding, and coolant temperatures during the blowdown phase of the LOCA. The RELAX hot channel nodalization is shown in Figure 4.4 for a top-peaked power shape, and in Figure 4.5 for a mid-peaked axial power shape. The hot channel analysis is performed using the system blowdown results to supply the core power and the system boundary conditions at the core inlet and exit. The initial average fuel rod temperature at the limiting plane of the hot channel is conservative relative to the average fuel rod temperature calculated by RODEX2 for operation of the ATRIUM 10XM assembly at the MAPLHGR limit. The heat transfer coefficients and fluid conditions at the limiting plane of the RELAX hot channel calculation are used as input to the HUXY heatup analysis.

4.2 *Refill/Reflood Analysis*

The RELAX code is also used to compute the system and hot channel hydraulic response during the refill/reflood phase of the LOCA. The RELAX system and RELAX hot channel analyses continue beyond the end of blowdown to analyze system and hot channel responses during the refill and reflood phases. The refill phase is the period when the lower plenum is filling due to ECCS injection. The reflood phase is the period when some portions of the core and hot assembly are being cooled with ECCS water entering from the lower plenum. The purpose of the RELAX calculations beyond blowdown is to determine the time when the liquid flow via upward entrainment from the bottom of the core becomes high enough at the hot node in the hot assembly to end the temperature increase of the fuel rod cladding. This event time is called the time of hot node reflood. **[**

] The time when the core bypass mixture level rises to the elevation of the hot node in the hot assembly is also determined.

RELAX provides a prediction of fluid inventory during the ECCS injection period. Allowing for countercurrent flow through the core and bypass, RELAX determines the refill rate of the lower plenum due to ECCS water and the subsequent reflood times for the core, hot assembly, and the core bypass. The RELAX calculations provide HUXY with the time of hot node reflood and the time when the liquid has risen in the bypass to the height of the axial plane of interest (time of bypass reflood).

4.3 *Heatup Analysis*

The HUXY code (Reference 2) is used to perform heatup calculations for the entire LOCA transient and provides PCT and local clad oxidation at the axial plane of interest. The heat generated by metal-water reaction (MWR) is included in the HUXY analysis. HUXY is used to calculate the thermal response of each fuel rod in one axial plane of the hot channel assembly. These calculations consider thermal-mechanical interactions within the fuel rod. The clad swelling and rupture models from NUREG-0630 have been incorporated into HUXY (Reference 3). The HUXY code complies with the 10 CFR 50 Appendix K criteria for LOCA Evaluation Models.

HUXY uses the EOB time and the times of core bypass reflood and core reflood at the axial plane of interest from the RELAX analysis. Until the EOB, HUXY uses RELAX hot channel heat transfer coefficients, fluid temperatures, fluid qualities, and power. Throughout the calculations, decay power is determined based on the ANS 1971 decay heat curve plus 20% as described in Reference 1. After the EOB and prior to the time of hot node reflood, HUXY uses Appendix K spray heat transfer coefficients for the fuel rods, water channel and fuel channel. Experimental data for AREVA 10X10 fuel which supports the use of the convective heat transfer coefficients listed in Appendix K is documented in Reference 5. After the time of hot node reflood, Appendix K reflood heat transfer coefficients are used in the HUXY analysis. The principal results of a HUXY heatup analysis are the PCT and the percent local oxidation of the fuel cladding, often called the %MWR.

4.4 [ ]

[

]

4.4.1 Calculation Approach

[

]

4.5 Plant Parameters and Initial Conditions

The LOCA break spectrum analysis is performed using plant parameters provided by Exelon. Limiting reactor power and core flow conditions were selected with consideration for the EPU range of operating conditions. The control blades are modeled as inserting with the scram timing required by the Quad Cities Technical Specification. Table 4.1 provides a summary of reactor initial conditions used in the break spectrum analysis. Table 4.2 lists selected reactor system parameters.

AREVA uses a process for determining initial power distributions that produce conservative LOCA results compared to the power distributions that could exist during actual operation. The initial power distributions are based on a conservatively low MCPR operating limit and the MAPLHGR limit to be supported.

The radial peaking factor for the hot bundle is established through calculations that determine the maximum radial that could be achieved without violating a conservatively low MCPR operating limit when the highest planar power is at the MAPLHGR limit. The use of a low MCPR operating limit results in a high bundle power. Since MCPR depends on core flow and axial power shape, a different radial peaking factor is used for each combination of core flow and axial power shape. After the radial peaking is calculated, the axial peaking factor at the peak power plane is calculated to put the nodal power at the MAPLHGR limit. Table 4.1 summarizes the MAPLHGR limit and the MCPR operating limit that were used to establish the radial and axial peaking factors for each of the power/flow conditions that have been analyzed.

The rod average axial power profiles resulting from the application of this process for 102%P [ ] are shown in Figure 4.7. The axial power profiles for the other power and flow conditions are similar.

The break spectrum analysis is performed for a full core of ATRIUM 10XM fuel. Some of the key ATRIUM 10XM fuel parameters used in the break spectrum analysis are summarized in Table 4.3.

4.6 *ECCS Parameters*

The ECCS configuration is shown in Figure 4.6. Table 4.4 – Table 4.7 provide the important ECCS characteristics assumed in the analysis. The ECCS is modeled as fill junctions connected to the appropriate reactor locations: LPCS injects into the upper plenum, HPCI injects into the upper downcomer, and LPCI injects into the recirculation lines.

The flow through each ECCS valve is determined based on system pressure and valve position. Flow versus pressure for a fully open valve is obtained by linearly interpolating the pump capacity data provided in Table 4.4 – Table 4.6. No HPCI flow is credited until the injection valve is fully open. LPCI and LPCS flow is governed by the valve characteristics provided by Exelon. Also, no credit for ECCS flow is assumed until ECCS pumps reach rated speed.

[

]

The ADS valves are modeled as a junction connecting the reactor steam line to the suppression pool. The flow through the ADS valves is calculated based on pressure and valve flow characteristics. The valve flow characteristics are determined such that the calculated flow is equal to the rated capacity at the reference pressure shown in Table 4.7. All five ADS valves are assumed operable during the LOCA except when a single failure is assumed to prevent one ADS valve from opening.

In the AREVA LOCA analysis model, ECCS initiation is assumed to occur when the water level drops to the applicable level setpoint. No credit is assumed for the start of LPCS or LPCI due to high drywell pressure. **[**

]

Recirculation discharge isolation valve (RDIV) and loop selection logic parameters are presented in Table 4.8. For recirculation line breaks sizes ≥ 0.15 ft^2, the loop selection logic directs all available LPCI flow to the intact loop and closes the RDIV in the intact loop. For break sizes < 0.15 ft^2, all available LPCI flow is assumed to be injected into the broken loop and the RDIV in the broken loop is closed.

Table 4.1 Initial Conditions*

Reactor power (% of rated)	102	102
[		]
Reactor power (MWt)	3016.14	3016.14
[		]
[		]
Steam flow rate (Mlb/hr)	11.98	11.98
Steam dome pressure (psia)	1020	1020
Core inlet enthalpy (Btu/lb)	521.6	518.4
ATRIUM 10XM hot assembly MAPLHGR (kW/ft)	11.7	11.7
[		]

* The AREVA calculated heat balance is adjusted to match the heat balance at 100% power and 100% core flow. AREVA heat balance calculations establish these initial conditions at the stated power and flow. Initial conditions are based on nominal values, except for initial core power and dome pressure which use conservative values.

[ ]

Table 4.2 Reactor System Parameters

Parameter	Value
Vessel ID (in)	251
Number of fuel assemblies	724
Recirculation suction pipe area (ft^2)	3.581
Recirculation discharge pipe area (ft^2)	3.477

Table 4.3 ATRIUM 10XM Fuel Assembly Parameters

Parameter	Value
Fuel rod array	10x10
Number of fuel rods per assembly	79 (full-length rods) 12 (part-length rods)
Non-fuel rod type	Water channel replaces 9 fuel rods
Fuel rod OD (in)	0.4047
Active fuel length (in) (including blankets)	145.24 (full-length rods) 75.0 (part-length rods)
Water channel outside width (in)	1.378
Fuel channel thickness (in)	0. 075 (minimum wall) 0. 100 (corner)
Fuel channel internal width (in)	5.278

Table 4.4 High-Pressure Coolant Injection Parameters

Parameter	Value
Coolant temperature (maximum) (°F)	140
Initiating Signals and Setpoints	
Water level (in)*	444
High drywell pressure (psig)†	2.5
Time Delays	
Time for HPCI pump to reach rated speed and injection valve wide open (sec)	55
Coolant Flow Rate Vs. Pressure	
Vessel to Torus ΔP (psid)	Flow Rate (gpm)
0	0
150	5000
1120	5000

* Relative to vessel zero.

† The time when the drywell reaches 2.5 psig is not explicitly calculated in the EXEM BWR-2000 LOCA methodology. [ ]

Table 4.5 Low-Pressure Coolant Injection Parameters

Parameter		Value
Reactor pressure permissive for opening injection valves - analytical (psig)		300
Coolant temperature (maximum) (°F)		160
Initiating Signals and Setpoints		
Water level (in)*		444
High drywell pressure (psig)†		2.5
Time Delays		
Initiating signal processing delay		1
Time from EDG started and initiated signal to LPCI pump at rated speed (sec)		14
Time for power at the injection valve (sec)		26
LPCI injection valve stroke time (sec)		28
Coolant Flow Rate‡ Vs. Pressure		
Vessel to Torus ΔP (psid)	Flow Rate for 2 Pumps (gpm)	Flow Rate for 4 Pumps (gpm)
0	9,300	15,700
20	9,000	15,200
150	6,200	10,200
257	0	0

* Relative to vessel zero.

† The time when the drywell reaches 2.5 psig is not explicitly calculated in the EXEM BWR-2000 LOCA methodology. [ ]

‡ [ ]

Table 4.6 Low-Pressure Core Spray Parameters

Parameter	Value
Reactor pressure permissive for opening injection valves - analytical (psig)	300
Coolant temperature (maximum) (°F)	160
Initiating Signals and Setpoints	
Water level (in)*	444
High drywell pressure (psig)†	2.5
Time Delays	
Initiating signal processing delay	1
Time from EDG started and initiated signal to LPCS pump at rated speed (sec)	17
Time for power at the injection valve (sec)	17
LPCS injection valve stroke time (sec)	53
Coolant Flow Rate‡ Vs. Pressure	
Vessel to Torus ΔP (psid)	Flow Rate for 1 Pump (gpm)
0	5,650
90	4,500
200	3,000
325	0

* Relative to vessel zero.

† The time when the drywell reaches 2.5 psig is not explicitly calculated in the EXEM BWR-2000 LOCA methodology. **[ ]**

‡ **[ ]**

Table 4.7 Automatic Depressurization System Parameters

Parameter	Value
Number of valves installed	5
Number of valves available*	5
Minimum flow capacity per valve of available valves (4 relief) (lbm/hr at psig)	558,000 at 1120
Minimum flow capacity of available valves (1 Target Rock) (lbm/hr at psig)	598,000 at 1080
Initiating Signals and Setpoints	
Water level (in)†	444
High drywell pressure (psig)‡	2.5
Time Delays	
ADS timer (delay time from initiating signal to time valves are open) (sec)	120

* All 5 valves are assumed operable in the analyses except when analyzing the potential single failure of 1 ADS valve during the LOCA.

† Relative to vessel zero.

‡ The time when the drywell reaches 2.5 psig is not explicitly calculated in the EXEM BWR-2000 LOCA methodology. **[**

]

Table 4.8 Recirculation Discharge Isolation Valve and LPCI Loop Selection Logic Parameters

Parameter	Value
Minimum break area for loop selection logic to select intact loop* (ft^2)	0.15
LPCI loop selection logic pressure permissive (minimum) (psig)	860
Time to power at the RDIV (sec)	26
RDIV stroke time (sec)	48

* For break sizes ≥ 0.15 ft^2, loop selection logic opens the LPCI injection valves in the intact loop and closes the RDIV in the intact loop. For break sizes < 0.15 ft^2, the available LPCI flow is assumed to be injected into the broken loop and the RDIV in the broken loop is closed.

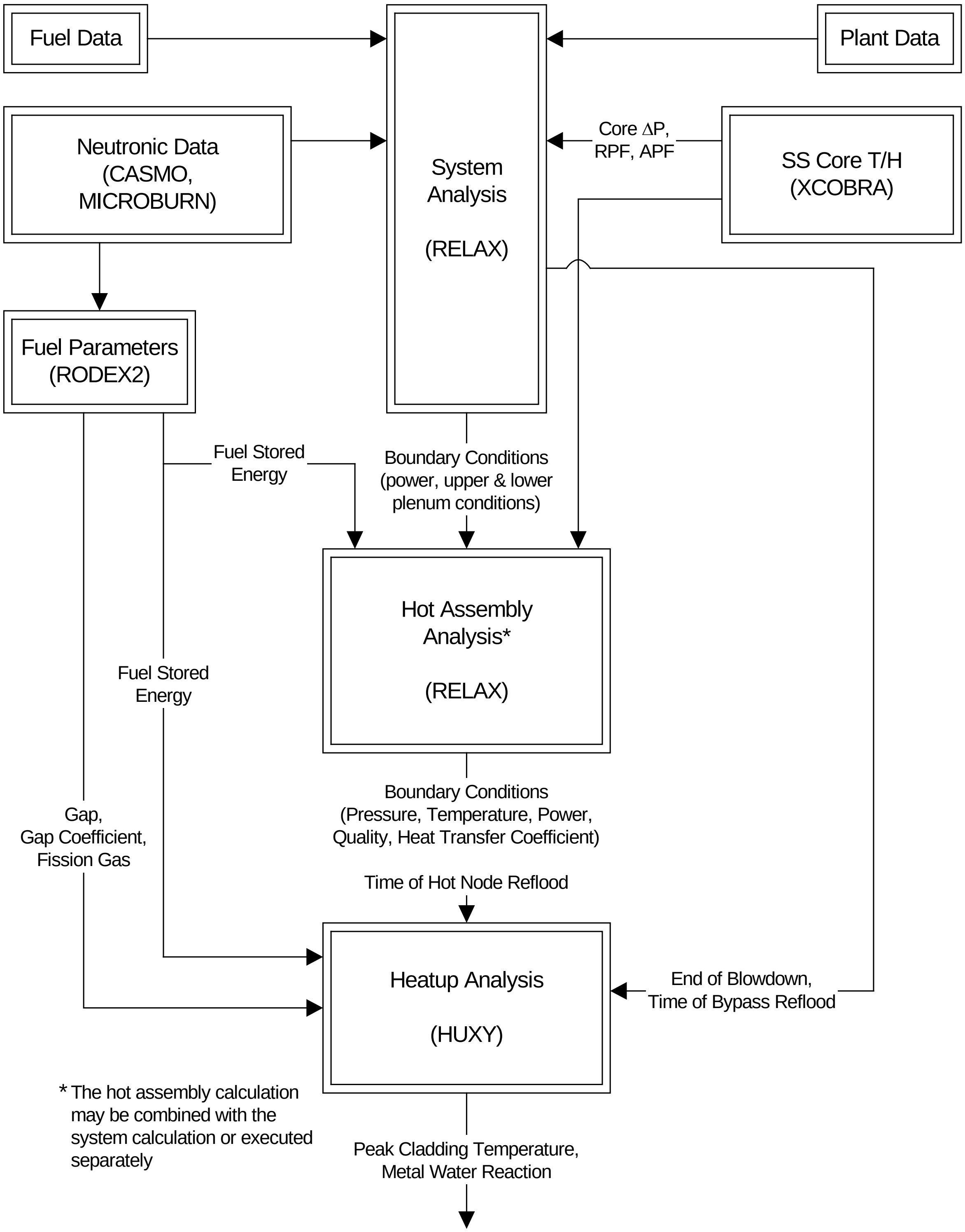


Figure 4.1 Flow Diagram for EXEM BWR-2000 ECCS Evaluation Model

Figure 4.2 [ ]

Figure 4.3 RELAX System Model

Figure 4.4 RELAX Hot Channel Model
Top-Peaked Axial

Figure 4.5 RELAX Hot Channel Model
Mid-Peaked Axial

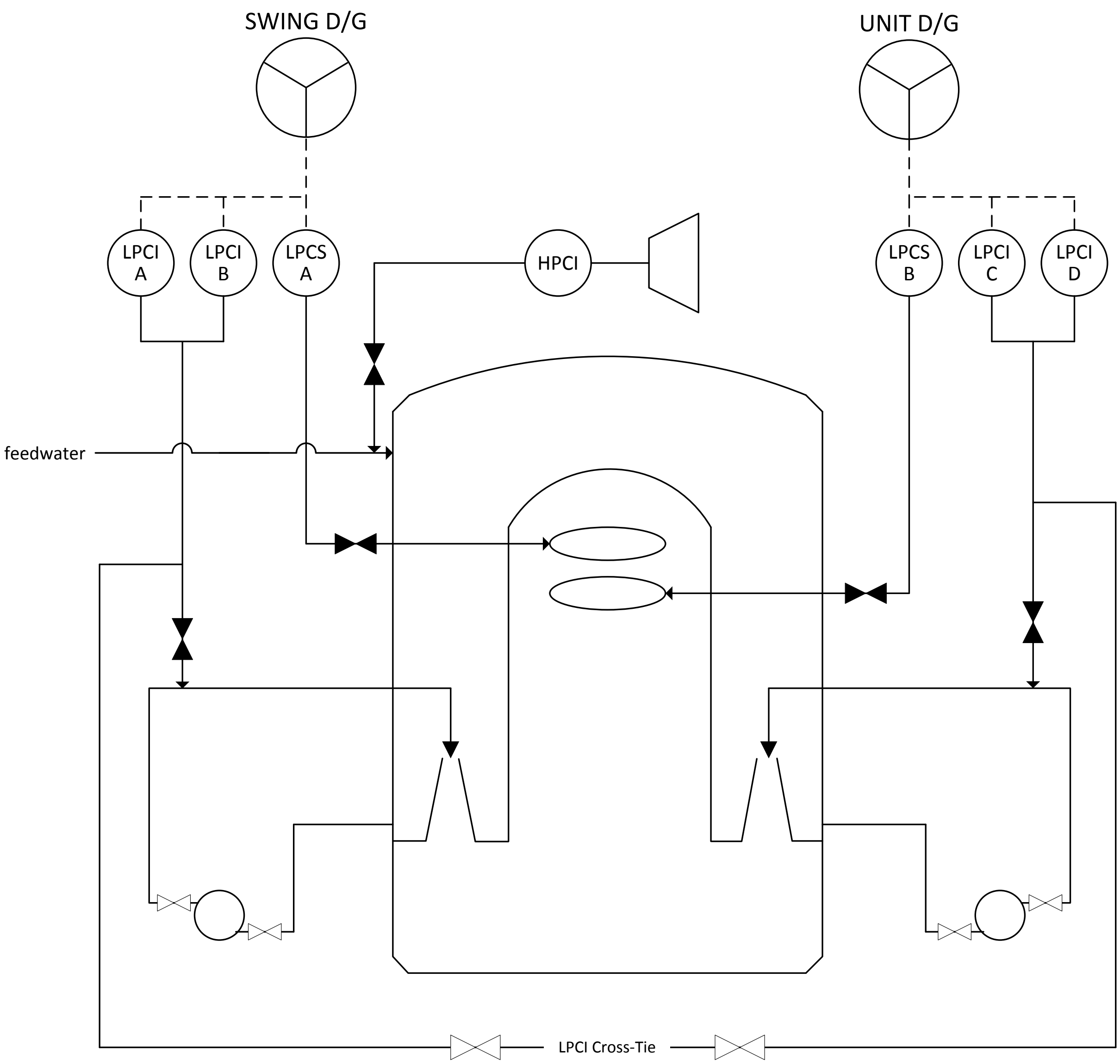


Figure 4.6 ECCS Schematic

Figure 4.7 Rod Average Power Distributions for 102%P [ ] Mid- and Top-Peaked

5.0 Break Spectrum Analysis Description

The objective of these LOCA analyses is to ensure that the limiting break location, break type, break size, and ECCS single failure are identified. The LOCA response scenario varies considerably over the spectrum of break locations. Potential break locations have been separated into two groups: recirculation line breaks and non-recirculation line breaks. The basis for the break locations and potentially limiting single failures analyzed in this report is described in the following sections.

5.1 *Limiting Single Failure*

Regulatory requirements specify that the LOCA analysis be performed assuming that all offsite power supplies are lost instantaneously and that only safety grade systems and components are available. In addition, regulatory requirements also specify that the most limiting single failure of ECCS equipment must be assumed in the LOCA analysis. The term "most limiting" refers to the ECCS equipment failure that produces the greatest challenge to event acceptance criteria. The limiting single failure can be a common power supply, an injection valve, a system pump, or system initiation logic. The most limiting single failure may vary with break size and location. The potential limiting single failures identified in the UFSAR (Reference 6) are shown below:

- LPCI injection valve (SF-LPCI))
- Diesel generator or 125-VDC (SF-DGEN)
- High-pressure coolant injection system (SF-HPCI)
- Loop Select Logic (SF-LSL)
- ADS valve (SF-ADS)

The single failures and the available ECCS for each failure assumed in these analyses are summarized in Table 5.1. Other potential failures are not specifically considered because they result in as much or more ECCS capacity.

The loop selection logic single failure results in all available LPCI flow directed to the default recirculation loop, and the closure of the RDIV in the same loop. The limiting scenario would be if the break is in the default loop, i.e., the loop in which the LPCI flow is injected. Depending on the break size and location relative to the RDIV, some or all of the LPCI flow will flow out the break, minimizing the amount of LPCI flow reaching the core. A comparison of the available ECCS equipment available for the other assumed failures shows that the loop selection logic failure has more or the same amount of ECCS capacity (even if one assumes no benefit from the LPCI system) as the SF-LPCI condition. Therefore, PCT results obtained for SF-LPCI are the same as or bound those obtained for loop selection logic failure. As a result, results reported here are those considering SF-LPCI, SF-DGEN, SF-HPCI and SF-ADS.

5.2 *Recirculation Line Breaks*

The response during a recirculation line LOCA is dependent on break size. The rate of reactor vessel depressurization decreases as the break size decreases. The high pressure ECCS and ADS will assist in reducing the reactor vessel pressure to the pressure where the LPCI and LPCS flows start. For large breaks, rated LPCS and LPCI flow is generally reached before or shortly after the time when the ADS valves open so the ADS system is not required to mitigate the LOCA. HPCI and ADS operation are important emergency systems for small breaks where they assist in depressurizing the reactor system faster, and thereby reduce the time required to reach rated LPCS and LPCI flow.

The two largest flow resistances in the recirculation piping are the recirculation pump and the jet pump nozzle. For breaks in the discharge piping (PD), there is a major flow resistance in both flow paths from the reactor vessel to the break. For breaks in the suction piping (PS), the major flow resistances are in the same flow path from the vessel to the break. As a result, pump suction side breaks experience a more rapid blowdown, which tends to make the large break events more severe. For recirculation line breaks with areas ≥ 0.15 ft^2, the LPCI loop selection logic directs all available LPCI flow to the intact loop and also closes the RDIV in the intact loop. For recirculation line break areas < 0.15 ft^2, the loop selection logic is not able to determine which loop is broken so the limiting scenario is injecting all available LPCI flow into the broken loop where at least some of the flow will exit out the break. In this limiting scenario, the RDIV in the broken loop will close. Both suction and discharge recirculation pipe breaks are considered in the break spectrum analysis.

Two break types (geometries) are considered for the recirculation line break. The two types are the double-ended guillotine (DEG) break and the split break.

For a DEG break, the piping is assumed to be completely severed resulting in two independent flow paths to the containment. The DEG break is modeled by setting the break area (at both ends of the pipe) equal to the full pipe cross-sectional area and varying the discharge coefficient between 1.0 and 0.4. The range of discharge coefficients is used to cover uncertainty in the actual geometry at the break. Discharge coefficients below 0.4 are unrealistic and not considered in the EXEM BWR-2000 methodology. The most limiting DEG break is determined by varying the discharge coefficient. The labeling convention for guillotine breaks is to list the discharge coefficient before DEG. For example, a guillotine break with a discharge coefficient of 0.8 on the suction side of the recirculation pump would be labeled as 0.8DEGPS.

A split type break is assumed to be a longitudinal opening or hole in the piping that results in a single break flow path to the containment. Appendix K of 10 CFR 50 defines the cross-sectional area of the piping as the maximum split break area required for analysis. The labeling convention for split breaks is to list the flow area using the letter "P" instead of a period. For example, a split break with a flow area of 3.5 ft^2 on the suction side of the recirculation pump would be labeled as 3P5FT2PS. These labeling conventions for double-ended guillotine and split breaks are typically used in figures such as those in Section 6.0 of this report.

Break types, break sizes and single failures are analyzed for both suction and discharge recirculation line breaks.

Section 6.0 provides a description and results summary for breaks in the recirculation line.

5.3 *Non-Recirculation Line Breaks*

In addition to breaks in the recirculation line, breaks in other reactor coolant system piping must be considered in the LOCA break spectrum analysis. Although the recirculation line large breaks result in the largest coolant inventory loss, they do not necessarily result in the most severe challenge to event acceptance criteria. The double-ended rupture of a main steam line is expected to result in the fastest depressurization of the reactor vessel. Special consideration is required when the postulated break occurs in ECCS piping. Although ECCS piping breaks are small relative to a recirculation pipe DEG break, the potential to disable an ECCS system increases their severity.

The following sections address potential LOCAs due to breaks in non-recirculation line piping.

Non-recirculation line breaks outside of the containment are inherently less challenging to fuel limits than breaks inside the containment. For breaks outside containment, isolation or check valve closure will terminate break flow prior to the loss of significant liquid inventory and the core will remain covered. If high-pressure coolant inventory makeup cannot be reestablished, ADS actuation may become necessary. **[**

] Although analyses of breaks outside containment may be required to address non-fuel related regulatory requirements, these breaks are not limiting relative to fuel acceptance criteria such as PCT.

5.3.1 Main Steam Line Breaks

A steam line break inside containment is assumed to occur between the reactor vessel and the inboard main steam line isolation valve (MSIV) upstream of the flow limiters. The break results in high steam flow out of the broken line and into the containment. Prior to MSIV closure, a steam line break also results in high steam flow in the intact steam lines as they feed the break via the steam line manifold. A steam line break inside containment results in a rapid depressurization of the reactor vessel. Initially the break flow will be high quality steam; however, the rapid depressurization produces a water level swell that results in liquid discharge at the break. For steam line breaks, the largest break size is most limiting because it results in the most level swell and liquid loss out of the break.

[

]

5.3.2 Feedwater Line Breaks

[

]

5.3.3 HPCI Line Breaks

The HPCI injection line is connected to the feedwater line outside of the containment.

[ ]

The HPCI steam supply line is connected to the main steam line inside of containment.

[ ]

HPCI and ADS are important for small breaks. With an HPCI line break, failure of an ADS valve is a potentially limiting single failure. [ ]

5.3.4 LPCS Line Breaks

A break in the LPCS line is expected to have many characteristics similar to [ ]. However, some characteristics of the LPCS line break are unique and are not addressed in other LOCA analyses. Two important differences from other LOCA analyses are that the break flow will exit from the region inside the core shroud and the break will disable one LPCS system. The LPCS line break is assumed to occur just outside the reactor vessel. [ ]

5.3.5 LPCI Line Breaks

The LPCI injection lines are connected to the larger recirculation discharge lines. [ ]

5.3.6 Reactor Water Cleanup Line Breaks

The extraction line is connected to a recirculation suction line with an additional connection to the vessel bottom head. [ ]

The return line is connected to the feedwater line; [ ].

5.3.7 Shutdown Cooling Line Breaks

The shutdown cooling suction piping is connected to a recirculation suction line and the shutdown cooling return line is connected to a recirculation discharge line. [ ]

5.3.8 Instrument Line Breaks

[ ]

Table 5.1 Available ECCS for Recirculation Line Break LOCAs

Assumed Failure	Systems Remaining*, †
LPCI injection valve (SF-LPCI)	2 LPCS + HPCI + 5 ADS
Diesel generator or 125-VDC (SF-DGEN)	1 LPCS + 2 LPCI + HPCI + 5 ADS
HPCI system (SF-HPCI)	2 LPCS + 4 LPCI + 5 ADS
Loop select logic (SF-LSL)	2 LPCS + 4 LPCI + HPCI + 5 ADS
ADS valve (SF-ADS)	2 LPCS + 4 LPCI + HPCI + 4 ADS

* Systems remaining, as identified in this table for recirculation line breaks, are applicable to all non-ECCS line breaks. For ECCS line breaks, the systems remaining are those listed for recirculation breaks, less the ECCS in which the break is assumed

† With loop selection logic operational, all available LPCI flow is directed to the intact loop for breaks ≥ 0.15 ft^2. All available LPCI flow is directed to the broken loop for breaks < 0.15 ft^2. The limiting condition for a loop selection logic failure would result in all available LPCI flow directed to the broken loop for all break sizes.

6.0 Recirculation Line Break LOCA Analyses

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 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 ft^2. As discussed in Section 5.0, the single failures considered in the recirculation line break analyses are SF-LPCI, SF-DGEN, SF-HPCI, SF-LSL and SF-ADS.

[

]

6.1 *Limiting Break Analysis Results*

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 single failure and a top-peaked axial power shape when operating at 102% rated core power **[ ]**. The PCT is 2127°F. The key results and event times for this limiting break are provided in Table 6.1 and Table 6.2, respectively. Figure 6.1 – Figure 6.23 provide plots of key parameters from the RELAX system and hot channel analyses. A plot of cladding temperature versus time in the hot assembly from the HUXY heatup analysis is provided in Figure 6.24.

Table 6.3 presents SF-HPCI results obtained from calculations performed for the range of break sizes, break locations, and axial power shapes that were considered when operating at 102% rated core power **[ ]**. Calculations were performed for this same range for each combination of initial operating state point and single failure. Table 6.4 provides a summary of the highest PCT recirculation line break calculations for each of the single failures, state points, and axial power shapes. The results of the break spectrum analyses are discussed in the following sections.

6.2 *Break Location Analysis Results*

Table 6.4 shows that the maximum PCT calculated for a recirculation line break occurs in the pump discharge piping.

6.3 *Break Geometry and Size Analysis Results*

Recirculation line break PCT results versus break geometry (DEG or split) and size were performed for

- DEG breaks with discharge coefficients of 1.0, 0.8, 0.6, and 0.4.
- Split breaks ranging in size from the full pipe diameter to 0.1 ft^2 in increments of 0.1 ft^2 and a final size of 0.05 ft^2. Based on the results for small breaks, for some conditions the break size increment was reduced to 0.01 ft^2 in order to assure the limiting break size has been identified.

Table 6.4 shows that the maximum PCT calculated for a recirculation line break occurs for a split break of 0.13 ft^2.

6.4 *Limiting Single-Failure Analysis Results*

The results in Table 6.4 show that the limiting single-failure is SF-HPCI.

6.5 *Axial Power Shape Analysis Results*

The results in Table 6.4 show that the top-peaked axial power shape is generally limiting compared to the mid-peaked shape analyses for the limiting break size.

6.6 *State Point Analysis*

Table 6.4 shows that 102% rated core power **[** **]** is generally the limiting state point for the recirculation line breaks.

**Table 6.1 Results for
Limiting TLO Recirculation Line Break
0.13 ft^2 Split Pump Discharge SF-HPCI
Top-Peaked Axial 102% Power [ ]**

PCT	2127°F
Maximum local cladding oxidation	3.02%
Maximum planar average cladding oxidation	2.40%

**Table 6.2 Event Times for
Limiting TLO Recirculation Line Break
0.13 ft^2 Split Pump Discharge SF-HPCI
Top-Peaked Axial 102% Power [ ]**

Event	Time (sec)
Initiate Break	0.0
Initiate Scram	0.6
Diesel Generators Started	17.0
Low-Low Liquid Level, L2 (444 in)	48.5
Power at LPCS Injection Valves	17.0
ADS with ECCS Pump(s) Running	56.5
LPCS Pump at Rated Speed	66.5
LPCS High-Pressure Cutoff	338.4
LPCS Valve Pressure Permissive	348.5
LPCS Valve Starts to Open	348.5
LPCS Flow Starts	351.2
Top Down Cooling Restriction Begins in the Hot Channel	351.2
LPCS Valve Fully Open	401.5
Rated LPCS Flow	549.1
LPCI Pump at Rated Speed	63.5
LPCI Valve Pressure Permissive	348.5
LPCI Valve Starts to Open	348.5
LPCI Flow Starts	366.7
LPCI Valve Fully Open	376.5
Jet Pump Uncovers	139.8
Recirculation Suction Uncovers	414.1
ADS Valves Open	169.5
RDIV Pressure Permissive	207.8
RDIV Starts to Close	207.8
RDIV Closed	255.8
PCT	452.8
Time of Bypass Reflood	513.1
Time of Rated Spray	549.1
Time of Hot Node Reflood	549.1

Table 6.3 TLO Recirculation Line Break Spectrum Results for 102% Power [ ] SF-HPCI

[ ]

[ ]

Table 6.4 Summary of TLO Recirculation Line Break Results Highest PCT Cases

Figure 6.1 Limiting TLO Recirculation Line Break
Upper Plenum Pressure

Figure 6.2 Limiting TLO Recirculation Line Break
Total Break Flow Rate

**Figure 6.3 Limiting TLO Recirculation Line Break
Core Inlet Flow Rate**

**Figure 6.4 Limiting TLO Recirculation Line Break
Core Outlet Flow Rate**

Figure 6.5 Limiting TLO Recirculation Line Break
Intact Loop Jet Pump Exit Flow Rate

Figure 6.6 Limiting TLO Recirculation Line Break
Broken Loop Jet Pump Exit Flow Rate

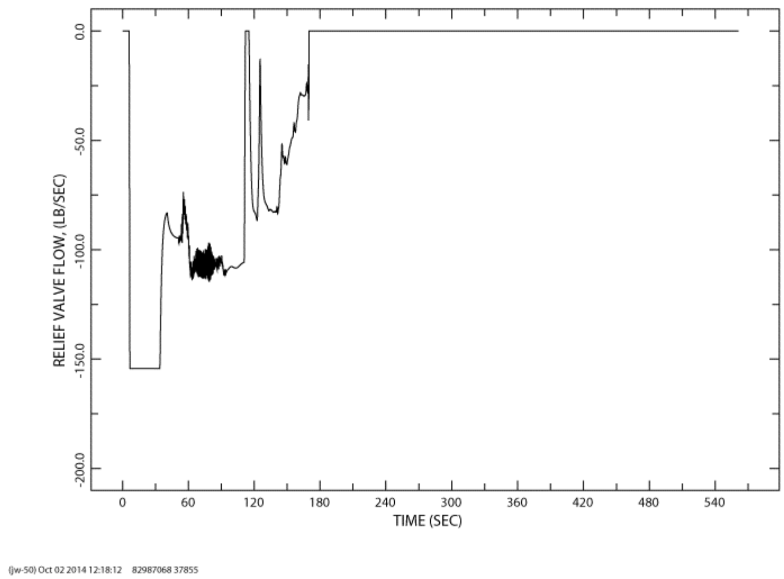


Figure 6.7 Limiting TLO Recirculation Line Break Relief Valve Flow Rate

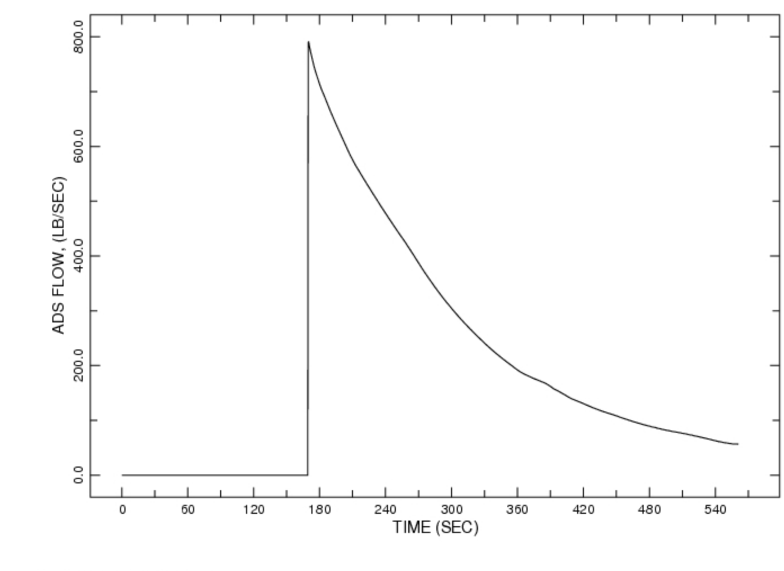
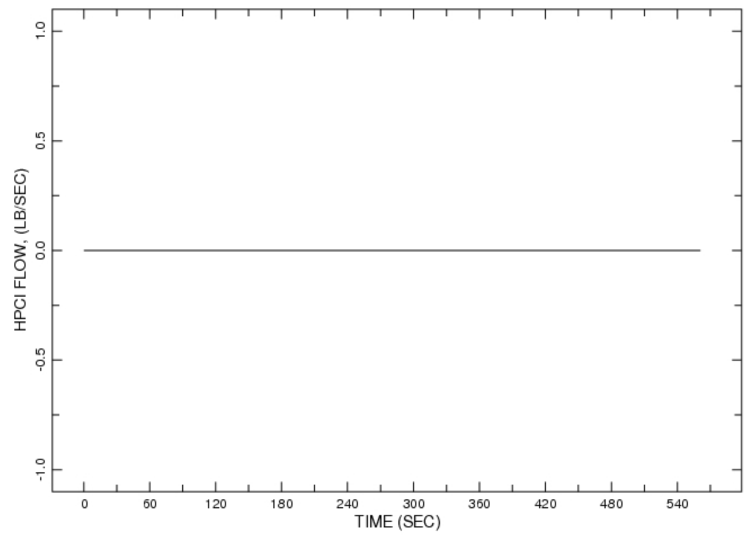
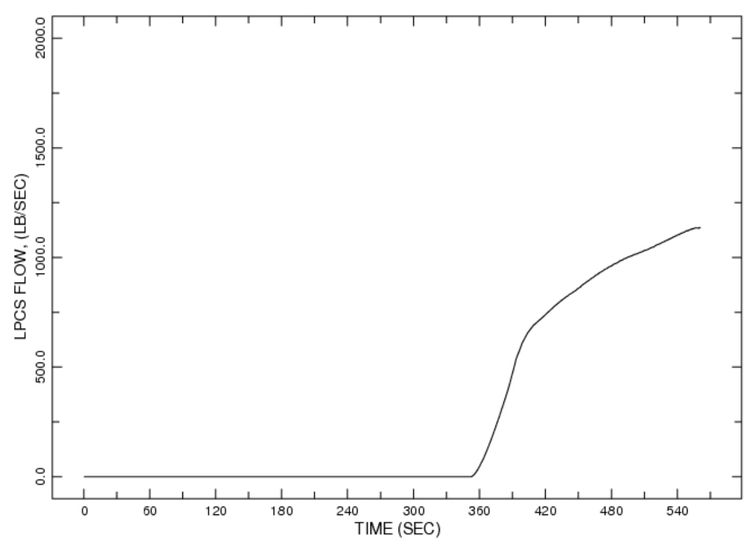


Figure 6.8 Limiting TLO Recirculation Line Break ADS Flow Rate



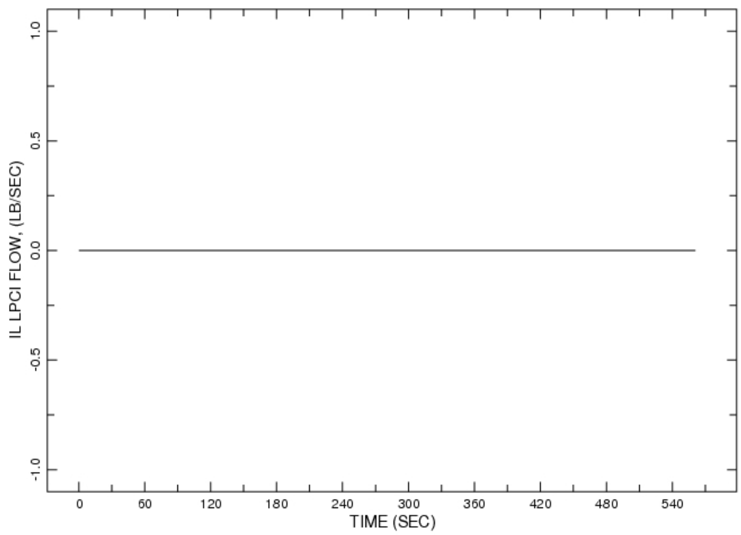
(jw-52) Oct 02 2014 12:18:12 82987068 37855

**Figure 6.9 Limiting TLO Recirculation Line Break
HPCI Flow Rate**



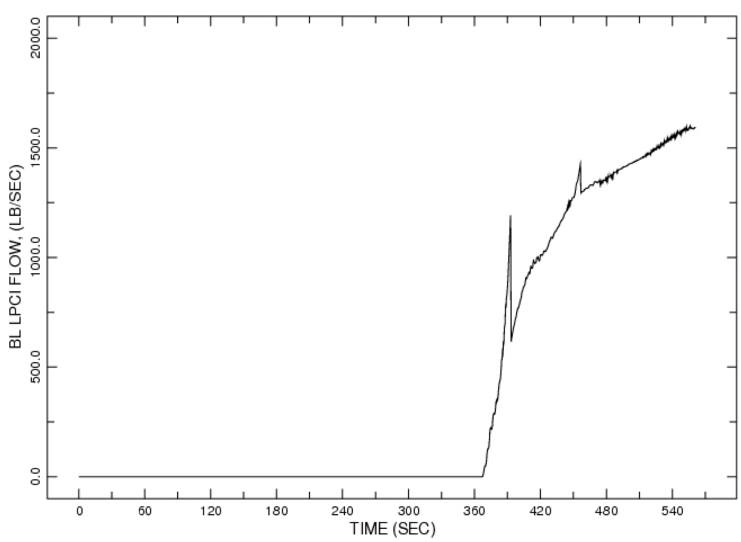
(jw-51) Oct 02 2014 12:18:12 82987068 37855

**Figure 6.10 Limiting TLO Recirculation Line Break
LPCS Flow Rate**



(jw-53) Oct 02 2014 12:18:12 82987068 37855

Figure 6.11 Limiting TLO Recirculation Line Break Intact Loop LPCI Flow Rate



(jw-54) Oct 02 2014 12:18:12 82987068 37855

Figure 6.12 Limiting TLO Recirculation Line Break Broken Loop LPCI Flow Rate

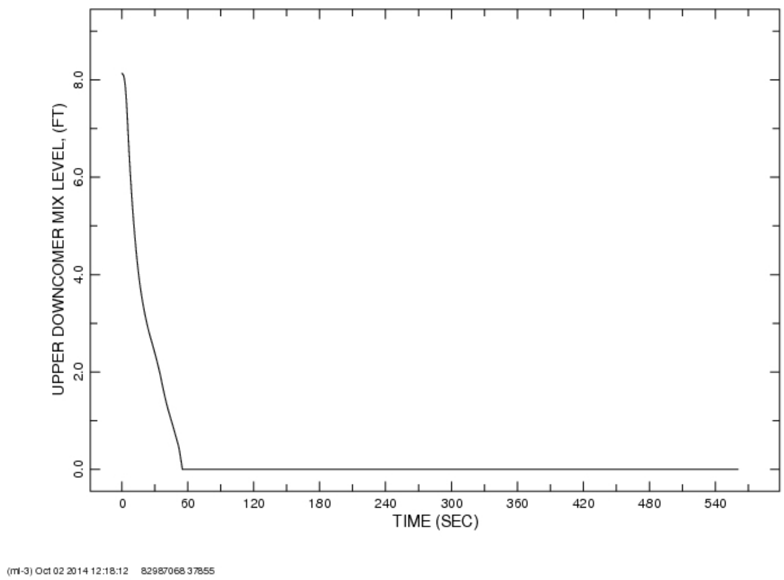


Figure 6.13 Limiting TLO Recirculation Line Break Upper Downcomer Mixture Level

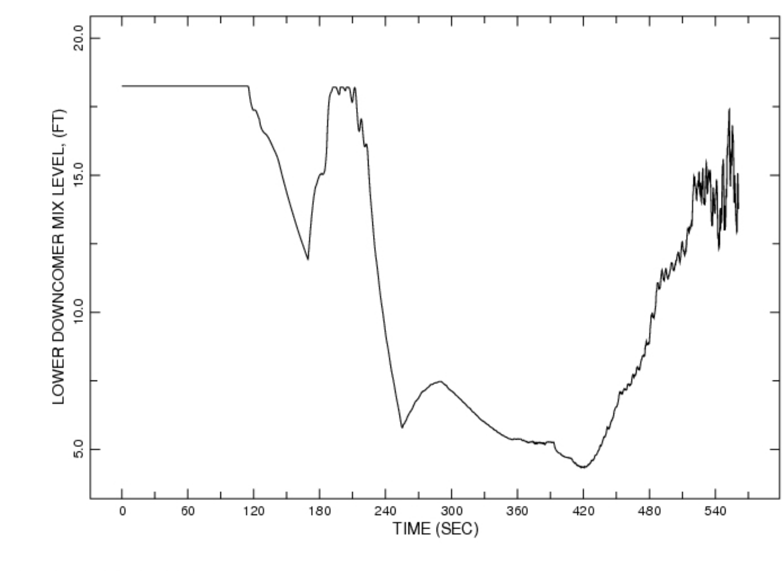


Figure 6.14 Limiting TLO Recirculation Line Break Lower Downcomer Mixture Level

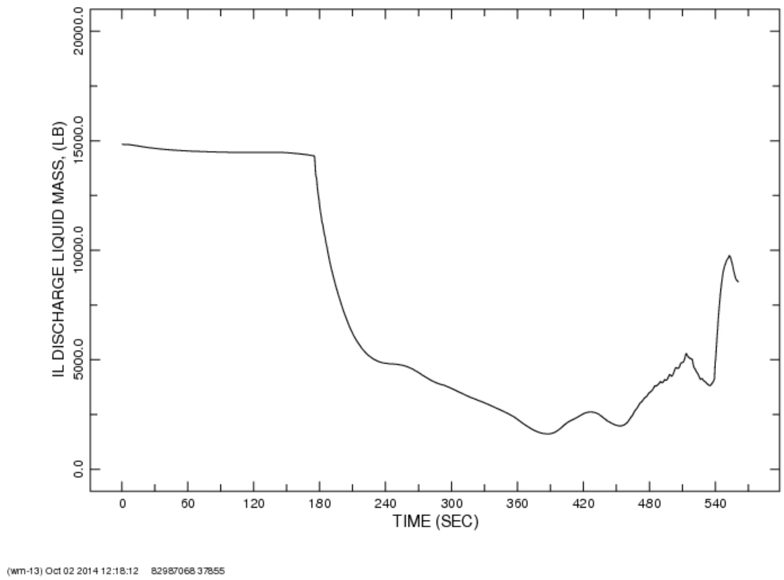


Figure 6.15 Limiting TLO Recirculation Line Break
Intact Loop Discharge Line Liquid Mass

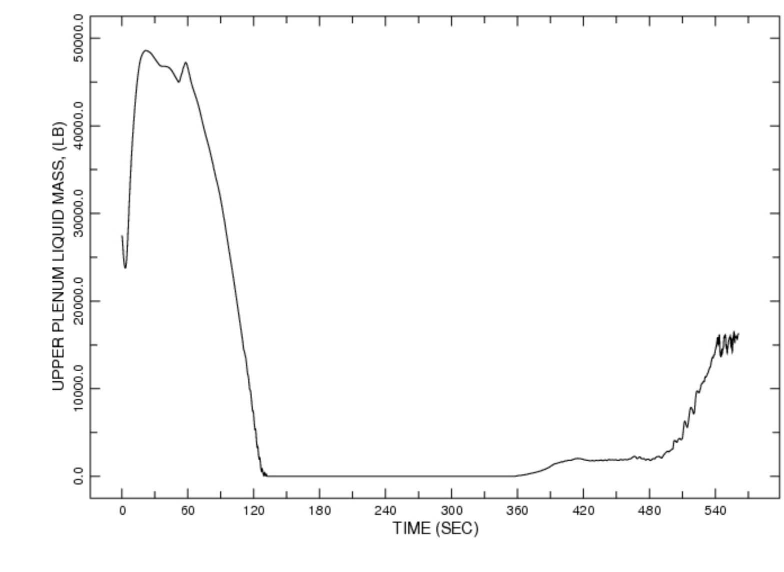


Figure 6.16 Limiting TLO Recirculation Line Break
Upper Plenum Liquid Mass

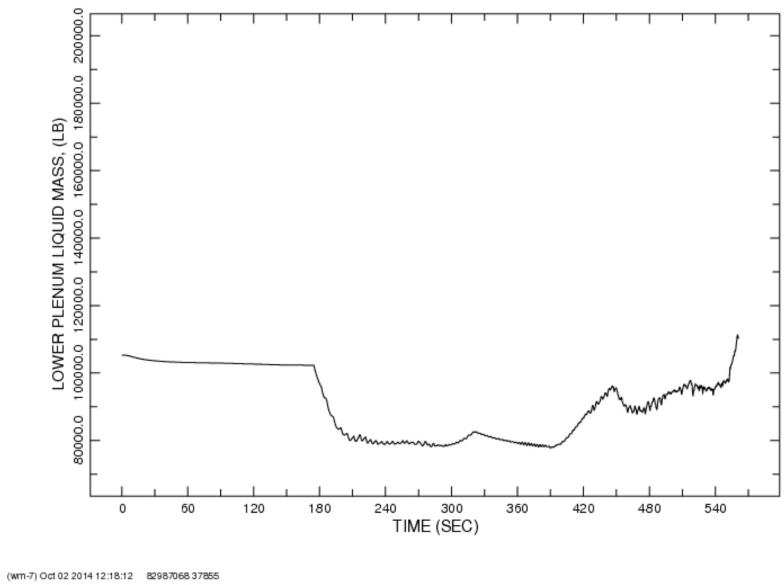


Figure 6.17 Limiting TLO Recirculation Line Break
Lower Plenum Liquid Mass

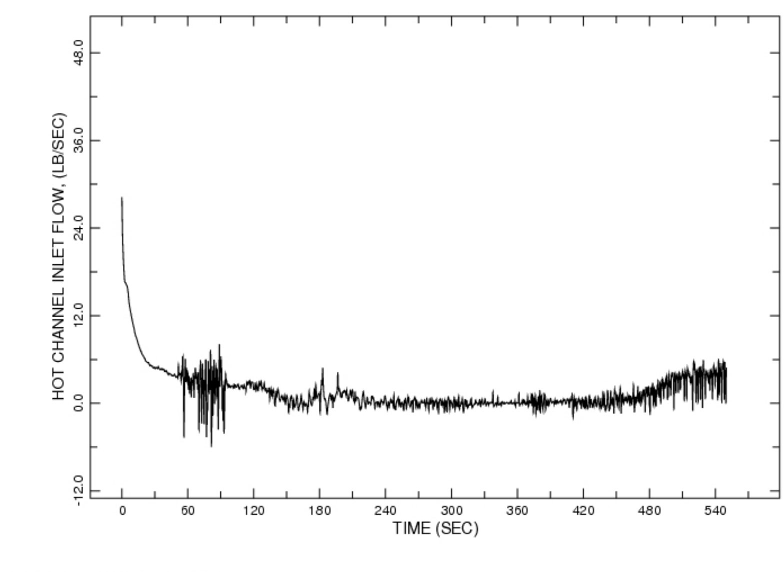
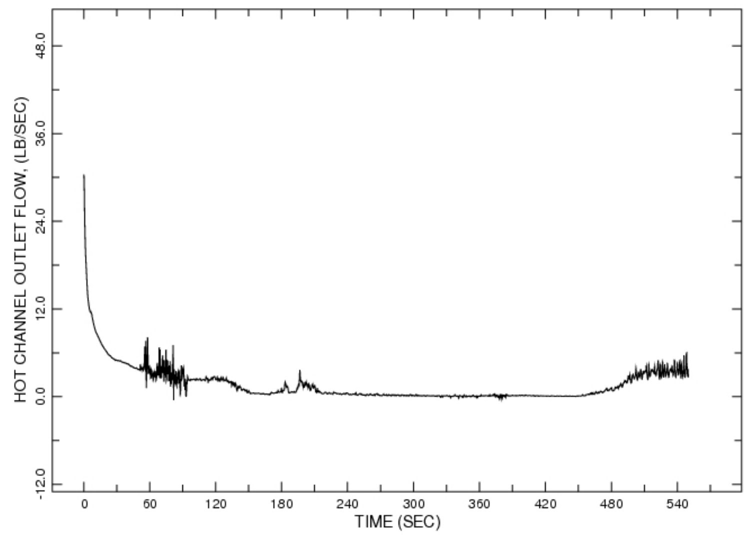


Figure 6.18 Limiting TLO Recirculation Line Break
Hot Channel Inlet Flow Rate



(jw-10) Oct 02 2014 12:20:17 18785392 26554

**Figure 6.19 Limiting TLO Recirculation Line Break
Hot Channel Outlet Flow Rate**

**Figure 6.20 Limiting TLO Recirculation Line Break
Hot Channel Coolant Temperature at the Hot Node at EOB**

**Figure 6.21 Limiting TLO Recirculation Line Break
Hot Channel Quality at the Hot Node at EOB**

**Figure 6.22 Limiting TLO Recirculation Line Break
Hot Channel Heat Transfer Coeff. at the Hot Node at EOB**

**Figure 6.23 Limiting TLO Recirculation Line Break
Hot Channel Reflood Junction Liquid Mass Flow Rate**

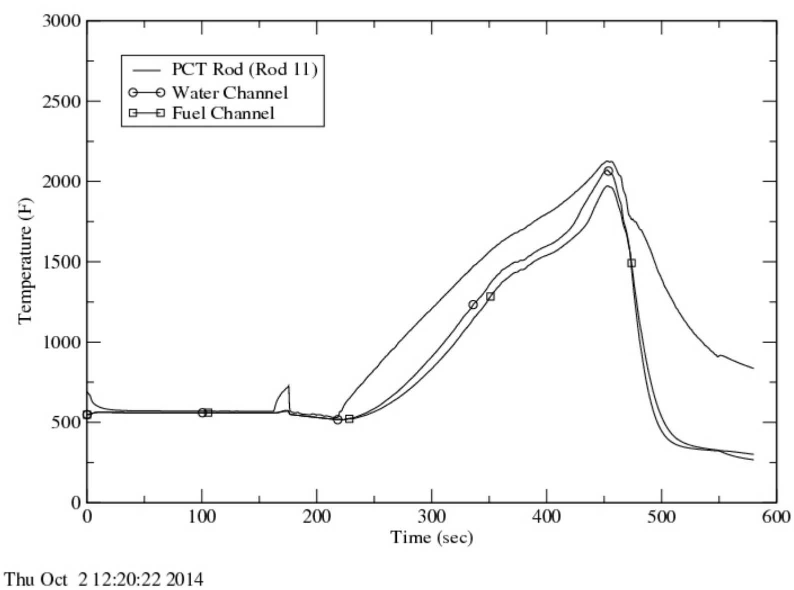


**Figure 6.24 Limiting TLO Recirculation Line Break
Cladding Temperatures**

7.0 Single-Loop Operation LOCA Analysis

During SLO, the pump in one recirculation loop is not operating. A break may occur in either loop, but results from a break in the inactive loop would be similar to those from a two-loop operation break. If a break occurs in the inactive loop during SLO, the intact active loop flow to the reactor vessel would continue during the recirculation pump coastdown period and would provide core cooling similar to that which would occur in breaks during TLO. The system response would be similar to that resulting from an equal-sized break during two-loop operation. A break in the active loop during SLO results in a more rapid loss of core flow and earlier degraded core conditions relative to those from a break in the inactive loop. Therefore, only breaks in the active recirculation loop are analyzed.

A break in the active recirculation loop during SLO will result in an earlier loss of core heat transfer relative to a similar break occurring during two-loop operation. This occurs because there will be an immediate loss of jet pump drive flow. Therefore, fuel rod surface temperatures will increase faster in an SLO LOCA relative to a TLO LOCA. Also, the early loss of core heat transfer will result in higher stored energy in the fuel rods at the start of the heatup. The increased severity of an SLO LOCA can be reduced by applying an SLO multiplier to the two-loop MAPLHGR limits. **[**

]

7.1 *SLO Analysis Modeling Methodology*

[

]

[

]

7.2 *SLO Analysis Results*

[

]

The SLO analyses are performed with a 0.80 multiplier applied to the two-loop MAPLHGR limit resulting in an SLO MAPLHGR limit of 9.36 kW/ft. The analyses are 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. Other key results and event times for the limiting SLO LOCA are provided in Table 7.1 and Table 7.2 respectively. Figure 7.1 – Figure 7.23 show important RELAX system and hot channel results from the SLO limiting LOCA analysis. Figure 7.24 shows the cladding surface temperature for the limiting rod as calculated by HUXY.

A comparison of the limiting SLO and the limiting two-loop results is provided in Table 7.3. The results in Table 7.3 show that the limiting two-loop LOCA PCT bounds the limiting SLO PCT when a 0.80 multiplier is applied to the two-loop MAPLHGR limit.

**Table 7.1 Results for
Limiting SLO Recirculation Line Break
0.1 ft^2 Split Pump Discharge SF-HPCI
Top-Peaked Axial 102% Power [ ]**

PCT	2047°F
Maximum local cladding oxidation	2.21%
Maximum planar average cladding oxidation	1.81%

**Table 7.2 Event Times for
Limiting SLO Recirculation Line Break
0.1 ft^2 Split Pump Discharge SF-HPCI
Top-Peaked Axial 102% Power [ ]**

Event	Time (sec)
Initiate Break	0.0
Initiate Scram	0.6
Diesel Generators Started	17.0
Low-Low Liquid Level, L2 (444 in)	58.8
Power at LPCS Injection Valves	17.0
ADS with ECCS Pump(s) Running	66.8
LPCS Pump at Rated Speed	76.8
LPCS High-Pressure Cutoff	375.3
LPCS Valve Pressure Permissive	387.5
LPCS Valve Starts to Open	387.5
LPCS Flow Starts	390.2
Top Down Cooling Restriction Begins in the Hot Channel	390.2
LPCS Valve Fully Open	440.5
Rated LPCS Flow	704.3
Power at LPCI Injection Valves	26.0
LPCI Pump at Rated Speed	73.8
LPCI Valve Pressure Permissive	387.5
LPCI Valve Starts to Open	387.5
LPCI Flow Starts	410.9
LPCI Valve Fully Open	415.5
Jet Pump Uncovers	175.6
Recirculation Suction Uncovers	No Uncovery
ADS Valves Open	179.8
RDIV Pressure Permissive	219.8
RDIV Starts to Close	219.8
RDIV Closed	267.8
PCT	493.2
Time of Rated Spray	704.3
Time of Hot Node Reflood	704.3
Time of Bypass Reflood	538.1

Table 7.3 Single- and Two-Loop Operation PCT Summary

Operation	Limiting Case	PCT (°F)
Single-loop	0.1 ft^2 split pump discharge top-peaked SF-HPCI	2047
Two-loop	0.13 ft^2 split pump discharge top-peaked SF-HPCI	2127

Figure 7.1 Limiting SLO Recirculation Line Break Upper Plenum Pressure

Figure 7.2 Limiting SLO Recirculation Line Break Total Break Flow Rate

**Figure 7.3 Limiting SLO Recirculation Line Break
Core Inlet Flow Rate**

**Figure 7.4 Limiting SLO Recirculation Line Break
Core Outlet Flow Rate**

**Figure 7.5 Limiting SLO Recirculation Line Break
Intact Loop Jet Pump Exit Flow Rate**

**Figure 7.6 Limiting SLO Recirculation Line Break
Broken Loop Jet Pump Exit Flow Rate**

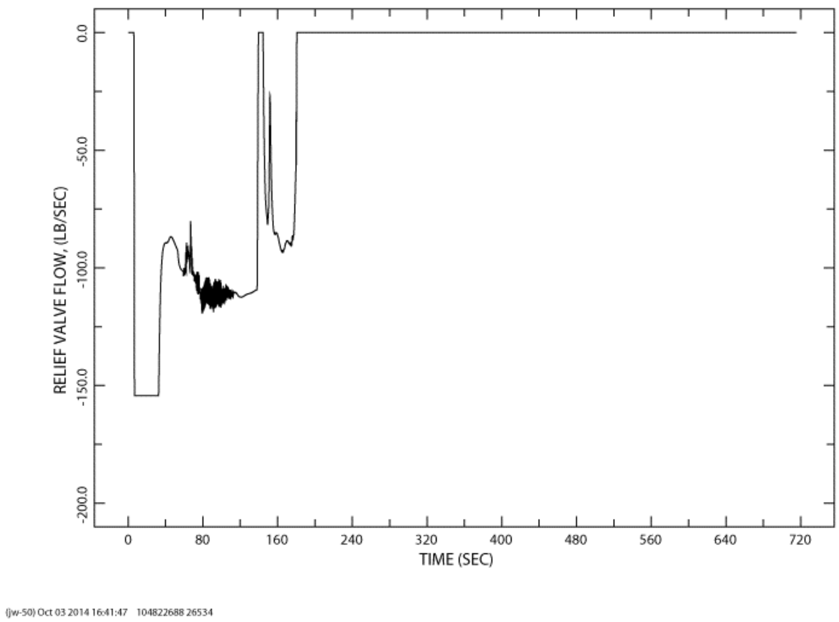


Figure 7.7 Limiting SLO Recirculation Line Break Relief Valve Flow Rate

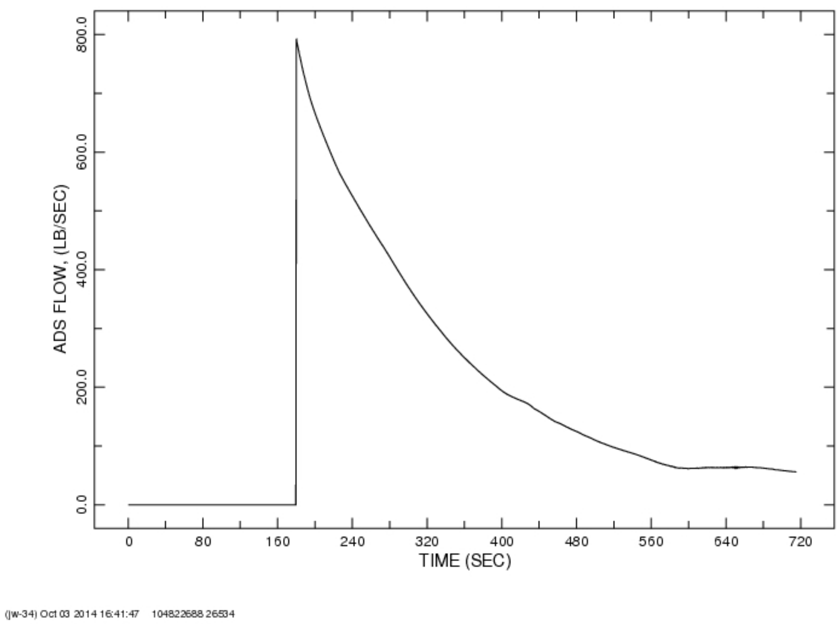


Figure 7.8 Limiting SLO Recirculation Line Break ADS Flow Rate

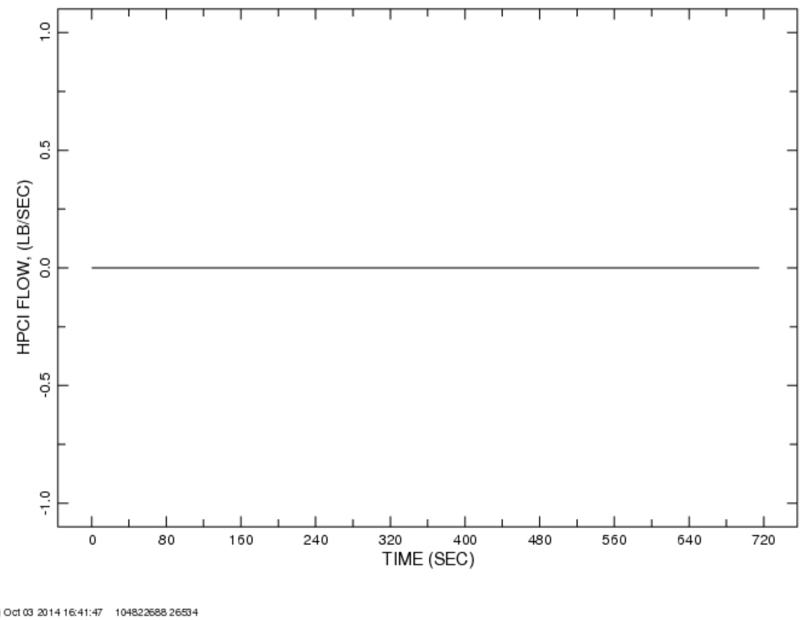


Figure 7.9 Limiting SLO Recirculation Line Break
HPCI Flow Rate

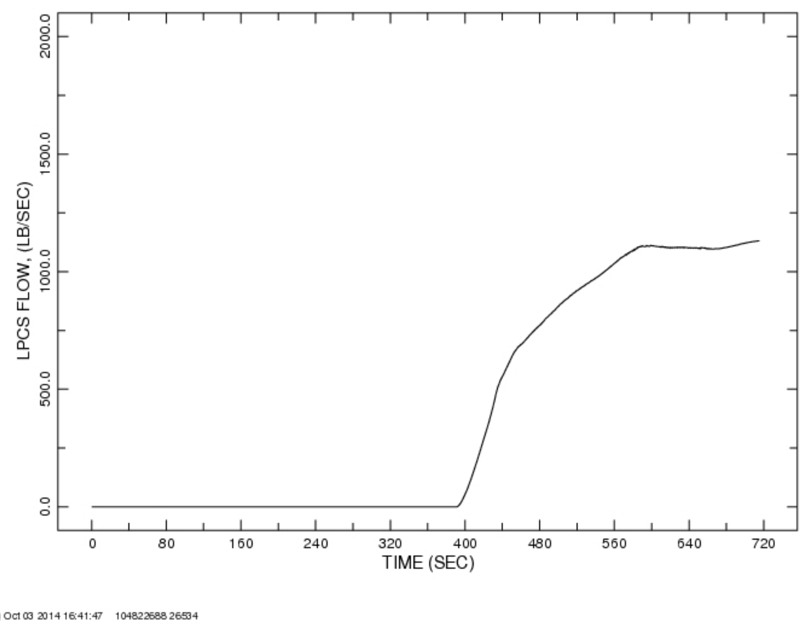


Figure 7.10 Limiting SLO Recirculation Line Break
LPCS Flow Rate

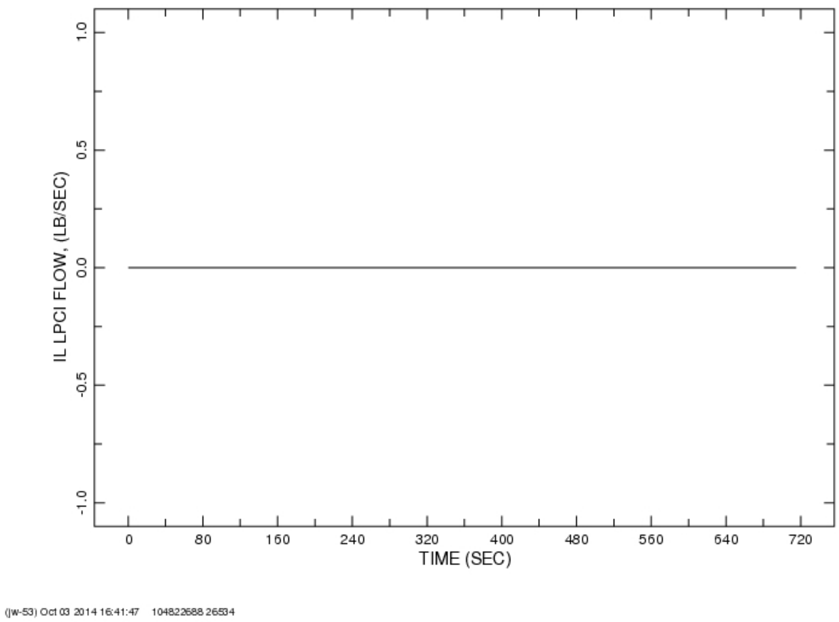


Figure 7.11 Limiting SLO Recirculation Line Break Intact Loop LPCI Flow Rate

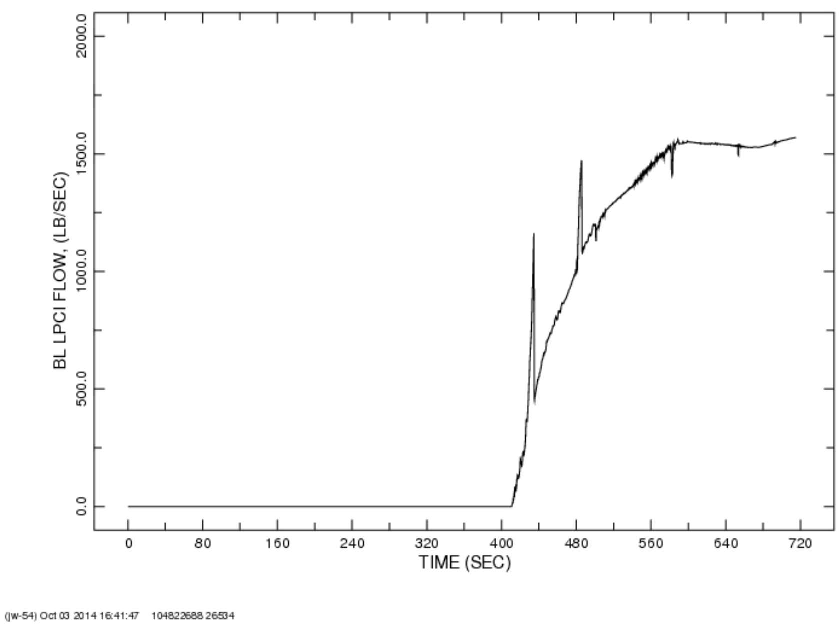


Figure 7.12 Limiting SLO Recirculation Line Break Broken Loop LPCI Flow Rate

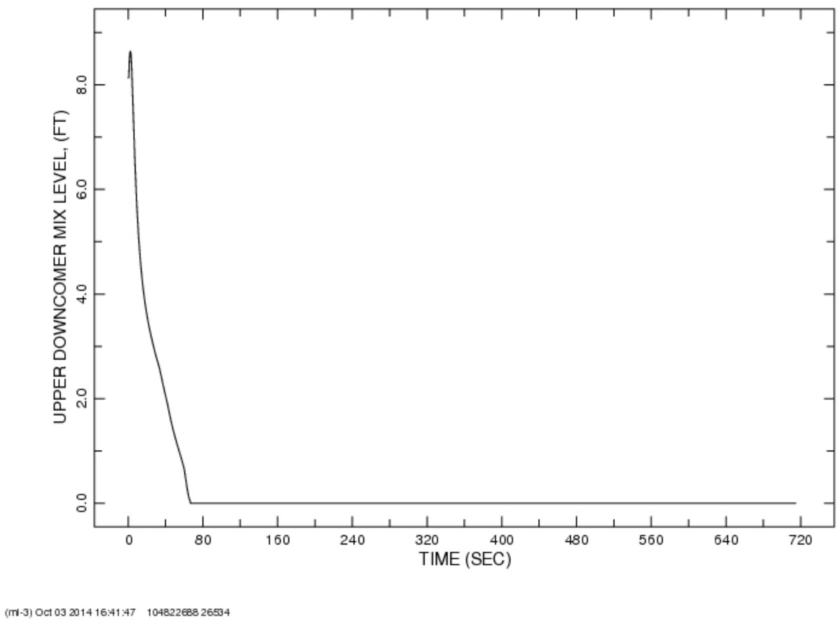


Figure 7.13 Limiting SLO Recirculation Line Break Upper Downcomer Mixture Level

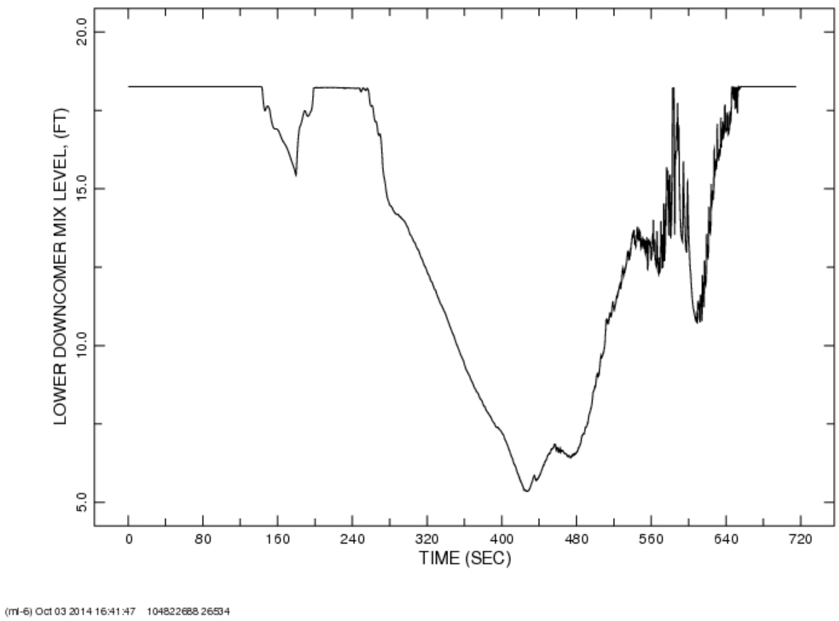


Figure 7.14 Limiting SLO Recirculation Line Break Lower Downcomer Mixture Level

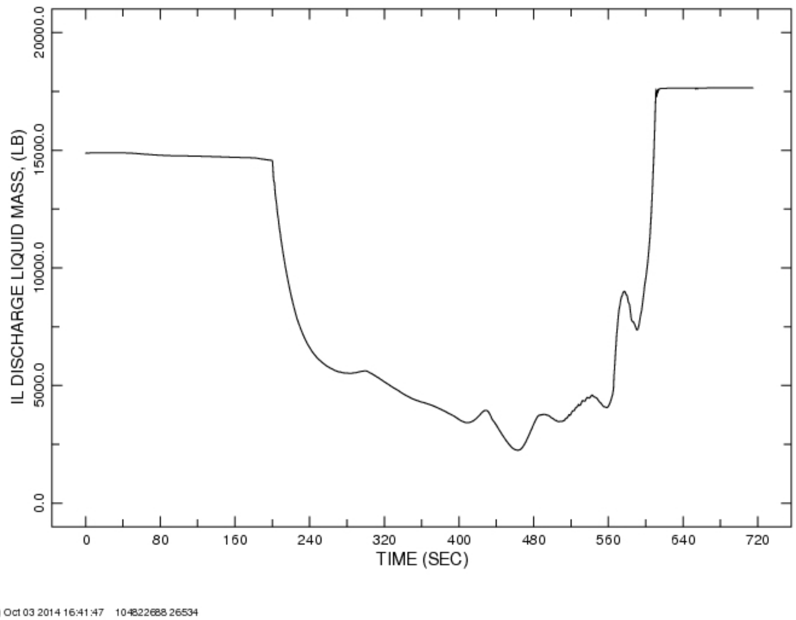


Figure 7.15 Limiting SLO Recirculation Line Break Intact Loop Discharge Line Liquid Mass

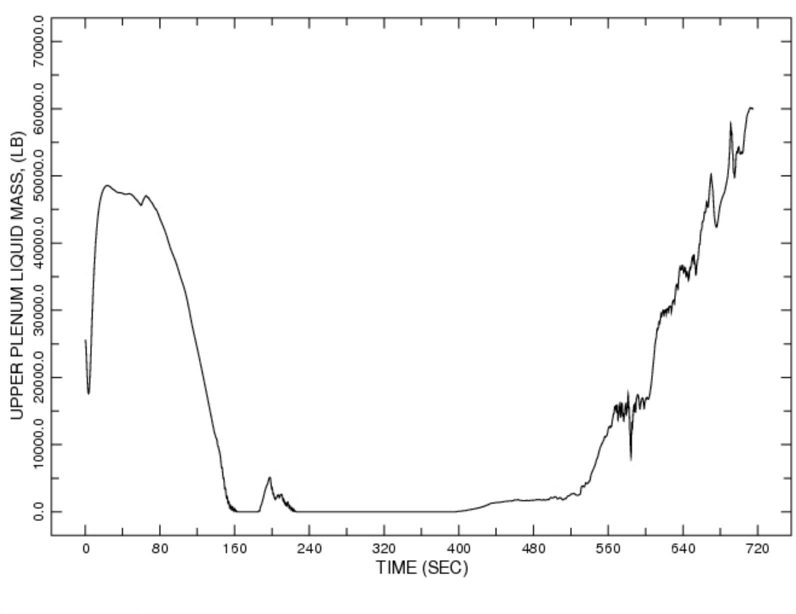


Figure 7.16 Limiting SLO Recirculation Line Break Upper Plenum Liquid Mass

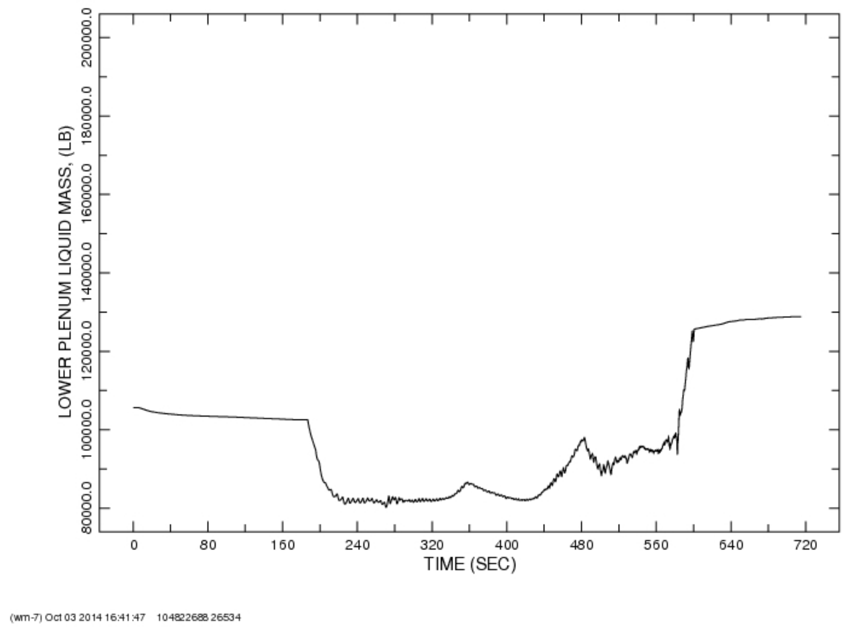


Figure 7.17 Limiting SLO Recirculation Line Break Lower Plenum Liquid Mass

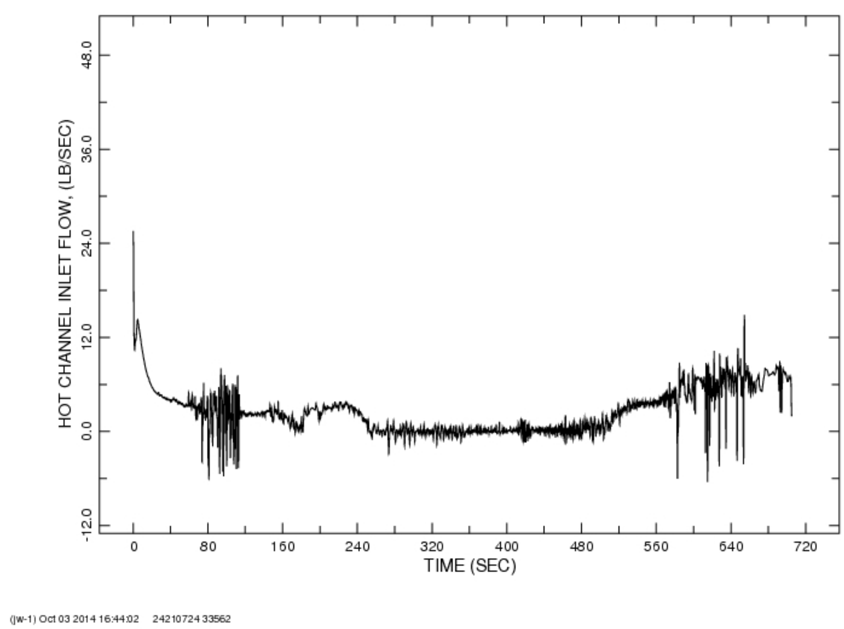


Figure 7.18 Limiting SLO Recirculation Line Break Hot Channel Inlet Flow Rate

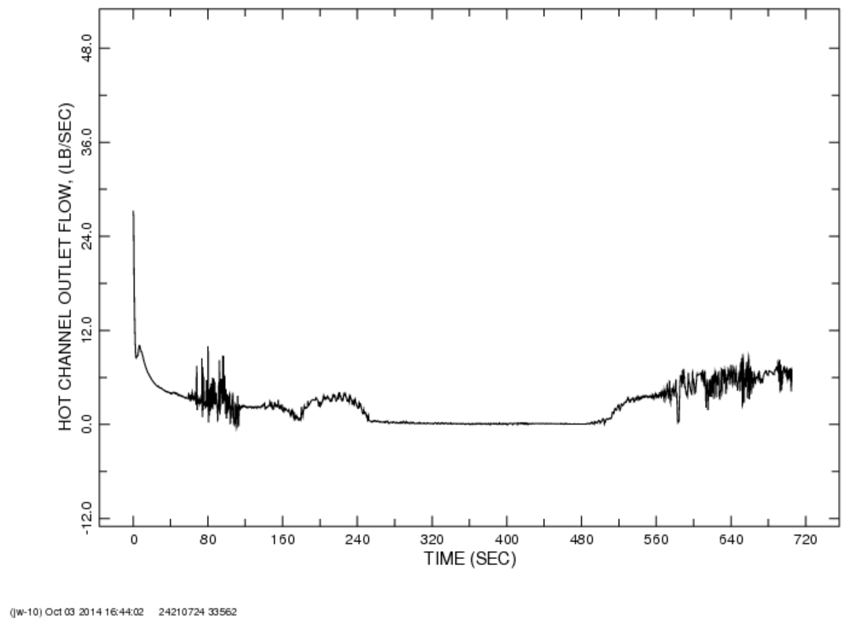


Figure 7.19 Limiting SLO Recirculation Line Break
Hot Channel Outlet Flow Rate

Figure 7.20 Limiting SLO Recirculation Line Break
Hot Channel Coolant Temperature at the Hot Node at EOB

**Figure 7.21 Limiting SLO Recirculation Line Break
Hot Channel Quality at the Hot Node at EOB**

**Figure 7.22 Limiting SLO Recirculation Line Break
Hot Channel Heat Transfer Coeff. at the Hot Node at EOB**

**Figure 7.23 Limiting SLO Recirculation Line Break
Hot Channel Reflood Junction Liquid Mass Flow Rate**

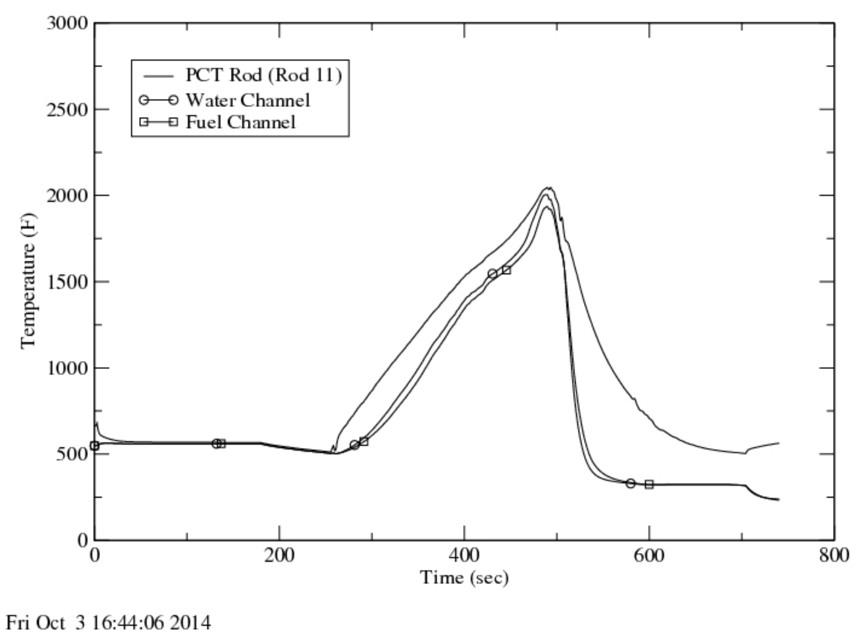


**Figure 7.24 Limiting SLO Recirculation Line Break
Cladding Temperatures**

8.0 Long-Term Coolability

Long-term coolability addresses the issue of reflooding the core and maintaining a water level adequate to cool the core and remove decay heat for an extended time period following a LOCA. For non-recirculation line breaks, the core can be reflooded to the top of the active fuel and be adequately cooled indefinitely. For recirculation line breaks, the core will initially remain covered following reflood due to the static head provided by the water filling the jet pumps to a level of approximately two-thirds core height. Eventually, the heat flux in the core will not be adequate to maintain a two-phase water level over the entire length of the core. Beyond this time, the upper third of the core will remain wetted and adequately cooled by core spray. Maintaining water level at two-thirds core height with one core spray system operating is sufficient to maintain long-term coolability as demonstrated by the NSSS vendor (Reference 7).

The first step in the long-term cooling analysis is to demonstrate that sufficient ECCS flow is always available to fill the inside of the core shroud to two-thirds core height. This was done by reviewing a subset of the breaks including the break that has the minimum mass inventory in the lower plenum region and the reactor core anytime during the event and the large break case with the minimum ECCS flow. The review considered the minimum number of ECCS available for any recirculation line break combined with the effects of axial power shape. In each case, the analysis was extended to 10 minutes and all potential leakage paths were conservatively accounted for. For the minimum ECCS flow case, the limiting break is the 1.0 DEG PS SF-LPCI. For this case, two core sprays are available and provide 9854 gpm at the core spray sparger. The results of these analyses show that in all cases the time to fill the core to the jet pump nozzle elevation (two-thirds core height) was less than 10 minutes.

The second step in the analysis is to demonstrate that after 10 minutes, sufficient ECCS flow is available to remove decay heat and maintain the water level at two-thirds core height. The results of the first step showed that at 10 minutes the vessel pressure dropped far enough to ensure rated core spray (i.e. 4500 gpm) is attained at the core spray sparger (even with leakages considered) and distribution of the core spray to all assemblies. ECCS flow of 3300 gpm is needed to remove decay heat and maintain the water level. This flow can be provided from one core spray; meeting the long-term coolability needs demonstrated by the NSSS vendor discussed above.

In summary, maintaining water level at two-thirds core height with one core spray operating or the core flooded to the top of active fuel is required for long-term cooling beyond 10 minutes for

ATRIUM 10XM fuel. When the core is not flooded to the top of active fuel, the conclusions demonstrate that as long as there is adequate water from LPCI or core spray to maintain the two-thirds core height water level, a single core spray of 4500 gpm to the top of the core is needed to meet the fuel safety limits for long-term core cooling.

9.0 Conclusions

The major conclusions of this LOCA break spectrum analysis are:

- The limiting recirculation line break is a 0.13 ft^2 split break in the pump discharge piping with single failure SF-HPCI and a top-peaked axial shape when operating at 102% rated core power **[ ]** .
- The limiting break analysis identified above satisfies all the acceptance criteria specified in 10 CFR 50.46. The analysis is performed in accordance with 10 CFR 50.46 Appendix K requirements.
- The MAPLHGR limit multiplier for SLO is 0.80 for ATRIUM 10XM fuel. This multiplier ensures that a LOCA from SLO is less limiting than a LOCA from two-loop operation.

The limiting break characteristics determined in this report can be referenced and used in future Quad Cities analyses to establish the MAPLHGR limit versus exposure for ATRIUM 10XM fuel.

10.0 References

1. EMF-2361(P)(A) Revision 0, *EXEM BWR-2000 ECCS Evaluation Model,* Framatome ANP, May 2001.

2. XN-CC-33(A) Revision 1, *HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual*, Exxon Nuclear Company, November 1975.

3. XN-NF-82-07(P)(A) Revision 1, *Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model*, Exxon Nuclear Company, November 1982.

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

5. EMF-2292(P)(A) Revision 0, *ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients*, Siemens Power Corporation, September 2000.

6. *Updated Final Safety Analysis Report*, Quad Cities Nuclear Station.

7. NEDO-20566A*, General Electric Company Analytical Model for Loss of Coolant Analysis in Accordance with 10CFR50 Appendix K*, September 1986.

8. Letter, P. Salas (AREVA) to Document Control Desk (NRC), “Proprietary Viewgraphs and Meeting Summary for Closed Meeting on Application of the EXEM BWR-2000 ECCS Evaluation Methodology,” NRC:11:096, September 22, 2011.

9. Letter, T.J. McGinty (NRC) to P. Salas (AREVA), “Response to AREVA NP, Inc. (AREVA) Proposed Analysis Approach for Its EXEM Boiling Water Reactor (BWR)-2000 Emergency Core Cooling System (ECCS) Evaluation Model,” July 5, 2012.

ATTACHMENT 24

LOCA-ECCS Analysis MAPLHGR Limit Report
(Non-Proprietary)

ANP-3356NP
Revision 0

Quad Cities Units 1 and 2 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel

December 2014

AREVA Inc.

ANP-3356NP
Revision 0

Quad Cities Units 1 and 2
LOCA-ECCS Analysis MAPLHGR
Limits for ATRIUM 10XM Fuel

AREVA Inc.

ANP-3356NP
Revision 0

Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial issue.

Contents

Tables

Figures

Nomenclature

ADS	automatic depressurization system
ANS	American Nuclear Society
BOL	beginning of life
BWR	boiling-water reactor
CFR	Code of Federal Regulations
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOB	end of blowdown
HPCI	high-pressure coolant injection
ID	inside diameter
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MWR	metal-water reaction
NRC	Nuclear Regulatory Commission, U.S.
PCT	peak cladding temperature
RDIV	recirculation discharge isolation valve
SF-ADS	single failure of ADS valve
SF-DGEN	single failure of diesel generator
SF-HPCI	single failure of the HPCI system
SF-LPCI	single failure of an LPCI injection valve
SF-LSL	single failure of the loop selection logic
TCD	thermal conductivity degradation
TCV	turbine control valve
TLO	two-loop operation
TSV	turbine stop valve
VDC	volts direct current

1.0 Introduction

The results of the loss-of-coolant accident emergency core cooling system (LOCA-ECCS) analyses for Quad Cities Units 1 and 2 are documented in this report. The results provide the maximum average planar linear heat generation rate (MAPLHGR) limits for ATRIUM™ 10XM* fuel as a function of exposure and demonstrate the MAPLHGR limits are adequate to ensure that the LOCA-ECCS criteria in 10 CFR 50.46 are satisfied for operation at or below the limit. MAPLHGR limits are established for normal (two-loop) operation and for operation when one of the recirculation loops is out-of-service (single-loop). The report also documents the licensing basis peak cladding temperature (PCT) and corresponding local cladding oxidation from the metal-water reaction (MWR) for ATRIUM 10XM fuel at Quad Cities Units 1 and 2.

The analyses documented in this report were performed with LOCA Evaluation Models developed by AREVA Inc. (AREVA) and approved for reactor licensing analyses by the U.S. Nuclear Regulatory Commission (NRC). The models and computer codes used by AREVA for LOCA analyses are collectively referred to as the EXEM BWR-2000 Evaluation Model. The EXEM BWR-2000 Evaluation Model and NRC approval are documented in Reference 2. A summary description of the LOCA analysis methodology is provided in Section 4.0.

[

]

The application of the EXEM BWR-2000 Evaluation Model for the Quad Cities LOCA break spectrum analysis is documented in Reference 1. The LOCA conditions evaluated in Reference 1 include break size, type, location, axial power shape, and ECCS single failure. The limiting LOCA break characteristics identified in Reference 1 are presented below:

Limiting LOCA Break Characteristics	
Location	Recirculation discharge pipe
Type / size	Split break / 0.13 ft^2
Single failure	HPCI
Axial power shape	Top-peaked
Initial state	102% power/ **[ ]**

* ATRIUM is a trademark of AREVA Inc.

The LOCA break spectrum analysis documented in Reference 1 was based on a generic ATRIUM 10XM neutronic design at beginning of life conditions. The PCT and MWR calculated for a fuel rod experiencing the fluid conditions during the limiting LOCA are affected by fuel characteristics that depend on the fuel assembly neutronic design and exposure (e.g., local rod power, stored energy). The fuel assembly heatup analysis results presented in this report are for the limiting (minimum margin to acceptance criteria) ATRIUM 10XM neutronic design currently designed for use at Quad Cities. This includes all of the ATRIUM 10XM enriched lattices in the representative Quad Cities Unit 2 cycle design. The heatup analyses were performed using the fluid conditions from the limiting LOCA identified in Reference 1. Cycle-specific heatup analyses are performed to confirm that the applicability of the MAPLHGR limits for future nuclear designs. The results of cycle-specific heatup analyses will be reported in the Quad Cities cycle-specific reload safety analysis report.

The impact on LOCA of operation with equipment out-of-service and their combinations as per the COLR have been considered. The LOCA analyses include the effects of 1 relief valve out-of-service. Operation with an ADS valve out-of-service is not currently allowed by the Quad Cities Technical Specification. Since the consequence of a LOCA would be more severe during operation with one of the recirculation lines out-of-service, a MAPLHGR multiplier is applied to support single-loop operation (SLO). Operation with other allowed equipment out-of-service conditions are supported (turbine bypass valves, feedwater heaters, one MSIV, power load unbalance, pressure controller, TCV slow closure, and one stuck closed TCV or TSV).

2.0 Summary

The MAPLHGR limits were determined by applying the EXEM BWR-2000 Evaluation Model for the analysis of the limiting LOCA event. The exposure-dependent MAPLHGR limits for ATRIUM 10XM fuel are shown in Figure 2.1. As indicated, one MAPLHGR limit is applied to the fully rodded lattices (91 fuel rods) and a slightly different MAPLHGR limit is applied to the partially rodded lattices (79 fuel rods). The fully rodded lattices (enriched and natural) are often referred to as bottom lattices and the partially rodded lattices (enriched and natural) are often referred to as top lattices. The results of these calculations confirm that the LOCA acceptance criteria in the Code of Federal Regulations (10 CFR 50.46) are met for operation at or below these MAPLHGR limits.

The MAPLHGR analysis results for the limiting lattice design are presented in Section 5.0. The peak cladding temperature (PCT) and the maximum local cladding oxidation results for the ATRIUM 10XM limiting lattice design are presented in Table 2.1.

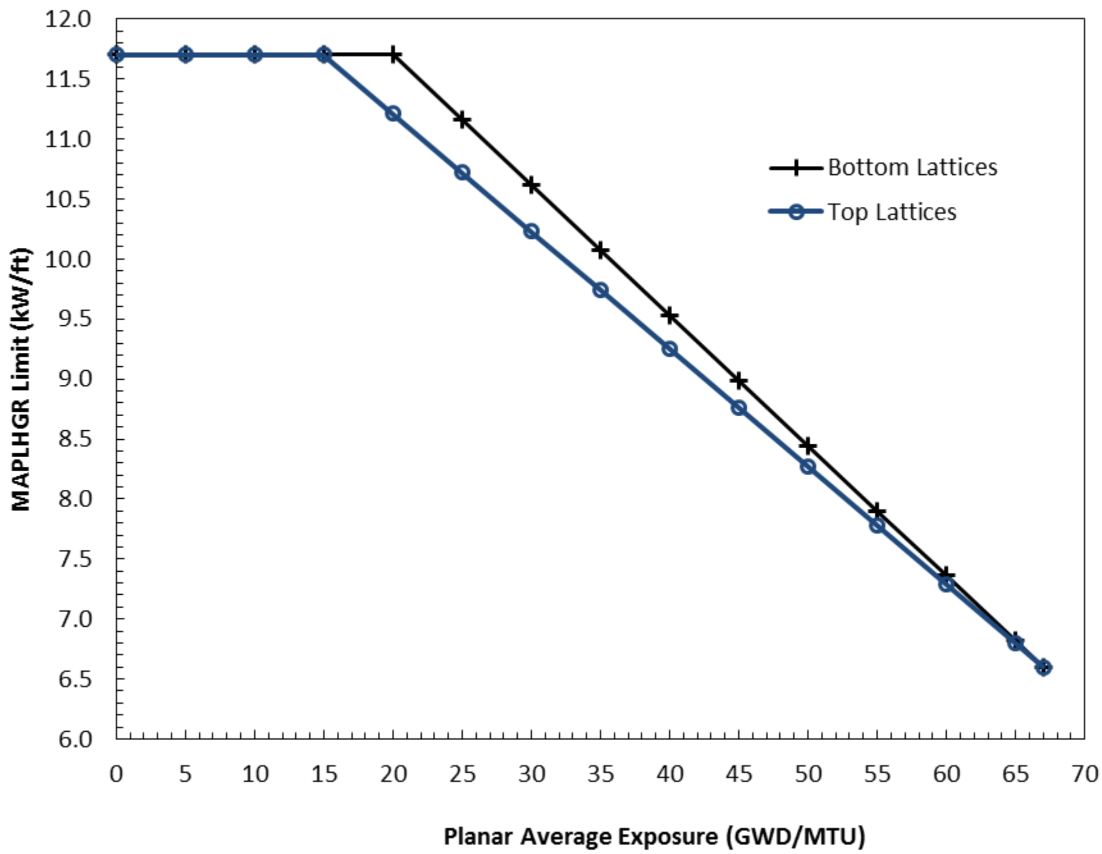
Operation with only one recirculation loop (single-loop operation) requires that a MAPLHGR multiplier of 0.80 be applied to the two-loop operation MAPLHGR limits.

[

]

**Table 2.1 LOCA Results for
PCT and Local Cladding Oxidation**

Parameter	ATRIUM 10XM
Peak cladding temperature (°F)	2138°F
Local cladding oxidation (max %)	4.05%
Total hydrogen generated (% of total hydrogen possible)	< 1.0%



Average Planar Exposure (GWd/MTU)	ATRIUM 10XM MAPLHGR (kW/ft) Bottom Lattices (91 fuel rods) (natural & enriched)
0.0	11.7
20.0	11.7
67.0	6.6

Average Planar Exposure (GWd/MTU)	ATRIUM 10XM MAPLHGR (kW/ft) Top Lattices (79 fuel rods) (natural & enriched)
0.0	11.7
15.0	11.7
67.0	6.6

Figure 2.1 MAPLHGR Limits for Two-Loop Operation*

* A MAPLHGR multiplier of 0.8 is required for single-loop operation.

3.0 LOCA Description

3.1 *Accident Description*

The LOCA is described in the Code of Federal Regulations 10 CFR 50.46 as a hypothetical accident that results in a loss of reactor coolant from breaks in reactor coolant pressure boundary piping up to and including a break equivalent in size to a double-ended rupture of the largest pipe in the reactor coolant system. There is not a specifically identified cause that results in the pipe break. However, for the purpose of identifying a design basis accident, the pipe break is postulated to occur inside the primary containment before the first isolation valve.

For a boiling water reactor (BWR), a LOCA may occur over a wide spectrum of break locations and sizes. Responses to the break vary significantly over the break spectrum. The largest possible break is a double-ended rupture of a recirculation pipe; however, this is not necessarily the most severe challenge to the emergency core cooling system (ECCS). A double-ended rupture of a main steam line causes the most rapid primary system depressurization, but because of other phenomena, steam line breaks are seldom limiting with respect to the event acceptance criteria (10 CFR 50.46). In addition to break location dependence, different break sizes in the same pipe produce quite different event responses, and the largest break area is not necessarily the most severe challenge to the event acceptance criteria. Because of these complexities, an analysis covering the full range of break sizes and locations is required. The results of the Quad Cities Units 1 and 2 ATRIUM 10XM break spectrum calculations using the EXEM BWR-2000 LOCA methodology are summarized in Reference 1.

Regardless of the initiating break characteristics, the event response is conveniently separated into three phases: the blowdown phase, the refill phase, and the reflood phase. The relative duration of each phase is strongly dependent upon the break size and location. The last two phases are often combined and will be discussed together in this report.

During the blowdown phase of a LOCA, there is a net loss-of-coolant inventory, an increase in fuel cladding temperature due to core flow degradation, and for the larger breaks, the core becomes fully or partially uncovered. There is a rapid decrease in pressure during the blowdown phase. During the early phase of the depressurization, the exiting coolant provides core cooling. Later in the blowdown, core cooling is provided by the injection of ECCS flows and by lower plenum flashing as the system continues to depressurize. The blowdown phase is defined to end when rated LPCS flow is attained.

In the refill phase of a LOCA, the ECCS is functioning and there is a net increase of coolant inventory. During this phase the core sprays provide core cooling and, along with low-pressure and high-pressure coolant injection (LPCI and HPCI), supply liquid to refill the lower portion of the reactor vessel. In general, the core heat transfer to the coolant is less than the fuel decay heat rate and the fuel cladding temperature continues to increase during the refill phase.

In the reflood phase, the coolant inventory has increased to the point where the mixture level reenters the core region. During the core reflood phase, cooling is provided above the mixture level by entrained reflood liquid and below the mixture level by pool boiling. Sufficient coolant eventually reaches the core hot node and the fuel cladding temperature decreases.

3.2 *Acceptance Criteria*

A LOCA is a potentially limiting event that may place constraints on fuel design, local power peaking, and in some cases, acceptable core power level. During a LOCA, the normal transfer of heat from the fuel to the coolant is disrupted. As the liquid inventory in the reactor decreases, the decay heat and stored energy of the fuel cause a heatup of the undercooled fuel assembly. In order to limit the amount of heat that can contribute to the heatup of the fuel assembly during a LOCA, an operating limit on the MAPLHGR is applied to each fuel assembly in the core.

The Code of Federal Regulations prescribes specific acceptance criteria (10 CFR 50.46) for a LOCA event as well as specific requirements and acceptable features for Evaluation Models (10 CFR 50 Appendix K). The conformance of the EXEM BWR-2000 LOCA Evaluation Models to Appendix K is described in Reference 2. The ECCS must be designed such that the plant response to a LOCA meets the following acceptance criteria specified in 10 CFR 50.46:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- The calculated local oxidation of the cladding shall nowhere exceed 0.17 times the local cladding thickness.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume, were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria are commonly referred to as the peak cladding temperature (PCT) criterion, the local oxidation criterion, the hydrogen generation criterion, the coolable geometry criterion, and the long-term cooling criterion. MAPLHGR limits are established to ensure that these criteria are met. For jet pump BWRs, the most challenging criterion is that PCT must not exceed 2200°F.

LOCA analysis results demonstrating that the PCT, local oxidation, and hydrogen generation criteria are met are provided in Section 5.0. Compliance with these three criteria ensures that a coolable geometry is maintained. Compliance with the long-term coolability criterion is discussed in Reference 1.

4.0 LOCA Analysis Description

The Evaluation Model used for the break spectrum analysis is the EXEM BWR-2000 LOCA analysis methodology described in Reference 2. The EXEM BWR-2000 methodology employs three major computer codes to evaluate the system and fuel response during all phases of a LOCA. These are the RELAX, HUXY, and RODEX2 computer codes. RELAX is used to calculate the system and hot channel response during the blowdown, refill, and reflood phases of the LOCA. The HUXY code is used to perform heatup calculations for the entire LOCA, and calculates the PCT and local clad oxidation at the axial plane of interest. RODEX2 is used to determine fuel parameters (such as stored energy) for input to the other LOCA codes. The code interfaces for the LOCA methodology are illustrated in Figure 4.1.

A complete analysis for a given break size starts with the specification of fuel parameters using RODEX2 (Reference 3). RODEX2 is used to determine the initial stored energy for both the blowdown analysis (RELAX hot channel) and the heatup analysis (HUXY). This is accomplished by ensuring that the initial stored energy in RELAX and HUXY is the same or higher than that calculated by RODEX2 for the power, exposure, and fuel design being considered.

4.1 *Blowdown Analysis*

The RELAX code (Reference 2) is used to calculate the system thermal-hydraulic response during the blowdown phase of the LOCA. For the system blowdown analysis, the core is represented by an average core channel. The reactor core is modeled with heat generation rates determined from reactor kinetics equations with reactivity feedback and with decay heating as required by Appendix K of 10 CFR 50. The reactor vessel nodalization for the system blowdown analysis is shown in Figure 4.3. This nodalization is consistent with that used in the topical report submitted to the NRC (Reference 2).

The RELAX analysis is performed from the time of the break initiation through the end of blowdown (EOB). The system blowdown calculation provides the upper and lower plenum transient boundary conditions for the hot channel analysis.

Following the system blowdown calculation, another RELAX analysis is performed to analyze the maximum power assembly (hot channel) of the core. The RELAX hot channel blowdown calculation determines hot channel fuel, cladding, and coolant temperatures during the blowdown phase of the LOCA. The RELAX hot channel nodalization in Figure 4.4 is used to model a top-peaked power shape, which is the axial power shape for the limiting break conditions identified in the break spectrum analysis (Reference 1). The hot channel analysis is performed using the system blowdown results to supply the core power and the system boundary conditions at the core inlet and exit. The initial average fuel rod temperature at the limiting plane of the hot channel is conservative relative to the average fuel rod temperature calculated by RODEX2 for operation of the ATRIUM 10XM assembly at the MAPLHGR limit. The heat transfer coefficients and fluid conditions at the limiting plane of the RELAX hot channel calculation are used as input to the HUXY heatup analysis.

4.2 *Refill / Reflood Analysis*

The RELAX code is also used to compute the system and hot channel hydraulic response during the refill/reflood phase of the LOCA. The RELAX system and RELAX hot channel analyses continue beyond the end of blowdown to analyze system and hot channel responses during the refill and reflood phases. The refill phase is the period when the lower plenum is filling due to ECCS injection. The reflood phase is the period when some portions of the core and hot assembly are being cooled with ECCS water entering from the lower plenum. The purpose of the RELAX calculations beyond blowdown is to determine the time when the liquid flow via upward entrainment from the bottom of the core becomes high enough at the hot node in the hot assembly to end the temperature increase of the fuel rod cladding. This event time is called the time of hot node reflood. **[**

] The time when the core bypass mixture level rises to the elevation of the hot node in the hot assembly is also determined.

RELAX provides a prediction of fluid inventory during the ECCS injection period. Allowing for countercurrent flow through the core and bypass, RELAX determines the refill rate of the lower plenum due to ECCS water and the subsequent reflood times for the core, hot assembly, and the core bypass. The RELAX calculations provide HUXY with the time of hot node reflood and the time when the liquid has risen in the bypass to the height of the axial plane of interest (time of bypass reflood).

4.3 *Heatup Analysis*

The HUXY code (Reference 4) is used to perform heatup calculations for the entire LOCA transient and provides PCT and local clad oxidation at the axial plane of interest. The heat generated by metal-water reaction (MWR) is included in the HUXY analysis. HUXY is used to calculate the thermal response of each fuel rod in one axial plane of the hot channel assembly.

These calculations consider thermal-mechanical interactions within the fuel rod. The clad swelling and rupture models from NUREG-0630 have been incorporated into HUXY (Reference 5). The HUXY code complies with the 10 CFR 50 Appendix K criteria for LOCA Evaluation Models.

[

]

HUXY uses the EOB time and the times of core bypass reflood and core reflood at the axial plane of interest from the RELAX analysis. Until the EOB, HUXY uses RELAX hot channel heat transfer coefficients, fluid temperatures, fluid qualities, and power. Throughout the calculations, decay power is determined based on the ANS 1971 decay heat curve plus 20% as described in Reference 2. After the EOB and prior to the time of hot node reflood, HUXY uses Appendix K spray heat transfer coefficients for the fuel rods, water canister and fuel channel. Experimental data for AREVA 10x10 fuel which supports the use of the convective heat transfer coefficients listed in Appendix K is documented in Reference 6. After the time of hot node reflood, Appendix K reflood heat transfer coefficients are used in the HUXY analysis. The principal results of a HUXY heatup analysis are the PCT and the percent local oxidation of the fuel cladding, often called the percent maximum local metal-water reaction (%MWR). An evaluation of the total hydrogen generated is also performed.

4.4 [ ]

[

]

[

]

4.4.1 Calculation Approach

[

]

4.5 *Plant Parameters*

The LOCA break spectrum analysis is performed using plant parameters provided by Exelon. Table 4.1 provides a summary of reactor initial conditions that were determined to be limiting in the break spectrum analysis. Table 4.2 lists selected reactor system parameters.

The break spectrum analysis is performed for a full core of ATRIUM 10XM fuel. Some of the key fuel parameters used in the break spectrum analysis are summarized in Table 4.3. The break spectrum calculations identified that the most limiting conditions include a top-peaked axial power shape (Reference 1). The boundary conditions from the limiting break conditions are used during the exposure dependent MAPLHGR analyses for bottom and top lattices. In order to put the bottom and top lattices at 102% of the MAPLHGR limit in the HUXY heatup calculations, bottom lattices use a mid-peaked power shape and top lattices use a top-peaked power shape (Figure 4.6).

4.6 *ECCS Parameters*

The ECCS configuration is shown in Figure 4.5. Table 4.4 –Table 4.7 provide the important ECCS characteristics assumed in the LOCA break spectrum analysis. The ECCS is modeled as fill junctions connected to the appropriate reactor locations: LPCS injects into the upper plenum, HPCI injects into the upper downcomer, and LPCI injects into the recirculation lines.

The flow through each ECCS valve is determined based on system pressure and valve position. Flow versus pressure for a fully open valve is obtained by linearly interpolating the pump capacity data provided in Table 4.4 – Table 4.6. No HPCI flow is credited until the injection valve is fully open. LPCI and LPCS flow is governed by the valve characteristics provided by Exelon. Also, no credit for ECCS flow is assumed until ECCS pumps reach rated speed.

[

]

The automatic depressurization system (ADS) valves are modeled as a junction connecting the reactor steam line to the suppression pool. The flow through the ADS valves is calculated based on pressure and valve flow characteristics. The valve flow characteristics are determined such that the calculated flow is equal to the rated capacity at the reference pressure shown in Table 4.7. All five ADS valves are assumed operable during the LOCA except when a single failure is assumed to prevent one ADS valve from opening.

In the AREVA LOCA analysis model, ECCS initiation is assumed to occur when the water level drops to the applicable level setpoint. No credit is assumed for the start of LPCS or LPCI due to high drywell pressure. [

]

Recirculation discharge isolation valve (RDIV) and loop selection logic parameters are presented in Table 4.8. For recirculation line breaks sizes ≥ 0.15 ft^2, the loop selection logic directs all available LPCI flow to the intact loop and closes the RDIV in the intact loop. For break sizes < 0.15 ft^2, all available LPCI flow is assumed to be injected into the broken loop and the RDIV in the broken loop is closed.

The single failures and the available ECCS for each failure are summarized in Table 4.9. The potential single failures are discussed in Section 5.1 of Reference 1.

Table 4.1 Initial Conditions*

Reactor power (% of rated)	102
[	]
Reactor power (MWt)	3016.14
[	]
[	]
Steam flow rate (Mlb/hr)	11.98
Steam dome pressure (psia)	1020
Core inlet enthalpy (Btu/lb)	521.6
ATRIUM 10XM hot assembly MAPLHGR (kW/ft)	11.7
[	]

* The AREVA calculated heat balance is adjusted to match the heat balance at 100% power and 100% core flow. AREVA heat balance calculations establish these initial conditions at the stated power and flow. Initial conditions are based on nominal values, except for initial core power and dome pressure which use conservative values.

[ ]

Table 4.2 Reactor System Parameters

Parameter	Value
Vessel ID (in)	251
Number of fuel assemblies	724
Recirculation suction pipe area (ft^2)	3.581
Recirculation discharge pipe area (ft^2)	3.477

Table 4.3 ATRIUM 10XM Fuel Assembly Parameters

Parameter	Value
Fuel rod array	10x10
Number of fuel rods per assembly	79 (full-length rods) 12 (part-length rods)
Non-fuel rod type	Water channel replaces 9 fuel rods
Fuel rod OD (in)	0.4047
Active fuel length (in) (including blankets)	145.24 (full-length rods) 75.0 (part-length rods)
Water channel outside width (in)	1.378
Fuel channel thickness (in)	0.075 (minimum wall) 0.100 (corner)
Fuel channel internal width (in)	5.278

Table 4.4 High-Pressure Coolant Injection Parameters

Parameter	Value
Coolant temperature (maximum) (°F)	140
Initiating Signals and Setpoints	
Water level (in)*	444
High drywell pressure (psig)†	2.5
Time Delays	
Time for HPCI pump to reach rated speed and injection valve wide open (sec)	55
Coolant Flow Rate Vs. Pressure	
Vessel to Torus ΔP (psid)	Flow Rate (gpm)
0	0
150	5000
1120	5000

* Relative to vessel zero.

† The time when the drywell reaches 2.5 psig is not explicitly calculated in the EXEM BWR-2000 LOCA methodology. [ ]

Table 4.5 Low-Pressure Coolant Injection Parameters

Parameter	Value
Reactor pressure permissive for opening injection valves - analytical (psig)	300
Coolant temperature (maximum) (°F)	160
Initiating Signals and Setpoints	
Water level (in)*	444
High drywell pressure (psig)†	2.5
Time Delays	
Initiating signal processing delay	1
Time from EDG started and initiating signal to LPCI pump at rated speed (sec)	14
Time for power at the injection valve (sec)	26
LPCI injection valve stroke time (sec)	28

Coolant Flow Rate‡ Vs. Pressure

Vessel to Torus ΔP (psid)	Flow Rate for 2 Pumps (gpm)	Flow Rate for 4 Pumps (gpm)
0	9,300	15,700
20	9,000	15,200
150	6,200	10,200
257	0	0

* Relative to vessel zero.

† The time when the drywell reaches 2.5 psig is not explicitly calculated in the EXEM BWR-2000 LOCA methodology. [ ]

‡ [ ]

Table 4.6 Low-Pressure Core Spray Parameters

Parameter	Value
Reactor pressure permissive for opening injection valves - analytical (psig)	300
Coolant temperature (maximum) (°F)	160
Initiating Signals and Setpoints	
Water level (in)*	444
High drywell pressure (psig)†	2.5
Time Delays	
Initiating signal processing delay	1
Time from EDG started and initiating signal to LPCS pump at rated speed (sec)	17
Time for power at the injection valve (sec)	17
LPCS injection valve stroke time (sec)	53
Coolant Flow Rate‡ Vs. Pressure	
Vessel to Torus ΔP (psid)	Flow Rate for 1 Pump (gpm)
0	5,650
90	4,500
200	3,000
325	0

* Relative to vessel zero.

† The time when the drywell reaches 2.5 psig is not explicitly calculated in the EXEM BWR-2000 LOCA methodology. **[** **]**

‡ **[** **]**

Table 4.7 Automatic Depressurization System Parameters

Parameter	Value
Number of valves installed	5
Number of valves available*	5
Minimum flow capacity per valve of available valves (4 relief) (lbm/hr at psig)	558,000 at 1120
Minimum flow capacity of available valves (1 Target Rock) (lbm/hr at psig)	598,000 at 1080
Initiating Signals and Setpoints	
Water level (in)†	444
High drywell pressure (psig)‡	2.5
Time Delays	
ADS timer (delay time from initiating signal to time valves are open) (sec)	120

* All 5 valves are assumed operable in the analyses except when analyzing the potential single failure of 1 ADS valve during the LOCA.

† Relative to vessel zero.

‡ The time when the drywell reaches 2.5 psig is not explicitly calculated in the EXEM BWR-2000 LOCA methodology. **[**

]

Table 4.8 Recirculation Discharge Isolation Valve and LPCI Loop Selection Logic Parameters

Parameter	Value
Minimum break area for loop selection logic to select intact loop* (ft^2)	0.15
LPCI loop selection logic pressure permissive (minimum) (psig)	860
Time to power at the RDIV (sec)	26
RDIV stroke time (sec)	48

* For break sizes ≥ 0.15 ft^2, loop selection logic opens the LPCI injection valves in the intact loop and closes the RDIV in the intact loop. For break sizes < 0.15 ft^2, the available LPCI flow is assumed to be injected into the broken loop and the RDIV in the broken loop is closed.

Table 4.9 Available ECCS for Recirculation Line Break LOCAs

Assumed Failure	Systems Remaining*, †
LPCI injection valve (SF-LPCI)	2 LPCS + HPCI + 5 ADS
Diesel generator or 125-VDC (SF-DGEN)	1 LPCS + 2 LPCI + HPCI + 5 ADS
HPCI system (SF-HPCI)	2 LPCS + 4 LPCI + 5 ADS
Loop select logic (SF-LSL)	2 LPCS + 4 LPCI + HPCI + 5 ADS
ADS valve (SF-ADS)	2 LPCS + 4 LPCI + HPCI + 4 ADS

* Systems remaining, as identified in this table for recirculation line breaks, are applicable to all non-ECCS line breaks. For ECCS line breaks, the systems remaining are those listed for recirculation breaks, less the ECCS in which the break is assumed

† With loop selection logic operational, all available LPCI flow is directed to the intact loop for breaks ≥ 0.15 ft^2. All available LPCI flow is directed to the broken loop for breaks < 0.15 ft^2. The limiting condition for a loop selection logic failure would result in all available LPCI flow directed to the broken loop for all break sizes.

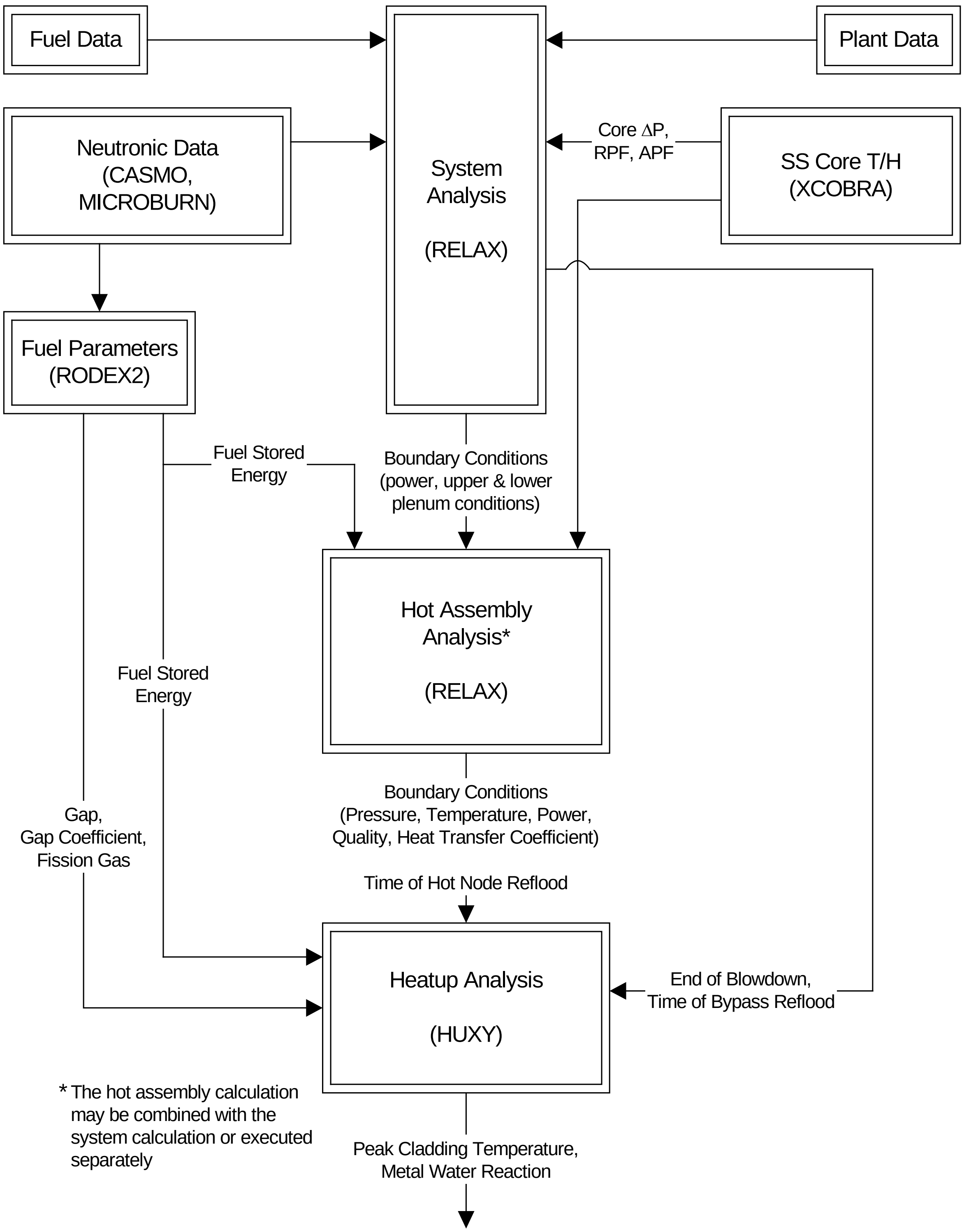


Figure 4.1 Flow Diagram for EXEM BWR-2000 ECCS Evaluation Model

Figure 4.2 [

]

Figure 4.3 RELAX System Blowdown Model

Figure 4.4 RELAX Hot Channel Model
Top-Peaked Axial

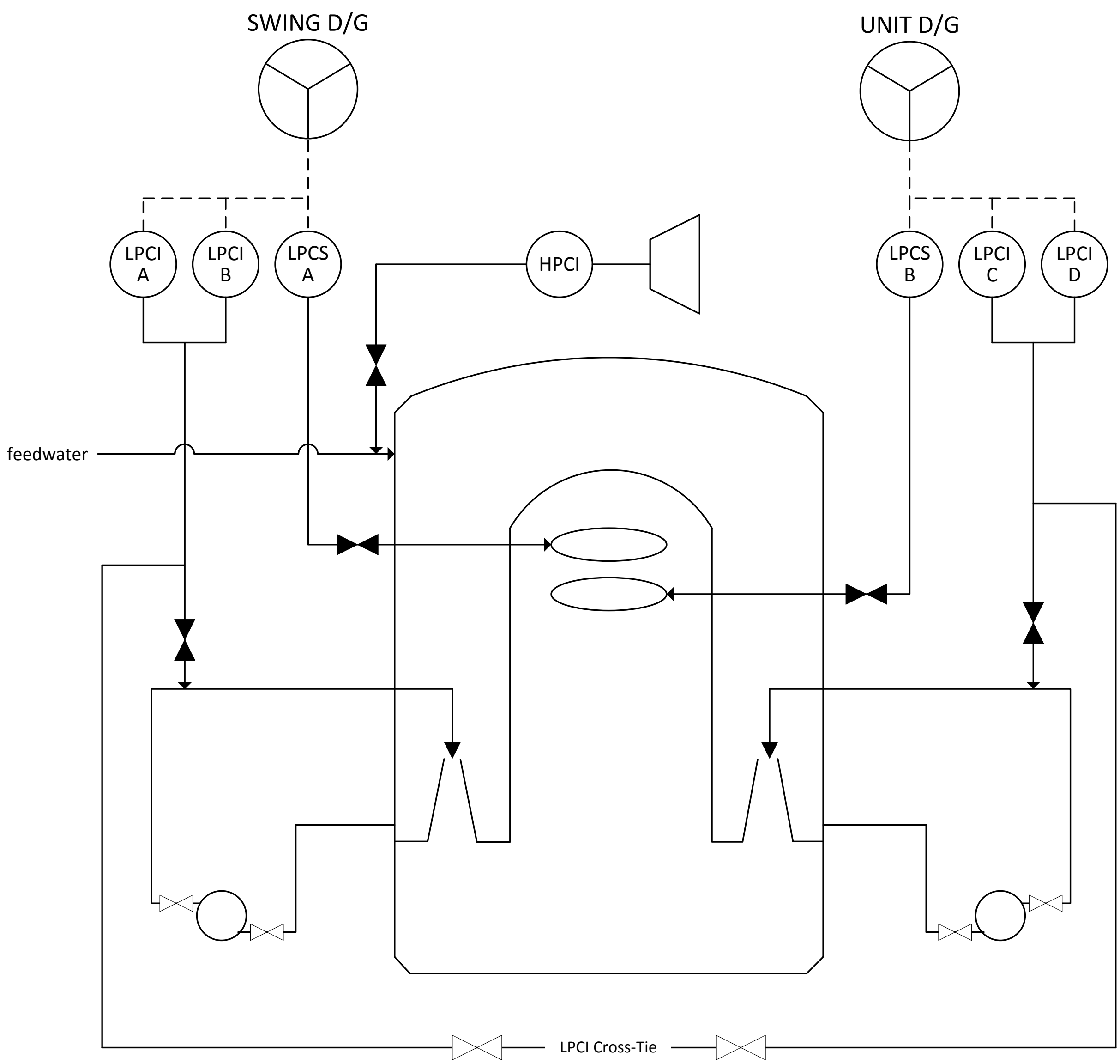


Figure 4.5 ECCS Schematic

Figure 4.6 Rod Average Power Distributions for 102%P [ ] Mid- and Top-Peaked

5.0 MAPLHGR Analysis Results

Exposure-dependent MAPLHGR limits for ATRIUM 10XM fuel are obtained by performing HUXY heatup analyses using results from the limiting LOCA analysis case identified in Reference 1. The break characteristics for the limiting analysis are summarized in Section 1.0. Table 5.1 shows event times for the analysis. The response of the reactor system is shown in Figures 5.1 – 5.23. In the MAPLHGR analysis, the ATRIUM 10XM fuel rod stored energy is set to be bounding at all exposures and the RELAX hot channel peak power node is modeled at a MAPLHGR at or above 102% of the MAPLHGR limit for a given average planar exposure. For example, the RELAX hot channel peak power node is modeled at 102% of 11.7 kW/ft for the bottom peaked lattices from 0 – 20 GWd/MTU. **[**

]

Table 5.2 shows the MAPLHGR analysis results for the ATRIUM 10XM fuel. The HUXY model of the ATRIUM 10XM fuel is applied to obtain these results as described in Section 4.3. The HUXY analysis is performed at average planar exposure intervals between 0 and 67 GWd/MTU. Some of the Quad Cities neutronic lattice designs experienced higher PCTs than the generic ATRIUM 10XM lattice design used in the Quad Cities break spectrum analyses reported in Reference 1, but remain within 10 CFR 50.46 limits. The HUXY MAPLHGR input is consistent with the data in Figure 2.1. Exposure-dependent ATRIUM 10XM fuel rod data is provided from RODEX2 results and includes gap coefficient, hot gap thickness, cold gap thickness, gas moles, fuel rod plenum length, and spring relaxation time. This data is provided as a function of linear heat generation rate at each exposure analyzed.

The ATRIUM 10XM limiting PCT is 2138°F at the 0.0 GWd/MTU exposure. The corresponding maximum local cladding oxidation at the PCT limiting exposure is 3.20%. The maximum local cladding oxidation of 4.05% occurred at 30 GWd/MTU. The limiting results are for a fully-rodded or bottom lattice. Analysis results show the total hydrogen generated is less than 1.0% of the hypothetical amount that would be generated if all cladding surrounding the fuel were to react.

Figure 5.24 shows temperatures for the limiting ATRIUM 10XM neutronic lattice as a function of time for the limiting break. The PCT is 2138°F and occurs at 452.5 seconds. These results demonstrate the acceptability of the ATRIUM 10XM MAPLHGR limit shown in Figure 2.1.

5.1 *Thermal Conductivity Degradation*

The RODEX2 code was approved by the NRC in the early 1980s. At that time, thermal conductivity degradation with burnup was not well characterized by irradiation or post-irradiation testing. As a result, fuel codes at that time did not account for thermal conductivity degradation (TCD). In the past 20 years, requests to the NRC have been made for commercial fuel operation to increasingly higher burnup levels. This has resulted in renewed interest in the degree and nature of burnup-induced TCD. Interactions between AREVA and the NRC on this topic are summarized in Appendix F of Reference 9, which describes how TCD is addressed in the AREVA analyses for Quad Cities.

[

]

The impact of TCD is included in the results summarized in Table 5.2. The assessment for Quad Cities ATRIUM 10XM fuel shows that the PCT calculated at 0.0 GWd/MTU, which is not affected by TCD, is still the highest exposure-dependent PCT when the effects of TCD are included.

Table 5.1 Event Times for Limiting TLO Recirculation Line Break 0.13 ft^2 Split Pump Discharge SF-HPCI Top-Peaked Axial 102% Power [ ]

Event	Time (sec)
Initiate Break	0.0
Initiate Scram	0.6
Diesel Generators Started	17.0
Low-Low Liquid Level, L2 (444 in)	48.5
Power at LPCS Injection Valves	17.0
ADS with ECCS Pump(s) Running	56.5
LPCS Pump at Rated Speed	66.5
LPCS High-Pressure Cutoff	338.4
LPCS Valve Pressure Permissive	348.5
LPCS Valve Starts to Open	348.5
LPCS Flow Starts	351.2
Top Down Cooling Restriction Begins in the Hot Channel	351.2
LPCS Valve Fully Open	401.5
Rated LPCS Flow	549.1
LPCI Pump at Rated Speed	63.5
LPCI Valve Pressure Permissive	348.5
LPCI Valve Starts to Open	348.5
LPCI Flow Starts	366.7
LPCI Valve Fully Open	376.5
Jet Pump Uncovers	139.8
Recirculation Suction Uncovers	414.1
ADS Valves Open	169.5
RDIV Pressure Permissive	207.8
RDIV Starts to Close	207.8
RDIV Closed	255.8
PCT	452.5
Time of Bypass Reflood	513.1
Time of Rated Spray	549.1
Time of Hot Node Reflood	549.1

Table 5.2 ATRIUM 10XM MAPLHGR Analysis Results

Average Planar Exposure (GWd/MTU)	PCT (ºF)	Local Cladding Oxidation (%)
0	2138	3.20*
5	2118	2.93
10	2105	2.71
15	2097	2.56
20	2092	2.52
25	2018	2.00
30	2008	4.05
40	1945	3.46
50	1952	2.82
60	1844	2.24
67	1804	1.87

* The corresponding planar average cladding oxidation is 2.35%.

Figure 5.1 Limiting TLO Recirculation Line Break Upper Plenum Pressure

Figure 5.2 Limiting TLO Recirculation Line Break Total Break Flow Rate

**Figure 5.3 Limiting TLO Recirculation Line Break
Core Inlet Flow Rate**

**Figure 5.4 Limiting TLO Recirculation Line Break
Core Outlet Flow Rate**

**Figure 5.5 Limiting TLO Recirculation Line Break
Intact Loop Jet Pump Exit Flow Rate**

**Figure 5.6 Limiting TLO Recirculation Line Break
Broken Loop Jet Pump Exit Flow Rate**

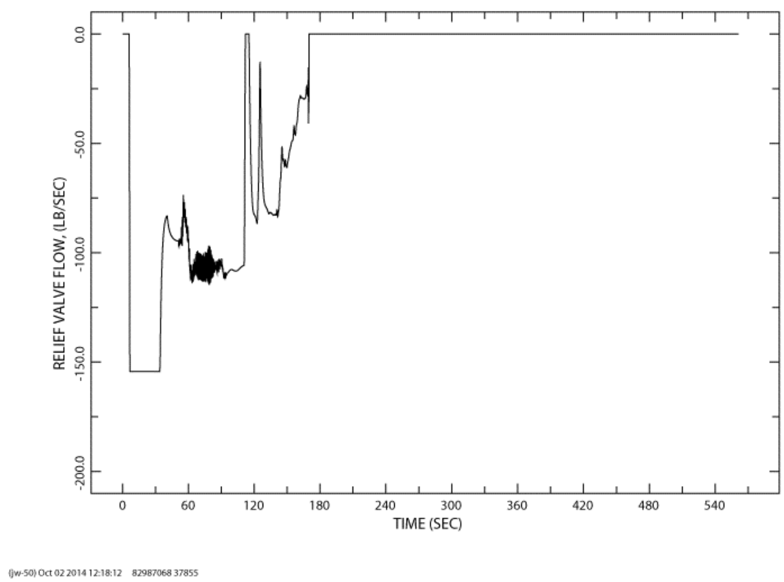


Figure 5.7 Limiting TLO Recirculation Line Break Relief Valve Flow Rate

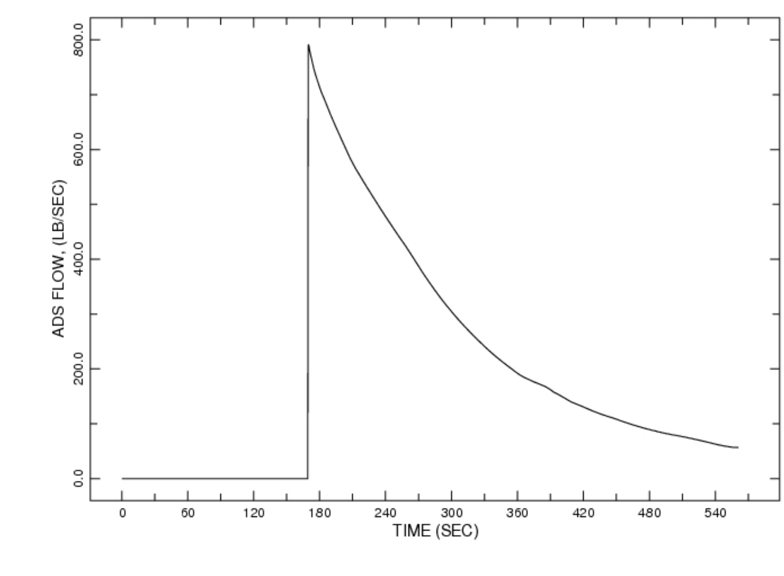
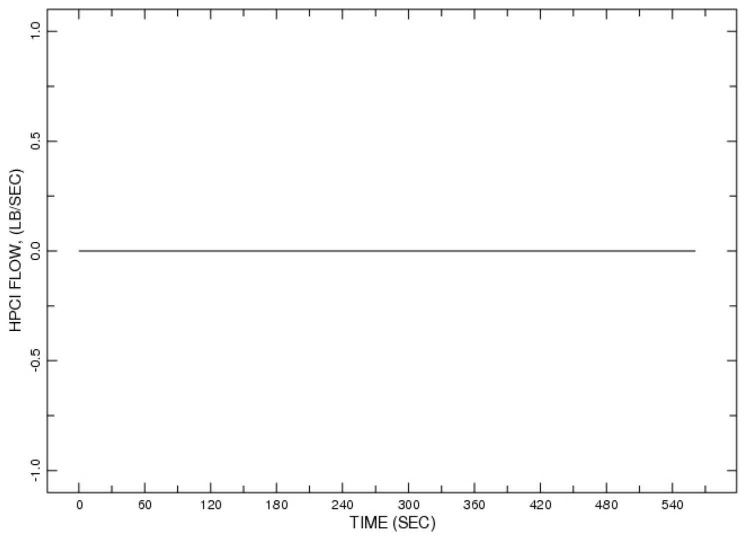
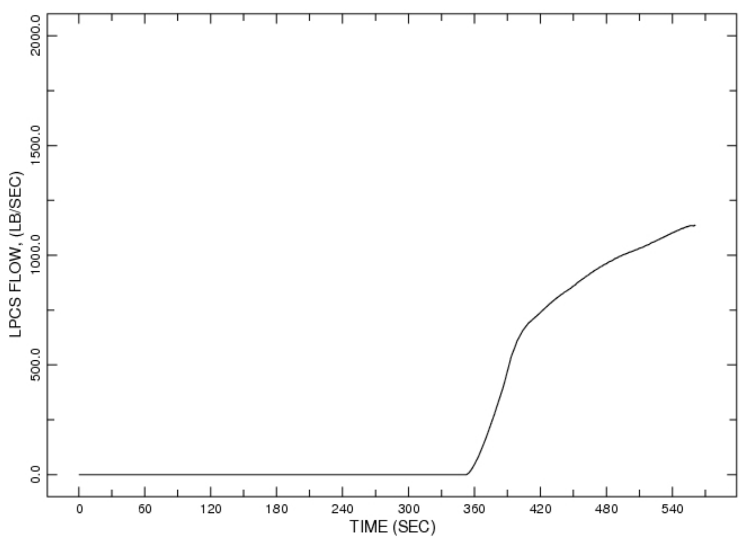


Figure 5.8 Limiting TLO Recirculation Line Break ADS Flow Rate



(jw-52) Oct 02 2014 12:18:12 82987068 37855

Figure 5.9 Limiting TLO Recirculation Line Break
HPCI Flow Rate



(jw-51) Oct 02 2014 12:18:12 82987068 37855

Figure 5.10 Limiting TLO Recirculation Line Break
LPCS Flow Rate

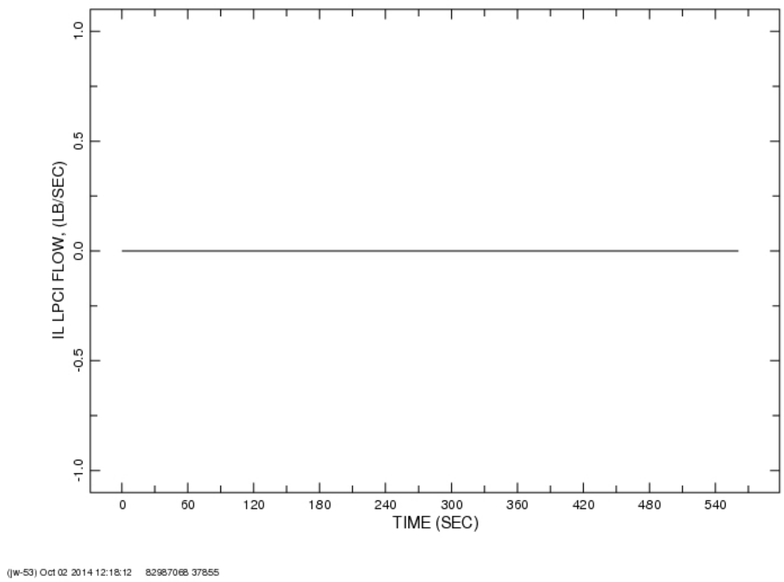


Figure 5.11 Limiting TLO Recirculation Line Break Intact Loop LPCI Flow Rate

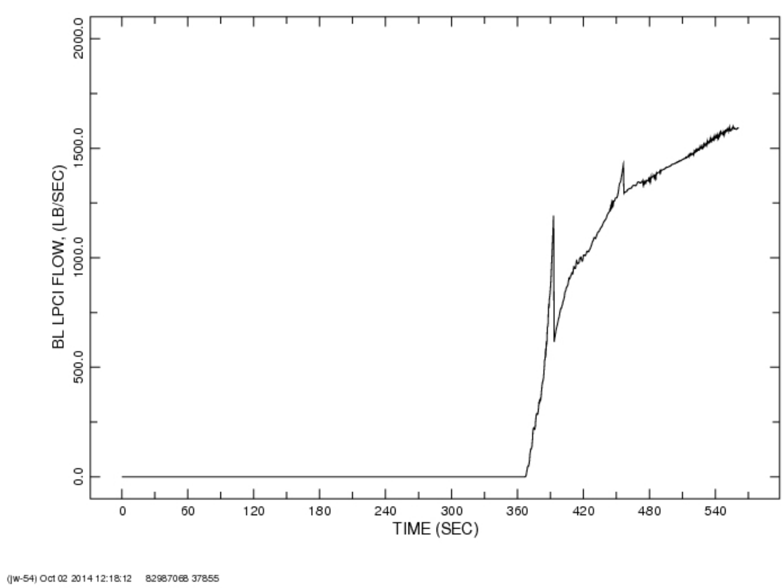


Figure 5.12 Limiting TLO Recirculation Line Break Broken Loop LPCI Flow Rate

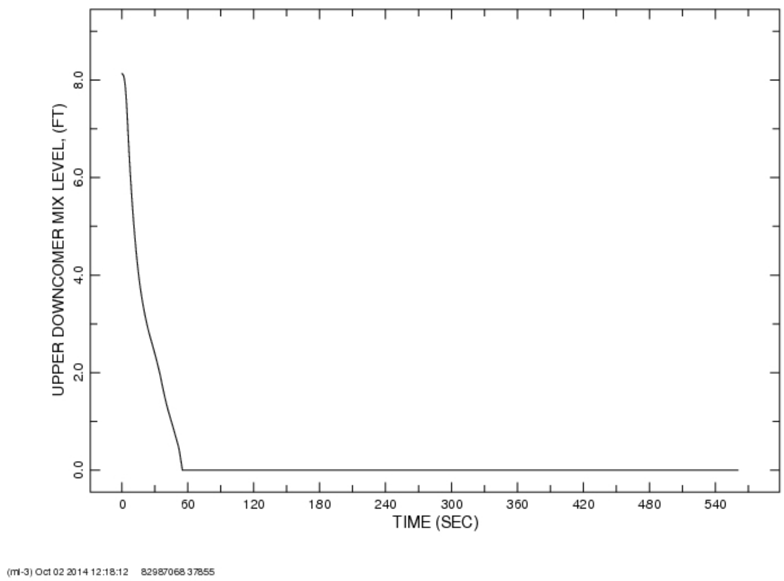


Figure 5.13 Limiting TLO Recirculation Line Break Upper Downcomer Mixture Level

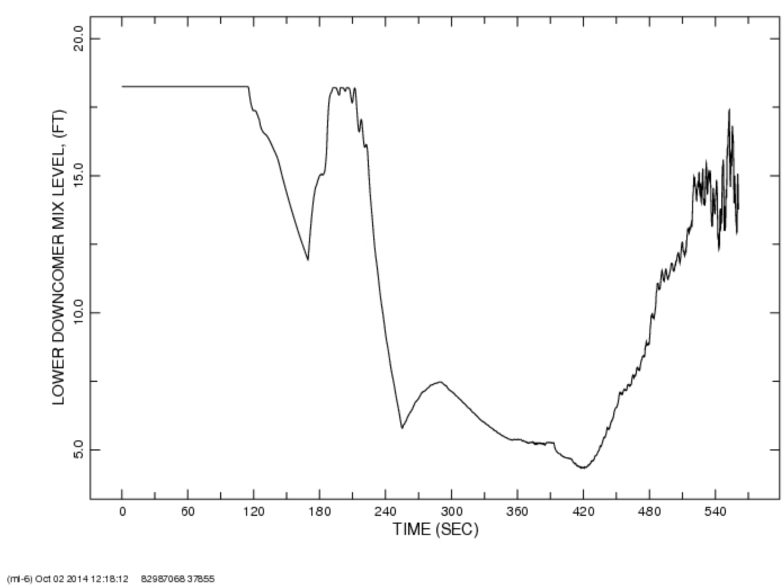


Figure 5.14 Limiting TLO Recirculation Line Break Lower Downcomer Mixture Level

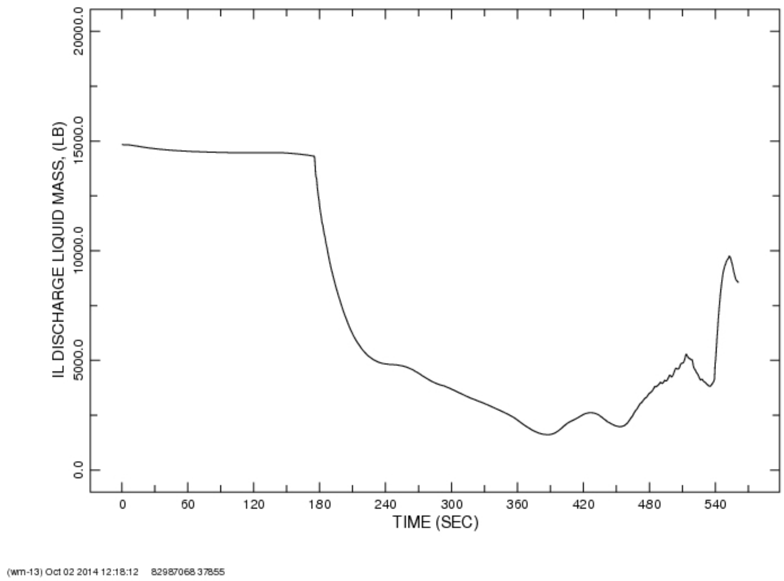


Figure 5.15 Limiting TLO Recirculation Line Break Intact Loop Discharge Line Liquid Mass

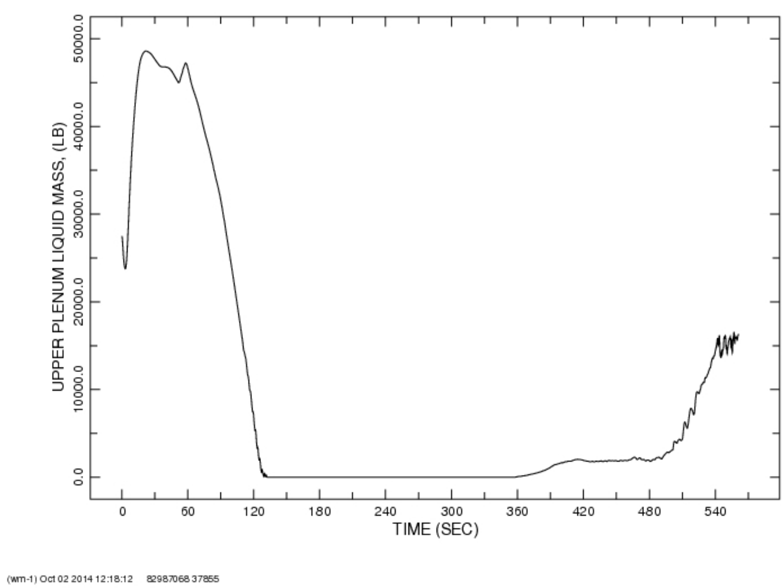


Figure 5.16 Limiting TLO Recirculation Line Break Upper Plenum Liquid Mass

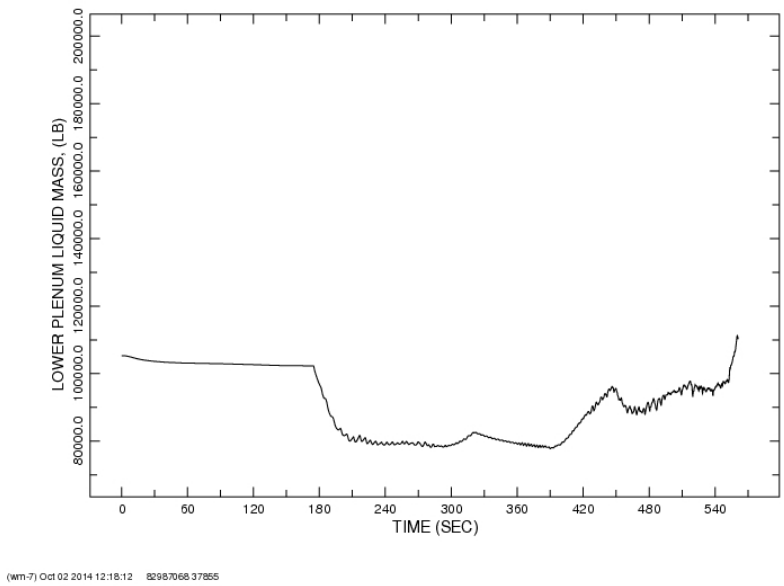


Figure 5.17 Limiting TLO Recirculation Line Break Lower Plenum Liquid Mass

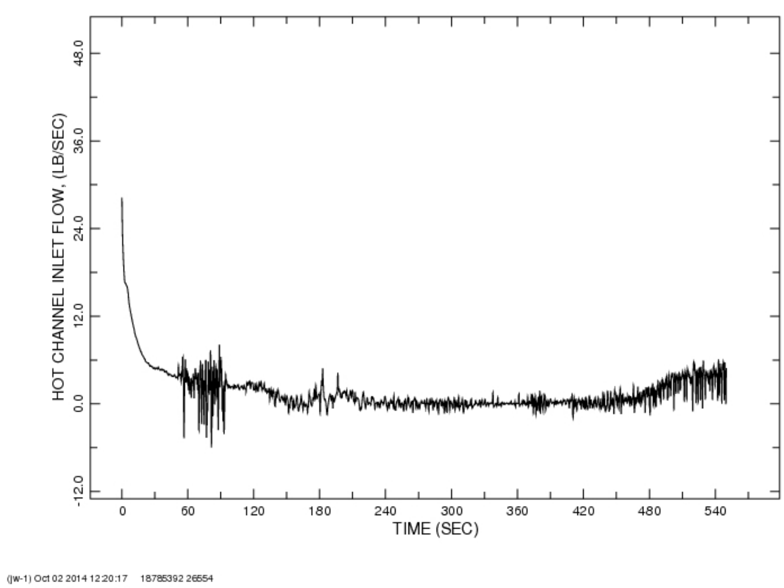


Figure 5.18 Limiting TLO Recirculation Line Break Hot Channel Inlet Flow Rate

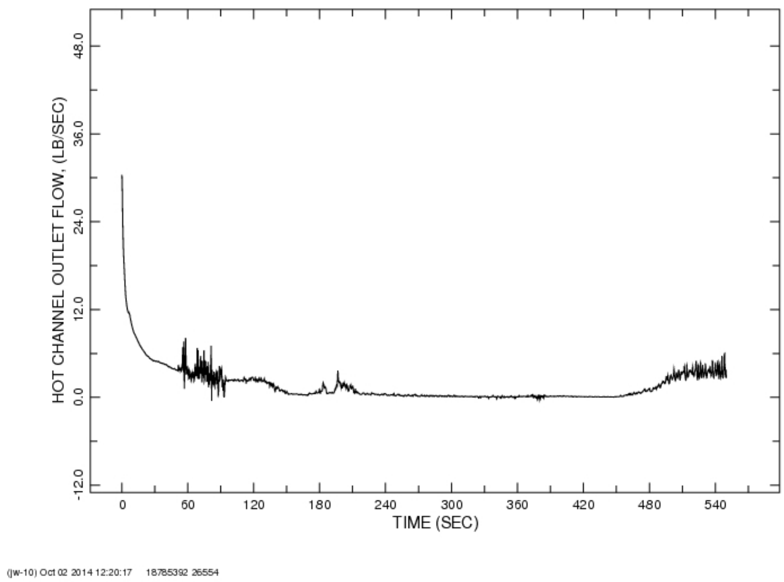


Figure 5.19 Limiting TLO Recirculation Line Break Hot Channel Outlet Flow Rate

Figure 5.20 Limiting TLO Recirculation Line Break Hot Channel Coolant Temperature at the Hot Node at EOB

Figure 5.21 Limiting TLO Recirculation Line Break
Hot Channel Quality at the Hot Node at EOB

Figure 5.22 Limiting TLO Recirculation Line Break
Hot Channel Heat Transfer Coeff. at the Hot Node at EOB

**Figure 5.23 Limiting TLO Recirculation Line Break
Hot Channel Reflood Junction Liquid Mass Flow Rate**

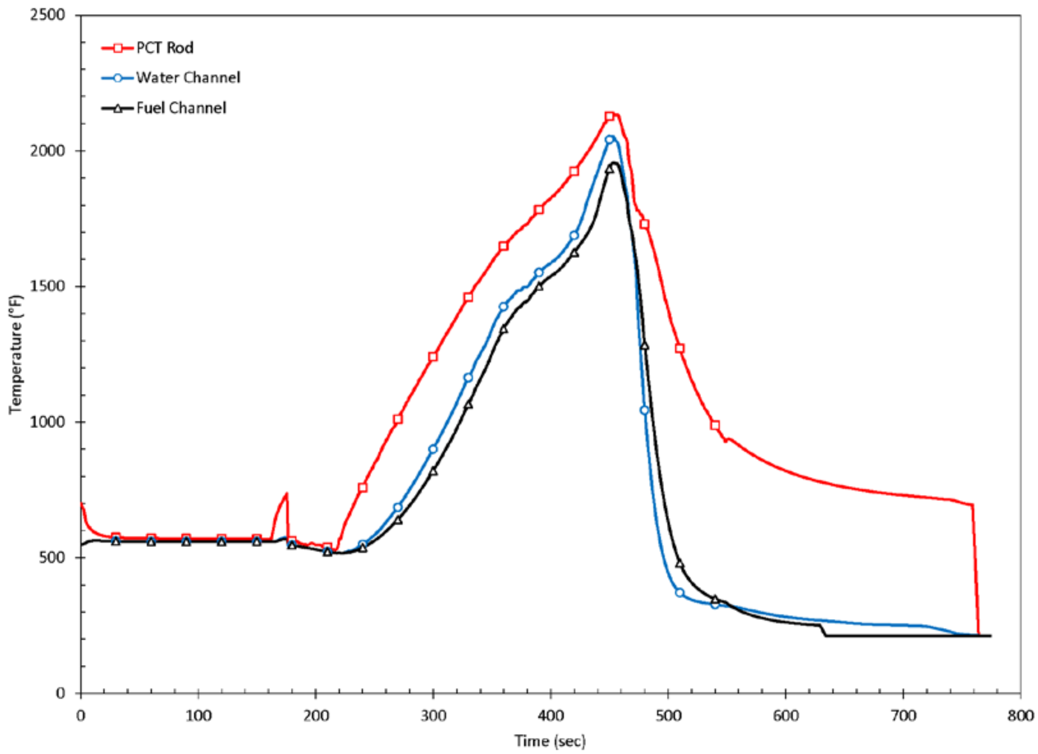


**Figure 5.24 Limiting TLO Recirculation Line Break
Cladding Temperatures**

6.0 Conclusions

The EXEM BWR-2000 Evaluation Model was applied to confirm the acceptability of the ATRIUM 10XM MAPLHGR limits for Quad Cities Units 1 and 2. The following conclusions were made from the analyses presented.

- The acceptance criteria of the Code of Federal Regulations (10 CFR 50.46) are met for operation at or below the ATRIUM 10XM MAPLHGR limits presented in Figure 2.1.
    - Peak PCT < 2200°F.
    - Local cladding oxidation thickness < 0.17.
    - Total hydrogen generation < 0.01.
    - Coolable geometry satisfied by meeting peak PCT, local cladding oxidation, and total hydrogen generation criteria.
    - Core long-term cooling is satisfied by concluding the core is flooded to the top of active fuel or the core is flooded to the elevation of the jet pump suction elevation (approximately two-thirds core height) with one core spray operating (Reference 1).
- The MAPLHGR limit is applicable for ATRIUM 10XM full cores as well as transition cores containing ATRIUM 10XM fuel.
- [ ]

[ ]

7.0 References

1. ANP-3328P Revision 0, *Quad Cities Units 1 and 2 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel*, AREVA, December 2014.

2. EMF-2361(P)(A) Revision 0, *EXEM BWR-2000 ECCS Evaluation Model,* Framatome ANP, May 2001.

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

4. XN-CC-33(A) Revision 1, *HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual*, Exxon Nuclear Company, November 1975.

5. XN-NF-82-07(P)(A) Revision 1, *Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model*, Exxon Nuclear Company, November 1982.

6. EMF-2292(P)(A) Revision 0, *ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients*, Siemens Power Corporation, September 2000.

7. Letter, P. Salas (AREVA) to Document Control Desk, U.S. Nuclear Regulatory Commission, “Proprietary Viewgraphs and Meeting Summary for Closed Meeting on Application of the EXEM BWR-2000 ECCS Evaluation Methodology,” NRC:11:096, September 22, 2011.

8. Letter, T.J. McGinty (NRC) to P. Salas (AREVA), “Response to AREVA NP, Inc. (AREVA) Proposed Analysis Approach for Its EXEM Boiling Water Reactor (BWR)-2000 Emergency Core Cooling System (ECCS) Evaluation Model,” July 5, 2012.

9. ANP-3338P Revision 0, *Applicability of AREVA BWR Methods to the Dresden and Quad Cities Reactors Operating at Extended Power Uprate,* AREVA, November 2014.

10. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, AREVA NP, February 2008.

ATTACHMENT 25

Nuclear Fuel Design Report
(Non-Proprietary)

Nuclear Fuel Design Report Quad Cities Unit 2 Cycle 24 Representative Cycle ATRIUM 10XM Fuel

ANP-3294NP
Revision 0

July 2014

AREVA Inc.

ANP-3294NP
Revision 0

Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial release.

Contents

Tables

Figures

Nomenclature

BOL	beginning of life
BWR	boiling water reactor
EVC	plenum region in a fuel pin modeled as an evacuated section
GE	General Electric
GWd/MTU	gigawatt days per metric ton of initial uranium
kg/MTU	kilograms per metric ton of initial uranium
LAR	license amendment request
LHGR	linear heat generation rate
LPF	local peaking factor
MCPR	minimum critical power ratio
MWd/MTU	megawatt days per metric ton of initial uranium
NRC	Nuclear Regulatory Commission, U. S.
PLFR	part-length fuel rod

1.0 Introduction

This report provides results of the neutronic design analyses performed by AREVA Inc. (AREVA) for the Quad Cities Unit 2 fabrication batch QCI2-24 ATRIUM™* 10XM boiling water reactor (BWR) fuel assemblies for Cycle 24. The fuel designs presented within will be used for the reference cycle to support the License Amendment Request for the inclusion of AREVA's methodology in the Technical Specifications for both Dresden and Quad Cities to support the introduction of the ATRIUM 10XM fuel design. The mechanical design parameters for the QCI2-24 fuel are summarized in Table 2.1.

Applicable neutronic design criteria are provided in the approved topical report ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 2). Neutronic design analysis methodology used to determine conformance to design criteria has been reviewed and approved by the NRC in the topical report EMF-2158(P)(A) (Reference 3).

The fuel design includes AREVA advanced fuel channels. Mechanical design criteria applicable to the design of these channels have been reviewed and approved by the NRC in Reference 1.

The neutronic design for the fabrication batch includes axially-varying enrichment and gadolinia designs with natural UO_2 blankets at the top and bottom of the assembly. The fabrication batch consists of 96 **[ ]**, 64 **[ ]** and 88 **[ ]** assemblies. Pertinent fuel and reactor core design information associated with this fabrication batch is given in Section 2.0 and in Appendices A through D.

* ATRIUM is a trademark of AREVA Inc.

2.0 Neutronic Design

The results of the Quad Cities Unit 2 fabrication batch QCI2-24 ATRIUM 10XM neutronic design analyses are presented in this section. The fuel was designed to meet applicable design criteria, as well as reactivity and control requirements. Reactor core loading patterns and the number of assemblies to be loaded will depend upon final cycle energy requirements as specified by the utility. Applicable neutronic design criteria outlined in Reference 2 are summarized below:

- **Power Distribution.** The local power distribution in the fuel assembly combined with the core power distribution shall result in Linear Heat Generation Rate (LHGR) and Minimum Critical Power Ratio (MCPR) values that are within the limits established for each fuel design.
- **Kinetics Parameters.** The moderator void reactivity coefficient due to boiling in the active channels and the Doppler fuel temperature reactivity coefficient shall be negative. The negative void and Doppler reactivity coefficients ensure a negative power coefficient during reactor operation. Additional calculations were performed to show that the assembly average Doppler and void reactivity coefficients remain negative for the life of the assembly. These results demonstrate that the Reference 2 kinetics criteria are met on a bundle average basis.
- **Control Blade Reactivity.** The design of the fuel assembly and the reactor core loading shall be such that the technical specification shutdown margin requirement is met for all reactor conditions.

2.1 *Neutronic Design Description*

The neutronic design parameters for fabrication batch QCI2-24 are presented in Table 2.1. The key ATRIUM 10XM reload assembly nuclear design characteristics are summarized below:

- The fuel assembly contains **[**

  ]. The water channel also serves the function of the spacer capture rod.
- Each fuel assembly has top and bottom natural uranium blankets.
- The plenum/spring region above the PLFRs will be explicitly modeled.
- The enrichments are designed to yield a local power distribution which results in a balanced design relative to MCPR, LHGR, and other reactor operating requirements, e.g., power peaking.
- Gadolinia (Gd_2O_3 blended with UO_2) rods are designed to control assembly reactivity in order to meet reactivity control requirements in the reactor, e.g. cold shutdown margin.

- The reload batch consists of 3 assembly designs which vary axially in enrichment and/or gadolinia. The axial distributions of the lattices in the assemblies are shown in Figures 2.1, 2.2, and 2.3. The fuel rod distribution and axial descriptions are presented in Figures 2.4 through 2.7. The enrichment and gadolinia distribution maps for each of the reload assembly lattices are displayed in Appendix D.
- The fuel assembly incorporates the AREVA advanced fuel channel which improves uranium utilization. For D-lattice plants, the fuel assembly is offset 40 mils toward the control blade.

2.2 ***Lattice Control Blade Worths and Kinetics Parameters***

Beginning of life (BOL) lattice reactivities (k_∞) have been calculated for moderator and fuel conditions ranging from cold to hot operating conditions. From these reactivities, BOL control blade worths and kinetics parameters have been determined based on the following control designs:

- GE DuraLife D100(OEM)/D120, D160, D190 and D230
- GE Marathon and Marathon Ultra HD
- Westinghouse CR82M-1/CR82B
- Westinghouse CR99

The DuraLife D-100 (OEM) and DuraLife D-120 control blades were found to have equivalent neutronic performance and due to the prevalence of the D-100 control blades, it will be used to model both blade types. The same conclusion was reached for the CR82B blade which will be modeled as the more prevalent CR82M-1 blade design. Certain zones of the Marathon Ultra HD were determined to have equivalent neutronic performance as the standard Marathon control blade and have been modeled as such. The other axial zones of the Marathon Ultra HD blade are explicitly modeled and are presented in this report as HD2 and HD3.

Due to the 3rd party proprietary nature of the control blade designs, control blade dimensional information has been withheld from this report. Kinetics parameters are calculated for fuel temperature (Doppler), moderator void, and moderator temperature. The Doppler reactivity was calculated over a fuel temperature range from hot standby to hot operating. The moderator void reactivity was evaluated between the 0% and 40% voided hot operating cases, and the moderator temperature reactivity was calculated from the cold to hot standby condition. The calculations neglect the spacer material and assume zero void in the coolant outside the fuel assembly channel as well as inside the internal water channel. The results of these calculations are presented in Tables 2.2 through 2.197.

2.3 *Enriched Lattice Uncontrolled Reactivities and Isotopic Data*

The enriched lattice exposure-dependent uncontrolled reactivities calculated at three void fractions are presented graphically in Appendix A, and in tabular format in Appendix B. The enriched lattice exposure-dependent isotopic data calculated at three void fractions are presented in Appendix C.

Table 2.1 Neutronic Design Parameters

[

]

* [ ]

† [ ]

Table 2.1 Neutronic Design Parameters *(Continued)*

[

]

Table 2.1 Neutronic Design Parameters *(Continued)*

Core Data*	
Number of fuel assemblies in the core	724
Rated thermal power level, MWt	2957
Rated core flow, Mlbm/hr	98.00
Inlet subcooling, Btu/lbm	24.21
Dome pressure, psia	1015.0
Boron concentration, PPM	918.0 (equivalent natural ^{10}B)
Intermediate temperature, °F	200.00
Warm temperature, °F	358.30

* Some values are representative of rated conditions and may vary depending on the core statepoint.

Table 2.2 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92163	1.08439	-0.1501
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90643	1.07403	-0.1560
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85268	1.04096	-0.1809
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.84899	1.03648	-0.1809
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.84906	1.03648	-0.1808
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.81356	1.02079	-0.2030
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.77089	1.00157	-0.2303

Table 2.3 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91987	1.08439	-0.1517
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90473	1.07403	-0.1576
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85087	1.04096	-0.1826
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.84720	1.03648	-0.1826
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.84726	1.03648	-0.1826
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.81117	1.02079	-0.2054
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.76757	1.00157	-0.2336

Table 2.4 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91662	1.08439	-0.1547
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90125	1.07403	-0.1609
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84574	1.04096	-0.1875
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.84211	1.03648	-0.1875
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.84211	1.03648	-0.1875
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.80456	1.02079	-0.2118
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.75901	1.00157	-0.2422

Table 2.5 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90909	1.08439	-0.1617
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89357	1.07403	-0.1680
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83686	1.04096	-0.1961
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.83327	1.03648	-0.1961
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.83333	1.03648	-0.1960
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.79432	1.02079	-0.2219
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.74689	1.00157	-0.2543

Table 2.6 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91254	1.08439	-0.1585
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89692	1.07403	-0.1649
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84024	1.04096	-0.1928
Hot Operating_{NoXe}	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.83663	1.03648	-0.1928
Hot Operating_0	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.83663	1.03648	-0.1928
Hot Operating_{40}	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.79840	1.02079	-0.2179
Hot Operating_{80}	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.75221	1.00157	-0.2490

Table 2.7 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92381	1.08439	-0.1481
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90880	1.07403	-0.1538
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85477	1.04096	-0.1789
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.85109	1.03648	-0.1789
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.85114	1.03648	-0.1788
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.81524	1.02079	-0.2014
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.77190	1.00157	-0.2293

Table 2.8 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91638	1.08439	-0.1549
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90115	1.07403	-0.1610
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84557	1.04096	-0.1877
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.84194	1.03648	-0.1877
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.84200	1.03648	-0.1876
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.80406	1.02079	-0.2123
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.75803	1.00157	-0.2432

Table 2.9 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91744	1.08439	-0.1540
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90242	1.07403	-0.1598
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84768	1.04096	-0.1857
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.84404	1.03648	-0.1857
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.84411	1.03648	-0.1856
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.80712	1.02079	-0.2093
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.76192	1.00157	-0.2393

Table 2.10 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93855	1.08439	-0.1345
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92425	1.07403	-0.1395
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87450	1.04096	-0.1599
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.87069	1.03648	-0.1600
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.87069	1.03648	-0.1600
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.83893	1.02079	-0.1782
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.80063	1.00157	-0.2006

Table 2.11 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91546	1.08439	-0.1558
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90020	1.07403	-0.1618
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84454	1.04096	-0.1887
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.84090	1.03648	-0.1887
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.84098	1.03648	-0.1886
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.80292	1.02079	-0.2134
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.75614	1.00157	-0.2450

Table 2.12 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91390	1.08439	-0.1572
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89861	1.07403	-0.1633
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84256	1.04096	-0.1906
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.83894	1.03648	-0.1906
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.83902	1.03648	-0.1905
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.80050	1.02079	-0.2158
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.75308	1.00157	-0.2481

Table 2.13 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93694	1.08439	-0.1360
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92232	1.07403	-0.1412
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87056	1.04096	-0.1637
Hot Operating_{NoXe}	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.86679	1.03648	-0.1637
Hot Operating_{0}	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.86684	1.03648	-0.1637
Hot Operating_{40}	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.83298	1.02079	-0.1840
Hot Operating_{80}	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.79177	1.00157	-0.2095

Table 2.14 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95946	1.08439	-0.1152
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.94554	1.07403	-0.1196
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.89776	1.04096	-0.1376
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.89382	1.03648	-0.1376
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.89387	1.03648	-0.1376
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.86394	1.02079	-0.1537
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.82730	1.00157	-0.1740

Table 2.15 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter ($\Delta k/k$)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.03648	1.04096	-0.0043
$Moderator_{Void}$ = $\frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	1.02079	1.03648	-0.0151
$Moderator_{Temperature}$ = $\frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.04096	1.08439	-0.0401

Table 2.16 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91223	1.07416	-0.1507
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89663	1.06324	-0.1567
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84066	1.02690	-0.1814
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.83698	1.02242	-0.1814
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.83704	1.02242	-0.1813
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.80062	1.00489	-0.2033
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.75747	0.98363	-0.2299

Table 2.17 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91050	1.07416	-0.1524
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89495	1.06324	-0.1583
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83889	1.02690	-0.1831
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.83523	1.02242	-0.1831
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.83528	1.02242	-0.1830
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.79829	1.00489	-0.2056
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.75424	0.98363	-0.2332

Table 2.18 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90725	1.07416	-0.1554
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89150	1.06324	-0.1615
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83384	1.02690	-0.1880
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.83021	1.02242	-0.1880
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.83021	1.02242	-0.1880
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.79180	1.00489	-0.2120
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.74592	0.98363	-0.2417

Table 2.19 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89983	1.07416	-0.1623
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88392	1.06324	-0.1686
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82510	1.02690	-0.1965
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.82152	1.02242	-0.1965
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.82157	1.02242	-0.1965
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.78177	1.00489	-0.2220
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.73408	0.98363	-0.2537

Table 2.20 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90320	1.07416	-0.1592
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88721	1.06324	-0.1656
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82842	1.02690	-0.1933
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.82481	1.02242	-0.1933
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.82481	1.02242	-0.1933
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.78577	1.00489	-0.2180
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.73931	0.98363	-0.2484

Table 2.21 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91443	1.07416	-0.1487
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89902	1.06324	-0.1545
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84279	1.02690	-0.1793
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.83912	1.02242	-0.1793
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.83917	1.02242	-0.1792
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.80236	1.00489	-0.2015
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.75859	0.98363	-0.2288

Table 2.22 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90706	1.07416	-0.1556
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89143	1.06324	-0.1616
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83369	1.02690	-0.1882
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.83007	1.02242	-0.1881
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.83012	1.02242	-0.1881
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.79135	1.00489	-0.2125
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.74500	0.98363	-0.2426

Table 2.23 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90817	1.07416	-0.1545
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89276	1.06324	-0.1603
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83589	1.02690	-0.1860
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.83225	1.02242	-0.1860
Hot $Operating_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.83232	1.02242	-0.1859
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.79449	1.00489	-0.2094
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.74892	0.98363	-0.2386

Table 2.24 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92901	1.07416	-0.1351
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91429	1.06324	-0.1401
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86223	1.02690	-0.1604
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.85843	1.02242	-0.1604
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.85843	1.02242	-0.1604
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.82562	1.00489	-0.1784
Hot Operating$_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.78665	0.98363	-0.2003

Table 2.25 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90614	1.07416	-0.1564
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89050	1.06324	-0.1625
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83270	1.02690	-0.1891
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.82907	1.02242	-0.1891
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.82914	1.02242	-0.1890
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.79024	1.00489	-0.2136
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.74313	0.98363	-0.2445

Table 2.26 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90461	1.07416	-0.1578
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88893	1.06324	-0.1639
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83075	1.02690	-0.1910
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.82714	1.02242	-0.1910
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.82722	1.02242	-0.1909
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.78787	1.00489	-0.2160
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.74014	0.98363	-0.2475

Table 2.27 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92739	1.07416	-0.1366
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91234	1.06324	-0.1419
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85826	1.02690	-0.1642
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.85449	1.02242	-0.1643
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.85454	1.02242	-0.1642
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.81965	1.00489	-0.1843
Hot Operating$_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.77781	0.98363	-0.2092

Table 2.28 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94964	1.07416	-0.1159
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93528	1.06324	-0.1203
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88500	1.02690	-0.1382
Hot Operating_{NoXe}	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.88106	1.02242	-0.1383
Hot Operating_{0}	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.88111	1.02242	-0.1382
Hot Operating_{40}	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.84995	1.00489	-0.1542
Hot Operating_{80}	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.81238	0.98363	-0.1741

Table 2.29 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter (Δk/k)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.02242	1.02690	-0.0044
Moderator_{Void} = $\frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	1.00489	1.02242	-0.0171
$\text{Moderator}_{Temperature}$ = $\frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.02690	1.07416	-0.0440

Table 2.30 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.83558	1.00046	-0.1648
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.82081	0.99033	-0.1712
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.77149	0.95996	-0.1963
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.76793	0.95554	-0.1963
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.76799	0.95554	-0.1963
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.73622	0.94208	-0.2185
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.69768	0.92519	-0.2459

Table 2.31 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.83402	1.00046	-0.1664
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.81932	0.99033	-0.1727
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.76994	0.95996	-0.1979
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.76639	0.95554	-0.1979
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.76644	0.95554	-0.1979
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.73412	0.94208	-0.2208
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.69471	0.92519	-0.2491

Table 2.32 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.83073	1.00046	-0.1696
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.81583	0.99033	-0.1762
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.76502	0.95996	-0.2031
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.76151	0.95554	-0.2031
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.76151	0.95554	-0.2031
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.72783	0.94208	-0.2274
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.68661	0.92519	-0.2579

Table 2.33 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.82347	1.00046	-0.1769
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.80841	0.99033	-0.1837
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.75642	0.95996	-0.2120
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.75296	0.95554	-0.2120
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.75301	0.95554	-0.2119
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.71795	0.94208	-0.2379
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.67493	0.92519	-0.2705

Table 2.34 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.82668	1.00046	-0.1737
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.81154	0.99033	-0.1805
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.75961	0.95996	-0.2087
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.75612	0.95554	-0.2087
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.75612	0.95554	-0.2087
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.72180	0.94208	-0.2338
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.67997	0.92519	-0.2651

Table 2.35 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.83759	1.00046	-0.1628
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.82308	0.99033	-0.1689
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.77370	0.95996	-0.1940
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.77014	0.95554	-0.1940
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.77019	0.95554	-0.1940
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.73807	0.94208	-0.2165
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.69893	0.92519	-0.2446

Table 2.36 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.83037	1.00046	-0.1700
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.81562	0.99033	-0.1764
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.76481	0.95996	-0.2033
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.76131	0.95554	-0.2033
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.76136	0.95554	-0.2032
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.72733	0.94208	-0.2280
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.68568	0.92519	-0.2589

Table 2.37 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.83158	1.00046	-0.1688
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.81704	0.99033	-0.1750
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.76700	0.95996	-0.2010
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.76349	0.95554	-0.2010
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.76356	0.95554	-0.2009
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.73042	0.94208	-0.2247
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.68950	0.92519	-0.2547

Table 2.38 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.85199	1.00046	-0.1484
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.83815	0.99033	-0.1537
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.79282	0.95996	-0.1741
Hot Operating_{NoXe}	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.78914	0.95554	-0.1741
Hot Operating_{0}	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.78914	0.95554	-0.1741
Hot Operating_{40}	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.76090	0.94208	-0.1923
Hot Operating_{80}	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72653	0.92519	-0.2147

Table 2.39 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.82982	1.00046	-0.1706
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.81503	0.99033	-0.1770
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.76407	0.95996	-0.2041
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.76056	0.95554	-0.2041
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.76063	0.95554	-0.2040
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.72648	0.94208	-0.2289
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.68409	0.92519	-0.2606

Table 2.40 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.82830	1.00046	-0.1721
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.81348	0.99033	-0.1786
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.76217	0.95996	-0.2060
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.75868	0.95554	-0.2060
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.75875	0.95554	-0.2059
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.72416	0.94208	-0.2313
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.68116	0.92519	-0.2638

Table 2.41 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.85060	1.00046	-0.1498
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.83644	0.99033	-0.1554
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.78913	0.95996	-0.1780
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.78548	0.95554	-0.1780
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.78553	0.95554	-0.1779
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.75531	0.94208	-0.1983
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.71816	0.92519	-0.2238

Table 2.42 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.87287	1.00046	-0.1275
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.85936	0.99033	-0.1323
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.81576	0.95996	-0.1502
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.81193	0.95554	-0.1503
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.81197	0.95554	-0.1502
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78542	0.94208	-0.1663
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75256	0.92519	-0.1866

Table 2.43 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter (Δk/k)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	0.95554	0.95996	-0.0046
$Moderator_{Void} = \frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	0.94208	0.95554	-0.0141
$Moderator_{Temperature} = \frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	0.95996	1.00046	-0.0405

Table 2.44 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.84699	1.00447	-0.1568
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.83226	0.99467	-0.1633
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.78393	0.96653	-0.1889
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.78047	0.96226	-0.1889
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.78053	0.96226	-0.1889
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.74807	0.94979	-0.2124
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.70599	0.93242	-0.2428

Table 2.45 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.84552	1.00447	-0.1582
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.83086	0.99467	-0.1647
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.78248	0.96653	-0.1904
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.77903	0.96226	-0.1904
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.77908	0.96226	-0.1904
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.74605	0.94979	-0.2145
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.70307	0.93242	-0.2460

Table 2.46 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.84243	1.00447	-0.1613
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.82760	0.99467	-0.1680
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.77787	0.96653	-0.1952
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.77445	0.96226	-0.1952
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.77445	0.96226	-0.1952
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.74002	0.94979	-0.2209
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.69506	0.93242	-0.2546

Table 2.47 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.83560	1.00447	-0.1681
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.82056	0.99467	-0.1750
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.76957	0.96653	-0.2038
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.76619	0.96226	-0.2038
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.76624	0.96226	-0.2037
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.73033	0.94979	-0.2311
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.68339	0.93242	-0.2671

Table 2.48 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.83862	1.00447	-0.1651
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.82351	0.99467	-0.1721
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.77262	0.96653	-0.2006
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.76923	0.96226	-0.2006
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.76923	0.96226	-0.2006
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.73408	0.94979	-0.2271
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.68839	0.93242	-0.2617

Table 2.49 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.84900	1.00447	-0.1548
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.83457	0.99467	-0.1610
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.78636	0.96653	-0.1864
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.78290	0.96226	-0.1864
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.78295	0.96226	-0.1863
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.75018	0.94979	-0.2102
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.70745	0.93242	-0.2413

Table 2.50 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.84210	1.00447	-0.1616
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.82743	0.99467	-0.1681
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.77771	0.96653	-0.1954
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.77430	0.96226	-0.1953
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.77435	0.96226	-0.1953
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.73957	0.94979	-0.2213
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.69416	0.93242	-0.2555

Table 2.51 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.84330	1.00447	-0.1605
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.82884	0.99467	-0.1667
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.77998	0.96653	-0.1930
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.77656	0.96226	-0.1930
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.77662	0.96226	-0.1929
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.74280	0.94979	-0.2179
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.69812	0.93242	-0.2513

Table 2.52 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.86249	1.00447	-0.1414
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.84876	0.99467	-0.1467
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.80465	0.96653	-0.1675
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.80108	0.96226	-0.1675
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.80108	0.96226	-0.1675
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.77243	0.94979	-0.1867
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73494	0.93242	-0.2118

Table 2.53 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.84161	1.00447	-0.1621
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.82689	0.99467	-0.1687
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.77705	0.96653	-0.1960
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.77364	0.96226	-0.1960
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.77370	0.96226	-0.1960
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.73886	0.94979	-0.2221
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.69270	0.93242	-0.2571

Table 2.54 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.84019	1.00447	-0.1636
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.82543	0.99467	-0.1701
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.77524	0.96653	-0.1979
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.77184	0.96226	-0.1979
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.77191	0.96226	-0.1978
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.73660	0.94979	-0.2245
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.68978	0.93242	-0.2602

Table 2.55 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.86138	1.00447	-0.1424
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.84730	0.99467	-0.1482
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.80124	0.96653	-0.1710
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.79769	0.96226	-0.1710
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.79774	0.96226	-0.1710
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.76708	0.94979	-0.1924
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72665	0.93242	-0.2207

Table 2.56 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88248	1.00447	-0.1215
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.86911	0.99467	-0.1262
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82689	0.96653	-0.1445
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82318	0.96226	-0.1445
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82322	0.96226	-0.1445
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79656	0.94979	-0.1613
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76097	0.93242	-0.1839

Table 2.57 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter ($\Delta k/k$)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	0.96226	0.96653	-0.0044
$Moderator_{Void} = \frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	0.94979	0.96226	-0.0130
$Moderator_{Temperature} = \frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	0.96653	1.00447	-0.0378

Table 2.58 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89564	1.04805	-0.1454
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88221	1.03979	-0.1515
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83156	1.01261	-0.1788
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82785	1.00810	-0.1788
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82791	1.00810	-0.1787
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78978	0.99291	-0.2046
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73936	0.97092	-0.2385

Table 2.59 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89413	1.04805	-0.1469
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88076	1.03979	-0.1529
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83003	1.01261	-0.1803
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82634	1.00810	-0.1803
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82638	1.00810	-0.1803
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78765	0.99291	-0.2067
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73625	0.97092	-0.2417

Table 2.60 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89122	1.04805	-0.1496
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87766	1.03979	-0.1559
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82545	1.01261	-0.1848
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82179	1.00810	-0.1848
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82179	1.00810	-0.1848
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78153	0.99291	-0.2129
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72801	0.97092	-0.2502

Table 2.61 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88447	1.04805	-0.1561
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87072	1.03979	-0.1626
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.81714	1.01261	-0.1930
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.81352	1.00810	-0.1930
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.81357	1.00810	-0.1930
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.77170	0.99291	-0.2228
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.71605	0.97092	-0.2625

Table 2.62 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88766	1.04805	-0.1530
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87382	1.03979	-0.1596
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82036	1.01261	-0.1899
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.81672	1.00810	-0.1898
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.81672	1.00810	-0.1898
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.77565	0.99291	-0.2188
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72128	0.97092	-0.2571

Table 2.63 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89760	1.04805	-0.1436
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88443	1.03979	-0.1494
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83395	1.01261	-0.1764
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83024	1.00810	-0.1764
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83029	1.00810	-0.1764
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79189	0.99291	-0.2025
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74086	0.97092	-0.2369

Table 2.64 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89077	1.04805	-0.1501
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87738	1.03979	-0.1562
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82519	1.01261	-0.1851
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82154	1.00810	-0.1851
Hot $Operating_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82159	1.00810	-0.1850
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78100	0.99291	-0.2134
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72709	0.97092	-0.2511

Table 2.65 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89199	1.04805	-0.1489
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87879	1.03979	-0.1548
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82754	1.01261	-0.1828
Hot Operating_{NoXe}	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82387	1.00810	-0.1827
Hot Operating_{0}	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82394	1.00810	-0.1827
Hot Operating_{40}	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78439	0.99291	-0.2100
Hot Operating_{80}	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73127	0.97092	-0.2468

Table 2.66 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91097	1.04805	-0.1308
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89855	1.03979	-0.1358
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85243	1.01261	-0.1582
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84861	1.00810	-0.1582
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84861	1.00810	-0.1582
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81469	0.99291	-0.1795
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76931	0.97092	-0.2077

Table 2.67 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89046	1.04805	-0.1504
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87703	1.03979	-0.1565
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82472	1.01261	-0.1855
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82106	1.00810	-0.1855
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82113	1.00810	-0.1855
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78049	0.99291	-0.2139
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72582	0.97092	-0.2524

Table 2.68 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88904	1.04805	-0.1517
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87557	1.03979	-0.1579
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82288	1.01261	-0.1874
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.81923	1.00810	-0.1873
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.81930	1.00810	-0.1873
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.77817	0.99291	-0.2163
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72278	0.97092	-0.2556

Table 2.69 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90970	1.04805	-0.1320
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89690	1.03979	-0.1374
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84881	1.01261	-0.1618
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84501	1.00810	-0.1618
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84505	1.00810	-0.1617
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80904	0.99291	-0.1852
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76060	0.97092	-0.2166

Table 2.70 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93037	1.04805	-0.1123
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91834	1.03979	-0.1168
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87448	1.01261	-0.1364
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87053	1.00810	-0.1365
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87056	1.00810	-0.1364
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83903	0.99291	-0.1550
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.79596	0.97092	-0.1802

Table 2.71 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter (Δk/k)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.00810	1.01261	-0.0044
$Moderator_{Void}$ = $\frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	0.99291	1.00810	-0.0151
$Moderator_{Temperature}$ = $\frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.01261	1.04805	-0.0338

Table 2.72 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92595	1.08128	-0.1436
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91455	1.07576	-0.1499
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87375	1.06264	-0.1778
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87002	1.05813	-0.1778
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87009	1.05813	-0.1777
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83526	1.04960	-0.2042
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.78671	1.03566	-0.2404

Table 2.73 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92437	1.08128	-0.1451
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91303	1.07576	-0.1513
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87212	1.06264	-0.1793
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86841	1.05813	-0.1793
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86846	1.05813	-0.1792
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83295	1.04960	-0.2064
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.78333	1.03566	-0.2436

Table 2.74 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92138	1.08128	-0.1479
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90986	1.07576	-0.1542
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86729	1.06264	-0.1838
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86362	1.05813	-0.1838
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86362	1.05813	-0.1838
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82641	1.04960	-0.2126
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77428	1.03566	-0.2524

Table 2.75 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91436	1.08128	-0.1544
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90261	1.07576	-0.1610
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85846	1.06264	-0.1921
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.85483	1.05813	-0.1921
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.85488	1.05813	-0.1921
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81586	1.04960	-0.2227
Hot Operating$_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76127	1.03566	-0.2649

Table 2.76 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91772	1.08128	-0.1513
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90590	1.07576	-0.1579
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86190	1.06264	-0.1889
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.85825	1.05813	-0.1889
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.85825	1.05813	-0.1889
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82009	1.04960	-0.2187
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76685	1.03566	-0.2596

Table 2.77 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92783	1.08128	-0.1419
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91671	1.07576	-0.1479
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87604	1.06264	-0.1756
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87233	1.05813	-0.1756
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87238	1.05813	-0.1756
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83721	1.04960	-0.2024
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.78795	1.03566	-0.2392

Table 2.78 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92083	1.08128	-0.1484
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90945	1.07576	-0.1546
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86694	1.06264	-0.1842
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86328	1.05813	-0.1842
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86333	1.05813	-0.1841
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82575	1.04960	-0.2133
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77318	1.03566	-0.2534

Table 2.79 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92189	1.08128	-0.1474
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91070	1.07576	-0.1534
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86900	1.06264	-0.1822
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86533	1.05813	-0.1822
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86540	1.05813	-0.1821
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82886	1.04960	-0.2103
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77722	1.03566	-0.2495

Table 2.80 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94170	1.08128	-0.1291
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93133	1.07576	-0.1343
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.89550	1.06264	-0.1573
Hot Operating_{NoXe}	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.89166	1.05813	-0.1573
Hot Operating_{0}	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.89166	1.05813	-0.1573
Hot Operating_{40}	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.86150	1.04960	-0.1792
Hot Operating_{80}	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.81883	1.03566	-0.2094

Table 2.81 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92048	1.08128	-0.1487
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90905	1.07576	-0.1550
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86634	1.06264	-0.1847
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86267	1.05813	-0.1847
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86275	1.05813	-0.1847
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82512	1.04960	-0.2139
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77186	1.03566	-0.2547

Table 2.82 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91899	1.08128	-0.1501
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90751	1.07576	-0.1564
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86437	1.06264	-0.1866
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86072	1.05813	-0.1866
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86079	1.05813	-0.1865
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82261	1.04960	-0.2163
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76855	1.03566	-0.2579

Table 2.83 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94052	1.08128	-0.1302
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92990	1.07576	-0.1356
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.89204	1.06264	-0.1605
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.88823	1.05813	-0.1606
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.88828	1.05813	-0.1605
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.85596	1.04960	-0.1845
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.81002	1.03566	-0.2179

Table 2.84 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.96199	1.08128	-0.1103
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.95214	1.07576	-0.1149
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.91931	1.06264	-0.1349
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.91534	1.05813	-0.1349
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.91538	1.05813	-0.1349
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.88824	1.04960	-0.1537
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.84889	1.03566	-0.1803

Table 2.85 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\dfrac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter ($\Delta k/k$)
Doppler = $\dfrac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.05813	1.06264	-0.0042
Moderator$_{Void}$ = $\dfrac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	1.04960	1.05813	-0.0081
Moderator$_{Temperature}$ = $\dfrac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.06264	1.08128	-0.0172

Table 2.86 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95930	1.11765	-0.1417
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.94431	1.10727	-0.1472
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88618	1.07039	-0.1721
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.88224	1.06567	-0.1721
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.88231	1.06567	-0.1721
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.84000	1.04412	-0.1955
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.78885	1.01855	-0.2255

Table 2.87 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95738	1.11765	-0.1434
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.94245	1.10727	-0.1489
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88420	1.07039	-0.1739
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.88028	1.06567	-0.1740
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.88033	1.06567	-0.1739
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.83741	1.04412	-0.1980
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.78533	1.01855	-0.2290

Table 2.88 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95421	1.11765	-0.1462
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93902	1.10727	-0.1520
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87902	1.07039	-0.1788
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.87514	1.06567	-0.1788
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.87514	1.06567	-0.1788
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.83069	1.04412	-0.2044
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.77664	1.01855	-0.2375

Table 2.89 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94652	1.11765	-0.1531
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93117	1.10727	-0.1590
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86996	1.07039	-0.1872
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.86613	1.06567	-0.1872
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.86619	1.06567	-0.1872
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.82026	1.04412	-0.2144
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.76429	1.01855	-0.2496

Table 2.90 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95023	1.11765	-0.1498
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93477	1.10727	-0.1558
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87352	1.07039	-0.1839
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.86966	1.06567	-0.1839
Hot $Operating_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.86966	1.06567	-0.1839
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.82450	1.04412	-0.2103
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.76974	1.01855	-0.2443

Table 2.91 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.96076	1.11765	-0.1404
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.94591	1.10727	-0.1457
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88736	1.07039	-0.1710
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.88343	1.06567	-0.1710
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.88349	1.06567	-0.1709
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.84068	1.04412	-0.1948
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.78887	1.01855	-0.2255

Table 2.92 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95378	1.11765	-0.1466
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93874	1.10727	-0.1522
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87867	1.07039	-0.1791
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.87479	1.06567	-0.1791
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.87485	1.06567	-0.1791
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.82997	1.04412	-0.2051
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.77538	1.01855	-0.2387

Table 2.93 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95332	1.11765	-0.1470
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93834	1.10727	-0.1526
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87880	1.07039	-0.1790
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.87492	1.06567	-0.1790
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.87500	1.06567	-0.1789
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.83098	1.04412	-0.2041
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.77737	1.01855	-0.2368

Table 2.94 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.97503	1.11765	-0.1276
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.96078	1.10727	-0.1323
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.90632	1.07039	-0.1533
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.90227	1.06567	-0.1533
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.90227	1.06567	-0.1533
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.86357	1.04412	-0.1729
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.81687	1.01855	-0.1980

Table 2.95 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95248	1.11765	-0.1478
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93737	1.10727	-0.1534
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87709	1.07039	-0.1806
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.87322	1.06567	-0.1806
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.87330	1.06567	-0.1805
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.82828	1.04412	-0.2067
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.77306	1.01855	-0.2410

Table 2.96 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95092	1.11765	-0.1492
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93578	1.10727	-0.1549
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87510	1.07039	-0.1825
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.87124	1.06567	-0.1825
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.87132	1.06567	-0.1824
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.82584	1.04412	-0.2091
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.76997	1.01855	-0.2440

Table 2.97 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.97457	1.11765	-0.1280
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.96017	1.10727	-0.1328
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.90403	1.07039	-0.1554
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.90000	1.06567	-0.1555
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.90006	1.06567	-0.1554
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.85943	1.04412	-0.1769
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.80982	1.01855	-0.2049

Table 2.98 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.99721	1.11765	-0.1078
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.98350	1.10727	-0.1118
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.93145	1.07039	-0.1298
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.92726	1.06567	-0.1299
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.92730	1.06567	-0.1298
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.89072	1.04412	-0.1469
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.84579	1.01855	-0.1696

Table 2.99 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter ($\Delta k/k$)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.06567	1.07039	-0.0044
$Moderator_{Void}$ = $\frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	1.04412	1.06567	-0.0202
$Moderator_{Temperature}$ = $\frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.07039	1.11765	-0.0423

Table 2.100 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89052	1.05078	-0.1525
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87612	1.04104	-0.1584
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82362	1.00975	-0.1843
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.81975	1.00503	-0.1844
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.81982	1.00503	-0.1843
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78126	0.98706	-0.2085
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73345	0.96512	-0.2400

Table 2.101 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88874	1.05078	-0.1542
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87440	1.04104	-0.1601
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82181	1.00975	-0.1861
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.81796	1.00503	-0.1861
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.81801	1.00503	-0.1861
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.77887	0.98706	-0.2109
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73016	0.96512	-0.2434

Table 2.102 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

$\text{Blade Worth} = \dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88553	1.05078	-0.1573
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87096	1.04104	-0.1634
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.81675	1.00975	-0.1911
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.81294	1.00503	-0.1911
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.81294	1.00503	-0.1911
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.77232	0.98706	-0.2176
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72164	0.96512	-0.2523

Table 2.103 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.87798	1.05078	-0.1645
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.86323	1.04104	-0.1708
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.80777	1.00975	-0.2000
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.80401	1.00503	-0.2000
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.80406	1.00503	-0.2000
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.76197	0.98706	-0.2280
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.70936	0.96512	-0.2650

Table 2.104 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88156	1.05078	-0.1610
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.86671	1.04104	-0.1675
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.81125	1.00975	-0.1966
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.80746	1.00503	-0.1966
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.80746	1.00503	-0.1966
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.76610	0.98706	-0.2239
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.71466	0.96512	-0.2595

Table 2.105 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89163	1.05078	-0.1515
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87742	1.04104	-0.1572
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82471	1.00975	-0.1832
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82086	1.00503	-0.1832
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82091	1.00503	-0.1832
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78190	0.98706	-0.2079
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73347	0.96512	-0.2400

Table 2.106 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88489	1.05078	-0.1579
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87049	1.04104	-0.1638
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.81630	1.00975	-0.1916
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.81250	1.00503	-0.1916
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.81256	1.00503	-0.1915
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.77152	0.98706	-0.2184
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72032	0.96512	-0.2536

Table 2.107 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88422	1.05078	-0.1585
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.86987	1.04104	-0.1644
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.81610	1.00975	-0.1918
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.81229	1.00503	-0.1918
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.81237	1.00503	-0.1917
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.77216	0.98706	-0.2177
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72195	0.96512	-0.2520

Table 2.108 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90560	1.05078	-0.1382
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89197	1.04104	-0.1432
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84328	1.00975	-0.1649
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83930	1.00503	-0.1649
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83930	1.00503	-0.1649
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80432	0.98706	-0.1851
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76105	0.96512	-0.2114

Table 2.109 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88386	1.05078	-0.1588
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.86937	1.04104	-0.1649
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.81488	1.00975	-0.1930
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.81108	1.00503	-0.1930
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.81115	1.00503	-0.1929
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.77000	0.98706	-0.2199
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.71821	0.96512	-0.2558

Table 2.110 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.88231	1.05078	-0.1603
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.86779	1.04104	-0.1664
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.81292	1.00975	-0.1949
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.80914	1.00503	-0.1949
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.80922	1.00503	-0.1948
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.76760	0.98706	-0.2223
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.71517	0.96512	-0.2590

Table 2.111 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90569	1.05078	-0.1381
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89190	1.04104	-0.1433
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84159	1.00975	-0.1665
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83763	1.00503	-0.1666
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83768	1.00503	-0.1665
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80089	0.98706	-0.1886
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75473	0.96512	-0.2180

Table 2.112 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92840	1.05078	-0.1165
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91531	1.04104	-0.1208
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86904	1.00975	-0.1394
Hot Operating_{NoXe}	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86490	1.00503	-0.1394
Hot Operating_{0}	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86495	1.00503	-0.1394
Hot Operating_{40}	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83219	0.98706	-0.1569
Hot Operating_{80}	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.79080	0.96512	-0.1806

Table 2.113 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter ($\Delta k/k$)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.00503	1.00975	-0.0047
Moderator_{Void} = $\frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	0.98706	1.00503	-0.0179
$\text{Moderator}_{Temperature}$ = $\frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.00975	1.05078	-0.0391

Table 2.114 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

$\text{Blade Worth} = \dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90635	1.05942	-0.1445
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89205	1.05008	-0.1505
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84055	1.02089	-0.1767
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83678	1.01633	-0.1767
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83684	1.01633	-0.1766
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79676	0.99848	-0.2020
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74354	0.97433	-0.2369

Table 2.115 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90466	1.05942	-0.1461
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89044	1.05008	-0.1520
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83885	1.02089	-0.1783
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83509	1.01633	-0.1783
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83515	1.01633	-0.1783
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79444	0.99848	-0.2044
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74028	0.97433	-0.2402

Table 2.116 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90166	1.05942	-0.1489
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88725	1.05008	-0.1551
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83410	1.02089	-0.1830
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83038	1.01633	-0.1830
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83038	1.01633	-0.1830
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78814	0.99848	-0.2107
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73185	0.97433	-0.2489

Table 2.117 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89450	1.05942	-0.1557
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.87989	1.05008	-0.1621
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82541	1.02089	-0.1915
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82173	1.01633	-0.1915
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82178	1.01633	-0.1914
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.77798	0.99848	-0.2208
Hot Operating$_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.71955	0.97433	-0.2615

Table 2.118 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89791	1.05942	-0.1525
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88320	1.05008	-0.1589
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.82876	1.02089	-0.1882
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82506	1.01633	-0.1882
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82506	1.01633	-0.1882
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78201	0.99848	-0.2168
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72482	0.97433	-0.2561

Table 2.119 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90752	1.05942	-0.1434
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89345	1.05008	-0.1492
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84191	1.02089	-0.1753
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83815	1.01633	-0.1753
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83820	1.01633	-0.1753
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79767	0.99848	-0.2011
Hot Operating$_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74376	0.97433	-0.2366

Table 2.120 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90108	1.05942	-0.1495
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88681	1.05008	-0.1555
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83373	1.02089	-0.1833
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83002	1.01633	-0.1833
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83007	1.01633	-0.1833
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78742	0.99848	-0.2114
Hot Operating$_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73057	0.97433	-0.2502

Table 2.121 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90050	1.05942	-0.1500
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88628	1.05008	-0.1560
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83364	1.02089	-0.1834
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82992	1.01633	-0.1834
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82999	1.01633	-0.1834
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78818	0.99848	-0.2106
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73228	0.97433	-0.2484

Table 2.122 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92068	1.05942	-0.1310
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90721	1.05008	-0.1361
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85970	1.02089	-0.1579
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.85583	1.01633	-0.1579
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.85583	1.01633	-0.1579
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81954	0.99848	-0.1792
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77124	0.97433	-0.2084

Table 2.123 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90015	1.05942	-0.1503
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88578	1.05008	-0.1565
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83239	1.02089	-0.1846
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82867	1.01633	-0.1846
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82875	1.01633	-0.1846
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78601	0.99848	-0.2128
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72857	0.97433	-0.2522

Table 2.124 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.89869	1.05942	-0.1517
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88429	1.05008	-0.1579
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83052	1.02089	-0.1865
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.82682	1.01633	-0.1865
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.82689	1.01633	-0.1864
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.78367	0.99848	-0.2151
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.72553	0.97433	-0.2554

Table 2.125 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92089	1.05942	-0.1308
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90727	1.05008	-0.1360
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85820	1.02089	-0.1594
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.85434	1.01633	-0.1594
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.85439	1.01633	-0.1593
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81630	0.99848	-0.1825
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76502	0.97433	-0.2148

Table 2.126 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94244	1.05942	-0.1104
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92955	1.05008	-0.1148
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88466	1.02089	-0.1334
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.88064	1.01633	-0.1335
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.88068	1.01633	-0.1335
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.84698	0.99848	-0.1517
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.80105	0.97433	-0.1779

Table 2.127 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter ($\Delta k/k$)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.01633	1.02089	-0.0045
$Moderator_{Void} = \frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	0.99848	1.01633	-0.0176
$Moderator_{Temperature} = \frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.02089	1.05942	-0.0364

Table 2.128 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91544	1.06926	-0.1439
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90164	1.06059	-0.1499
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85280	1.03526	-0.1762
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84903	1.03070	-0.1763
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84910	1.03070	-0.1762
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81022	1.01507	-0.2018
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75790	0.99362	-0.2372

Table 2.129 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91374	1.06926	-0.1454
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90000	1.06059	-0.1514
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85107	1.03526	-0.1779
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84732	1.03070	-0.1779
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84737	1.03070	-0.1779
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80786	1.01507	-0.2041
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75455	0.99362	-0.2406

Table 2.130 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91071	1.06926	-0.1483
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89678	1.06059	-0.1545
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84625	1.03526	-0.1826
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84254	1.03070	-0.1826
Hot $Operating_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84254	1.03070	-0.1826
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80143	1.01507	-0.2105
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74588	0.99362	-0.2493

Table 2.131 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90349	1.06926	-0.1550
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.88933	1.06059	-0.1615
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.83742	1.03526	-0.1911
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83375	1.03070	-0.1911
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83380	1.03070	-0.1910
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79106	1.01507	-0.2207
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73328	0.99362	-0.2620

Table 2.132 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90694	1.06926	-0.1518
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89270	1.06059	-0.1583
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84083	1.03526	-0.1878
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83714	1.03070	-0.1878
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83714	1.03070	-0.1878
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79518	1.01507	-0.2166
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73867	0.99362	-0.2566

Table 2.133 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

$\text{Blade Worth} = \dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91660	1.06926	-0.1428
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90303	1.06059	-0.1486
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85415	1.03526	-0.1749
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.85039	1.03070	-0.1749
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.85045	1.03070	-0.1749
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81111	1.01507	-0.2009
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75806	0.99362	-0.2371

Table 2.134 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91011	1.06926	-0.1488
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89632	1.06059	-0.1549
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84585	1.03526	-0.1830
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84215	1.03070	-0.1829
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84220	1.03070	-0.1829
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80066	1.01507	-0.2112
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74455	0.99362	-0.2507

Table 2.135 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90951	1.06926	-0.1494
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89578	1.06059	-0.1554
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84574	1.03526	-0.1831
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84204	1.03070	-0.1830
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84211	1.03070	-0.1830
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80143	1.01507	-0.2105
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74631	0.99362	-0.2489

Table 2.136 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92991	1.06926	-0.1303
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91695	1.06059	-0.1354
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87225	1.03526	-0.1575
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86837	1.03070	-0.1575
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86837	1.03070	-0.1575
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83345	1.01507	-0.1789
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.78630	0.99362	-0.2087

Table 2.137 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90918	1.06926	-0.1497
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89529	1.06059	-0.1559
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84450	1.03526	-0.1843
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84079	1.03070	-0.1842
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84087	1.03070	-0.1842
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79925	1.01507	-0.2126
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74256	0.99362	-0.2527

Table 2.138 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.90770	1.06926	-0.1511
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89377	1.06059	-0.1573
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84260	1.03526	-0.1861
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83890	1.03070	-0.1861
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83898	1.03070	-0.1860
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79686	1.01507	-0.2150
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73945	0.99362	-0.2558

Table 2.139 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93013	1.06926	-0.1301
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91701	1.06059	-0.1354
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87074	1.03526	-0.1589
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86688	1.03070	-0.1589
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86693	1.03070	-0.1589
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83018	1.01507	-0.1822
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77998	0.99362	-0.2150

Table 2.140 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95189	1.06926	-0.1098
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93955	1.06059	-0.1141
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.89764	1.03526	-0.1329
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.89363	1.03070	-0.1330
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.89367	1.03070	-0.1330
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.86151	1.01507	-0.1513
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.81705	0.99362	-0.1777

Table 2.141 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter ($\Delta k/k$)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.03070	1.03526	-0.0044
Moderator$_{Void}$ = $\frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	1.01507	1.03070	-0.0152
Moderator$_{Temperature}$ = $\frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.03526	1.06926	-0.0318

Table 2.142 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.99162	1.14917	-0.1371
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.97775	1.14039	-0.1426
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.91925	1.10578	-0.1687
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.91518	1.10092	-0.1687
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.91525	1.10092	-0.1686
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.87072	1.07923	-0.1932
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.81619	1.05218	-0.2243

Table 2.143 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.98976	1.14917	-0.1387
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.97589	1.14039	-0.1442
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.91728	1.10578	-0.1705
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.91323	1.10092	-0.1705
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.91329	1.10092	-0.1704
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.86811	1.07923	-0.1956
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.81259	1.05218	-0.2277

Table 2.144 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.98669	1.14917	-0.1414
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.97257	1.14039	-0.1472
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.91214	1.10578	-0.1751
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.90813	1.10092	-0.1751
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.90813	1.10092	-0.1751
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.86136	1.07923	-0.2019
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.80375	1.05218	-0.2361

Table 2.145 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.97914	1.14917	-0.1480
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.96486	1.14039	-0.1539
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.90312	1.10578	-0.1833
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.89915	1.10092	-0.1833
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.89921	1.10092	-0.1832
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.85085	1.07923	-0.2116
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.79122	1.05218	-0.2480

Table 2.146 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.98283	1.14917	-0.1447
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.96845	1.14039	-0.1508
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.90672	1.10578	-0.1800
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.90273	1.10092	-0.1800
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.90273	1.10092	-0.1800
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.85519	1.07923	-0.2076
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.79680	1.05218	-0.2427

Table 2.147 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.99372	1.14917	-0.1353
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.98000	1.14039	-0.1406
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.92125	1.10578	-0.1669
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.91719	1.10092	-0.1669
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.91724	1.10092	-0.1668
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.87231	1.07923	-0.1917
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.81717	1.05218	-0.2234

Table 2.148 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.98638	1.14917	-0.1417
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.97243	1.14039	-0.1473
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.91195	1.10578	-0.1753
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.90794	1.10092	-0.1753
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.90800	1.10092	-0.1752
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.86086	1.07923	-0.2023
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.80281	1.05218	-0.2370

Table 2.149 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.98710	1.14917	-0.1410
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.97332	1.14039	-0.1465
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.91369	1.10578	-0.1737
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.90967	1.10092	-0.1737
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.90976	1.10092	-0.1736
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.86362	1.07923	-0.1998
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.80651	1.05218	-0.2335

Table 2.150 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	1.00837	1.14917	-0.1225
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.99534	1.14039	-0.1272
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.94110	1.10578	-0.1489
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.93691	1.10092	-0.1490
Hot $Operating_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.93691	1.10092	-0.1490
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.89646	1.07923	-0.1694
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.84672	1.05218	-0.1953

Table 2.151 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.98536	1.14917	-0.1426
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.97138	1.14039	-0.1482
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.91076	1.10578	-0.1764
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.90675	1.10092	-0.1764
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.90683	1.10092	-0.1763
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.85958	1.07923	-0.2035
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.80083	1.05218	-0.2389

Table 2.152 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.98379	1.14917	-0.1439
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.96976	1.14039	-0.1496
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.90872	1.10578	-0.1782
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.90473	1.10092	-0.1782
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.90481	1.10092	-0.1781
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.85704	1.07923	-0.2059
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.79760	1.05218	-0.2419

Table 2.153 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	1.00696	1.14917	-0.1237
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.99369	1.14039	-0.1286
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.93742	1.10578	-0.1522
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.93326	1.10092	-0.1523
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.93331	1.10092	-0.1522
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.89069	1.07923	-0.1747
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.83791	1.05218	-0.2036

Table 2.154 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	1.02933	1.14917	-0.1043
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	1.01676	1.14039	-0.1084
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.96493	1.10578	-0.1274
Hot Operating_{NoXe}	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void, No xenon	0.96060	1.10092	-0.1275
Hot Operating_{0}	$T_{Moderator}$=546° F T_{Fuel}=930° F 0% Void	0.96065	1.10092	-0.1274
Hot Operating_{40}	$T_{Moderator}$=546° F T_{Fuel}=930° F 40% Void	0.92240	1.07923	-0.1453
Hot Operating_{80}	$T_{Moderator}$=546° F T_{Fuel}=930° F 80% Void	0.87465	1.05218	-0.1687

Table 2.155 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter (Δk/k)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.10092	1.10578	-0.0044
Moderator_{Void} = $\frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	1.07923	1.10092	-0.0197
$\text{Moderator}_{Temperature}$ = $\frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.10578	1.14917	-0.0378

Table 2.156 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92547	1.08498	-0.1470
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91218	1.07695	-0.1530
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85911	1.04807	-0.1803
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.85509	1.04319	-0.1803
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.85516	1.04319	-0.1802
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81437	1.02535	-0.2058
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76290	1.00194	-0.2386

Table 2.157 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92373	1.08498	-0.1486
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91050	1.07695	-0.1546
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85731	1.04807	-0.1820
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.85332	1.04319	-0.1820
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.85337	1.04319	-0.1820
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81194	1.02535	-0.2081
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75950	1.00194	-0.2420

Table 2.158 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92066	1.08498	-0.1515
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90721	1.07695	-0.1576
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85233	1.04807	-0.1868
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84837	1.04319	-0.1867
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84837	1.04319	-0.1867
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80538	1.02535	-0.2145
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75083	1.00194	-0.2506

Table 2.159 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91325	1.08498	-0.1583
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.89962	1.07695	-0.1647
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84337	1.04807	-0.1953
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.83945	1.04319	-0.1953
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.83951	1.04319	-0.1953
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79493	1.02535	-0.2247
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.73833	1.00194	-0.2631

Table 2.160 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91682	1.08498	-0.1550
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90309	1.07695	-0.1614
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84689	1.04807	-0.1920
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84295	1.04319	-0.1920
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84295	1.04319	-0.1920
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.79917	1.02535	-0.2206
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74379	1.00194	-0.2576

Table 2.161 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92741	1.08498	-0.1452
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91437	1.07695	-0.1510
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86120	1.04807	-0.1783
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.85720	1.04319	-0.1783
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.85725	1.04319	-0.1782
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81609	1.02535	-0.2041
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76405	1.00194	-0.2374

Table 2.162 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92024	1.08498	-0.1518
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90695	1.07695	-0.1578
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85212	1.04807	-0.1870
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84816	1.04319	-0.1870
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84822	1.04319	-0.1869
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80488	1.02535	-0.2150
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74992	1.00194	-0.2515

Table 2.163 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92089	1.08498	-0.1512
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90777	1.07695	-0.1571
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85368	1.04807	-0.1855
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84972	1.04319	-0.1855
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84979	1.04319	-0.1854
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80742	1.02535	-0.2125
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75338	1.00194	-0.2481

Table 2.164 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94174	1.08498	-0.1320
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92939	1.07695	-0.1370
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88072	1.04807	-0.1597
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87658	1.04319	-0.1597
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87658	1.04319	-0.1597
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83987	1.02535	-0.1809
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.79333	1.00194	-0.2082

Table 2.165 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91945	1.08498	-0.1526
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90611	1.07695	-0.1586
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.85106	1.04807	-0.1880
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84710	1.04319	-0.1880
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84718	1.04319	-0.1879
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80374	1.02535	-0.2161
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74813	1.00194	-0.2533

Table 2.166 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.91790	1.08498	-0.1540
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.90452	1.07695	-0.1601
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.84905	1.04807	-0.1899
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.84511	1.04319	-0.1899
Hot $Operating_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.84519	1.04319	-0.1898
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.80125	1.02535	-0.2186
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74494	1.00194	-0.2565

Table 2.167 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94078	1.08498	-0.1329
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92815	1.07695	-0.1382
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87752	1.04807	-0.1627
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87340	1.04319	-0.1628
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87346	1.04319	-0.1627
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83466	1.02535	-0.1860
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.78509	1.00194	-0.2164

Table 2.168 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.96313	1.08498	-0.1123
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.95123	1.07695	-0.1167
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.90502	1.04807	-0.1365
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.90073	1.04319	-0.1366
Hot $Operating_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.90077	1.04319	-0.1365
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.86640	1.02535	-0.1550
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.82203	1.00194	-0.1796

Table 2.169 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter (Δk/k)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.04319	1.04807	-0.0047
$Moderator_{Void}$ = $\frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	1.02535	1.04319	-0.0171
$Moderator_{Temperature}$ = $\frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.04807	1.08498	-0.0340

Table 2.170 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94095	1.09259	-0.1388
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92801	1.08514	-0.1448
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87659	1.05906	-0.1723
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87268	1.05435	-0.1723
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87274	1.05435	-0.1722
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83059	1.03705	-0.1991
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77362	1.01170	-0.2353

Table 2.171 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93931	1.09259	-0.1403
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92644	1.08514	-0.1462
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87492	1.05906	-0.1739
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87102	1.05435	-0.1739
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87107	1.05435	-0.1738
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82826	1.03705	-0.2013
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77025	1.01170	-0.2387

Table 2.172 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93646	1.09259	-0.1429
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92339	1.08514	-0.1491
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87026	1.05906	-0.1783
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86639	1.05435	-0.1783
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86639	1.05435	-0.1783
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82195	1.03705	-0.2074
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76166	1.01170	-0.2472

Table 2.173 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.92948	1.09259	-0.1493
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91619	1.08514	-0.1557
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86161	1.05906	-0.1864
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.85779	1.05435	-0.1864
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.85784	1.05435	-0.1864
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81171	1.03705	-0.2173
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.74915	1.01170	-0.2595

Table 2.174 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93284	1.09259	-0.1462
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.91949	1.08514	-0.1527
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86500	1.05906	-0.1832
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86116	1.05435	-0.1832
Hot $Operating_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86116	1.05435	-0.1832
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81585	1.03705	-0.2133
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75460	1.01170	-0.2541

Table 2.175 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94290	1.09259	-0.1370
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93024	1.08514	-0.1427
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87892	1.05906	-0.1701
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87502	1.05435	-0.1701
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87506	1.05435	-0.1701
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83259	1.03705	-0.1972
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77497	1.01170	-0.2340

Table 2.176 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93609	1.09259	-0.1432
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92317	1.08514	-0.1493
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87009	1.05906	-0.1784
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86624	1.05435	-0.1784
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86629	1.05435	-0.1784
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82149	1.03705	-0.2079
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76079	1.01170	-0.2480

Table 2.177 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93672	1.09259	-0.1427
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92399	1.08514	-0.1485
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87174	1.05906	-0.1769
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86787	1.05435	-0.1769
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86794	1.05435	-0.1768
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82419	1.03705	-0.2053
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76439	1.01170	-0.2445

Table 2.178 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

$\text{Blade Worth} = \dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95628	1.09259	-0.1248
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.94433	1.08514	-0.1298
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.89756	1.05906	-0.1525
Hot Operating_{NoXe}	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.89353	1.05435	-0.1525
Hot Operating_{0}	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.89353	1.05435	-0.1525
Hot Operating_{40}	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.85577	1.03705	-0.1748
Hot Operating_{80}	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.80417	1.01170	-0.2051

Table 2.179 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		**$k_{Controlled}$**	**$k_{Uncontrolled}$**	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93537	1.09259	-0.1439
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92238	1.08514	-0.1500
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86912	1.05906	-0.1794
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86526	1.05435	-0.1793
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86533	1.05435	-0.1793
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82050	1.03705	-0.2088
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75913	1.01170	-0.2497

Table 2.180 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93390	1.09259	-0.1452
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92088	1.08514	-0.1514
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.86721	1.05906	-0.1812
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86337	1.05435	-0.1811
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86344	1.05435	-0.1811
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.81808	1.03705	-0.2112
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.75594	1.01170	-0.2528

Table 2.181 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95559	1.09259	-0.1254
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.94335	1.08514	-0.1307
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.89463	1.05906	-0.1553
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.89062	1.05435	-0.1553
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.89067	1.05435	-0.1552
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.85079	1.03705	-0.1796
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.79602	1.01170	-0.2132

Table 2.182 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.97669	1.09259	-0.1061
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.96522	1.08514	-0.1105
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.92106	1.05906	-0.1303
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.91690	1.05435	-0.1304
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.91694	1.05435	-0.1303
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.88186	1.03705	-0.1497
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.83292	1.01170	-0.1767

Table 2.183 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter ($\Delta k/k$)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.05435	1.05906	-0.0044
Moderator$_{Void}$ = $\frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	1.03705	1.05435	-0.0164
Moderator$_{Temperature}$ = $\frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.05906	1.09259	-0.0307

Table 2.184 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE OEM and D120

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94929	1.10181	-0.1384
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93680	1.09498	-0.1445
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88788	1.07252	-0.1722
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.88398	1.06781	-0.1722
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.88404	1.06781	-0.1721
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.84305	1.05261	-0.1991
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.78693	1.02986	-0.2359

Table 2.185 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D160 and D190 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94763	1.10181	-0.1399
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93521	1.09498	-0.1459
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88618	1.07252	-0.1737
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.88228	1.06781	-0.1737
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.88233	1.06781	-0.1737
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.84067	1.05261	-0.2014
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.78348	1.02986	-0.2392

Table 2.186 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D190 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94475	1.10181	-0.1426
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93213	1.09498	-0.1487
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88145	1.07252	-0.1782
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87759	1.06781	-0.1781
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87759	1.06781	-0.1781
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83424	1.05261	-0.2075
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77466	1.02986	-0.2478

Table 2.187 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Lower Blade

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.93769	1.10181	-0.1489
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92484	1.09498	-0.1554
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87266	1.07252	-0.1863
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.86885	1.06781	-0.1863
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.86890	1.06781	-0.1863
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82379	1.05261	-0.2174
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76186	1.02986	-0.2602

Table 2.188 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE D230 Blade Tip

Blade Worth = $\dfrac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth ($\Delta k/k$)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94110	1.10181	-0.1459
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92819	1.09498	-0.1523
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87611	1.07252	-0.1831
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87228	1.06781	-0.1831
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87228	1.06781	-0.1831
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.82801	1.05261	-0.2134
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76739	1.02986	-0.2549

Table 2.189 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Bottom Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.95120	1.10181	-0.1367
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93900	1.09498	-0.1424
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.89018	1.07252	-0.1700
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.88628	1.06781	-0.1700
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.88633	1.06781	-0.1700
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.84499	1.05261	-0.1972
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.78819	1.02986	-0.2347

Table 2.190 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Blade Main Segment

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94434	1.10181	-0.1429
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93188	1.09498	-0.1489
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88126	1.07252	-0.1783
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87741	1.06781	-0.1783
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87746	1.06781	-0.1783
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83374	1.05261	-0.2079
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77372	1.02986	-0.2487

Table 2.191 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94494	1.10181	-0.1424
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93265	1.09498	-0.1482
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88283	1.07252	-0.1769
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87897	1.06781	-0.1769
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87904	1.06781	-0.1768
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83637	1.05261	-0.2054
Hot Operating$_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77730	1.02986	-0.2452

Table 2.192 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR82M1 and CR82B Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.96474	1.10181	-0.1244
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.95325	1.09498	-0.1294
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.90910	1.07252	-0.1524
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.90507	1.06781	-0.1524
Hot $Operating_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.90507	1.06781	-0.1524
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.86861	1.05261	-0.1748
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.81810	1.02986	-0.2056

Table 2.193 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Tip

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth (Δk/k)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94362	1.10181	-0.1436
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.93109	1.09498	-0.1497
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.88025	1.07252	-0.1793
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87640	1.06781	-0.1792
Hot Operating$_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87647	1.06781	-0.1792
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83273	1.05261	-0.2089
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.77207	1.02986	-0.2503

Table 2.194 Lattice [ ] Control Blade Worths at BOL for Control Blade Type CR99 Blade Main

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.94214	1.10181	-0.1449
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.92956	1.09498	-0.1511
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.87831	1.07252	-0.1811
Hot Operating$_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.87447	1.06781	-0.1811
Hot Operating$_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.87454	1.06781	-0.1810
Hot Operating$_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.83025	1.05261	-0.2112
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.76880	1.02986	-0.2535

Table 2.195 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD2

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	Blade Worth ($\Delta k/k$)
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.96406	1.10181	-0.1250
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.95230	1.09498	-0.1303
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.90620	1.07252	-0.1551
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.90220	1.06781	-0.1551
Hot $Operating_{0}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.90226	1.06781	-0.1550
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.86365	1.05261	-0.1795
Hot $Operating_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.80990	1.02986	-0.2136

Table 2.196 Lattice [ ] Control Blade Worths at BOL for Control Blade Type GE Marathon Ultra HD3

Blade Worth = $\frac{k_{Controlled} - k_{Uncontrolled}}{k_{Uncontrolled}}$		$k_{Controlled}$	$k_{Uncontrolled}$	**Blade Worth (Δk/k)**
Cold	$T_{Moderator}$=68° F T_{Fuel}=68° F 0% Void, No xenon	0.98540	1.10181	-0.1056
Intermediate	$T_{Moderator}$=200° F T_{Fuel}=200° F 0% Void, No xenon	0.97444	1.09498	-0.1101
Hot Standby	$T_{Moderator}$=546° F T_{Fuel}=546° F 0% Void, No xenon	0.93309	1.07252	-0.1300
Hot $Operating_{NoXe}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void, No xenon	0.92892	1.06781	-0.1301
Hot $Operating_0$	$T_{Moderator}$=546° F T_{Fuel}=989° F 0% Void	0.92897	1.06781	-0.1300
Hot $Operating_{40}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 40% Void	0.89537	1.05261	-0.1494
Hot Operating $_{80}$	$T_{Moderator}$=546° F T_{Fuel}=989° F 80% Void	0.84782	1.02986	-0.1768

Table 2.197 Lattice [ ] Kinetics Parameters at BOL

Kinetics Parameter = $\frac{k_\alpha - k_\beta}{k_\beta}$	k_α	k_β	Kinetics Parameter (Δk/k)
Doppler = $\frac{k_{HotOperatingNoXe} - k_{HotStandby}}{k_{HotStandby}}$	1.06781	1.07252	-0.0044
$Moderator_{Void} = \frac{k_{HotOperating40} - k_{HotOperating0}}{k_{HotOperating0}}$	1.05261	1.06781	-0.0142
$Moderator_{Temperature} = \frac{k_{HotStandby} - k_{Cold}}{k_{Cold}}$	1.07252	1.10181	-0.0266

[

]

Figure 2.1 Assembly Type [ ]

[

]

Figure 2.2 Assembly Type [ ]

[

]

Figure 2.3 Assembly Type [ ]

1	2	3	4	5	6	5	4	3	7
2	4	5	8	9	8	9	8	10	3
3	5	9	9	11	9	11	9	8	6
4	8	9	12	9	9	9	8	9	10
5	9	11	9				11	8	10
6	8	9	9		Water Channel		10	9	10
5	9	11	9				11	8	10
4	8	9	8	11	10	11	9	10	6
3	10	8	9	8	9	8	10	12	4
7	3	6	10	10	10	10	6	4	7

Fuel Rod Type	Quantity
1	1
2	2
3	6
4	7
5	6
6	6

Fuel Rod Type	Quantity
7	3
8	14
9	22
10	14
11	8
12	2

Figure 2.4 [ ] Fuel Rod Distribution

1	2	3	4	5	5	5	4	3	7
2	4	13	9	9	13	9	13	10	3
3	13	10	9	11	9	11	9	13	5
4	9	9	5	9	9	9	10	9	10
5	9	11	9				11	13	10
5	13	9	9		Water Channel		10	9	10
5	9	11	9				11	13	10
4	13	9	10	11	10	11	9	10	10
3	10	13	9	13	9	13	10	13	4
7	3	5	10	10	10	10	10	4	7

Fuel Rod Type	Quantity	Fuel Rod Type	Quantity
1	1	7	3
2	2	9	23
3	6	10	19
4	7	11	8
5	9	13	13

Figure 2.5 [ ] Fuel Rod Distribution

1	2	3	4	6	6	6	5	3	7
2	3	5	13	9	13	9	13	10	4
3	5	10	9	11	9	11	9	13	6
4	13	9	5	9	9	9	10	9	10
6	9	11	9				11	13	10
6	13	9	9	Water Channel			10	9	10
6	9	11	9				11	13	10
5	13	9	10	11	10	11	9	10	6
3	10	13	9	13	9	13	10	9	4
7	4	6	10	10	10	10	6	4	7

Fuel Rod Type	Quantity	Fuel Rod Type	Quantity
1	1	7	3
2	2	9	22
3	5	10	17
4	6	11	8
5	5	13	12
6	10		

Figure 2.6 [ ] Fuel Rod Distribution

[

]

Figure 2.7 Fuel Rod Axial Description

3.0 References

1. EMF-93-177(P)(A) Revision 1, *Mechanical Design for BWR Fuel Channels,* Framatome ANP, August 2005.

2. ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995.

3. 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, October 1999.

Appendix A Enriched Lattice Hot Uncontrolled Reactivity and LPF Plots

The results in this appendix are based on hot operating and equilibrium xenon conditions.

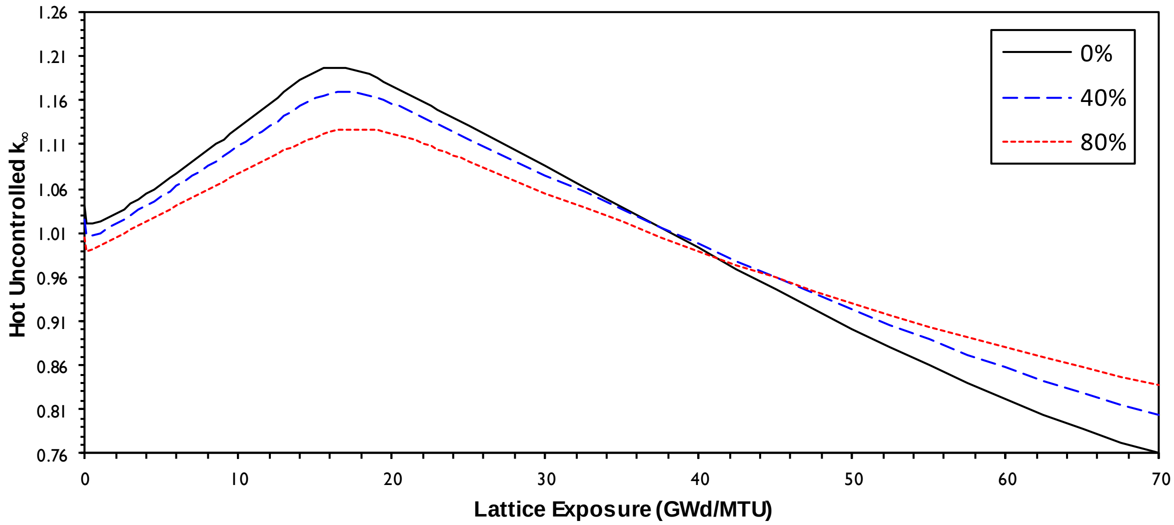


Figure A.1 [ ] Hot Uncontrolled k_∞

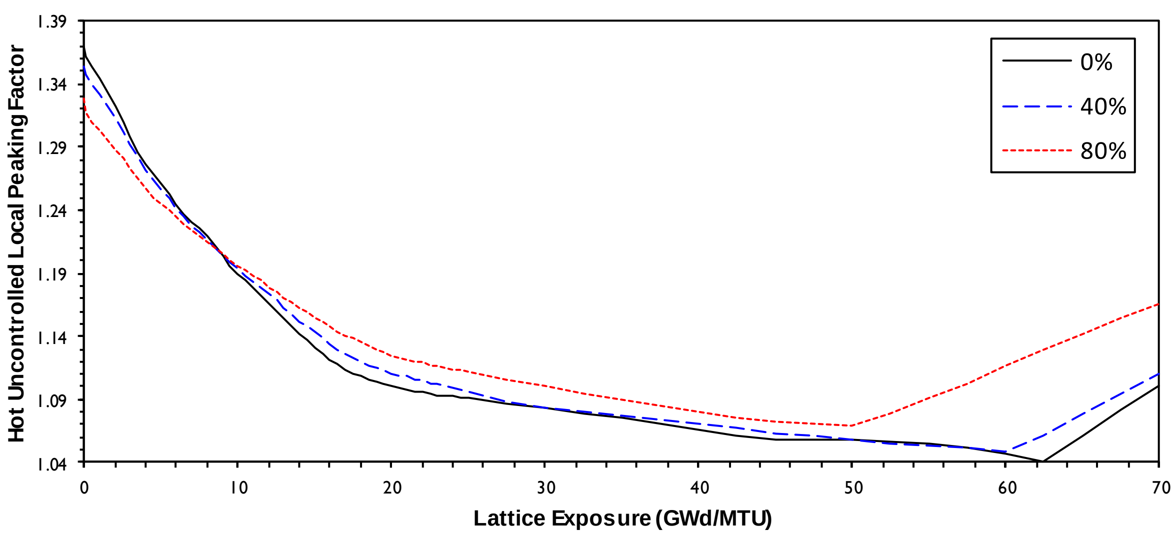


Figure A.2 [ ] Hot Uncontrolled LPF

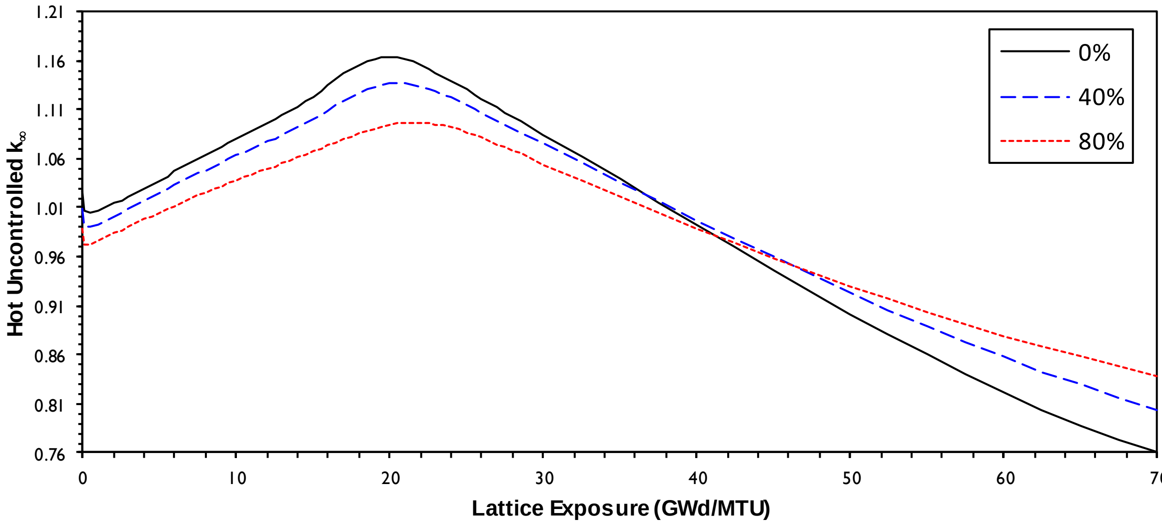


Figure A.3 [ ] Hot Uncontrolled k_∞

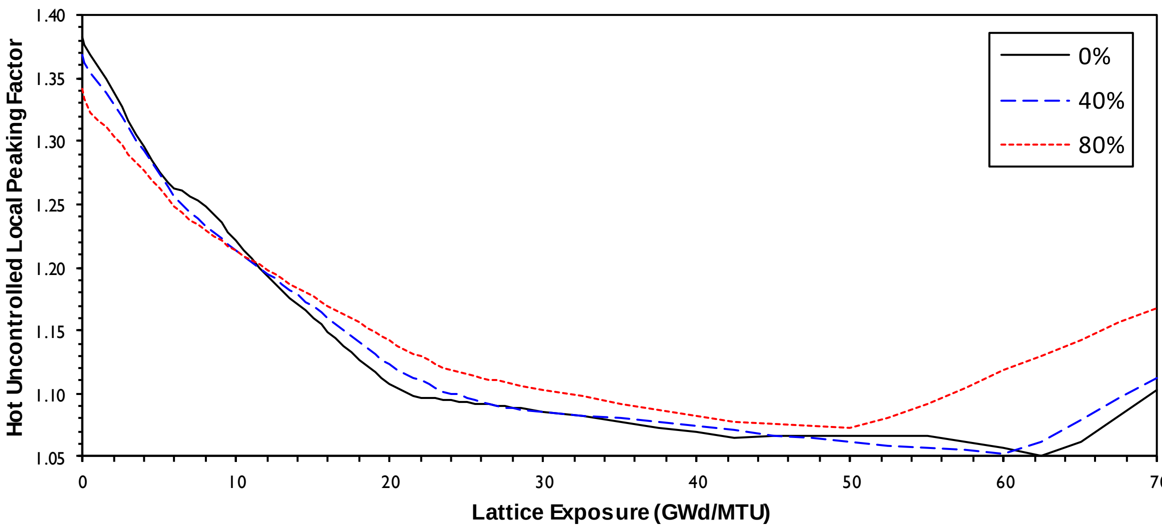


Figure A.4 [ ] Hot Uncontrolled LPF

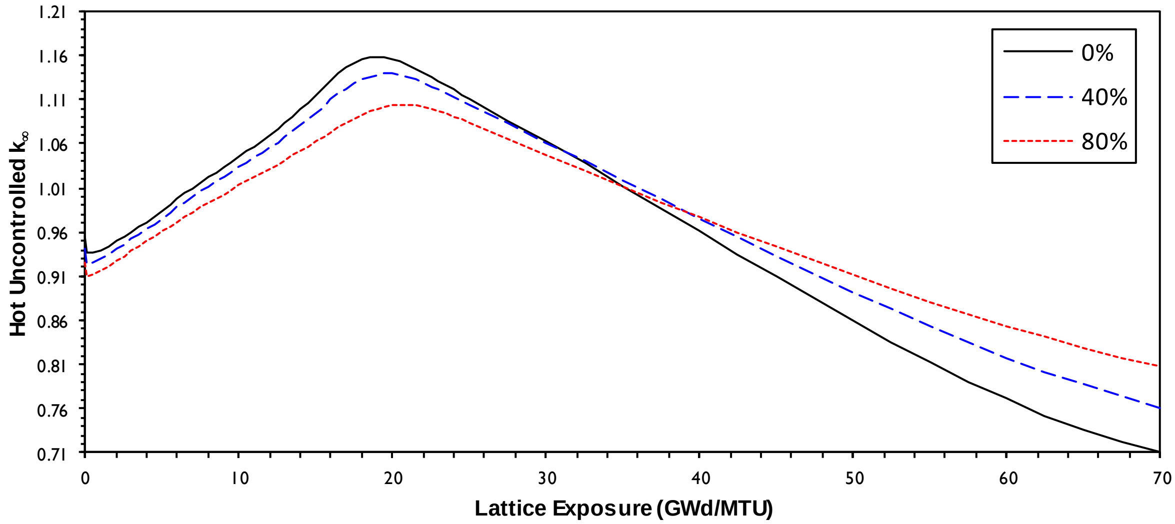


Figure A.5 [ ] Hot Uncontrolled k_∞

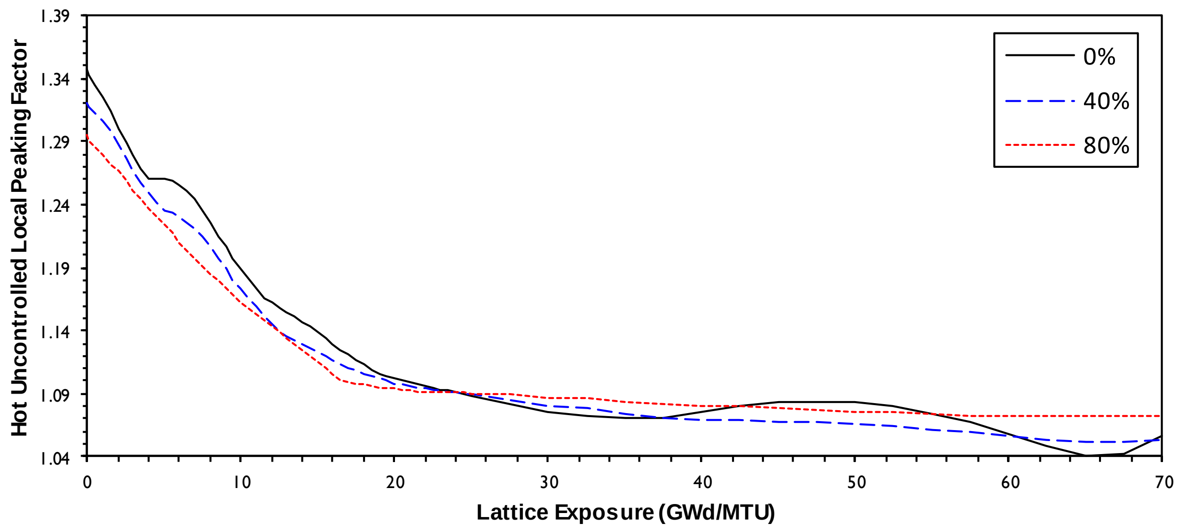


Figure A.6 [ ] Hot Uncontrolled LPF

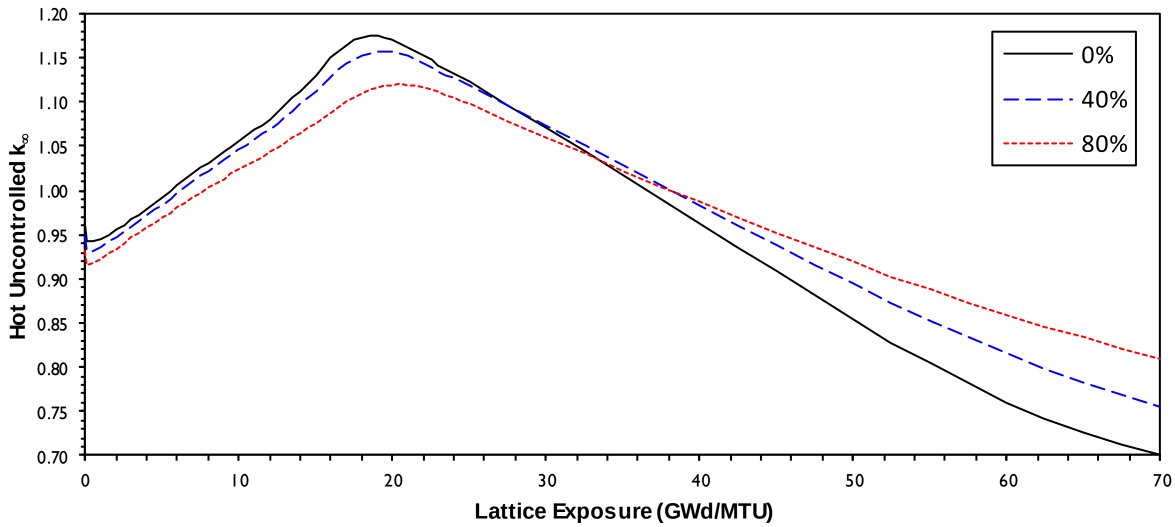


Figure A.7 [ ] Hot Uncontrolled k_∞

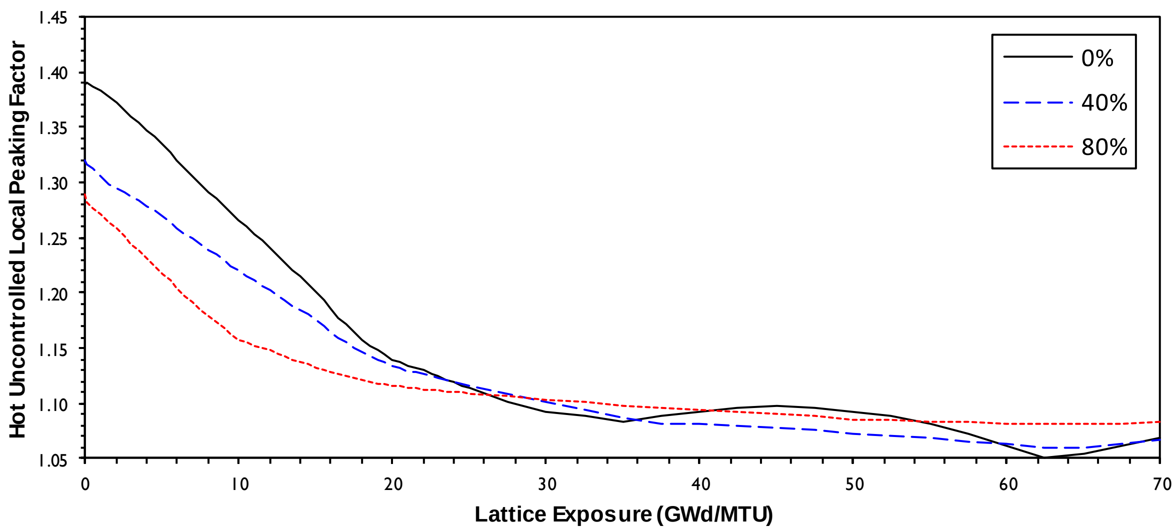


Figure A.8 [ ] Hot Uncontrolled LPF

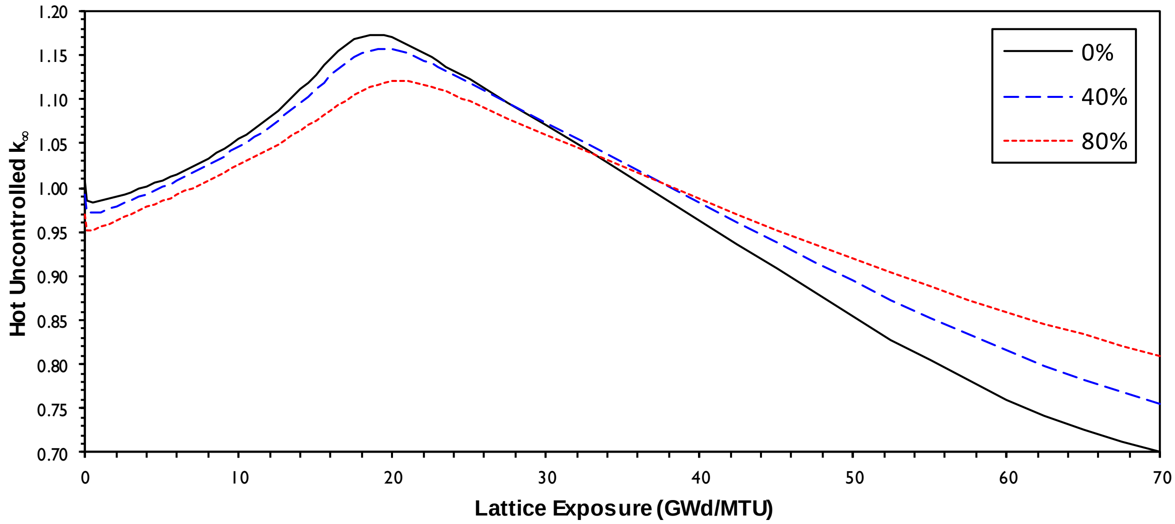


Figure A.9 [ ] Hot Uncontrolled k_∞

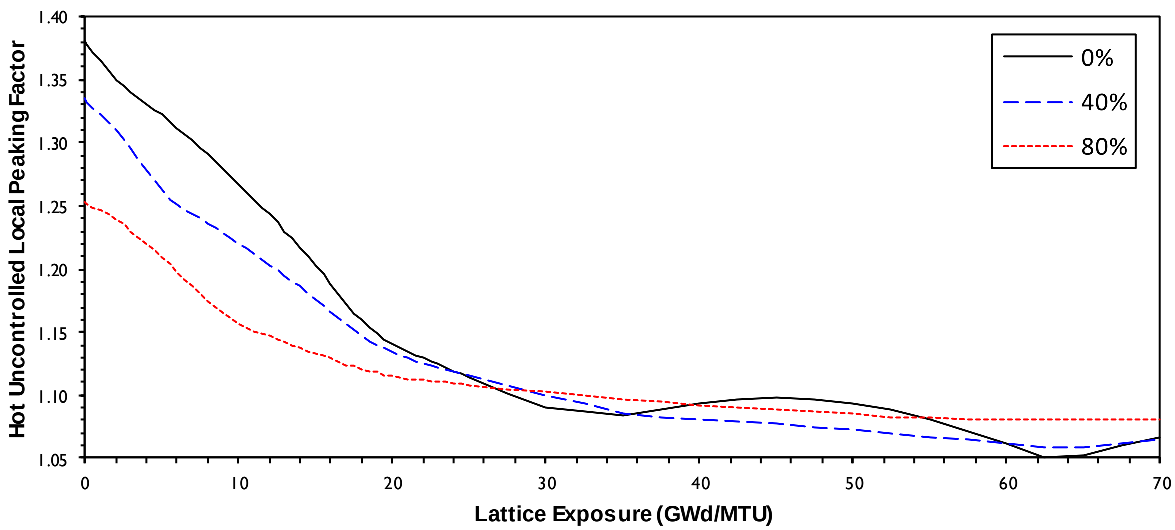


Figure A.10 [ ] Hot Uncontrolled LPF

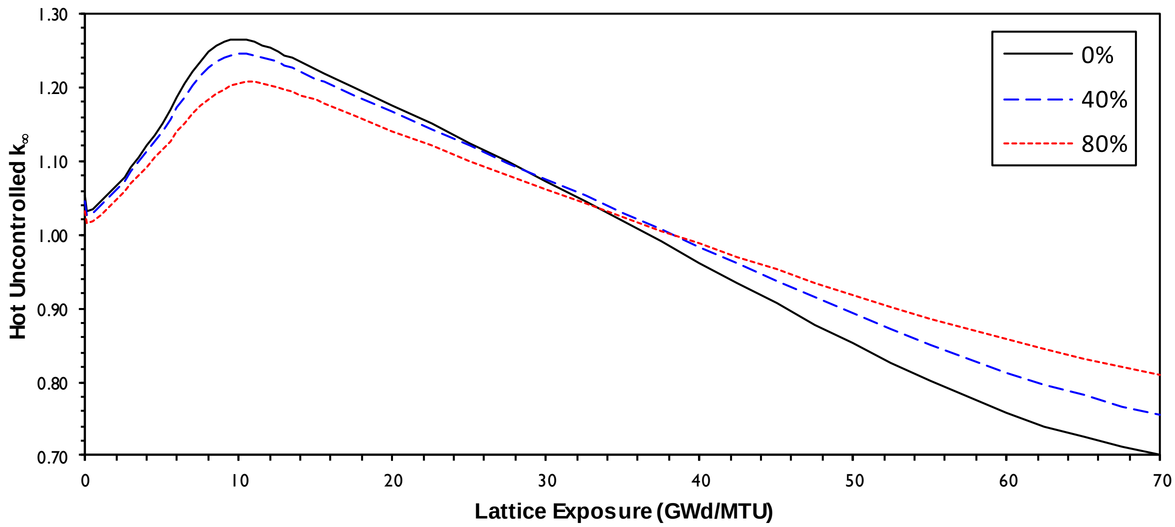


Figure A.11 [ ] Hot Uncontrolled k_∞

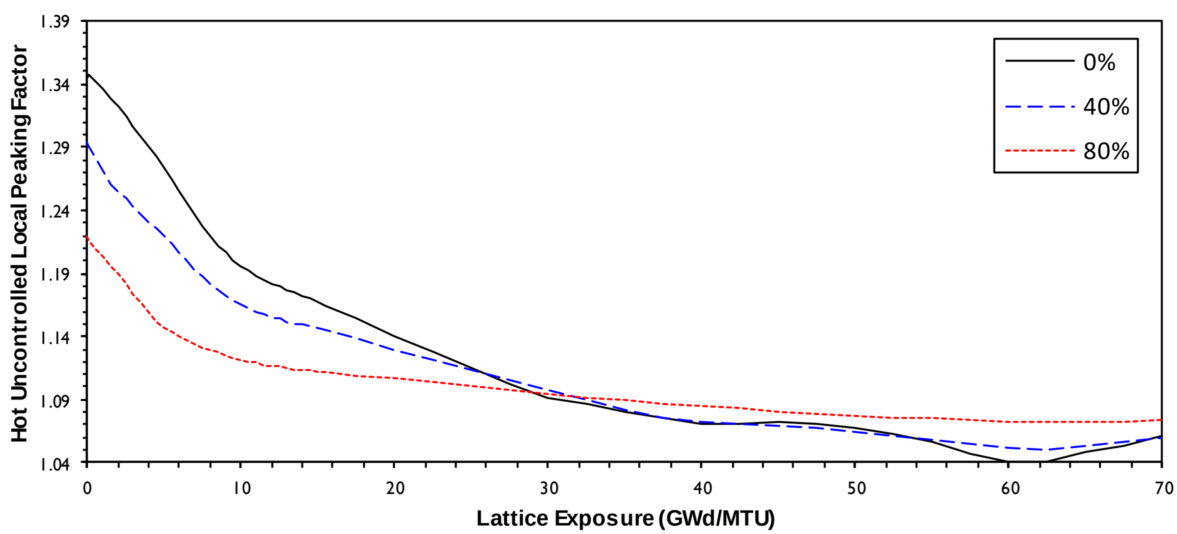


Figure A.12 [ ] Hot Uncontrolled LPF

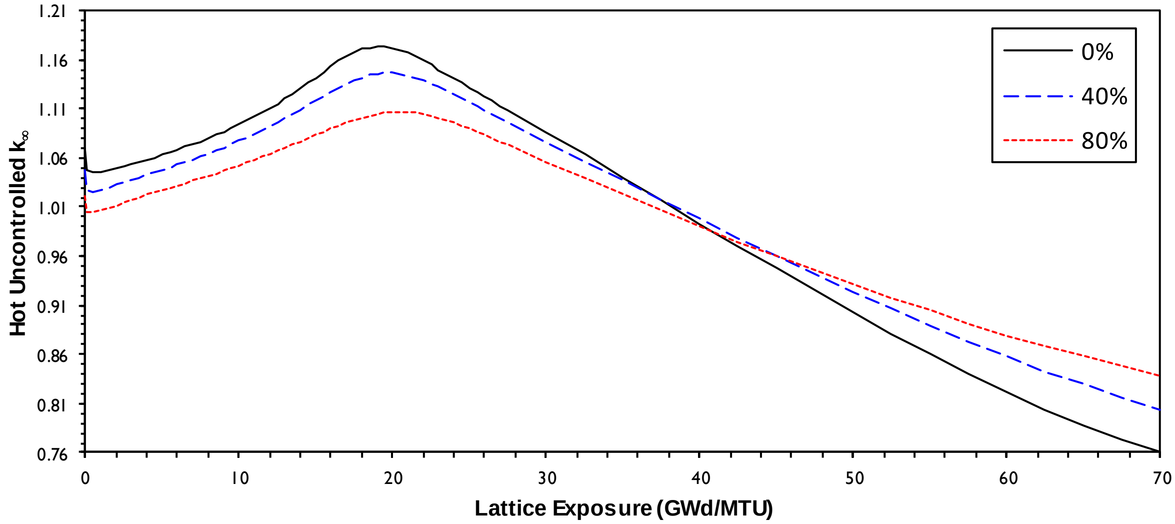


Figure A.13 [ ] Hot Uncontrolled k_∞

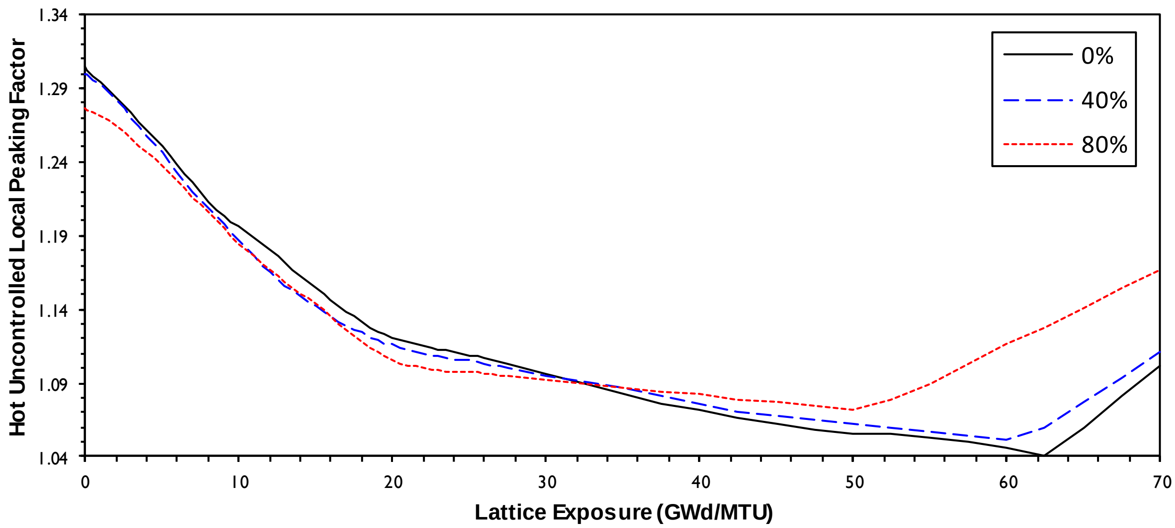


Figure A.14 [ ] Hot Uncontrolled LPF

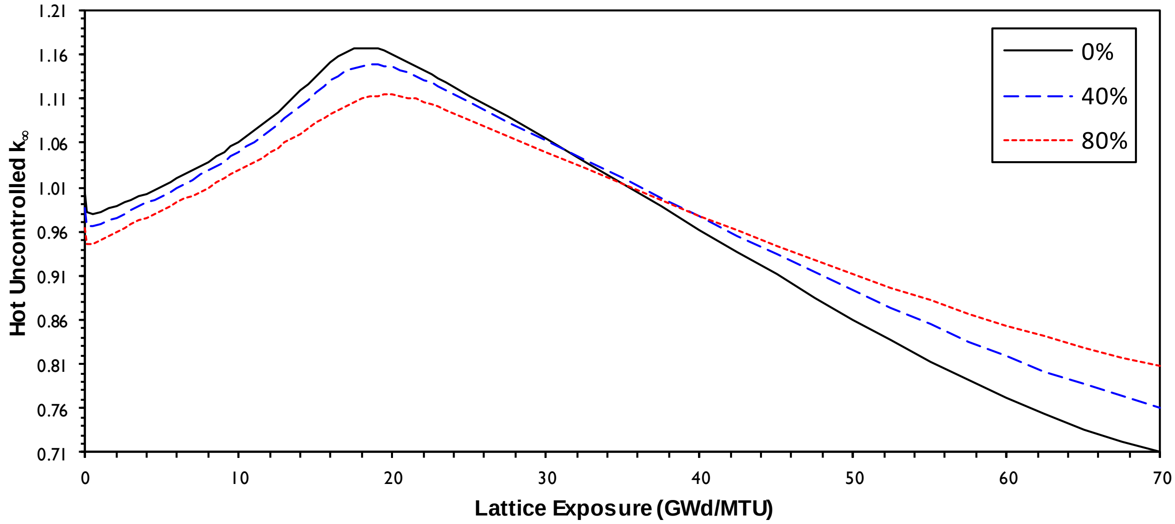


Figure A.15 [ ] Hot Uncontrolled k_∞

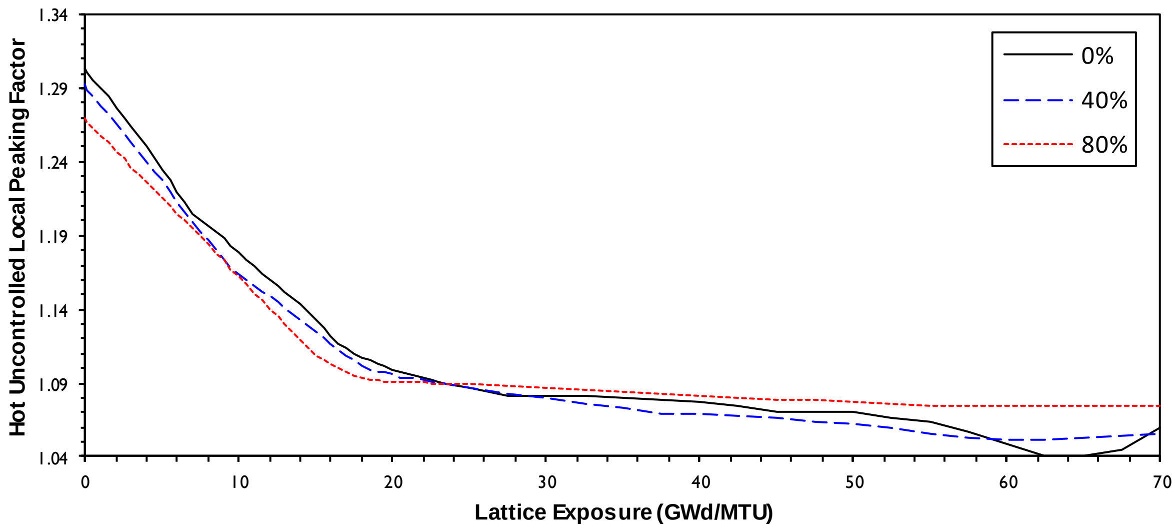


Figure A.16 [ ] Hot Uncontrolled LPF

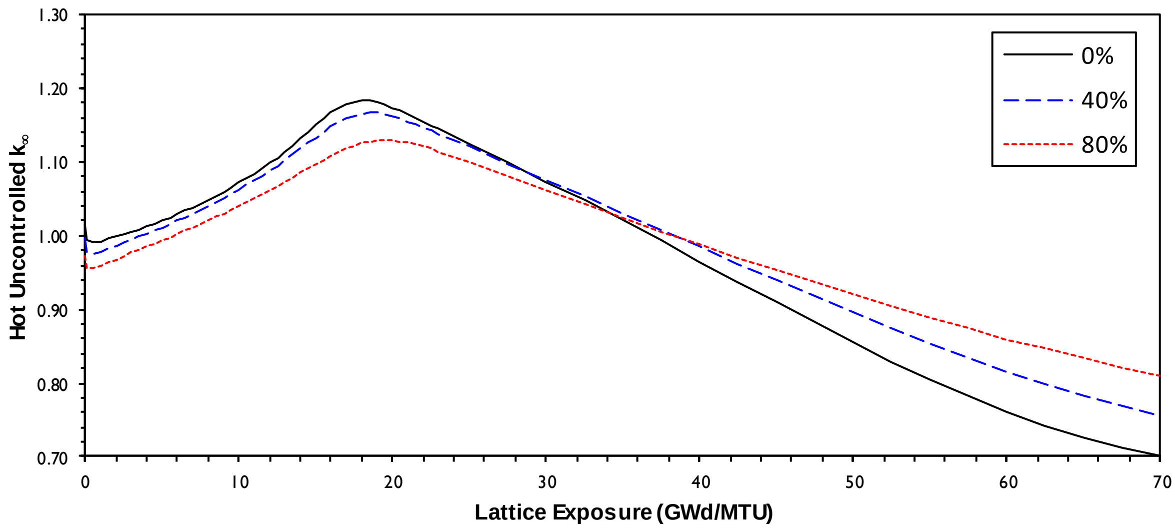


Figure A.17 [ ] Hot Uncontrolled k_∞

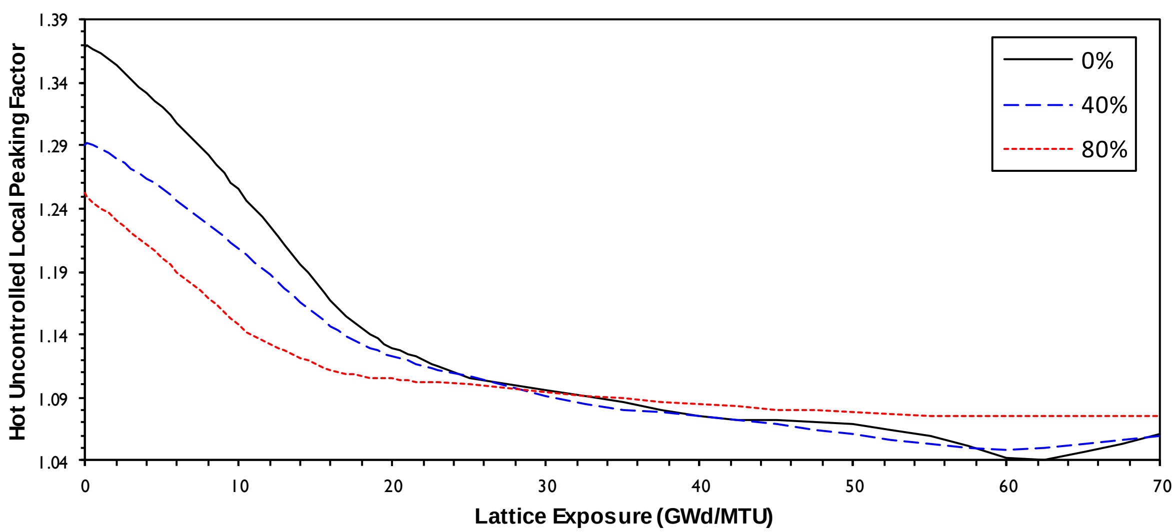


Figure A.18 [ ] Hot Uncontrolled LPF

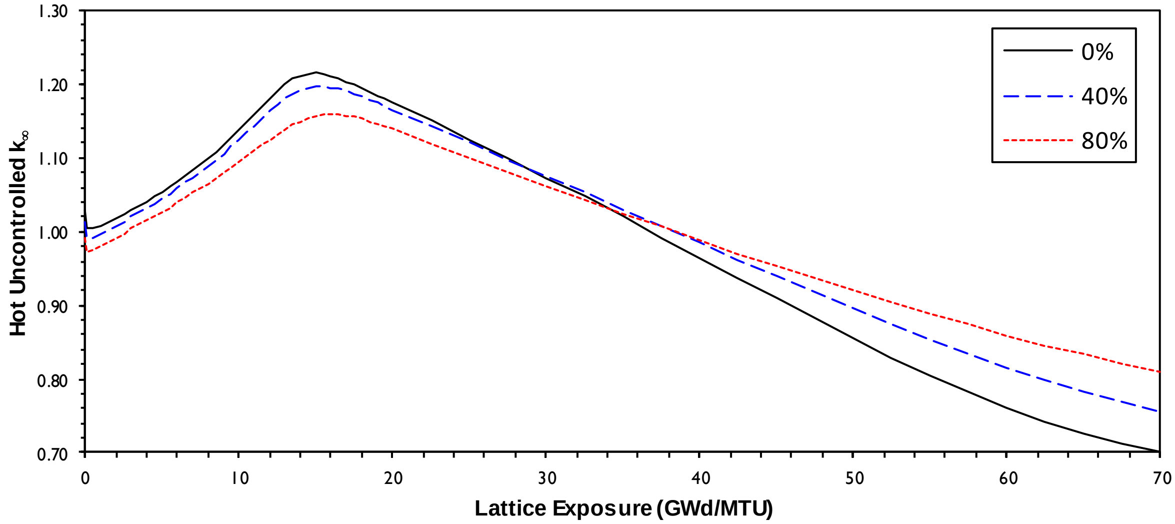


Figure A.19 [ ] Hot Uncontrolled k$_\infty$

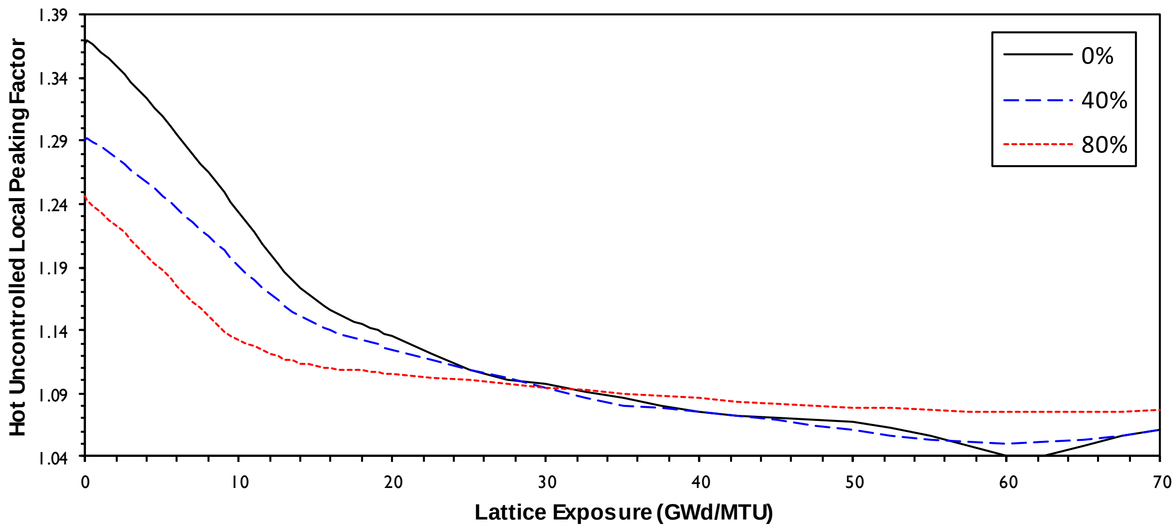


Figure A.20 [ ] Hot Uncontrolled LPF

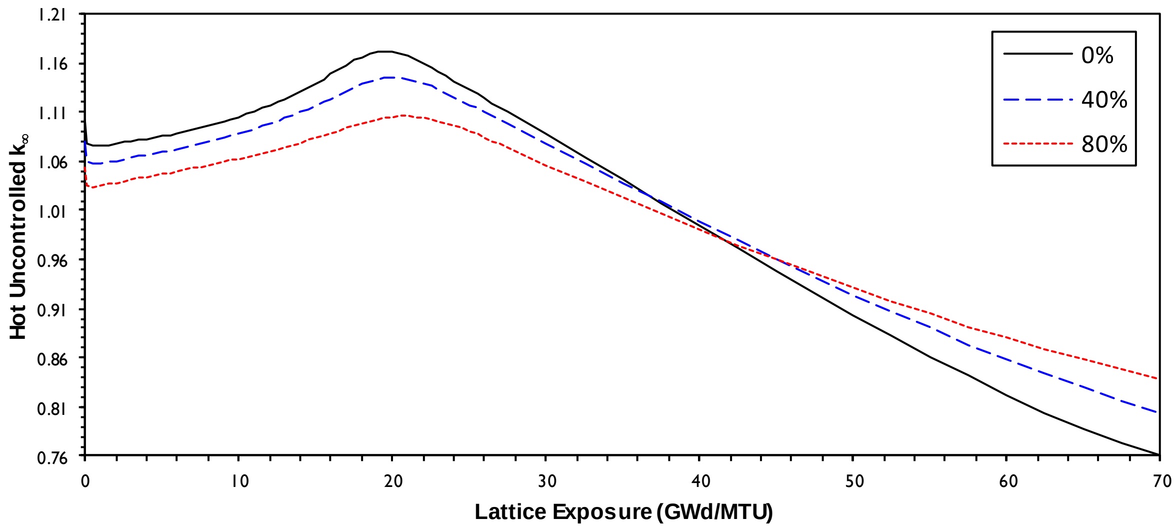


Figure A.21 [ ] Hot Uncontrolled k_∞

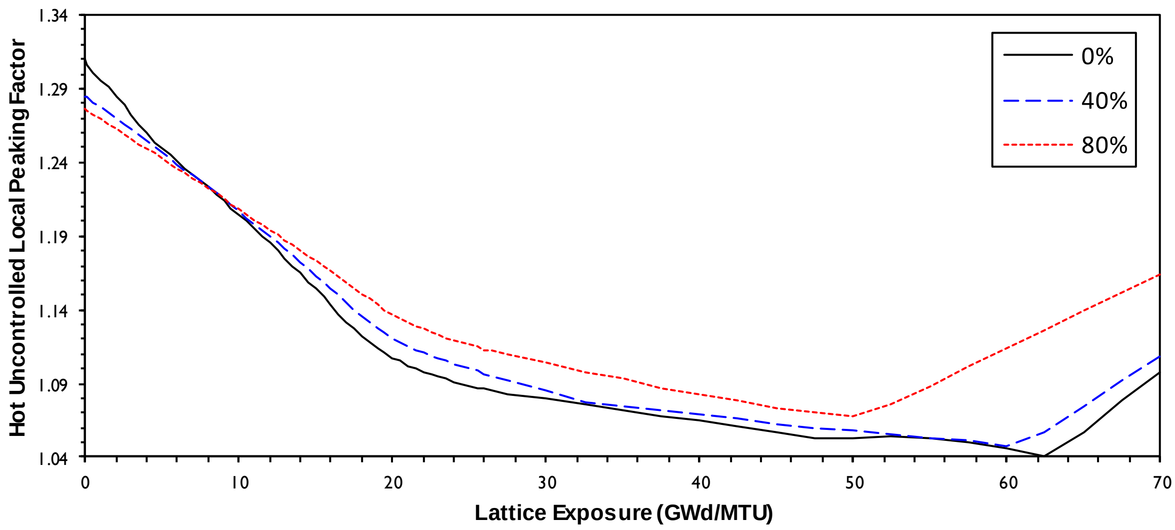


Figure A.22 [ ] Hot Uncontrolled LPF

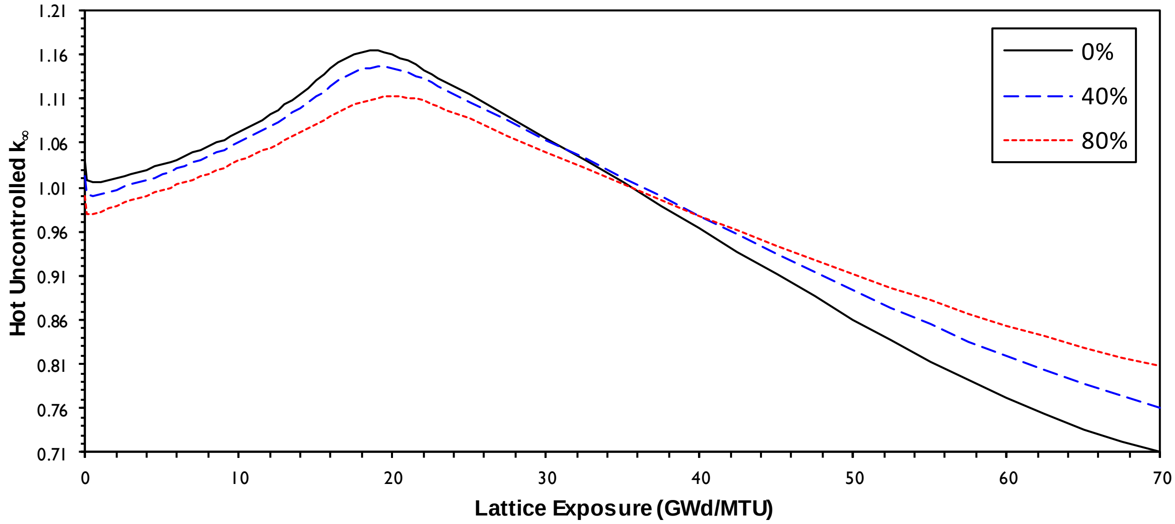


Figure A.23 [ ] Hot Uncontrolled k_∞

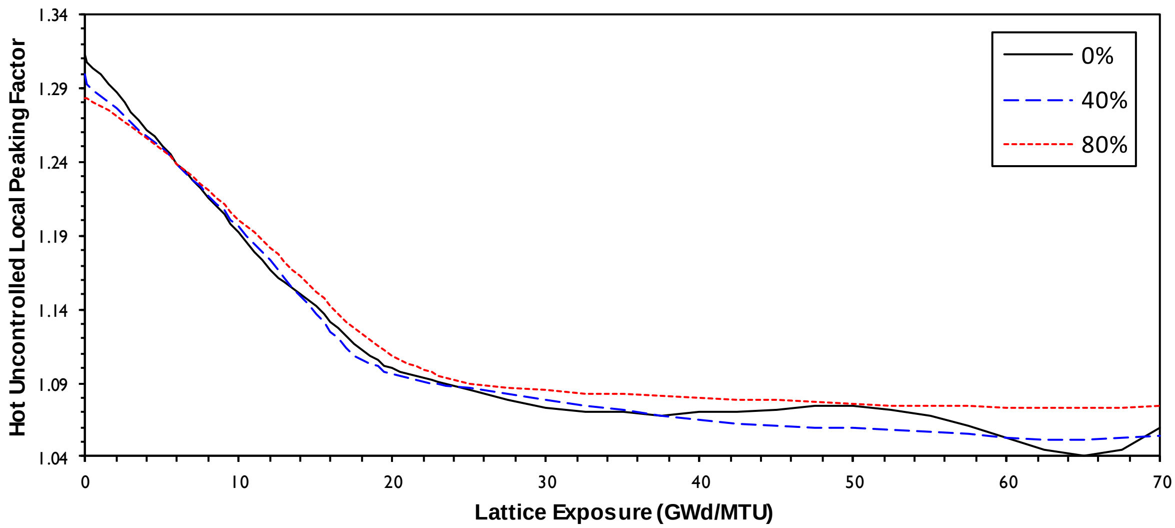


Figure A.24 [ ] Hot Uncontrolled LPF

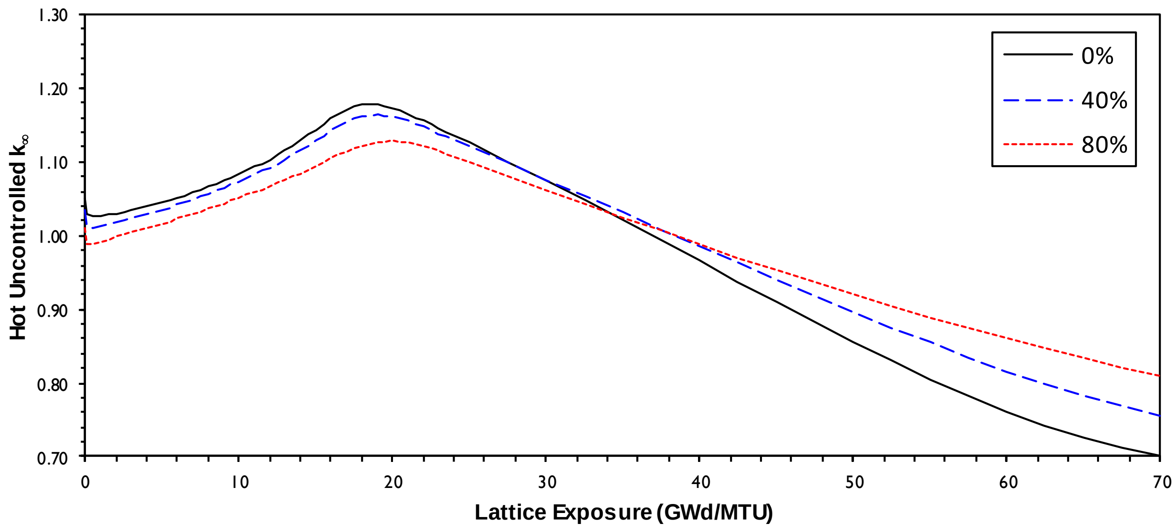


Figure A.25 [ ] Hot Uncontrolled k_∞

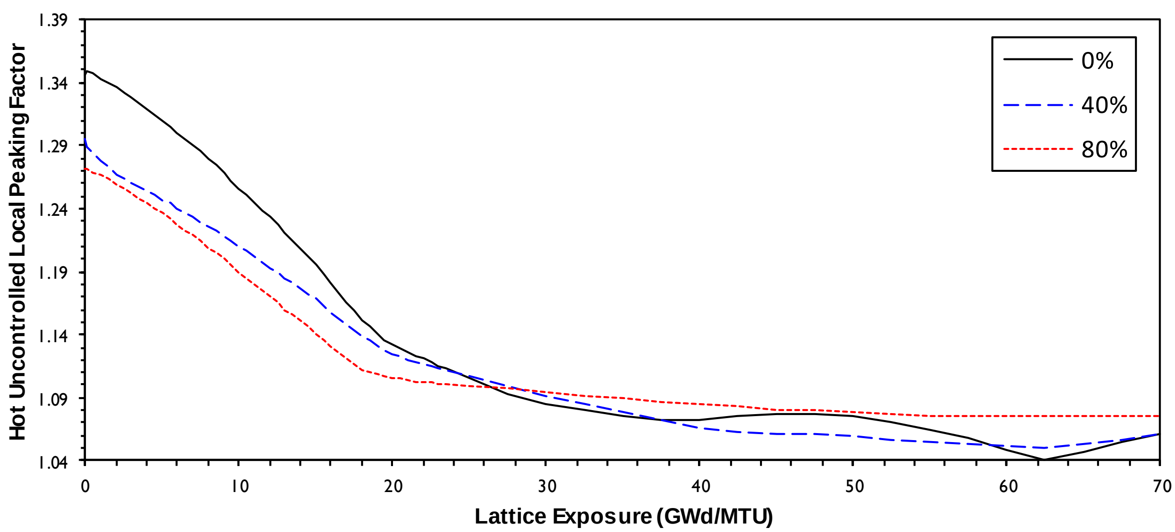


Figure A.26 [ ] Hot Uncontrolled LPF

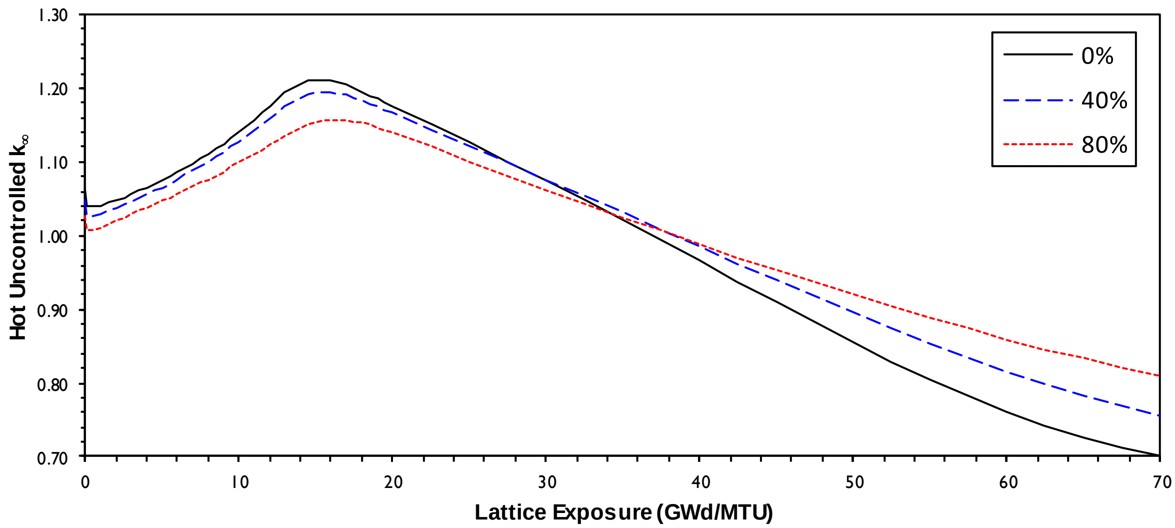


Figure A.27 [ ] Hot Uncontrolled k_{∞}

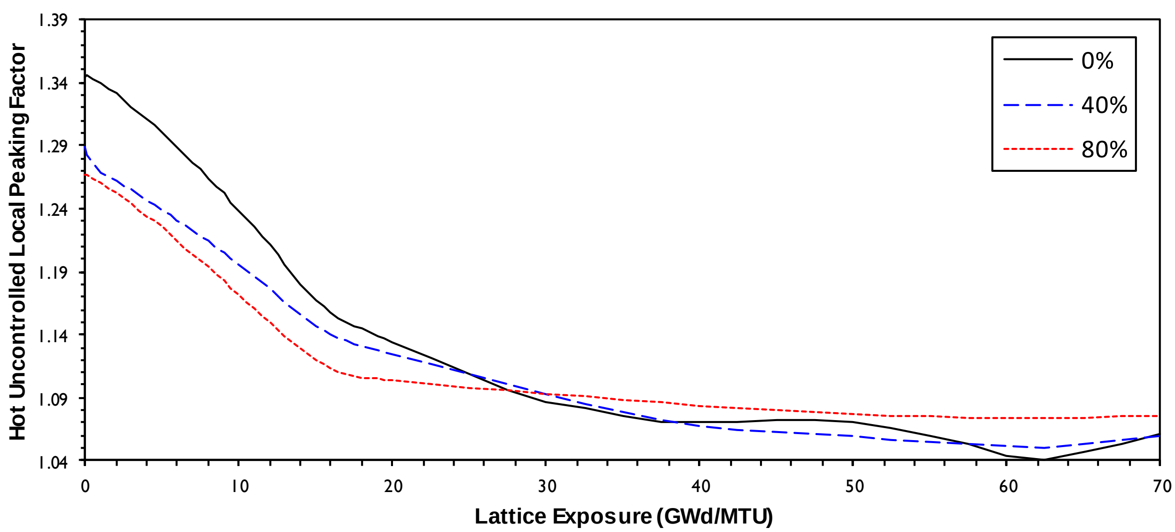


Figure A.28 [ ] Hot Uncontrolled LPF

Appendix B Enriched Lattice Hot Uncontrolled Reactivity and LPF Tables

The results in this appendix are based on hot operating and equilibrium xenon conditions.

Table B.1 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.03648	1.02079	1.00157
100.	1.01611	1.00216	0.98536
500.	1.01579	1.00267	0.98676
1000.	1.01919	1.00660	0.99102
1500.	1.02374	1.01142	0.99580
2000.	1.02865	1.01648	1.00063
2500.	1.03371	1.02158	1.00537
3000.	1.03889	1.02669	1.00999
3500.	1.04422	1.03184	1.01454
4000.	1.04975	1.03706	1.01905
4500.	1.05552	1.04239	1.02355
5000.	1.06153	1.04786	1.02806
5500.	1.06777	1.05346	1.03262
6000.	1.07417	1.05916	1.03721
6500.	1.08063	1.06491	1.04185
7000.	1.08707	1.07064	1.04650
7500.	1.09345	1.07630	1.05113
8000.	1.09978	1.08190	1.05572
8500.	1.10608	1.08747	1.06026
9000.	1.11235	1.09300	1.06474
9500.	1.11864	1.09852	1.06918
10000.	1.12502	1.10405	1.07358
10500.	1.13155	1.10964	1.07795
11000.	1.13829	1.11531	1.08229
11500.	1.14526	1.12109	1.08663
12000.	1.15242	1.12696	1.09096
12500.	1.15968	1.13288	1.09528
13000.	1.16683	1.13873	1.09954
13500.	1.17359	1.14438	1.10370
14000.	1.17974	1.14963	1.10768
14500.	1.18499	1.15434	1.11138
15000.	1.18915	1.15833	1.11474
15500.	1.19205	1.16152	1.11766
16000.	1.19363	1.16379	1.12009
16500.	1.19392	1.16510	1.12196
17000.	1.19304	1.16546	1.12322
17500.	1.19115	1.16494	1.12386
18000.	1.18846	1.16361	1.12389
18500.	1.18515	1.16159	1.12334
19000.	1.18141	1.15901	1.12225
19500.	1.17735	1.15598	1.12069
20000.	1.17309	1.15260	1.11869
20500.	1.16853	1.14897	1.11634
21000.	1.16397	1.14517	1.11368
21500.	1.15940	1.14124	1.11077
22000.	1.15484	1.13724	1.10767
22500.	1.15028	1.13313	1.10441
23000.	1.14571	1.12901	1.10105
23500.	1.14114	1.12489	1.09761
24000.	1.13656	1.12076	1.09411
24500.	1.13199	1.11664	1.09058
25000.	1.12742	1.11252	1.08700
27500.	1.10449	1.09219	1.06921
30000.	1.08149	1.07206	1.05177
32500.	1.05840	1.05211	1.03471
35000.	1.03523	1.03234	1.01802
37500.	1.01201	1.01277	1.00173
40000.	0.98880	0.99343	0.98582
42500.	0.96568	0.97436	0.97034
45000.	0.94274	0.95561	0.95530
47500.	0.92011	0.93724	0.94072
50000.	0.89792	0.91930	0.92665
52500.	0.87636	0.90187	0.91310
55000.	0.85558	0.88502	0.90008
57500.	0.83576	0.86885	0.88763
60000.	0.81708	0.85342	0.87575
62500.	0.79967	0.83879	0.86446
65000.	0.78366	0.82503	0.85377
67500.	0.76914	0.81219	0.84368
70000.	0.75613	0.80029	0.83421

Table B.2 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.366	1.351	1.325
100.	1.359	1.345	1.315
500.	1.351	1.337	1.306
1000.	1.341	1.329	1.300
1500.	1.330	1.319	1.293
2000.	1.319	1.309	1.285
2500.	1.307	1.299	1.278
3000.	1.295	1.289	1.270
3500.	1.283	1.279	1.262
4000.	1.274	1.269	1.255
4500.	1.266	1.260	1.247
5000.	1.258	1.253	1.242
5500.	1.250	1.246	1.237
6000.	1.242	1.239	1.232
6500.	1.234	1.232	1.226
7000.	1.228	1.225	1.221
7500.	1.223	1.219	1.216
8000.	1.216	1.213	1.211
8500.	1.209	1.207	1.207
9000.	1.201	1.201	1.202
9500.	1.193	1.196	1.198
10000.	1.187	1.191	1.193
10500.	1.181	1.185	1.189
11000.	1.175	1.180	1.185
11500.	1.169	1.175	1.181
12000.	1.163	1.170	1.176
12500.	1.157	1.165	1.172
13000.	1.151	1.160	1.168
13500.	1.145	1.155	1.164
14000.	1.139	1.149	1.160
14500.	1.134	1.145	1.156
15000.	1.128	1.140	1.152
15500.	1.123	1.135	1.149
16000.	1.119	1.131	1.145
16500.	1.115	1.127	1.141
17000.	1.111	1.123	1.138
17500.	1.108	1.120	1.135
18000.	1.106	1.117	1.132
18500.	1.103	1.114	1.129
19000.	1.101	1.112	1.126
19500.	1.099	1.110	1.124
20000.	1.098	1.108	1.122
20500.	1.097	1.106	1.120
21000.	1.095	1.105	1.118
21500.	1.094	1.103	1.117
22000.	1.092	1.102	1.116
22500.	1.091	1.100	1.114
23000.	1.090	1.099	1.113
23500.	1.090	1.097	1.112
24000.	1.089	1.096	1.111
24500.	1.089	1.094	1.110
25000.	1.088	1.093	1.109
27500.	1.084	1.085	1.103
30000.	1.081	1.080	1.098
32500.	1.076	1.077	1.092
35000.	1.072	1.074	1.087
37500.	1.067	1.071	1.082
40000.	1.063	1.068	1.077
42500.	1.059	1.064	1.073
45000.	1.055	1.060	1.069
47500.	1.055	1.058	1.068
50000.	1.055	1.055	1.066
52500.	1.054	1.052	1.076
55000.	1.052	1.050	1.088
57500.	1.049	1.048	1.100
60000.	1.044	1.046	1.113
62500.	1.037	1.058	1.126
65000.	1.058	1.075	1.139
67500.	1.079	1.092	1.151
70000.	1.098	1.108	1.163

Table B.3 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.02242	1.00489	0.98363
100.	1.00211	0.98635	0.96753
500.	1.00101	0.98616	0.96831
1000.	1.00325	0.98906	0.97171
1500.	1.00647	0.99271	0.97555
2000.	1.01003	0.99656	0.97941
2500.	1.01373	1.00045	0.98319
3000.	1.01748	1.00430	0.98684
3500.	1.02132	1.00813	0.99038
4000.	1.02529	1.01200	0.99386
4500.	1.02940	1.01591	0.99730
5000.	1.03366	1.01988	1.00071
5500.	1.03806	1.02392	1.00413
6000.	1.04257	1.02802	1.00755
6500.	1.04708	1.03214	1.01098
7000.	1.05153	1.03622	1.01441
7500.	1.05586	1.04021	1.01781
8000.	1.06005	1.04409	1.02116
8500.	1.06409	1.04788	1.02443
9000.	1.06800	1.05156	1.02763
9500.	1.07181	1.05515	1.03075
10000.	1.07559	1.05868	1.03380
10500.	1.07938	1.06217	1.03677
11000.	1.08322	1.06566	1.03969
11500.	1.08712	1.06918	1.04256
12000.	1.09112	1.07273	1.04541
12500.	1.09525	1.07634	1.04825
13000.	1.09955	1.08004	1.05109
13500.	1.10406	1.08384	1.05395
14000.	1.10883	1.08777	1.05684
14500.	1.11389	1.09185	1.05978
15000.	1.11923	1.09609	1.06278
15500.	1.12481	1.10050	1.06583
16000.	1.13056	1.10501	1.06892
16500.	1.13632	1.10955	1.07205
17000.	1.14186	1.11399	1.07515
17500.	1.14694	1.11824	1.07818
18000.	1.15139	1.12215	1.08107
18500.	1.15497	1.12556	1.08376
19000.	1.15752	1.12836	1.08621
19500.	1.15895	1.13045	1.08834
20000.	1.15924	1.13178	1.09008
20500.	1.15847	1.13232	1.09139
21000.	1.15675	1.13206	1.09222
21500.	1.15425	1.13107	1.09257
22000.	1.15114	1.12940	1.09242
22500.	1.14754	1.12717	1.09179
23000.	1.14360	1.12447	1.09069
23500.	1.13943	1.12139	1.08918
24000.	1.13491	1.11802	1.08729
24500.	1.13040	1.11443	1.08507
25000.	1.12588	1.11068	1.08257
25500.	1.12132	1.10682	1.07984
26000.	1.11676	1.10278	1.07691
26500.	1.11219	1.09875	1.07383
27000.	1.10763	1.09471	1.07063
27500.	1.10307	1.09068	1.06736
28000.	1.09850	1.08668	1.06402
28500.	1.09393	1.08268	1.06065
30000.	1.08021	1.07069	1.05032
32500.	1.05727	1.05086	1.03340
35000.	1.03427	1.03123	1.01682
37500.	1.01122	1.01179	1.00062
40000.	0.98818	0.99259	0.98481
42500.	0.96521	0.97365	0.96941
45000.	0.94242	0.95501	0.95446
47500.	0.91991	0.93674	0.93996
50000.	0.89782	0.91890	0.92596
52500.	0.87633	0.90154	0.91246
55000.	0.85559	0.88476	0.89949
57500.	0.83578	0.86862	0.88707
60000.	0.81707	0.85320	0.87521
62500.	0.79960	0.83856	0.86393
65000.	0.78351	0.82478	0.85324
67500.	0.76887	0.81188	0.84314
70000.	0.75573	0.79992	0.83365

Table B.4 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.378	1.363	1.337
100.	1.371	1.357	1.328
500.	1.363	1.350	1.318
1000.	1.354	1.342	1.312
1500.	1.344	1.333	1.306
2000.	1.334	1.324	1.299
2500.	1.323	1.315	1.292
3000.	1.312	1.306	1.285
3500.	1.301	1.296	1.278
4000.	1.290	1.287	1.271
4500.	1.279	1.278	1.264
5000.	1.270	1.268	1.257
5500.	1.263	1.259	1.250
6000.	1.258	1.251	1.243
6500.	1.256	1.245	1.238
7000.	1.252	1.239	1.233
7500.	1.248	1.234	1.229
8000.	1.243	1.228	1.224
8500.	1.237	1.223	1.220
9000.	1.230	1.218	1.216
9500.	1.223	1.213	1.212
10000.	1.217	1.208	1.208
10500.	1.209	1.204	1.204
11000.	1.202	1.199	1.200
11500.	1.195	1.195	1.197
12000.	1.188	1.190	1.193
12500.	1.182	1.186	1.189
13000.	1.177	1.181	1.186
13500.	1.171	1.177	1.182
14000.	1.166	1.173	1.179
14500.	1.161	1.168	1.175
15000.	1.155	1.164	1.172
15500.	1.150	1.159	1.168
16000.	1.144	1.154	1.165
16500.	1.139	1.150	1.161
17000.	1.133	1.145	1.158
17500.	1.128	1.140	1.154
18000.	1.122	1.136	1.151
18500.	1.117	1.131	1.147
19000.	1.112	1.127	1.144
19500.	1.108	1.122	1.140
20000.	1.103	1.118	1.137
20500.	1.100	1.114	1.133
21000.	1.096	1.111	1.130
21500.	1.093	1.108	1.127
22000.	1.092	1.105	1.124
22500.	1.091	1.102	1.121
23000.	1.091	1.100	1.119
23500.	1.090	1.097	1.116
24000.	1.089	1.095	1.114
24500.	1.089	1.094	1.112
25000.	1.088	1.092	1.111
25500.	1.087	1.090	1.109
26000.	1.087	1.089	1.107
26500.	1.086	1.087	1.106
27000.	1.086	1.086	1.105
27500.	1.085	1.084	1.104
28000.	1.084	1.083	1.102
28500.	1.083	1.083	1.101
30000.	1.081	1.081	1.098
32500.	1.077	1.078	1.093
35000.	1.073	1.075	1.087
37500.	1.068	1.072	1.082
40000.	1.064	1.069	1.078
42500.	1.060	1.066	1.073
45000.	1.061	1.062	1.071
47500.	1.062	1.060	1.070
50000.	1.062	1.057	1.068
52500.	1.062	1.054	1.075
55000.	1.061	1.052	1.087
57500.	1.057	1.050	1.100
60000.	1.052	1.048	1.113
62500.	1.045	1.057	1.125
65000.	1.057	1.074	1.138
67500.	1.078	1.091	1.151
70000.	1.098	1.108	1.163

Table B.5 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	0.95554	0.94208	0.92519
100.	0.93632	0.92444	0.90974
500.	0.93616	0.92526	0.91159
1000.	0.93967	0.92956	0.91650
1500.	0.94433	0.93478	0.92201
2000.	0.94944	0.94032	0.92765
2500.	0.95475	0.94595	0.93325
3000.	0.96021	0.95160	0.93876
3500.	0.96589	0.95735	0.94421
4000.	0.97181	0.96321	0.94965
4500.	0.97797	0.96921	0.95510
5000.	0.98437	0.97533	0.96057
5500.	0.99093	0.98154	0.96607
6000.	0.99750	0.98777	0.97158
6500.	1.00395	0.99391	0.97704
7000.	1.01024	0.99993	0.98243
7500.	1.01634	1.00580	0.98771
8000.	1.02225	1.01152	0.99289
8500.	1.02802	1.01711	0.99797
9000.	1.03372	1.02260	1.00294
9500.	1.03942	1.02805	1.00783
10000.	1.04518	1.03350	1.01265
10500.	1.05102	1.03900	1.01744
11000.	1.05699	1.04457	1.02220
11500.	1.06312	1.05023	1.02698
12000.	1.06949	1.05601	1.03179
12500.	1.07614	1.06196	1.03664
13000.	1.08314	1.06810	1.04156
13500.	1.09054	1.07450	1.04658
14000.	1.09839	1.08118	1.05171
14500.	1.10667	1.08816	1.05698
15000.	1.11520	1.09536	1.06239
15500.	1.12378	1.10267	1.06789
16000.	1.13212	1.10987	1.07343
16500.	1.13984	1.11676	1.07887
17000.	1.14664	1.12309	1.08410
17500.	1.15211	1.12864	1.08900
18000.	1.15600	1.13318	1.09343
18500.	1.15827	1.13650	1.09728
19000.	1.15892	1.13856	1.10039
19500.	1.15809	1.13938	1.10269
20000.	1.15599	1.13903	1.10415
20500.	1.15292	1.13766	1.10478
21000.	1.14917	1.13542	1.10460
21500.	1.14461	1.13252	1.10369
22000.	1.14006	1.12912	1.10213
22500.	1.13550	1.12538	1.10002
23000.	1.13068	1.12114	1.09747
23500.	1.12587	1.11690	1.09456
24000.	1.12105	1.11267	1.09139
24500.	1.11624	1.10843	1.08802
25000.	1.11142	1.10419	1.08453
27500.	1.08705	1.08276	1.06603
30000.	1.06241	1.06129	1.04765
32500.	1.03745	1.03983	1.02950
35000.	1.01218	1.01839	1.01162
37500.	0.98665	0.99698	0.99403
40000.	0.96095	0.97564	0.97673
42500.	0.93520	0.95443	0.95976
45000.	0.90954	0.93344	0.94315
47500.	0.88417	0.91275	0.92696
50000.	0.85935	0.89247	0.91124
52500.	0.83534	0.87274	0.89601
55000.	0.81241	0.85369	0.88133
57500.	0.79084	0.83547	0.86724
60000.	0.77089	0.81820	0.85378
62500.	0.75274	0.80200	0.84100
65000.	0.73652	0.78698	0.82893
67500.	0.72229	0.77320	0.81758
70000.	0.70999	0.76070	0.80700

Table B.6 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.349	1.323	1.297
100.	1.345	1.320	1.293
500.	1.337	1.315	1.288
1000.	1.328	1.309	1.281
1500.	1.316	1.300	1.274
2000.	1.302	1.290	1.268
2500.	1.291	1.279	1.261
3000.	1.281	1.269	1.253
3500.	1.271	1.260	1.246
4000.	1.263	1.252	1.239
4500.	1.263	1.243	1.232
5000.	1.262	1.237	1.226
5500.	1.261	1.235	1.219
6000.	1.258	1.232	1.212
6500.	1.253	1.228	1.206
7000.	1.246	1.223	1.199
7500.	1.237	1.216	1.193
8000.	1.227	1.208	1.187
8500.	1.217	1.200	1.182
9000.	1.209	1.191	1.176
9500.	1.200	1.182	1.171
10000.	1.192	1.175	1.165
10500.	1.184	1.168	1.160
11000.	1.176	1.161	1.155
11500.	1.168	1.154	1.150
12000.	1.164	1.147	1.145
12500.	1.160	1.140	1.140
13000.	1.156	1.137	1.136
13500.	1.153	1.134	1.131
14000.	1.149	1.131	1.126
14500.	1.145	1.128	1.121
15000.	1.140	1.125	1.117
15500.	1.136	1.122	1.112
16000.	1.132	1.119	1.107
16500.	1.127	1.116	1.103
17000.	1.123	1.113	1.101
17500.	1.118	1.111	1.100
18000.	1.115	1.108	1.099
18500.	1.111	1.106	1.098
19000.	1.108	1.104	1.097
19500.	1.106	1.102	1.096
20000.	1.104	1.100	1.096
20500.	1.102	1.099	1.095
21000.	1.101	1.098	1.095
21500.	1.100	1.097	1.094
22000.	1.098	1.096	1.094
22500.	1.097	1.095	1.094
23000.	1.096	1.094	1.093
23500.	1.094	1.093	1.093
24000.	1.093	1.093	1.093
24500.	1.091	1.092	1.093
25000.	1.090	1.091	1.092
27500.	1.084	1.087	1.091
30000.	1.078	1.083	1.089
32500.	1.074	1.080	1.088
35000.	1.073	1.076	1.086
37500.	1.073	1.072	1.084
40000.	1.078	1.071	1.083
42500.	1.083	1.071	1.082
45000.	1.086	1.070	1.080
47500.	1.086	1.069	1.079
50000.	1.085	1.068	1.078
52500.	1.082	1.066	1.077
55000.	1.076	1.064	1.076
57500.	1.069	1.061	1.075
60000.	1.060	1.058	1.075
62500.	1.050	1.055	1.074
65000.	1.042	1.053	1.074
67500.	1.044	1.054	1.074
70000.	1.059	1.055	1.074

Table B.7 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	0.96226	0.94979	0.93242
100.	0.94267	0.93178	0.91672
500.	0.94237	0.93251	0.91854
1000.	0.94585	0.93683	0.92353
1500.	0.95056	0.94217	0.92917
2000.	0.95578	0.94787	0.93497
2500.	0.96123	0.95370	0.94076
3000.	0.96689	0.95960	0.94647
3500.	0.97281	0.96563	0.95215
4000.	0.97901	0.97181	0.95783
4500.	0.98551	0.97816	0.96353
5000.	0.99228	0.98466	0.96927
5500.	0.99924	0.99127	0.97505
6000.	1.00619	0.99789	0.98083
6500.	1.01302	1.00441	0.98656
7000.	1.01965	1.01079	0.99221
7500.	1.02604	1.01701	0.99774
8000.	1.03220	1.02305	1.00318
8500.	1.03820	1.02895	1.00849
9000.	1.04414	1.03474	1.01371
9500.	1.05009	1.04050	1.01883
10000.	1.05612	1.04629	1.02390
10500.	1.06226	1.05213	1.02893
11000.	1.06856	1.05807	1.03396
11500.	1.07509	1.06413	1.03901
12000.	1.08190	1.07034	1.04411
12500.	1.08907	1.07677	1.04926
13000.	1.09666	1.08344	1.05450
13500.	1.10474	1.09043	1.05985
14000.	1.11331	1.09776	1.06535
14500.	1.12232	1.10542	1.07101
15000.	1.13159	1.11332	1.07682
15500.	1.14088	1.12132	1.08274
16000.	1.14980	1.12914	1.08869
16500.	1.15794	1.13656	1.09451
17000.	1.16482	1.14328	1.10008
17500.	1.17012	1.14899	1.10526
18000.	1.17364	1.15341	1.10989
18500.	1.17527	1.15641	1.11382
19000.	1.17509	1.15800	1.11690
19500.	1.17332	1.15822	1.11909
20000.	1.17037	1.15717	1.12033
20500.	1.16660	1.15507	1.12067
21000.	1.16184	1.15215	1.12018
21500.	1.15707	1.14866	1.11893
22000.	1.15231	1.14454	1.11702
22500.	1.14755	1.14042	1.11458
23000.	1.14255	1.13601	1.11172
23500.	1.13755	1.13160	1.10854
24000.	1.13256	1.12718	1.10514
24500.	1.12756	1.12277	1.10158
25000.	1.12256	1.11836	1.09786
27500.	1.09716	1.09611	1.07884
30000.	1.07132	1.07378	1.06003
32500.	1.04502	1.05137	1.04144
35000.	1.01828	1.02888	1.02309
37500.	0.99113	1.00633	1.00500
40000.	0.96369	0.98376	0.98717
42500.	0.93609	0.96126	0.96965
45000.	0.90854	0.93890	0.95247
47500.	0.88129	0.91682	0.93570
50000.	0.85465	0.89515	0.91937
52500.	0.82896	0.87405	0.90354
55000.	0.80458	0.85371	0.88827
57500.	0.78186	0.83429	0.87359
60000.	0.76111	0.81597	0.85958
62500.	0.74253	0.79891	0.84626
65000.	0.72623	0.78321	0.83369
67500.	0.71222	0.76896	0.82190
70000.	0.70038	0.75619	0.81092

Table B.8 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.385	1.317	1.286
100.	1.387	1.314	1.279
500.	1.384	1.309	1.274
1000.	1.380	1.302	1.268
1500.	1.375	1.296	1.261
2000.	1.369	1.292	1.255
2500.	1.363	1.288	1.248
3000.	1.357	1.284	1.241
3500.	1.351	1.280	1.235
4000.	1.345	1.275	1.228
4500.	1.338	1.271	1.221
5000.	1.331	1.266	1.214
5500.	1.324	1.261	1.208
6000.	1.317	1.256	1.201
6500.	1.310	1.251	1.195
7000.	1.303	1.246	1.188
7500.	1.296	1.241	1.182
8000.	1.289	1.236	1.176
8500.	1.283	1.232	1.171
9000.	1.276	1.227	1.165
9500.	1.270	1.222	1.160
10000.	1.263	1.218	1.154
10500.	1.257	1.213	1.152
11000.	1.250	1.208	1.149
11500.	1.244	1.204	1.147
12000.	1.238	1.199	1.145
12500.	1.231	1.195	1.142
13000.	1.225	1.191	1.140
13500.	1.218	1.186	1.137
14000.	1.212	1.181	1.135
14500.	1.205	1.177	1.133
15000.	1.198	1.172	1.130
15500.	1.190	1.167	1.128
16000.	1.183	1.162	1.126
16500.	1.175	1.157	1.124
17000.	1.168	1.153	1.122
17500.	1.161	1.148	1.120
18000.	1.155	1.144	1.118
18500.	1.149	1.140	1.117
19000.	1.145	1.137	1.115
19500.	1.141	1.134	1.114
20000.	1.137	1.131	1.113
20500.	1.134	1.129	1.112
21000.	1.132	1.126	1.111
21500.	1.129	1.125	1.111
22000.	1.126	1.123	1.110
22500.	1.124	1.121	1.109
23000.	1.121	1.119	1.109
23500.	1.119	1.118	1.108
24000.	1.116	1.116	1.107
24500.	1.114	1.115	1.107
25000.	1.111	1.113	1.106
27500.	1.099	1.105	1.104
30000.	1.090	1.098	1.101
32500.	1.085	1.091	1.098
35000.	1.081	1.084	1.095
37500.	1.086	1.079	1.093
40000.	1.090	1.078	1.091
42500.	1.093	1.077	1.089
45000.	1.094	1.075	1.087
47500.	1.093	1.073	1.085
50000.	1.090	1.070	1.083
52500.	1.085	1.068	1.082
55000.	1.078	1.065	1.081
57500.	1.069	1.062	1.080
60000.	1.058	1.060	1.079
62500.	1.047	1.057	1.079
65000.	1.051	1.057	1.079
67500.	1.058	1.060	1.079
70000.	1.065	1.064	1.080

Table B.9 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.00810	0.99291	0.97092
100.	0.98616	0.97266	0.95317
500.	0.98403	0.97158	0.95334
1000.	0.98530	0.97376	0.95642
1500.	0.98770	0.97690	0.96014
2000.	0.99046	0.98030	0.96400
2500.	0.99330	0.98372	0.96779
3000.	0.99618	0.98711	0.97145
3500.	0.99916	0.99051	0.97502
4000.	1.00228	0.99398	0.97856
4500.	1.00558	0.99754	0.98210
5000.	1.00907	1.00124	0.98567
5500.	1.01279	1.00510	0.98930
6000.	1.01674	1.00913	0.99302
6500.	1.02093	1.01335	0.99683
7000.	1.02533	1.01773	1.00076
7500.	1.02990	1.02226	1.00478
8000.	1.03466	1.02692	1.00890
8500.	1.03959	1.03171	1.01310
9000.	1.04473	1.03663	1.01737
9500.	1.05009	1.04171	1.02173
10000.	1.05567	1.04696	1.02618
10500.	1.06150	1.05239	1.03071
11000.	1.06758	1.05801	1.03534
11500.	1.07396	1.06384	1.04007
12000.	1.08067	1.06988	1.04491
12500.	1.08776	1.07619	1.04986
13000.	1.09527	1.08277	1.05494
13500.	1.10324	1.08968	1.06017
14000.	1.11169	1.09693	1.06557
14500.	1.12054	1.10451	1.07116
15000.	1.12962	1.11233	1.07692
15500.	1.13878	1.12025	1.08281
16000.	1.14770	1.12800	1.08873
16500.	1.15588	1.13542	1.09455
17000.	1.16293	1.14220	1.10013
17500.	1.16853	1.14804	1.10532
18000.	1.17244	1.15264	1.10999
18500.	1.17452	1.15588	1.11399
19000.	1.17477	1.15772	1.11717
19500.	1.17338	1.15819	1.11943
20000.	1.17070	1.15739	1.12076
20500.	1.16711	1.15549	1.12119
21000.	1.16292	1.15272	1.12077
21500.	1.15807	1.14933	1.11960
22000.	1.15321	1.14526	1.11776
22500.	1.14836	1.14118	1.11537
23000.	1.14333	1.13677	1.11254
23500.	1.13830	1.13235	1.10938
24000.	1.13328	1.12794	1.10599
24500.	1.12825	1.12352	1.10244
25000.	1.12322	1.11911	1.09872
27500.	1.09775	1.09682	1.07967
30000.	1.07186	1.07443	1.06081
32500.	1.04551	1.05197	1.04217
35000.	1.01871	1.02942	1.02377
37500.	0.99151	1.00681	1.00562
40000.	0.96401	0.98420	0.98775
42500.	0.93637	0.96164	0.97018
45000.	0.90877	0.93925	0.95297
47500.	0.88149	0.91712	0.93616
50000.	0.85482	0.89542	0.91980
52500.	0.82912	0.87430	0.90394
55000.	0.80474	0.85394	0.88864
57500.	0.78204	0.83451	0.87395
60000.	0.76130	0.81620	0.85992
62500.	0.74275	0.79914	0.84659
65000.	0.72649	0.78345	0.83402
67500.	0.71251	0.76923	0.82223
70000.	0.70071	0.75649	0.81125

Table B.10 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.379	1.333	1.251
100.	1.376	1.330	1.249
500.	1.370	1.326	1.247
1000.	1.363	1.321	1.244
1500.	1.355	1.314	1.241
2000.	1.347	1.308	1.237
2500.	1.342	1.300	1.233
3000.	1.338	1.293	1.228
3500.	1.333	1.285	1.223
4000.	1.329	1.277	1.218
4500.	1.324	1.269	1.213
5000.	1.320	1.260	1.207
5500.	1.315	1.252	1.202
6000.	1.310	1.249	1.196
6500.	1.305	1.245	1.190
7000.	1.300	1.242	1.184
7500.	1.294	1.238	1.178
8000.	1.289	1.234	1.172
8500.	1.283	1.230	1.168
9000.	1.277	1.226	1.163
9500.	1.271	1.222	1.159
10000.	1.266	1.218	1.154
10500.	1.259	1.214	1.151
11000.	1.253	1.210	1.149
11500.	1.247	1.206	1.147
12000.	1.241	1.201	1.145
12500.	1.235	1.197	1.142
13000.	1.228	1.193	1.140
13500.	1.222	1.188	1.138
14000.	1.215	1.184	1.136
14500.	1.208	1.179	1.133
15000.	1.201	1.174	1.131
15500.	1.194	1.169	1.129
16000.	1.186	1.164	1.127
16500.	1.178	1.159	1.124
17000.	1.171	1.154	1.122
17500.	1.163	1.150	1.121
18000.	1.157	1.145	1.119
18500.	1.151	1.141	1.117
19000.	1.146	1.138	1.116
19500.	1.142	1.135	1.114
20000.	1.139	1.132	1.113
20500.	1.135	1.129	1.112
21000.	1.132	1.127	1.111
21500.	1.130	1.125	1.110
22000.	1.127	1.123	1.110
22500.	1.125	1.122	1.109
23000.	1.122	1.120	1.108
23500.	1.120	1.119	1.108
24000.	1.117	1.117	1.107
24500.	1.115	1.116	1.107
25000.	1.112	1.114	1.106
27500.	1.100	1.106	1.103
30000.	1.089	1.098	1.101
32500.	1.085	1.091	1.098
35000.	1.082	1.084	1.095
37500.	1.087	1.080	1.093
40000.	1.092	1.079	1.090
42500.	1.095	1.077	1.088
45000.	1.096	1.075	1.086
47500.	1.095	1.073	1.085
50000.	1.092	1.071	1.083
52500.	1.087	1.068	1.081
55000.	1.079	1.065	1.080
57500.	1.070	1.063	1.079
60000.	1.059	1.060	1.079
62500.	1.048	1.057	1.079
65000.	1.051	1.057	1.079
67500.	1.058	1.060	1.079
70000.	1.065	1.063	1.079

Table B.11 [ ] Hot Uncontrolled k_{∞}

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.05813	1.04960	1.03566
100.	1.03629	1.02933	1.01773
500.	1.03915	1.03298	1.02196
1000.	1.04786	1.04213	1.03091
1500.	1.05848	1.05293	1.04105
2000.	1.06993	1.06437	1.05157
2500.	1.08214	1.07633	1.06227
3000.	1.09513	1.08879	1.07314
3500.	1.10891	1.10179	1.08423
4000.	1.12353	1.11533	1.09555
4500.	1.13907	1.12946	1.10713
5000.	1.15555	1.14421	1.11898
5500.	1.17289	1.15953	1.13106
6000.	1.19077	1.17519	1.14329
6500.	1.20852	1.19074	1.15545
7000.	1.22512	1.20549	1.16720
7500.	1.23956	1.21861	1.17806
8000.	1.25117	1.22955	1.18757
8500.	1.25978	1.23805	1.19544
9000.	1.26544	1.24412	1.20157
9500.	1.26833	1.24787	1.20599
10000.	1.26885	1.24950	1.20879
10500.	1.26750	1.24934	1.21010
11000.	1.26483	1.24773	1.21010
11500.	1.26127	1.24507	1.20901
12000.	1.25715	1.24169	1.20701
12500.	1.25269	1.23781	1.20431
13000.	1.24778	1.23363	1.20111
13500.	1.24288	1.22907	1.19754
14000.	1.23797	1.22451	1.19373
14500.	1.23307	1.21995	1.18975
15000.	1.22816	1.21539	1.18568
15500.	1.22324	1.21081	1.18155
17500.	1.20354	1.19247	1.16473
20000.	1.17873	1.16969	1.14414
22500.	1.15363	1.14698	1.12398
25000.	1.12813	1.12427	1.10419
27500.	1.10220	1.10150	1.08468
30000.	1.07579	1.07865	1.06543
32500.	1.04889	1.05571	1.04640
35000.	1.02154	1.03268	1.02761
37500.	0.99379	1.00960	1.00908
40000.	0.96576	0.98653	0.99084
42500.	0.93763	0.96354	0.97292
45000.	0.90962	0.94075	0.95540
47500.	0.88200	0.91829	0.93831
50000.	0.85511	0.89632	0.92172
52500.	0.82933	0.87502	0.90567
55000.	0.80500	0.85457	0.89022
57500.	0.78249	0.83515	0.87543
60000.	0.76207	0.81694	0.86136
62500.	0.74394	0.80007	0.84805
65000.	0.72815	0.78466	0.83554
67500.	0.71467	0.77076	0.82386
70000.	0.70335	0.75839	0.81303

Table B.12 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.344	1.293	1.218
100.	1.346	1.289	1.215
500.	1.341	1.281	1.209
1000.	1.335	1.270	1.202
1500.	1.328	1.259	1.195
2000.	1.321	1.253	1.188
2500.	1.313	1.248	1.181
3000.	1.306	1.242	1.173
3500.	1.298	1.236	1.166
4000.	1.289	1.230	1.159
4500.	1.281	1.224	1.151
5000.	1.272	1.218	1.146
5500.	1.263	1.212	1.143
6000.	1.254	1.205	1.140
6500.	1.245	1.199	1.136
7000.	1.236	1.192	1.133
7500.	1.227	1.186	1.130
8000.	1.218	1.181	1.128
8500.	1.211	1.176	1.126
9000.	1.205	1.171	1.123
9500.	1.199	1.167	1.122
10000.	1.195	1.164	1.120
10500.	1.191	1.161	1.119
11000.	1.187	1.159	1.118
11500.	1.184	1.156	1.116
12000.	1.181	1.154	1.116
12500.	1.179	1.153	1.115
13000.	1.176	1.151	1.114
13500.	1.174	1.150	1.113
14000.	1.171	1.148	1.113
14500.	1.169	1.146	1.112
15000.	1.166	1.145	1.111
15500.	1.163	1.143	1.111
17500.	1.153	1.137	1.108
20000.	1.140	1.128	1.106
22500.	1.127	1.120	1.103
25000.	1.114	1.112	1.100
27500.	1.101	1.104	1.097
30000.	1.091	1.096	1.094
32500.	1.085	1.088	1.091
35000.	1.080	1.081	1.089
37500.	1.074	1.074	1.086
40000.	1.070	1.072	1.084
42500.	1.070	1.070	1.082
45000.	1.071	1.068	1.080
47500.	1.070	1.066	1.078
50000.	1.067	1.063	1.076
52500.	1.062	1.060	1.075
55000.	1.055	1.057	1.074
57500.	1.046	1.054	1.073
60000.	1.039	1.051	1.072
62500.	1.040	1.049	1.072
65000.	1.047	1.052	1.072
67500.	1.053	1.055	1.072
70000.	1.060	1.058	1.073

Table B.13 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.06567	1.04412	1.01855
100.	1.04363	1.02395	1.00093
500.	1.04128	1.02259	1.00065
1000.	1.04211	1.02420	1.00292
1500.	1.04398	1.02666	1.00575
2000.	1.04615	1.02931	1.00867
2500.	1.04836	1.03194	1.01146
3000.	1.05053	1.03447	1.01409
3500.	1.05273	1.03693	1.01658
4000.	1.05499	1.03940	1.01899
4500.	1.05732	1.04187	1.02135
5000.	1.05974	1.04439	1.02369
5500.	1.06227	1.04696	1.02602
6000.	1.06492	1.04961	1.02836
6500.	1.06771	1.05236	1.03075
7000.	1.07065	1.05521	1.03319
7500.	1.07371	1.05815	1.03572
8000.	1.07689	1.06117	1.03830
8500.	1.08018	1.06428	1.04094
9000.	1.08360	1.06746	1.04362
9500.	1.08715	1.07074	1.04635
10000.	1.09084	1.07412	1.04913
10500.	1.09470	1.07762	1.05196
11000.	1.09872	1.08123	1.05484
11500.	1.10293	1.08497	1.05778
12000.	1.10733	1.08884	1.06078
12500.	1.11193	1.09285	1.06385
13000.	1.11677	1.09702	1.06698
13500.	1.12187	1.10134	1.07018
14000.	1.12726	1.10580	1.07344
14500.	1.13291	1.11039	1.07675
15000.	1.13865	1.11503	1.08005
15500.	1.14442	1.11964	1.08332
16000.	1.15008	1.12415	1.08648
16500.	1.15544	1.12843	1.08953
17000.	1.16025	1.13240	1.09241
17500.	1.16425	1.13590	1.09508
18000.	1.16726	1.13881	1.09746
18500.	1.16920	1.14104	1.09949
19000.	1.17007	1.14252	1.10112
19500.	1.16983	1.14319	1.10232
20000.	1.16852	1.14308	1.10306
20500.	1.16625	1.14221	1.10331
21000.	1.16322	1.14064	1.10306
21500.	1.15962	1.13843	1.10233
22000.	1.15564	1.13569	1.10112
22500.	1.15133	1.13253	1.09949
23000.	1.14679	1.12907	1.09749
23500.	1.14224	1.12538	1.09516
24000.	1.13770	1.12139	1.09255
24500.	1.13315	1.11740	1.08971
25000.	1.12861	1.11341	1.08668
25500.	1.12404	1.10937	1.08351
26000.	1.11947	1.10533	1.08024
26500.	1.11491	1.10129	1.07689
27000.	1.11034	1.09725	1.07348
27500.	1.10577	1.09321	1.07001
30000.	1.08283	1.07313	1.05268
32500.	1.05981	1.05323	1.03566
35000.	1.03671	1.03351	1.01900
37500.	1.01356	1.01399	1.00272
40000.	0.99042	0.99469	0.98684
42500.	0.96734	0.97565	0.97136
45000.	0.94443	0.95692	0.95632
47500.	0.92180	0.93856	0.94175
50000.	0.89960	0.92062	0.92766
52500.	0.87798	0.90317	0.91409
55000.	0.85713	0.88629	0.90105
57500.	0.83721	0.87006	0.88857
60000.	0.81839	0.85456	0.87665
62500.	0.80084	0.83985	0.86531
65000.	0.78466	0.82600	0.85456
67500.	0.76996	0.81305	0.84442
70000.	0.75677	0.80104	0.83488

Table B.14 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.301	1.296	1.272
100.	1.298	1.295	1.271
500.	1.294	1.292	1.269
1000.	1.290	1.288	1.267
1500.	1.285	1.283	1.264
2000.	1.279	1.278	1.260
2500.	1.274	1.272	1.256
3000.	1.269	1.266	1.252
3500.	1.263	1.260	1.247
4000.	1.257	1.254	1.243
4500.	1.252	1.248	1.238
5000.	1.246	1.242	1.233
5500.	1.240	1.235	1.228
6000.	1.234	1.229	1.223
6500.	1.228	1.222	1.218
7000.	1.222	1.215	1.212
7500.	1.216	1.210	1.207
8000.	1.209	1.204	1.202
8500.	1.203	1.199	1.196
9000.	1.199	1.194	1.191
9500.	1.195	1.188	1.185
10000.	1.192	1.183	1.180
10500.	1.188	1.177	1.176
11000.	1.184	1.172	1.172
11500.	1.180	1.166	1.167
12000.	1.176	1.161	1.163
12500.	1.172	1.156	1.159
13000.	1.168	1.152	1.155
13500.	1.163	1.149	1.151
14000.	1.159	1.145	1.147
14500.	1.155	1.141	1.143
15000.	1.150	1.138	1.139
15500.	1.146	1.134	1.135
16000.	1.142	1.131	1.131
16500.	1.138	1.128	1.126
17000.	1.134	1.125	1.122
17500.	1.131	1.122	1.118
18000.	1.127	1.120	1.114
18500.	1.124	1.117	1.110
19000.	1.121	1.115	1.107
19500.	1.119	1.113	1.104
20000.	1.117	1.112	1.101
20500.	1.115	1.110	1.099
21000.	1.114	1.109	1.097
21500.	1.113	1.107	1.097
22000.	1.111	1.106	1.096
22500.	1.110	1.105	1.095
23000.	1.109	1.104	1.095
23500.	1.108	1.103	1.094
24000.	1.107	1.102	1.094
24500.	1.106	1.102	1.093
25000.	1.105	1.101	1.093
25500.	1.104	1.100	1.093
26000.	1.103	1.099	1.092
26500.	1.101	1.098	1.092
27000.	1.100	1.097	1.091
27500.	1.099	1.096	1.091
30000.	1.092	1.091	1.088
32500.	1.086	1.087	1.086
35000.	1.079	1.082	1.083
37500.	1.072	1.077	1.080
40000.	1.068	1.072	1.078
42500.	1.063	1.067	1.075
45000.	1.058	1.064	1.073
47500.	1.054	1.061	1.070
50000.	1.052	1.058	1.068
52500.	1.051	1.055	1.074
55000.	1.049	1.053	1.086
57500.	1.046	1.050	1.099
60000.	1.042	1.047	1.112
62500.	1.036	1.056	1.124
65000.	1.056	1.073	1.137
67500.	1.077	1.090	1.150
70000.	1.097	1.107	1.162

Table B.15 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.00503	0.98706	0.96512
100.	0.98380	0.96752	0.94789
500.	0.98222	0.96700	0.94853
1000.	0.98408	0.96979	0.95212
1500.	0.98711	0.97357	0.95643
2000.	0.99050	0.97762	0.96089
2500.	0.99394	0.98165	0.96525
3000.	0.99742	0.98563	0.96946
3500.	1.00101	0.98963	0.97357
4000.	1.00473	0.99368	0.97764
4500.	1.00860	0.99781	0.98171
5000.	1.01264	1.00205	0.98579
5500.	1.01686	1.00641	0.98991
6000.	1.02127	1.01091	0.99410
6500.	1.02587	1.01556	0.99836
7000.	1.03064	1.02035	1.00273
7500.	1.03559	1.02526	1.00718
8000.	1.04068	1.03028	1.01171
8500.	1.04595	1.03542	1.01630
9000.	1.05139	1.04067	1.02097
9500.	1.05703	1.04606	1.02571
10000.	1.06287	1.05161	1.03053
10500.	1.06894	1.05732	1.03544
11000.	1.07526	1.06322	1.04044
11500.	1.08185	1.06932	1.04554
12000.	1.08873	1.07563	1.05075
12500.	1.09597	1.08216	1.05607
13000.	1.10365	1.08896	1.06152
13500.	1.11177	1.09605	1.06709
14000.	1.12029	1.10338	1.07275
14500.	1.12898	1.11079	1.07845
15000.	1.13765	1.11815	1.08410
15500.	1.14603	1.12529	1.08960
16000.	1.15361	1.13199	1.09488
16500.	1.15999	1.13793	1.09983
17000.	1.16487	1.14286	1.10432
17500.	1.16811	1.14661	1.10821
18000.	1.16970	1.14913	1.11136
18500.	1.16967	1.15041	1.11369
19000.	1.16820	1.15048	1.11518
19500.	1.16558	1.14943	1.11583
20000.	1.16212	1.14740	1.11570
20500.	1.15811	1.14461	1.11484
21000.	1.15342	1.14126	1.11330
21500.	1.14873	1.13754	1.11119
22000.	1.14404	1.13339	1.10859
22500.	1.13935	1.12925	1.10563
23000.	1.13453	1.12494	1.10239
23500.	1.12970	1.12064	1.09896
25000.	1.11523	1.10772	1.08789
27500.	1.09083	1.08619	1.06924
30000.	1.06613	1.06467	1.05076
32500.	1.04111	1.04314	1.03254
35000.	1.01576	1.02162	1.01459
37500.	0.99014	1.00012	0.99690
40000.	0.96433	0.97869	0.97951
42500.	0.93845	0.95738	0.96244
45000.	0.91264	0.93627	0.94573
47500.	0.88712	0.91545	0.92945
50000.	0.86213	0.89505	0.91362
52500.	0.83793	0.87519	0.89829
55000.	0.81482	0.85600	0.88351
57500.	0.79307	0.83765	0.86932
60000.	0.77293	0.82025	0.85577
62500.	0.75462	0.80392	0.84290
65000.	0.73825	0.78878	0.83073
67500.	0.72388	0.77489	0.81931
70000.	0.71147	0.76230	0.80865

Table B.16 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.302	1.291	1.268
100.	1.299	1.287	1.265
500.	1.294	1.282	1.261
1000.	1.288	1.276	1.256
1500.	1.282	1.270	1.251
2000.	1.275	1.264	1.245
2500.	1.268	1.257	1.240
3000.	1.262	1.251	1.234
3500.	1.255	1.244	1.229
4000.	1.248	1.238	1.224
4500.	1.241	1.231	1.219
5000.	1.233	1.225	1.214
5500.	1.226	1.218	1.208
6000.	1.218	1.211	1.203
6500.	1.211	1.204	1.198
7000.	1.203	1.197	1.193
7500.	1.199	1.191	1.187
8000.	1.194	1.185	1.182
8500.	1.190	1.178	1.176
9000.	1.186	1.172	1.171
9500.	1.181	1.166	1.165
10000.	1.177	1.162	1.160
10500.	1.172	1.158	1.155
11000.	1.167	1.154	1.149
11500.	1.162	1.150	1.144
12000.	1.158	1.147	1.138
12500.	1.154	1.143	1.133
13000.	1.150	1.139	1.128
13500.	1.146	1.135	1.122
14000.	1.141	1.131	1.117
14500.	1.136	1.127	1.112
15000.	1.131	1.123	1.107
15500.	1.126	1.119	1.104
16000.	1.120	1.114	1.101
16500.	1.115	1.110	1.098
17000.	1.112	1.106	1.096
17500.	1.108	1.103	1.093
18000.	1.105	1.099	1.091
18500.	1.103	1.097	1.090
19000.	1.101	1.096	1.090
19500.	1.099	1.095	1.089
20000.	1.097	1.094	1.089
20500.	1.095	1.092	1.089
21000.	1.094	1.092	1.089
21500.	1.092	1.091	1.089
22000.	1.091	1.090	1.089
22500.	1.090	1.089	1.088
23000.	1.089	1.088	1.088
23500.	1.088	1.087	1.088
25000.	1.084	1.085	1.087
27500.	1.079	1.081	1.086
30000.	1.079	1.078	1.085
32500.	1.079	1.074	1.083
35000.	1.078	1.071	1.082
37500.	1.077	1.067	1.080
40000.	1.075	1.067	1.079
42500.	1.072	1.066	1.078
45000.	1.069	1.064	1.077
47500.	1.068	1.062	1.076
50000.	1.068	1.060	1.075
52500.	1.065	1.057	1.074
55000.	1.061	1.054	1.073
57500.	1.055	1.051	1.073
60000.	1.047	1.050	1.072
62500.	1.038	1.050	1.072
65000.	1.039	1.051	1.072
67500.	1.042	1.052	1.072
70000.	1.058	1.053	1.072

Table B.17 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.01633	0.99848	0.97433
100.	0.99451	0.97841	0.95676
500.	0.99264	0.97767	0.95730
1000.	0.99428	0.98032	0.96088
1500.	0.99715	0.98404	0.96523
2000.	1.00040	0.98805	0.96976
2500.	1.00372	0.99207	0.97420
3000.	1.00711	0.99606	0.97850
3500.	1.01063	1.00011	0.98272
4000.	1.01431	1.00422	0.98691
4500.	1.01816	1.00844	0.99111
5000.	1.02220	1.01278	0.99534
5500.	1.02645	1.01727	0.99961
6000.	1.03090	1.02192	1.00397
6500.	1.03556	1.02673	1.00841
7000.	1.04041	1.03170	1.01296
7500.	1.04545	1.03680	1.01761
8000.	1.05065	1.04203	1.02233
8500.	1.05604	1.04738	1.02714
9000.	1.06162	1.05287	1.03202
9500.	1.06742	1.05852	1.03698
10000.	1.07345	1.06434	1.04204
10500.	1.07971	1.07035	1.04720
11000.	1.08625	1.07657	1.05245
11500.	1.09308	1.08300	1.05782
12000.	1.10025	1.08967	1.06332
12500.	1.10783	1.09661	1.06893
13000.	1.11589	1.10387	1.07468
13500.	1.12446	1.11147	1.08058
14000.	1.13350	1.11935	1.08658
14500.	1.14280	1.12735	1.09265
15000.	1.15212	1.13531	1.09866
15500.	1.16103	1.14301	1.10453
16000.	1.16902	1.15013	1.11015
16500.	1.17563	1.15636	1.11539
17000.	1.18055	1.16141	1.12009
17500.	1.18362	1.16510	1.12411
18000.	1.18483	1.16741	1.12727
18500.	1.18431	1.16833	1.12953
19000.	1.18233	1.16795	1.13088
19500.	1.17925	1.16639	1.13135
20000.	1.17542	1.16387	1.13099
20500.	1.17061	1.16067	1.12986
21000.	1.16581	1.15697	1.12804
21500.	1.16100	1.15269	1.12565
22000.	1.15620	1.14841	1.12280
22500.	1.15139	1.14413	1.11962
23000.	1.14640	1.13969	1.11620
25000.	1.12642	1.12192	1.10114
27500.	1.10099	1.09961	1.08210
30000.	1.07511	1.07723	1.06321
32500.	1.04876	1.05476	1.04455
35000.	1.02195	1.03219	1.02612
37500.	0.99472	1.00956	1.00793
40000.	0.96717	0.98691	0.99001
42500.	0.93946	0.96430	0.97239
45000.	0.91177	0.94183	0.95511
47500.	0.88436	0.91962	0.93823
50000.	0.85754	0.89782	0.92181
52500.	0.83167	0.87659	0.90587
55000.	0.80709	0.85611	0.89049
57500.	0.78418	0.83655	0.87572
60000.	0.76323	0.81809	0.86160
62500.	0.74447	0.80089	0.84819
65000.	0.72801	0.78507	0.83553
67500.	0.71385	0.77070	0.82366
70000.	0.70189	0.75783	0.81260

Table B.18 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.370	1.293	1.254
100.	1.372	1.294	1.251
500.	1.369	1.292	1.247
1000.	1.365	1.289	1.242
1500.	1.361	1.286	1.238
2000.	1.356	1.282	1.233
2500.	1.350	1.278	1.228
3000.	1.345	1.274	1.223
3500.	1.339	1.270	1.218
4000.	1.334	1.266	1.213
4500.	1.328	1.262	1.208
5000.	1.322	1.257	1.203
5500.	1.316	1.253	1.198
6000.	1.310	1.248	1.192
6500.	1.303	1.244	1.187
7000.	1.297	1.239	1.182
7500.	1.291	1.234	1.177
8000.	1.284	1.230	1.171
8500.	1.277	1.225	1.166
9000.	1.270	1.220	1.160
9500.	1.263	1.215	1.155
10000.	1.257	1.210	1.150
10500.	1.249	1.205	1.144
11000.	1.242	1.200	1.140
11500.	1.235	1.195	1.137
12000.	1.228	1.190	1.135
12500.	1.220	1.184	1.132
13000.	1.213	1.179	1.129
13500.	1.206	1.174	1.126
14000.	1.198	1.168	1.124
14500.	1.191	1.163	1.121
15000.	1.183	1.158	1.119
15500.	1.176	1.154	1.116
16000.	1.169	1.149	1.114
16500.	1.163	1.145	1.113
17000.	1.157	1.141	1.111
17500.	1.152	1.138	1.110
18000.	1.147	1.135	1.109
18500.	1.143	1.132	1.108
19000.	1.139	1.129	1.108
19500.	1.135	1.127	1.107
20000.	1.132	1.125	1.107
20500.	1.129	1.123	1.106
21000.	1.127	1.121	1.106
21500.	1.124	1.119	1.105
22000.	1.122	1.118	1.105
22500.	1.119	1.116	1.104
23000.	1.117	1.115	1.104
25000.	1.107	1.109	1.102
27500.	1.102	1.101	1.099
30000.	1.098	1.094	1.096
32500.	1.093	1.087	1.094
35000.	1.088	1.083	1.091
37500.	1.083	1.081	1.089
40000.	1.077	1.078	1.087
42500.	1.075	1.075	1.085
45000.	1.074	1.071	1.083
47500.	1.073	1.067	1.082
50000.	1.071	1.063	1.080
52500.	1.067	1.059	1.079
55000.	1.061	1.055	1.078
57500.	1.054	1.052	1.077
60000.	1.045	1.050	1.077
62500.	1.042	1.052	1.077
65000.	1.049	1.055	1.077
67500.	1.056	1.058	1.077
70000.	1.063	1.062	1.078

Table B.19 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.03070	1.01507	0.99362
100.	1.00885	0.99494	0.97595
500.	1.00807	0.99524	0.97746
1000.	1.01136	0.99946	0.98244
1500.	1.01607	1.00493	0.98832
2000.	1.02117	1.01070	0.99436
2500.	1.02643	1.01655	1.00031
3000.	1.03187	1.02247	1.00618
3500.	1.03755	1.02851	1.01201
4000.	1.04350	1.03471	1.01787
4500.	1.04974	1.04112	1.02379
5000.	1.05629	1.04775	1.02981
5500.	1.06315	1.05460	1.03595
6000.	1.07029	1.06166	1.04221
6500.	1.07772	1.06893	1.04858
7000.	1.08547	1.07641	1.05508
7500.	1.09355	1.08413	1.06170
8000.	1.10199	1.09213	1.06848
8500.	1.11083	1.10041	1.07544
9000.	1.12009	1.10899	1.08256
9500.	1.12984	1.11791	1.08986
10000.	1.14010	1.12717	1.09733
10500.	1.15088	1.13678	1.10494
11000.	1.16208	1.14662	1.11262
11500.	1.17343	1.15648	1.12027
12000.	1.18442	1.16608	1.12775
12500.	1.19450	1.17505	1.13491
13000.	1.20311	1.18298	1.14153
13500.	1.20985	1.18952	1.14745
14000.	1.21450	1.19446	1.15242
14500.	1.21705	1.19775	1.15632
15000.	1.21760	1.19945	1.15908
15500.	1.21645	1.19963	1.16075
16000.	1.21398	1.19850	1.16137
16500.	1.21057	1.19629	1.16105
17000.	1.20656	1.19328	1.15987
17500.	1.20207	1.18973	1.15796
18000.	1.19719	1.18580	1.15545
18500.	1.19232	1.18140	1.15249
19000.	1.18744	1.17700	1.14919
19500.	1.18257	1.17260	1.14565
20000.	1.17769	1.16820	1.14196
22500.	1.15295	1.14584	1.12236
25000.	1.12781	1.12345	1.10287
27500.	1.10224	1.10101	1.08362
30000.	1.07621	1.07849	1.06461
32500.	1.04969	1.05586	1.04583
35000.	1.02271	1.03315	1.02728
37500.	0.99531	1.01036	1.00897
40000.	0.96760	0.98756	0.99093
42500.	0.93973	0.96482	0.97319
45000.	0.91191	0.94222	0.95581
47500.	0.88441	0.91991	0.93884
50000.	0.85753	0.89802	0.92234
52500.	0.83164	0.87674	0.90635
55000.	0.80710	0.85623	0.89092
57500.	0.78427	0.83669	0.87612
60000.	0.76344	0.81827	0.86199
62500.	0.74484	0.80115	0.84858
65000.	0.72854	0.78542	0.83595
67500.	0.71457	0.77118	0.82412
70000.	0.70278	0.75844	0.81313

Table B.20 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.366	1.289	1.245
100.	1.368	1.291	1.242
500.	1.365	1.288	1.238
1000.	1.359	1.284	1.233
1500.	1.354	1.280	1.227
2000.	1.348	1.275	1.222
2500.	1.342	1.271	1.216
3000.	1.335	1.266	1.210
3500.	1.329	1.261	1.204
4000.	1.322	1.256	1.198
4500.	1.315	1.251	1.192
5000.	1.308	1.246	1.186
5500.	1.301	1.241	1.180
6000.	1.294	1.235	1.174
6500.	1.287	1.230	1.168
7000.	1.279	1.224	1.162
7500.	1.271	1.219	1.156
8000.	1.264	1.213	1.150
8500.	1.256	1.207	1.144
9000.	1.248	1.202	1.138
9500.	1.240	1.196	1.135
10000.	1.232	1.190	1.132
10500.	1.224	1.184	1.129
11000.	1.216	1.179	1.126
11500.	1.208	1.173	1.123
12000.	1.200	1.168	1.121
12500.	1.192	1.163	1.118
13000.	1.185	1.158	1.116
13500.	1.179	1.154	1.115
14000.	1.173	1.150	1.113
14500.	1.168	1.147	1.112
15000.	1.163	1.144	1.111
15500.	1.159	1.141	1.110
16000.	1.155	1.139	1.109
16500.	1.152	1.136	1.108
17000.	1.149	1.134	1.108
17500.	1.146	1.133	1.107
18000.	1.144	1.131	1.107
18500.	1.141	1.129	1.106
19000.	1.139	1.128	1.106
19500.	1.136	1.126	1.105
20000.	1.134	1.124	1.105
22500.	1.121	1.116	1.102
25000.	1.108	1.108	1.100
27500.	1.100	1.101	1.097
30000.	1.096	1.093	1.094
32500.	1.091	1.086	1.092
35000.	1.086	1.080	1.089
37500.	1.080	1.078	1.087
40000.	1.074	1.075	1.085
42500.	1.071	1.072	1.083
45000.	1.069	1.068	1.081
47500.	1.068	1.064	1.079
50000.	1.066	1.060	1.078
52500.	1.062	1.056	1.077
55000.	1.056	1.053	1.076
57500.	1.048	1.051	1.075
60000.	1.039	1.049	1.075
62500.	1.040	1.050	1.074
65000.	1.047	1.053	1.075
67500.	1.055	1.056	1.075
70000.	1.061	1.060	1.076

Table B.21 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.10092	1.07923	1.05218
100.	1.07733	1.05753	1.03312
500.	1.07412	1.05522	1.03184
1000.	1.07401	1.05579	1.03300
1500.	1.07493	1.05720	1.03476
2000.	1.07612	1.05881	1.03659
2500.	1.07737	1.06040	1.03833
3000.	1.07857	1.06189	1.03992
3500.	1.07976	1.06331	1.04139
4000.	1.08098	1.06470	1.04277
4500.	1.08226	1.06610	1.04411
5000.	1.08362	1.06754	1.04542
5500.	1.08509	1.06904	1.04675
6000.	1.08669	1.07063	1.04811
6500.	1.08842	1.07232	1.04952
7000.	1.09029	1.07411	1.05100
7500.	1.09228	1.07599	1.05255
8000.	1.09437	1.07795	1.05418
8500.	1.09656	1.07998	1.05585
9000.	1.09883	1.08207	1.05758
9500.	1.10121	1.08425	1.05934
10000.	1.10370	1.08650	1.06115
10500.	1.10633	1.08885	1.06300
11000.	1.10910	1.09129	1.06489
11500.	1.11200	1.09383	1.06682
12000.	1.11504	1.09648	1.06880
12500.	1.11823	1.09922	1.07083
13000.	1.12158	1.10207	1.07291
13500.	1.12512	1.10503	1.07504
14000.	1.12889	1.10812	1.07724
14500.	1.13294	1.11136	1.07949
15000.	1.13727	1.11478	1.08182
15500.	1.14188	1.11837	1.08421
16000.	1.14669	1.12210	1.08668
16500.	1.15159	1.12592	1.08920
17000.	1.15639	1.12969	1.09173
17500.	1.16080	1.13331	1.09422
18000.	1.16455	1.13658	1.09658
18500.	1.16740	1.13935	1.09875
19000.	1.16918	1.14146	1.10065
19500.	1.16982	1.14280	1.10218
20000.	1.16929	1.14332	1.10328
20500.	1.16767	1.14303	1.10389
21000.	1.16511	1.14193	1.10397
21500.	1.16185	1.14010	1.10353
22000.	1.15809	1.13765	1.10260
22500.	1.15392	1.13470	1.10120
23000.	1.14942	1.13138	1.09938
23500.	1.14492	1.12779	1.09719
24000.	1.14041	1.12402	1.09468
24500.	1.13591	1.12001	1.09191
25000.	1.13141	1.11601	1.08893
25500.	1.12685	1.11196	1.08579
26000.	1.12228	1.10791	1.08253
26500.	1.11772	1.10387	1.07919
27000.	1.11315	1.09982	1.07579
27500.	1.10859	1.09577	1.07232
30000.	1.08567	1.07568	1.05496
32500.	1.06266	1.05576	1.03790
35000.	1.03956	1.03602	1.02119
37500.	1.01640	1.01647	1.00486
40000.	0.99324	0.99714	0.98893
42500.	0.97014	0.97806	0.97340
45000.	0.94719	0.95928	0.95831
47500.	0.92451	0.94086	0.94368
50000.	0.90224	0.92286	0.92954
52500.	0.88054	0.90535	0.91590
55000.	0.85959	0.88840	0.90280
57500.	0.83956	0.87209	0.89025
60000.	0.82062	0.85650	0.87826
62500.	0.80292	0.84170	0.86685
65000.	0.78660	0.82776	0.85604
67500.	0.77174	0.81471	0.84583
70000.	0.75840	0.80259	0.83622

Table B.22 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.306	1.281	1.272
100.	1.302	1.280	1.271
500.	1.297	1.277	1.268
1000.	1.292	1.274	1.265
1500.	1.287	1.270	1.262
2000.	1.281	1.266	1.259
2500.	1.275	1.262	1.255
3000.	1.268	1.259	1.252
3500.	1.262	1.255	1.248
4000.	1.256	1.251	1.245
4500.	1.249	1.247	1.242
5000.	1.245	1.243	1.239
5500.	1.241	1.239	1.235
6000.	1.237	1.235	1.232
6500.	1.232	1.231	1.229
7000.	1.228	1.227	1.225
7500.	1.223	1.223	1.222
8000.	1.219	1.219	1.218
8500.	1.214	1.215	1.215
9000.	1.210	1.211	1.212
9500.	1.205	1.207	1.208
10000.	1.200	1.203	1.204
10500.	1.196	1.198	1.201
11000.	1.191	1.194	1.197
11500.	1.186	1.190	1.194
12000.	1.181	1.186	1.190
12500.	1.176	1.181	1.187
13000.	1.171	1.177	1.183
13500.	1.166	1.173	1.180
14000.	1.161	1.168	1.176
14500.	1.155	1.164	1.172
15000.	1.150	1.159	1.169
15500.	1.145	1.155	1.165
16000.	1.139	1.150	1.162
16500.	1.133	1.146	1.158
17000.	1.128	1.141	1.155
17500.	1.123	1.136	1.151
18000.	1.118	1.132	1.147
18500.	1.114	1.128	1.144
19000.	1.110	1.124	1.140
19500.	1.107	1.120	1.136
20000.	1.103	1.117	1.133
20500.	1.101	1.114	1.130
21000.	1.098	1.111	1.128
21500.	1.096	1.109	1.125
22000.	1.094	1.107	1.123
22500.	1.092	1.105	1.121
23000.	1.090	1.103	1.119
23500.	1.089	1.101	1.117
24000.	1.087	1.099	1.115
24500.	1.086	1.097	1.114
25000.	1.084	1.096	1.112
25500.	1.083	1.094	1.111
26000.	1.082	1.093	1.109
26500.	1.081	1.091	1.108
27000.	1.080	1.090	1.107
27500.	1.079	1.088	1.106
30000.	1.076	1.081	1.100
32500.	1.072	1.073	1.094
35000.	1.068	1.071	1.089
37500.	1.064	1.068	1.083
40000.	1.061	1.065	1.078
42500.	1.057	1.062	1.074
45000.	1.053	1.058	1.069
47500.	1.049	1.056	1.066
50000.	1.049	1.054	1.064
52500.	1.050	1.051	1.072
55000.	1.049	1.049	1.084
57500.	1.046	1.047	1.097
60000.	1.042	1.044	1.110
62500.	1.036	1.053	1.122
65000.	1.053	1.071	1.135
67500.	1.075	1.088	1.148
70000.	1.094	1.104	1.160

Table B.23 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.04319	1.02535	1.00194
100.	1.02026	1.00412	0.98313
500.	1.01776	1.00259	0.98271
1000.	1.01860	1.00425	0.98512
1500.	1.02057	1.00687	0.98824
2000.	1.02288	1.00973	0.99150
2500.	1.02523	1.01260	0.99468
3000.	1.02758	1.01538	0.99770
3500.	1.03000	1.01815	1.00062
4000.	1.03251	1.02096	1.00349
4500.	1.03515	1.02383	1.00635
5000.	1.03795	1.02680	1.00923
5500.	1.04093	1.02990	1.01216
6000.	1.04409	1.03314	1.01517
6500.	1.04745	1.03653	1.01827
7000.	1.05096	1.04005	1.02146
7500.	1.05462	1.04368	1.02473
8000.	1.05840	1.04740	1.02809
8500.	1.06231	1.05121	1.03150
9000.	1.06636	1.05511	1.03496
9500.	1.07057	1.05913	1.03849
10000.	1.07496	1.06327	1.04208
10500.	1.07953	1.06756	1.04575
11000.	1.08427	1.07198	1.04949
11500.	1.08921	1.07655	1.05330
12000.	1.09439	1.08127	1.05719
12500.	1.09983	1.08615	1.06116
13000.	1.10558	1.09124	1.06522
13500.	1.11171	1.09657	1.06939
14000.	1.11821	1.10217	1.07367
14500.	1.12511	1.10806	1.07811
15000.	1.13228	1.11418	1.08269
15500.	1.13957	1.12045	1.08740
16000.	1.14668	1.12665	1.09217
16500.	1.15319	1.13257	1.09686
17000.	1.15875	1.13788	1.10135
17500.	1.16306	1.14236	1.10545
18000.	1.16589	1.14578	1.10900
18500.	1.16716	1.14800	1.11186
19000.	1.16690	1.14897	1.11393
19500.	1.16528	1.14872	1.11514
20000.	1.16256	1.14738	1.11551
20500.	1.15906	1.14512	1.11507
21000.	1.15503	1.14217	1.11388
21500.	1.15037	1.13871	1.11204
22000.	1.14570	1.13492	1.10965
22500.	1.14104	1.13082	1.10684
23000.	1.13620	1.12651	1.10371
23500.	1.13136	1.12221	1.10035
25000.	1.11685	1.10929	1.08936
27500.	1.09241	1.08772	1.07071
30000.	1.06768	1.06615	1.05218
32500.	1.04261	1.04457	1.03390
35000.	1.01722	1.02300	1.01589
37500.	0.99156	1.00146	0.99815
40000.	0.96571	0.97997	0.98070
42500.	0.93978	0.95861	0.96359
45000.	0.91393	0.93745	0.94684
47500.	0.88837	0.91659	0.93050
50000.	0.86333	0.89614	0.91463
52500.	0.83909	0.87624	0.89925
55000.	0.81593	0.85702	0.88443
57500.	0.79413	0.83862	0.87021
60000.	0.77395	0.82118	0.85662
62500.	0.75559	0.80482	0.84372
65000.	0.73917	0.78965	0.83153
67500.	0.72476	0.77573	0.82008
70000.	0.71230	0.76311	0.80940

Table B.24 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.310	1.296	1.280
100.	1.305	1.289	1.280
500.	1.301	1.285	1.278
1000.	1.296	1.281	1.275
1500.	1.290	1.277	1.272
2000.	1.284	1.273	1.268
2500.	1.277	1.268	1.264
3000.	1.270	1.264	1.261
3500.	1.265	1.259	1.257
4000.	1.259	1.255	1.253
4500.	1.254	1.250	1.249
5000.	1.248	1.246	1.245
5500.	1.242	1.241	1.241
6000.	1.236	1.236	1.236
6500.	1.231	1.230	1.232
7000.	1.225	1.225	1.227
7500.	1.219	1.220	1.222
8000.	1.213	1.214	1.218
8500.	1.207	1.209	1.213
9000.	1.201	1.204	1.208
9500.	1.195	1.198	1.203
10000.	1.189	1.193	1.198
10500.	1.182	1.187	1.194
11000.	1.176	1.181	1.189
11500.	1.170	1.176	1.184
12000.	1.163	1.170	1.179
12500.	1.158	1.164	1.174
13000.	1.155	1.158	1.169
13500.	1.151	1.152	1.164
14000.	1.147	1.146	1.159
14500.	1.143	1.140	1.154
15000.	1.139	1.134	1.149
15500.	1.134	1.128	1.144
16000.	1.129	1.122	1.139
16500.	1.124	1.117	1.134
17000.	1.119	1.111	1.129
17500.	1.114	1.106	1.125
18000.	1.110	1.103	1.120
18500.	1.106	1.100	1.116
19000.	1.102	1.098	1.112
19500.	1.099	1.095	1.109
20000.	1.097	1.093	1.106
20500.	1.094	1.092	1.103
21000.	1.093	1.090	1.100
21500.	1.092	1.089	1.098
22000.	1.090	1.088	1.096
22500.	1.089	1.087	1.094
23000.	1.088	1.086	1.092
23500.	1.086	1.085	1.090
25000.	1.082	1.083	1.086
27500.	1.076	1.079	1.083
30000.	1.070	1.076	1.082
32500.	1.068	1.072	1.080
35000.	1.067	1.069	1.079
37500.	1.065	1.065	1.078
40000.	1.067	1.062	1.077
42500.	1.067	1.059	1.075
45000.	1.069	1.058	1.075
47500.	1.071	1.057	1.074
50000.	1.071	1.057	1.073
52500.	1.069	1.055	1.072
55000.	1.065	1.054	1.071
57500.	1.058	1.052	1.071
60000.	1.050	1.050	1.070
62500.	1.041	1.049	1.070
65000.	1.037	1.049	1.070
67500.	1.041	1.050	1.070
70000.	1.056	1.051	1.071

Table B.25 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.05435	1.03705	1.01170
100.	1.03084	1.01528	0.99251
500.	1.02809	1.01355	0.99200
1000.	1.02874	1.01508	0.99439
1500.	1.03059	1.01765	0.99754
2000.	1.03279	1.02051	1.00086
2500.	1.03507	1.02337	1.00410
3000.	1.03736	1.02618	1.00721
3500.	1.03974	1.02899	1.01022
4000.	1.04224	1.03186	1.01319
4500.	1.04490	1.03482	1.01616
5000.	1.04773	1.03790	1.01917
5500.	1.05076	1.04112	1.02224
6000.	1.05399	1.04451	1.02540
6500.	1.05743	1.04806	1.02865
7000.	1.06104	1.05175	1.03200
7500.	1.06480	1.05556	1.03545
8000.	1.06870	1.05948	1.03897
8500.	1.07275	1.06349	1.04256
9000.	1.07695	1.06762	1.04621
9500.	1.08133	1.07187	1.04994
10000.	1.08591	1.07627	1.05374
10500.	1.09068	1.08082	1.05761
11000.	1.09565	1.08553	1.06158
11500.	1.10085	1.09040	1.06562
12000.	1.10632	1.09544	1.06975
12500.	1.11209	1.10068	1.07397
13000.	1.11822	1.10617	1.07828
13500.	1.12475	1.11194	1.08272
14000.	1.13170	1.11802	1.08730
14500.	1.13904	1.12439	1.09205
15000.	1.14660	1.13102	1.09697
15500.	1.15423	1.13777	1.10202
16000.	1.16165	1.14439	1.10712
16500.	1.16833	1.15058	1.11213
17000.	1.17389	1.15606	1.11685
17500.	1.17808	1.16057	1.12110
18000.	1.18072	1.16385	1.12473
18500.	1.18171	1.16581	1.12758
19000.	1.18109	1.16643	1.12956
19500.	1.17908	1.16577	1.13062
20000.	1.17600	1.16399	1.13077
20500.	1.17216	1.16132	1.13009
21000.	1.16739	1.15800	1.12863
21500.	1.16262	1.15423	1.12652
22000.	1.15785	1.14999	1.12389
22500.	1.15308	1.14574	1.12086
23000.	1.14808	1.14129	1.11754
23500.	1.14307	1.13684	1.11402
25000.	1.12806	1.12348	1.10266
27500.	1.10259	1.10114	1.08355
30000.	1.07668	1.07871	1.06461
32500.	1.05029	1.05620	1.04589
35000.	1.02344	1.03358	1.02741
37500.	0.99617	1.01091	1.00917
40000.	0.96859	0.98820	0.99120
42500.	0.94083	0.96555	0.97353
45000.	0.91310	0.94304	0.95621
47500.	0.88565	0.92078	0.93928
50000.	0.85879	0.89894	0.92281
52500.	0.83286	0.87766	0.90683
55000.	0.80824	0.85714	0.89141
57500.	0.78527	0.83754	0.87660
60000.	0.76426	0.81904	0.86245
62500.	0.74544	0.80180	0.84901
65000.	0.72892	0.78594	0.83632
67500.	0.71471	0.77154	0.82443
70000.	0.70270	0.75864	0.81335

Table B.26 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.346	1.296	1.271
100.	1.349	1.289	1.271
500.	1.347	1.284	1.269
1000.	1.343	1.278	1.266
1500.	1.340	1.273	1.263
2000.	1.336	1.267	1.259
2500.	1.332	1.263	1.256
3000.	1.328	1.260	1.252
3500.	1.323	1.257	1.248
4000.	1.319	1.254	1.244
4500.	1.314	1.251	1.240
5000.	1.310	1.247	1.236
5500.	1.305	1.244	1.232
6000.	1.300	1.240	1.228
6500.	1.295	1.237	1.223
7000.	1.290	1.233	1.219
7500.	1.285	1.229	1.214
8000.	1.279	1.226	1.209
8500.	1.274	1.222	1.205
9000.	1.268	1.218	1.200
9500.	1.262	1.214	1.195
10000.	1.256	1.210	1.190
10500.	1.251	1.206	1.185
11000.	1.245	1.202	1.180
11500.	1.239	1.198	1.175
12000.	1.233	1.193	1.170
12500.	1.227	1.189	1.165
13000.	1.221	1.185	1.160
13500.	1.215	1.181	1.156
14000.	1.209	1.176	1.151
14500.	1.202	1.172	1.146
15000.	1.195	1.168	1.141
15500.	1.188	1.163	1.136
16000.	1.181	1.158	1.131
16500.	1.173	1.153	1.126
17000.	1.166	1.148	1.121
17500.	1.159	1.144	1.117
18000.	1.152	1.139	1.112
18500.	1.146	1.135	1.110
19000.	1.141	1.131	1.109
19500.	1.136	1.128	1.107
20000.	1.132	1.125	1.106
20500.	1.129	1.123	1.105
21000.	1.126	1.120	1.104
21500.	1.123	1.118	1.103
22000.	1.121	1.117	1.102
22500.	1.118	1.115	1.102
23000.	1.116	1.113	1.101
23500.	1.113	1.112	1.101
25000.	1.106	1.107	1.099
27500.	1.093	1.099	1.097
30000.	1.085	1.092	1.094
32500.	1.080	1.085	1.091
35000.	1.075	1.078	1.089
37500.	1.072	1.072	1.087
40000.	1.072	1.066	1.085
42500.	1.075	1.063	1.083
45000.	1.077	1.062	1.081
47500.	1.077	1.061	1.080
50000.	1.075	1.059	1.078
52500.	1.071	1.057	1.077
55000.	1.065	1.055	1.076
57500.	1.058	1.053	1.075
60000.	1.048	1.051	1.075
62500.	1.040	1.050	1.075
65000.	1.047	1.053	1.075
67500.	1.055	1.057	1.076
70000.	1.062	1.061	1.076

Table B.27 [ ] Hot Uncontrolled k_∞

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.06781	1.05261	1.02986
100.	1.04424	1.03075	1.01055
500.	1.04242	1.02990	1.01085
1000.	1.04450	1.03278	1.01442
1500.	1.04793	1.03685	1.01887
2000.	1.05174	1.04121	1.02348
2500.	1.05565	1.04561	1.02800
3000.	1.05970	1.05005	1.03242
3500.	1.06395	1.05458	1.03681
4000.	1.06844	1.05927	1.04120
4500.	1.07321	1.06414	1.04567
5000.	1.07827	1.06924	1.05025
5500.	1.08359	1.07453	1.05493
6000.	1.08915	1.08000	1.05972
6500.	1.09494	1.08562	1.06460
7000.	1.10099	1.09141	1.06959
7500.	1.10732	1.09741	1.07468
8000.	1.11396	1.10363	1.07989
8500.	1.12090	1.11008	1.08525
9000.	1.12818	1.11677	1.09074
9500.	1.13582	1.12371	1.09639
10000.	1.14389	1.13093	1.10218
10500.	1.15240	1.13847	1.10811
11000.	1.16134	1.14634	1.11420
11500.	1.17060	1.15446	1.12041
12000.	1.17996	1.16270	1.12673
12500.	1.18902	1.17078	1.13304
13000.	1.19729	1.17830	1.13917
13500.	1.20428	1.18495	1.14486
14000.	1.20967	1.19040	1.14989
14500.	1.21331	1.19447	1.15408
15000.	1.21512	1.19706	1.15729
15500.	1.21518	1.19817	1.15949
16000.	1.21370	1.19787	1.16065
16500.	1.21102	1.19635	1.16082
17000.	1.20750	1.19387	1.16007
17500.	1.20342	1.19068	1.15851
18000.	1.19860	1.18700	1.15627
18500.	1.19378	1.18300	1.15351
19000.	1.18897	1.17860	1.15036
19500.	1.18415	1.17419	1.14692
20000.	1.17933	1.16979	1.14329
22500.	1.15459	1.14736	1.12379
25000.	1.12942	1.12494	1.10427
27500.	1.10382	1.10246	1.08497
30000.	1.07776	1.07990	1.06591
32500.	1.05122	1.05724	1.04709
35000.	1.02420	1.03449	1.02849
37500.	0.99677	1.01167	1.01014
40000.	0.96902	0.98883	0.99205
42500.	0.94111	0.96604	0.97428
45000.	0.91324	0.94340	0.95685
47500.	0.88569	0.92104	0.93985
50000.	0.85876	0.89911	0.92330
52500.	0.83281	0.87778	0.90727
55000.	0.80820	0.85723	0.89180
57500.	0.78530	0.83763	0.87696
60000.	0.76440	0.81917	0.86280
62500.	0.74572	0.80200	0.84936
65000.	0.72936	0.78623	0.83670
67500.	0.71532	0.77194	0.82484
70000.	0.70347	0.75917	0.81381

Table B.28 [ ] Hot Uncontrolled LPF

Exposure MWd/MTU	0.00 Void History	0.40 Void History	0.80 Void History
0.	1.343	1.288	1.266
100.	1.345	1.281	1.265
500.	1.342	1.275	1.263
1000.	1.339	1.268	1.259
1500.	1.334	1.264	1.255
2000.	1.330	1.261	1.251
2500.	1.325	1.257	1.247
3000.	1.320	1.254	1.243
3500.	1.315	1.250	1.238
4000.	1.310	1.246	1.233
4500.	1.305	1.242	1.229
5000.	1.299	1.238	1.224
5500.	1.293	1.234	1.219
6000.	1.288	1.230	1.214
6500.	1.282	1.226	1.208
7000.	1.276	1.222	1.203
7500.	1.270	1.217	1.198
8000.	1.263	1.213	1.193
8500.	1.257	1.208	1.187
9000.	1.251	1.204	1.182
9500.	1.244	1.199	1.176
10000.	1.238	1.194	1.171
10500.	1.231	1.190	1.165
11000.	1.224	1.185	1.160
11500.	1.217	1.180	1.154
12000.	1.210	1.175	1.148
12500.	1.202	1.170	1.143
13000.	1.194	1.165	1.138
13500.	1.187	1.160	1.133
14000.	1.179	1.155	1.128
14500.	1.172	1.151	1.123
15000.	1.166	1.146	1.119
15500.	1.161	1.143	1.115
16000.	1.156	1.139	1.112
16500.	1.152	1.136	1.109
17000.	1.149	1.134	1.107
17500.	1.146	1.132	1.106
18000.	1.143	1.130	1.105
18500.	1.141	1.128	1.104
19000.	1.138	1.126	1.104
19500.	1.136	1.125	1.103
20000.	1.133	1.123	1.103
22500.	1.120	1.115	1.100
25000.	1.107	1.107	1.097
27500.	1.095	1.099	1.095
30000.	1.086	1.092	1.092
32500.	1.081	1.084	1.090
35000.	1.075	1.078	1.087
37500.	1.070	1.071	1.085
40000.	1.069	1.066	1.083
42500.	1.069	1.063	1.081
45000.	1.071	1.062	1.079
47500.	1.071	1.060	1.078
50000.	1.069	1.058	1.076
52500.	1.065	1.056	1.075
55000.	1.059	1.054	1.074
57500.	1.052	1.052	1.073
60000.	1.043	1.050	1.073
62500.	1.039	1.049	1.073
65000.	1.046	1.052	1.073
67500.	1.053	1.055	1.074
70000.	1.060	1.059	1.074

Appendix C Enriched Lattice Isotopic Data Tables

The results in this appendix are based on hot operating and equilibrium xenon conditions.

Table C.1 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.358	44.780	0.000	954.862	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.358	44.659	0.023	954.809	0.000	0.000	0.021	0.000	0.000	0.000
500.0	0.356	44.182	0.113	954.595	0.002	0.000	0.197	0.002	0.000	0.000
1000.0	0.354	43.593	0.225	954.328	0.004	0.000	0.419	0.007	0.000	0.000
1500.0	0.352	43.013	0.334	954.060	0.006	0.000	0.629	0.015	0.001	0.000
2000.0	0.350	42.441	0.442	953.793	0.008	0.000	0.830	0.026	0.002	0.000
2500.0	0.348	41.876	0.548	953.527	0.011	0.000	1.021	0.040	0.005	0.000
3000.0	0.346	41.319	0.652	953.261	0.013	0.000	1.203	0.056	0.008	0.000
3500.0	0.344	40.768	0.755	952.995	0.016	0.001	1.377	0.074	0.011	0.000
4000.0	0.342	40.224	0.856	952.730	0.020	0.001	1.542	0.093	0.016	0.000
4500.0	0.340	39.686	0.956	952.465	0.023	0.001	1.699	0.114	0.022	0.001
5000.0	0.338	39.155	1.054	952.200	0.026	0.001	1.849	0.137	0.029	0.001
5500.0	0.336	38.629	1.151	951.936	0.030	0.001	1.991	0.160	0.036	0.001
6000.0	0.335	38.108	1.246	951.673	0.034	0.002	2.127	0.184	0.045	0.002
6500.0	0.333	37.594	1.340	951.410	0.038	0.002	2.256	0.210	0.054	0.002
7000.0	0.331	37.084	1.433	951.148	0.042	0.002	2.379	0.236	0.064	0.003
7500.0	0.329	36.579	1.525	950.886	0.046	0.003	2.496	0.263	0.075	0.004
8000.0	0.327	36.080	1.615	950.625	0.050	0.003	2.607	0.290	0.087	0.005
8500.0	0.325	35.585	1.704	950.364	0.054	0.004	2.712	0.318	0.099	0.006
9000.0	0.323	35.095	1.792	950.103	0.059	0.004	2.813	0.347	0.111	0.007
9500.0	0.322	34.609	1.879	949.843	0.063	0.005	2.908	0.376	0.125	0.009
10000.0	0.320	34.127	1.965	949.584	0.068	0.006	2.998	0.405	0.138	0.010
10500.0	0.318	33.650	2.050	949.324	0.073	0.006	3.084	0.435	0.152	0.012
11000.0	0.316	33.177	2.134	949.066	0.077	0.007	3.165	0.465	0.167	0.014
11500.0	0.314	32.708	2.216	948.808	0.082	0.008	3.242	0.495	0.182	0.016
12000.0	0.313	32.243	2.298	948.550	0.087	0.008	3.314	0.526	0.197	0.018
12500.0	0.311	31.781	2.378	948.293	0.092	0.009	3.383	0.557	0.213	0.020
13000.0	0.309	31.323	2.458	948.036	0.098	0.010	3.448	0.588	0.228	0.023
13500.0	0.307	30.869	2.537	947.779	0.103	0.011	3.509	0.619	0.244	0.025
14000.0	0.306	30.418	2.615	947.523	0.108	0.012	3.566	0.651	0.260	0.028
14500.0	0.304	29.971	2.691	947.267	0.113	0.013	3.620	0.682	0.276	0.031
15000.0	0.302	29.527	2.767	947.012	0.119	0.014	3.671	0.714	0.292	0.034
15500.0	0.301	29.087	2.843	946.756	0.124	0.015	3.720	0.745	0.308	0.038
16000.0	0.299	28.649	2.917	946.500	0.130	0.016	3.765	0.777	0.325	0.041
16500.0	0.297	28.216	2.990	946.243	0.135	0.018	3.808	0.809	0.341	0.045
17000.0	0.296	27.785	3.063	945.986	0.141	0.019	3.849	0.840	0.358	0.049
17500.0	0.294	27.358	3.135	945.728	0.147	0.020	3.887	0.872	0.374	0.053
18000.0	0.292	26.935	3.206	945.469	0.153	0.022	3.924	0.903	0.391	0.058
18500.0	0.291	26.514	3.276	945.209	0.158	0.023	3.959	0.935	0.408	0.062
19000.0	0.289	26.098	3.345	944.948	0.164	0.025	3.991	0.966	0.424	0.067
19500.0	0.287	25.684	3.414	944.685	0.170	0.026	4.022	0.998	0.441	0.072
20000.0	0.286	25.274	3.481	944.422	0.176	0.028	4.052	1.029	0.458	0.077
22500.0	0.277	23.276	3.808	943.083	0.207	0.037	4.176	1.183	0.543	0.106
25000.0	0.269	21.363	4.115	941.709	0.240	0.048	4.266	1.334	0.626	0.140
27500.0	0.261	19.536	4.402	940.299	0.274	0.060	4.326	1.480	0.705	0.180
30000.0	0.253	17.793	4.669	938.849	0.308	0.075	4.359	1.622	0.780	0.226
32500.0	0.244	16.135	4.917	937.358	0.344	0.091	4.371	1.757	0.849	0.277
35000.0	0.236	14.563	5.146	935.823	0.380	0.110	4.364	1.886	0.913	0.333
37500.0	0.228	13.077	5.355	934.242	0.416	0.130	4.341	2.009	0.969	0.395
40000.0	0.220	11.679	5.544	932.612	0.452	0.151	4.306	2.123	1.020	0.462
42500.0	0.212	10.369	5.714	930.931	0.487	0.175	4.260	2.229	1.064	0.534
45000.0	0.204	9.147	5.865	929.195	0.522	0.199	4.206	2.327	1.101	0.611
47500.0	0.196	8.016	5.995	927.403	0.556	0.225	4.146	2.416	1.132	0.693
50000.0	0.188	6.976	6.107	925.553	0.589	0.251	4.083	2.496	1.156	0.778
52500.0	0.180	6.026	6.199	923.642	0.621	0.278	4.018	2.568	1.176	0.868
55000.0	0.173	5.165	6.272	921.670	0.651	0.305	3.952	2.630	1.190	0.960
57500.0	0.165	4.394	6.326	919.636	0.680	0.332	3.888	2.683	1.199	1.056
60000.0	0.158	3.708	6.363	917.540	0.706	0.358	3.827	2.728	1.205	1.153
62500.0	0.151	3.106	6.382	915.385	0.730	0.384	3.769	2.765	1.207	1.253
65000.0	0.144	2.582	6.386	913.171	0.752	0.408	3.715	2.795	1.207	1.354
67500.0	0.137	2.131	6.375	910.903	0.772	0.430	3.665	2.818	1.204	1.455
70000.0	0.130	1.748	6.351	908.584	0.789	0.451	3.620	2.835	1.200	1.556

Table C.2 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.358	44.780	0.000	954.862	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.358	44.660	0.024	954.802	0.000	0.000	0.024	0.000	0.000	0.000
500.0	0.356	44.184	0.118	954.559	0.002	0.000	0.222	0.002	0.000	0.000
1000.0	0.354	43.599	0.233	954.255	0.004	0.000	0.472	0.008	0.000	0.000
1500.0	0.351	43.023	0.346	953.952	0.007	0.000	0.710	0.017	0.001	0.000
2000.0	0.349	42.456	0.458	953.649	0.010	0.000	0.936	0.030	0.003	0.000
2500.0	0.347	41.897	0.567	953.347	0.013	0.000	1.152	0.045	0.006	0.000
3000.0	0.345	41.346	0.674	953.045	0.016	0.000	1.357	0.062	0.010	0.000
3500.0	0.342	40.803	0.780	952.744	0.019	0.001	1.553	0.082	0.015	0.000
4000.0	0.340	40.267	0.884	952.444	0.023	0.001	1.739	0.103	0.021	0.001
4500.0	0.338	39.738	0.986	952.144	0.027	0.001	1.917	0.125	0.028	0.001
5000.0	0.336	39.216	1.086	951.844	0.031	0.001	2.086	0.149	0.036	0.001
5500.0	0.334	38.700	1.185	951.546	0.035	0.002	2.248	0.174	0.046	0.002
6000.0	0.332	38.191	1.283	951.248	0.040	0.002	2.402	0.200	0.056	0.002
6500.0	0.330	37.687	1.378	950.950	0.044	0.003	2.549	0.227	0.067	0.003
7000.0	0.328	37.189	1.473	950.653	0.049	0.003	2.689	0.255	0.079	0.004
7500.0	0.326	36.696	1.566	950.357	0.054	0.004	2.822	0.284	0.092	0.005
8000.0	0.323	36.209	1.658	950.062	0.059	0.004	2.949	0.313	0.106	0.006
8500.0	0.321	35.727	1.748	949.766	0.064	0.005	3.070	0.343	0.121	0.007
9000.0	0.319	35.250	1.837	949.472	0.070	0.006	3.186	0.373	0.136	0.009
9500.0	0.317	34.777	1.924	949.178	0.075	0.006	3.296	0.403	0.151	0.010
10000.0	0.315	34.310	2.011	948.885	0.081	0.007	3.400	0.435	0.168	0.012
10500.0	0.314	33.847	2.096	948.592	0.086	0.008	3.500	0.466	0.184	0.014
11000.0	0.312	33.388	2.180	948.300	0.092	0.009	3.594	0.498	0.201	0.016
11500.0	0.310	32.934	2.263	948.008	0.098	0.010	3.684	0.530	0.219	0.018
12000.0	0.308	32.484	2.345	947.717	0.104	0.011	3.769	0.562	0.236	0.021
12500.0	0.306	32.039	2.426	947.426	0.110	0.012	3.850	0.595	0.254	0.023
13000.0	0.304	31.597	2.505	947.136	0.116	0.013	3.927	0.628	0.273	0.026
13500.0	0.302	31.159	2.584	946.846	0.122	0.014	4.000	0.661	0.291	0.029
14000.0	0.300	30.724	2.661	946.557	0.128	0.016	4.069	0.694	0.310	0.032
14500.0	0.298	30.294	2.738	946.268	0.134	0.017	4.134	0.727	0.328	0.036
15000.0	0.297	29.867	2.814	945.979	0.141	0.018	4.196	0.761	0.347	0.039
15500.0	0.295	29.444	2.888	945.691	0.147	0.020	4.254	0.794	0.366	0.043
16000.0	0.293	29.024	2.962	945.402	0.153	0.021	4.310	0.827	0.385	0.047
16500.0	0.291	28.607	3.035	945.114	0.160	0.023	4.363	0.861	0.404	0.051
17000.0	0.289	28.194	3.107	944.825	0.166	0.025	4.413	0.894	0.422	0.055
17500.0	0.287	27.785	3.178	944.535	0.173	0.026	4.461	0.928	0.441	0.060
18000.0	0.286	27.379	3.248	944.245	0.179	0.028	4.506	0.961	0.461	0.064
18500.0	0.284	26.977	3.318	943.953	0.186	0.030	4.549	0.994	0.480	0.069
19000.0	0.282	26.578	3.386	943.661	0.193	0.032	4.591	1.027	0.499	0.074
19500.0	0.280	26.182	3.454	943.369	0.200	0.034	4.630	1.060	0.518	0.080
20000.0	0.279	25.790	3.520	943.074	0.207	0.036	4.667	1.093	0.537	0.085
20500.0	0.277	25.402	3.586	942.779	0.214	0.038	4.703	1.126	0.556	0.091
21000.0	0.275	25.017	3.651	942.483	0.221	0.040	4.737	1.159	0.575	0.097
21500.0	0.273	24.635	3.716	942.185	0.228	0.043	4.769	1.192	0.594	0.103
22000.0	0.272	24.257	3.779	941.887	0.235	0.045	4.800	1.224	0.613	0.109
22500.0	0.270	23.883	3.842	941.586	0.242	0.048	4.829	1.257	0.632	0.116
25000.0	0.261	22.060	4.142	940.065	0.279	0.061	4.955	1.416	0.726	0.151
27500.0	0.253	20.322	4.422	938.510	0.317	0.077	5.048	1.570	0.817	0.192
30000.0	0.244	18.668	4.682	936.919	0.356	0.095	5.113	1.720	0.904	0.238
32500.0	0.236	17.097	4.922	935.293	0.395	0.115	5.155	1.863	0.985	0.289
35000.0	0.228	15.606	5.143	933.628	0.434	0.137	5.175	2.000	1.060	0.344
37500.0	0.220	14.197	5.345	931.924	0.473	0.161	5.177	2.131	1.129	0.405
40000.0	0.212	12.868	5.529	930.180	0.512	0.188	5.164	2.254	1.191	0.469
42500.0	0.204	11.619	5.693	928.394	0.550	0.216	5.139	2.370	1.247	0.538
45000.0	0.196	10.449	5.840	926.566	0.587	0.245	5.103	2.477	1.296	0.610
47500.0	0.189	9.357	5.968	924.693	0.624	0.276	5.059	2.577	1.339	0.686
50000.0	0.181	8.343	6.079	922.776	0.659	0.308	5.009	2.668	1.375	0.765
52500.0	0.174	7.405	6.172	920.814	0.692	0.340	4.954	2.751	1.406	0.847
55000.0	0.167	6.542	6.249	918.806	0.724	0.373	4.896	2.826	1.432	0.931
57500.0	0.160	5.753	6.309	916.752	0.755	0.406	4.836	2.893	1.452	1.018
60000.0	0.154	5.035	6.354	914.653	0.783	0.439	4.776	2.951	1.468	1.106
62500.0	0.148	4.386	6.383	912.509	0.810	0.471	4.717	3.003	1.479	1.196
65000.0	0.141	3.804	6.399	910.322	0.835	0.502	4.659	3.047	1.488	1.287
67500.0	0.135	3.283	6.400	908.093	0.857	0.532	4.604	3.084	1.492	1.378
70000.0	0.130	2.822	6.390	905.824	0.878	0.561	4.551	3.115	1.495	1.470

Table C.3 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.358	44.780	0.000	954.862	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.358	44.660	0.025	954.792	0.000	0.000	0.027	0.000	0.000	0.000
500.0	0.356	44.188	0.123	954.511	0.002	0.000	0.254	0.002	0.000	0.000
1000.0	0.353	43.608	0.244	954.161	0.005	0.000	0.540	0.009	0.001	0.000
1500.0	0.350	43.039	0.362	953.811	0.008	0.000	0.812	0.019	0.002	0.000
2000.0	0.348	42.479	0.478	953.461	0.012	0.000	1.071	0.033	0.004	0.000
2500.0	0.345	41.928	0.592	953.113	0.015	0.000	1.318	0.049	0.008	0.000
3000.0	0.343	41.387	0.703	952.765	0.019	0.001	1.554	0.068	0.013	0.000
3500.0	0.340	40.853	0.813	952.418	0.024	0.001	1.779	0.089	0.019	0.000
4000.0	0.338	40.328	0.920	952.072	0.028	0.001	1.994	0.111	0.027	0.001
4500.0	0.335	39.810	1.026	951.726	0.033	0.002	2.200	0.135	0.035	0.001
5000.0	0.333	39.299	1.130	951.381	0.038	0.002	2.397	0.161	0.046	0.001
5500.0	0.330	38.795	1.232	951.036	0.043	0.003	2.585	0.187	0.057	0.002
6000.0	0.328	38.298	1.332	950.693	0.049	0.003	2.765	0.215	0.070	0.003
6500.0	0.325	37.807	1.430	950.349	0.054	0.004	2.938	0.243	0.083	0.004
7000.0	0.323	37.323	1.527	950.007	0.060	0.004	3.103	0.272	0.098	0.004
7500.0	0.321	36.844	1.622	949.665	0.066	0.005	3.261	0.302	0.113	0.006
8000.0	0.319	36.371	1.716	949.324	0.072	0.006	3.412	0.333	0.129	0.007
8500.0	0.316	35.904	1.808	948.984	0.079	0.007	3.557	0.364	0.146	0.008
9000.0	0.314	35.442	1.899	948.644	0.085	0.008	3.695	0.396	0.164	0.010
9500.0	0.312	34.985	1.989	948.304	0.092	0.009	3.828	0.429	0.182	0.012
10000.0	0.310	34.534	2.077	947.966	0.098	0.010	3.955	0.462	0.201	0.013
10500.0	0.307	34.087	2.164	947.628	0.105	0.011	4.076	0.495	0.220	0.016
11000.0	0.305	33.645	2.249	947.290	0.112	0.013	4.192	0.529	0.240	0.018
11500.0	0.303	33.208	2.333	946.954	0.119	0.014	4.303	0.563	0.260	0.020
12000.0	0.301	32.775	2.416	946.617	0.126	0.015	4.410	0.597	0.280	0.023
12500.0	0.299	32.347	2.498	946.282	0.133	0.017	4.511	0.632	0.301	0.026
13000.0	0.297	31.923	2.578	945.946	0.141	0.019	4.608	0.667	0.322	0.029
13500.0	0.295	31.503	2.658	945.611	0.148	0.020	4.701	0.702	0.343	0.032
14000.0	0.293	31.087	2.736	945.277	0.155	0.022	4.789	0.737	0.364	0.035
14500.0	0.290	30.675	2.813	944.944	0.163	0.024	4.874	0.773	0.385	0.039
15000.0	0.288	30.268	2.889	944.610	0.170	0.026	4.955	0.809	0.406	0.042
15500.0	0.286	29.864	2.964	944.277	0.178	0.028	5.032	0.844	0.428	0.046
16000.0	0.284	29.463	3.038	943.944	0.185	0.030	5.105	0.880	0.449	0.050
16500.0	0.282	29.067	3.111	943.611	0.193	0.032	5.176	0.916	0.471	0.055
17000.0	0.281	28.674	3.183	943.278	0.201	0.034	5.243	0.952	0.492	0.059
17500.0	0.279	28.284	3.254	942.945	0.209	0.037	5.308	0.989	0.514	0.064
18000.0	0.277	27.899	3.324	942.611	0.217	0.039	5.370	1.025	0.535	0.069
18500.0	0.275	27.517	3.393	942.277	0.224	0.042	5.429	1.061	0.557	0.073
19000.0	0.273	27.138	3.461	941.943	0.232	0.044	5.486	1.097	0.578	0.079
19500.0	0.271	26.763	3.528	941.607	0.240	0.047	5.541	1.133	0.600	0.084
20000.0	0.269	26.391	3.595	941.271	0.248	0.050	5.593	1.169	0.621	0.090
20500.0	0.267	26.023	3.660	940.934	0.256	0.053	5.644	1.205	0.643	0.095
21000.0	0.265	25.659	3.725	940.596	0.264	0.056	5.692	1.240	0.664	0.101
21500.0	0.263	25.297	3.788	940.258	0.272	0.059	5.739	1.276	0.686	0.107
22000.0	0.262	24.940	3.851	939.918	0.281	0.062	5.784	1.312	0.707	0.113
22500.0	0.260	24.585	3.913	939.577	0.289	0.065	5.827	1.347	0.728	0.120
23000.0	0.258	24.235	3.974	939.234	0.297	0.069	5.869	1.383	0.750	0.126
23500.0	0.256	23.887	4.034	938.891	0.305	0.072	5.909	1.418	0.771	0.133
24000.0	0.254	23.543	4.093	938.547	0.313	0.076	5.948	1.453	0.792	0.140
24500.0	0.252	23.202	4.152	938.201	0.322	0.079	5.985	1.488	0.813	0.147
25000.0	0.251	22.865	4.210	937.854	0.330	0.083	6.020	1.523	0.834	0.155
27500.0	0.242	21.229	4.485	936.100	0.372	0.104	6.179	1.694	0.938	0.194
30000.0	0.233	19.673	4.740	934.314	0.415	0.127	6.308	1.861	1.039	0.237
32500.0	0.224	18.198	4.976	932.496	0.457	0.152	6.409	2.022	1.134	0.284
35000.0	0.216	16.799	5.192	930.646	0.500	0.181	6.487	2.178	1.225	0.336
37500.0	0.208	15.477	5.388	928.764	0.542	0.211	6.544	2.328	1.311	0.390
40000.0	0.200	14.229	5.567	926.848	0.584	0.244	6.582	2.471	1.390	0.448
42500.0	0.193	13.053	5.727	924.899	0.625	0.279	6.606	2.608	1.464	0.510
45000.0	0.185	11.948	5.869	922.917	0.665	0.316	6.615	2.738	1.533	0.574
47500.0	0.178	10.912	5.995	920.902	0.704	0.354	6.613	2.861	1.595	0.641
50000.0	0.171	9.943	6.104	918.854	0.741	0.394	6.602	2.976	1.653	0.710
52500.0	0.165	9.040	6.197	916.773	0.777	0.435	6.582	3.084	1.704	0.782
55000.0	0.158	8.199	6.275	914.660	0.811	0.476	6.556	3.185	1.751	0.856
57500.0	0.152	7.420	6.338	912.514	0.844	0.519	6.524	3.279	1.792	0.931
60000.0	0.147	6.699	6.388	910.338	0.875	0.561	6.489	3.365	1.829	1.009
62500.0	0.141	6.035	6.424	908.132	0.903	0.604	6.450	3.444	1.861	1.088
65000.0	0.136	5.424	6.448	905.897	0.930	0.646	6.410	3.516	1.889	1.168
67500.0	0.131	4.865	6.460	903.633	0.955	0.687	6.368	3.582	1.913	1.250
70000.0	0.126	4.354	6.461	901.343	0.978	0.728	6.326	3.641	1.934	1.332

Table C.4 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.358	44.783	0.000	954.859	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.358	44.662	0.023	954.806	0.000	0.000	0.021	0.000	0.000	0.000
500.0	0.356	44.185	0.113	954.590	0.002	0.000	0.198	0.002	0.000	0.000
1000.0	0.354	43.597	0.225	954.321	0.004	0.000	0.421	0.007	0.000	0.000
1500.0	0.352	43.017	0.334	954.052	0.006	0.000	0.634	0.015	0.001	0.000
2000.0	0.350	42.445	0.442	953.783	0.008	0.000	0.836	0.026	0.002	0.000
2500.0	0.348	41.880	0.548	953.514	0.011	0.000	1.029	0.040	0.005	0.000
3000.0	0.346	41.323	0.653	953.245	0.014	0.000	1.213	0.056	0.008	0.000
3500.0	0.344	40.773	0.755	952.976	0.017	0.001	1.389	0.074	0.012	0.000
4000.0	0.342	40.229	0.857	952.708	0.020	0.001	1.557	0.094	0.016	0.000
4500.0	0.340	39.692	0.956	952.439	0.023	0.001	1.716	0.115	0.022	0.001
5000.0	0.338	39.161	1.055	952.171	0.027	0.001	1.869	0.137	0.029	0.001
5500.0	0.336	38.636	1.152	951.903	0.030	0.001	2.014	0.160	0.037	0.001
6000.0	0.334	38.117	1.247	951.636	0.034	0.002	2.153	0.185	0.045	0.002
6500.0	0.332	37.603	1.342	951.368	0.038	0.002	2.285	0.210	0.055	0.003
7000.0	0.331	37.095	1.435	951.101	0.042	0.002	2.411	0.236	0.065	0.003
7500.0	0.329	36.591	1.526	950.834	0.046	0.003	2.531	0.263	0.076	0.004
8000.0	0.327	36.093	1.617	950.568	0.051	0.003	2.645	0.291	0.088	0.005
8500.0	0.325	35.600	1.706	950.301	0.055	0.004	2.754	0.319	0.100	0.006
9000.0	0.323	35.111	1.794	950.035	0.060	0.004	2.858	0.347	0.113	0.007
9500.0	0.321	34.627	1.881	949.768	0.064	0.005	2.957	0.377	0.127	0.009
10000.0	0.319	34.148	1.967	949.502	0.069	0.006	3.051	0.406	0.141	0.010
10500.0	0.318	33.673	2.052	949.236	0.074	0.006	3.141	0.436	0.156	0.012
11000.0	0.316	33.203	2.136	948.970	0.079	0.007	3.226	0.466	0.171	0.014
11500.0	0.314	32.736	2.218	948.704	0.084	0.008	3.306	0.497	0.186	0.016
12000.0	0.312	32.274	2.300	948.438	0.090	0.009	3.383	0.528	0.202	0.018
12500.0	0.310	31.816	2.381	948.172	0.095	0.010	3.456	0.559	0.218	0.021
13000.0	0.309	31.362	2.460	947.906	0.100	0.011	3.524	0.590	0.235	0.023
13500.0	0.307	30.912	2.539	947.640	0.106	0.011	3.589	0.622	0.251	0.026
14000.0	0.305	30.466	2.616	947.375	0.111	0.013	3.651	0.653	0.268	0.029
14500.0	0.303	30.023	2.693	947.109	0.117	0.014	3.709	0.685	0.285	0.032
15000.0	0.302	29.584	2.769	946.844	0.123	0.015	3.763	0.717	0.302	0.036
15500.0	0.300	29.148	2.844	946.579	0.128	0.016	3.814	0.749	0.319	0.039
16000.0	0.298	28.716	2.918	946.315	0.134	0.017	3.862	0.782	0.336	0.043
16500.0	0.296	28.287	2.991	946.050	0.140	0.018	3.906	0.814	0.353	0.047
17000.0	0.295	27.862	3.063	945.786	0.146	0.020	3.948	0.846	0.370	0.051
17500.0	0.293	27.439	3.134	945.522	0.151	0.021	3.987	0.878	0.387	0.055
18000.0	0.291	27.020	3.205	945.259	0.157	0.023	4.023	0.911	0.404	0.060
18500.0	0.290	26.604	3.274	944.995	0.163	0.024	4.057	0.943	0.421	0.065
19000.0	0.288	26.191	3.343	944.731	0.169	0.026	4.088	0.975	0.438	0.069
19500.0	0.286	25.782	3.412	944.466	0.175	0.027	4.117	1.007	0.455	0.075
20000.0	0.285	25.375	3.479	944.201	0.181	0.029	4.144	1.039	0.471	0.080
20500.0	0.283	24.972	3.546	943.936	0.187	0.031	4.169	1.071	0.488	0.085
21000.0	0.281	24.571	3.611	943.669	0.193	0.033	4.193	1.103	0.505	0.091
21500.0	0.280	24.174	3.676	943.402	0.199	0.034	4.215	1.134	0.521	0.097
22000.0	0.278	23.780	3.741	943.133	0.206	0.036	4.235	1.166	0.538	0.103
22500.0	0.276	23.390	3.804	942.864	0.212	0.038	4.254	1.197	0.554	0.109
23000.0	0.275	23.002	3.867	942.593	0.218	0.040	4.272	1.228	0.571	0.116
23500.0	0.273	22.618	3.929	942.320	0.225	0.042	4.288	1.259	0.587	0.123
25000.0	0.268	21.486	4.110	941.494	0.244	0.049	4.331	1.350	0.635	0.144
27500.0	0.260	19.664	4.396	940.088	0.278	0.062	4.380	1.498	0.714	0.184
30000.0	0.252	17.925	4.663	938.643	0.312	0.077	4.405	1.640	0.789	0.230
32500.0	0.244	16.270	4.911	937.157	0.347	0.093	4.410	1.775	0.858	0.281
35000.0	0.236	14.699	5.140	935.627	0.383	0.111	4.397	1.904	0.922	0.338
37500.0	0.228	13.213	5.350	934.052	0.418	0.131	4.369	2.025	0.978	0.400
40000.0	0.219	11.812	5.540	932.428	0.454	0.153	4.330	2.138	1.029	0.467
42500.0	0.212	10.499	5.711	930.753	0.489	0.176	4.281	2.243	1.072	0.539
45000.0	0.204	9.273	5.862	929.025	0.524	0.201	4.225	2.340	1.109	0.616
47500.0	0.196	8.137	5.994	927.241	0.558	0.226	4.163	2.428	1.139	0.697
50000.0	0.188	7.090	6.107	925.399	0.591	0.253	4.098	2.507	1.164	0.783
52500.0	0.180	6.133	6.201	923.497	0.622	0.280	4.032	2.577	1.183	0.872
55000.0	0.173	5.265	6.275	921.534	0.652	0.307	3.965	2.638	1.196	0.964
57500.0	0.165	4.485	6.332	919.509	0.681	0.333	3.900	2.690	1.206	1.060
60000.0	0.158	3.790	6.370	917.423	0.707	0.359	3.837	2.734	1.211	1.157
62500.0	0.151	3.179	6.392	915.277	0.731	0.385	3.778	2.771	1.213	1.257
65000.0	0.144	2.646	6.397	913.073	0.753	0.408	3.723	2.800	1.212	1.357
67500.0	0.137	2.187	6.388	910.813	0.773	0.431	3.673	2.822	1.209	1.458
70000.0	0.130	1.796	6.366	908.503	0.790	0.452	3.627	2.839	1.204	1.559

Table C.5 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.358	44.783	0.000	954.859	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.358	44.663	0.024	954.798	0.000	0.000	0.024	0.000	0.000	0.000
500.0	0.356	44.188	0.118	954.553	0.002	0.000	0.224	0.002	0.000	0.000
1000.0	0.354	43.602	0.233	954.247	0.004	0.000	0.476	0.008	0.000	0.000
1500.0	0.351	43.027	0.347	953.941	0.007	0.000	0.716	0.017	0.001	0.000
2000.0	0.349	42.460	0.458	953.635	0.010	0.000	0.945	0.030	0.003	0.000
2500.0	0.347	41.901	0.568	953.330	0.013	0.000	1.163	0.045	0.006	0.000
3000.0	0.345	41.351	0.675	953.025	0.016	0.000	1.371	0.062	0.010	0.000
3500.0	0.342	40.808	0.781	952.721	0.020	0.001	1.569	0.082	0.015	0.000
4000.0	0.340	40.273	0.885	952.416	0.023	0.001	1.759	0.103	0.021	0.001
4500.0	0.338	39.745	0.987	952.112	0.027	0.001	1.939	0.126	0.028	0.001
5000.0	0.336	39.223	1.088	951.809	0.032	0.002	2.112	0.150	0.037	0.001
5500.0	0.334	38.708	1.187	951.505	0.036	0.002	2.277	0.175	0.046	0.002
6000.0	0.332	38.199	1.284	951.202	0.040	0.002	2.435	0.201	0.057	0.002
6500.0	0.329	37.696	1.380	950.900	0.045	0.003	2.585	0.228	0.068	0.003
7000.0	0.327	37.199	1.475	950.597	0.050	0.003	2.729	0.256	0.080	0.004
7500.0	0.325	36.708	1.568	950.295	0.055	0.004	2.866	0.284	0.094	0.005
8000.0	0.323	36.222	1.660	949.994	0.060	0.004	2.998	0.313	0.108	0.006
8500.0	0.321	35.742	1.750	949.692	0.065	0.005	3.123	0.343	0.123	0.007
9000.0	0.319	35.266	1.840	949.391	0.071	0.006	3.242	0.373	0.138	0.009
9500.0	0.317	34.796	1.927	949.090	0.076	0.007	3.357	0.404	0.154	0.010
10000.0	0.315	34.331	2.014	948.790	0.082	0.007	3.466	0.435	0.171	0.012
10500.0	0.313	33.870	2.099	948.489	0.088	0.008	3.569	0.467	0.188	0.014
11000.0	0.311	33.414	2.183	948.189	0.094	0.009	3.668	0.499	0.206	0.016
11500.0	0.309	32.963	2.266	947.889	0.100	0.010	3.763	0.531	0.224	0.019
12000.0	0.307	32.516	2.348	947.589	0.106	0.011	3.853	0.563	0.242	0.021
12500.0	0.305	32.073	2.429	947.289	0.112	0.013	3.938	0.596	0.261	0.024
13000.0	0.303	31.635	2.509	946.990	0.119	0.014	4.020	0.629	0.280	0.027
13500.0	0.301	31.201	2.587	946.690	0.125	0.015	4.097	0.662	0.299	0.030
14000.0	0.299	30.771	2.665	946.391	0.132	0.016	4.170	0.696	0.318	0.033
14500.0	0.297	30.344	2.741	946.093	0.138	0.018	4.240	0.729	0.338	0.037
15000.0	0.296	29.922	2.817	945.794	0.145	0.019	4.306	0.763	0.357	0.040
15500.0	0.294	29.503	2.891	945.496	0.151	0.021	4.368	0.797	0.377	0.044
16000.0	0.292	29.088	2.965	945.198	0.158	0.022	4.427	0.831	0.397	0.048
16500.0	0.290	28.677	3.037	944.900	0.164	0.024	4.482	0.865	0.417	0.052
17000.0	0.288	28.269	3.109	944.603	0.171	0.026	4.534	0.899	0.436	0.057
17500.0	0.286	27.865	3.180	944.306	0.178	0.028	4.583	0.933	0.456	0.062
18000.0	0.285	27.464	3.249	944.009	0.185	0.029	4.629	0.967	0.475	0.066
18500.0	0.283	27.066	3.318	943.712	0.192	0.031	4.672	1.001	0.495	0.072
19000.0	0.281	26.671	3.386	943.416	0.198	0.033	4.713	1.035	0.514	0.077
19500.0	0.279	26.280	3.454	943.119	0.205	0.035	4.751	1.069	0.534	0.082
20000.0	0.277	25.892	3.520	942.822	0.212	0.038	4.787	1.102	0.553	0.088
20500.0	0.276	25.507	3.585	942.525	0.219	0.040	4.821	1.136	0.572	0.094
21000.0	0.274	25.125	3.650	942.227	0.226	0.042	4.852	1.170	0.591	0.100
21500.0	0.272	24.747	3.714	941.929	0.233	0.044	4.882	1.203	0.610	0.106
22000.0	0.270	24.372	3.777	941.629	0.241	0.047	4.910	1.236	0.629	0.112
22500.0	0.269	24.000	3.839	941.329	0.248	0.049	4.936	1.270	0.647	0.119
23000.0	0.267	23.631	3.901	941.028	0.255	0.052	4.961	1.302	0.666	0.126
23500.0	0.265	23.265	3.961	940.726	0.262	0.055	4.984	1.335	0.685	0.133
24000.0	0.264	22.903	4.021	940.423	0.270	0.057	5.006	1.368	0.703	0.140
24500.0	0.262	22.544	4.080	940.118	0.277	0.060	5.027	1.400	0.721	0.148
25000.0	0.260	22.188	4.138	939.812	0.284	0.063	5.046	1.432	0.740	0.155
25500.0	0.258	21.836	4.196	939.505	0.292	0.066	5.064	1.464	0.758	0.163
27500.0	0.252	20.458	4.417	938.262	0.322	0.079	5.126	1.589	0.829	0.196
30000.0	0.243	18.809	4.677	936.677	0.360	0.097	5.181	1.739	0.915	0.242
32500.0	0.235	17.241	4.917	935.055	0.399	0.117	5.213	1.883	0.996	0.293
35000.0	0.227	15.753	5.138	933.396	0.438	0.140	5.226	2.020	1.071	0.349
37500.0	0.219	14.344	5.341	931.699	0.476	0.164	5.222	2.150	1.140	0.409
40000.0	0.211	13.014	5.525	929.961	0.515	0.190	5.204	2.273	1.202	0.474
42500.0	0.203	11.763	5.690	928.182	0.553	0.218	5.174	2.387	1.258	0.542
45000.0	0.196	10.590	5.838	926.360	0.590	0.247	5.135	2.494	1.307	0.615
47500.0	0.188	9.494	5.967	924.496	0.626	0.278	5.087	2.592	1.349	0.690
50000.0	0.181	8.474	6.079	922.586	0.661	0.310	5.034	2.683	1.386	0.769
52500.0	0.174	7.530	6.174	920.632	0.694	0.342	4.977	2.764	1.416	0.851
55000.0	0.167	6.661	6.253	918.633	0.726	0.375	4.917	2.838	1.442	0.935
57500.0	0.160	5.865	6.315	916.588	0.756	0.408	4.856	2.904	1.461	1.022
60000.0	0.154	5.139	6.361	914.497	0.785	0.440	4.795	2.962	1.477	1.110
62500.0	0.147	4.482	6.392	912.363	0.812	0.472	4.734	3.012	1.488	1.200
65000.0	0.141	3.891	6.409	910.185	0.836	0.504	4.675	3.055	1.496	1.291
67500.0	0.135	3.363	6.413	907.965	0.859	0.534	4.618	3.091	1.500	1.382
70000.0	0.130	2.894	6.404	905.705	0.879	0.562	4.564	3.122	1.502	1.474

Table C.6 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.358	44.783	0.000	954.859	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.358	44.664	0.025	954.789	0.000	0.000	0.027	0.000	0.000	0.000
500.0	0.356	44.192	0.123	954.505	0.002	0.000	0.256	0.002	0.000	0.000
1000.0	0.353	43.612	0.244	954.151	0.005	0.000	0.545	0.009	0.001	0.000
1500.0	0.350	43.042	0.363	953.798	0.008	0.000	0.820	0.019	0.002	0.000
2000.0	0.348	42.483	0.479	953.445	0.012	0.000	1.082	0.033	0.004	0.000
2500.0	0.345	41.933	0.593	953.092	0.016	0.000	1.333	0.049	0.008	0.000
3000.0	0.342	41.392	0.705	952.740	0.020	0.001	1.572	0.068	0.013	0.000
3500.0	0.340	40.858	0.814	952.389	0.024	0.001	1.801	0.089	0.019	0.000
4000.0	0.337	40.333	0.922	952.038	0.029	0.001	2.020	0.111	0.027	0.001
4500.0	0.335	39.816	1.028	951.687	0.033	0.002	2.230	0.135	0.036	0.001
5000.0	0.332	39.306	1.132	951.337	0.038	0.002	2.430	0.161	0.046	0.001
5500.0	0.330	38.803	1.234	950.987	0.044	0.003	2.623	0.187	0.058	0.002
6000.0	0.328	38.306	1.334	950.638	0.049	0.003	2.807	0.215	0.070	0.003
6500.0	0.325	37.817	1.433	950.289	0.055	0.004	2.984	0.243	0.084	0.004
7000.0	0.323	37.333	1.530	949.940	0.061	0.005	3.154	0.272	0.099	0.005
7500.0	0.320	36.856	1.626	949.592	0.067	0.005	3.317	0.302	0.114	0.006
8000.0	0.318	36.384	1.720	949.244	0.074	0.006	3.473	0.333	0.131	0.007
8500.0	0.316	35.918	1.812	948.897	0.080	0.007	3.622	0.364	0.148	0.008
9000.0	0.314	35.458	1.903	948.550	0.087	0.008	3.766	0.396	0.166	0.010
9500.0	0.311	35.003	1.993	948.203	0.093	0.009	3.903	0.428	0.185	0.012
10000.0	0.309	34.553	2.081	947.857	0.100	0.010	4.035	0.461	0.204	0.014
10500.0	0.307	34.108	2.168	947.511	0.107	0.012	4.162	0.495	0.224	0.016
11000.0	0.305	33.668	2.254	947.165	0.114	0.013	4.283	0.528	0.244	0.018
11500.0	0.302	33.233	2.338	946.820	0.122	0.014	4.399	0.562	0.265	0.021
12000.0	0.300	32.803	2.421	946.475	0.129	0.016	4.511	0.597	0.286	0.023
12500.0	0.298	32.378	2.503	946.130	0.136	0.017	4.618	0.632	0.307	0.026
13000.0	0.296	31.957	2.584	945.785	0.144	0.019	4.720	0.667	0.328	0.029
13500.0	0.294	31.540	2.663	945.441	0.151	0.021	4.818	0.702	0.350	0.032
14000.0	0.292	31.128	2.741	945.097	0.159	0.023	4.912	0.737	0.372	0.036
14500.0	0.290	30.719	2.819	944.753	0.167	0.025	5.001	0.773	0.394	0.040
15000.0	0.288	30.315	2.895	944.409	0.174	0.027	5.087	0.809	0.416	0.043
15500.0	0.285	29.915	2.970	944.066	0.182	0.029	5.169	0.845	0.439	0.047
16000.0	0.283	29.519	3.043	943.723	0.190	0.031	5.247	0.882	0.461	0.052
16500.0	0.281	29.127	3.116	943.380	0.198	0.033	5.321	0.918	0.483	0.056
17000.0	0.279	28.739	3.188	943.038	0.206	0.036	5.392	0.954	0.506	0.060
17500.0	0.277	28.354	3.259	942.696	0.214	0.038	5.460	0.991	0.528	0.065
18000.0	0.275	27.973	3.328	942.354	0.222	0.041	5.524	1.028	0.550	0.070
18500.0	0.274	27.595	3.397	942.012	0.230	0.043	5.585	1.064	0.572	0.075
19000.0	0.272	27.221	3.465	941.671	0.238	0.046	5.643	1.101	0.595	0.081
19500.0	0.270	26.850	3.532	941.330	0.246	0.049	5.698	1.138	0.617	0.086
20000.0	0.268	26.483	3.598	940.988	0.255	0.052	5.751	1.175	0.638	0.092
20500.0	0.266	26.119	3.663	940.647	0.263	0.055	5.801	1.211	0.660	0.097
21000.0	0.264	25.758	3.727	940.305	0.271	0.058	5.848	1.248	0.682	0.103
21500.0	0.262	25.401	3.790	939.964	0.279	0.061	5.894	1.285	0.703	0.110
22000.0	0.260	25.047	3.852	939.621	0.287	0.064	5.937	1.321	0.725	0.116
22500.0	0.258	24.696	3.914	939.279	0.295	0.068	5.978	1.357	0.746	0.123
23000.0	0.257	24.348	3.975	938.935	0.303	0.071	6.017	1.394	0.767	0.129
23500.0	0.255	24.003	4.034	938.591	0.312	0.075	6.055	1.430	0.789	0.136
24000.0	0.253	23.662	4.093	938.247	0.320	0.078	6.090	1.466	0.810	0.143
24500.0	0.251	23.324	4.152	937.901	0.328	0.082	6.125	1.501	0.831	0.150
25000.0	0.249	22.989	4.209	937.554	0.336	0.086	6.157	1.537	0.851	0.158
25500.0	0.248	22.657	4.265	937.206	0.345	0.090	6.188	1.572	0.872	0.165
26000.0	0.246	22.329	4.321	936.858	0.353	0.094	6.218	1.607	0.893	0.173
26500.0	0.244	22.004	4.376	936.508	0.361	0.098	6.247	1.642	0.913	0.181
27000.0	0.242	21.681	4.430	936.157	0.370	0.102	6.274	1.677	0.933	0.189
27500.0	0.240	21.362	4.483	935.805	0.378	0.107	6.300	1.712	0.954	0.197
28000.0	0.239	21.047	4.536	935.451	0.386	0.111	6.325	1.746	0.974	0.206
28500.0	0.237	20.734	4.587	935.097	0.395	0.116	6.349	1.780	0.994	0.214
30000.0	0.232	19.814	4.738	934.025	0.420	0.130	6.415	1.881	1.053	0.241
32500.0	0.223	18.344	4.973	932.213	0.462	0.156	6.504	2.044	1.148	0.288
35000.0	0.215	16.949	5.188	930.369	0.504	0.184	6.572	2.201	1.239	0.339
37500.0	0.207	15.628	5.386	928.493	0.546	0.214	6.620	2.351	1.325	0.394
40000.0	0.200	14.381	5.564	926.584	0.588	0.247	6.651	2.494	1.405	0.452
42500.0	0.192	13.204	5.725	924.642	0.628	0.282	6.668	2.631	1.479	0.514
45000.0	0.185	12.097	5.869	922.668	0.668	0.319	6.672	2.760	1.547	0.578
47500.0	0.178	11.059	5.995	920.660	0.707	0.357	6.665	2.882	1.610	0.645
50000.0	0.171	10.086	6.105	918.619	0.744	0.397	6.649	2.997	1.667	0.714
52500.0	0.164	9.179	6.200	916.546	0.780	0.438	6.626	3.104	1.718	0.786
55000.0	0.158	8.333	6.279	914.441	0.814	0.479	6.596	3.203	1.765	0.860
57500.0	0.152	7.548	6.344	912.304	0.846	0.522	6.562	3.296	1.806	0.935
60000.0	0.146	6.821	6.396	910.136	0.877	0.564	6.524	3.381	1.842	1.013
62500.0	0.141	6.151	6.433	907.938	0.906	0.606	6.483	3.459	1.874	1.092
65000.0	0.136	5.534	6.459	905.711	0.932	0.649	6.441	3.531	1.902	1.172
67500.0	0.131	4.968	6.473	903.456	0.957	0.690	6.397	3.596	1.926	1.253
70000.0	0.126	4.450	6.475	901.174	0.980	0.731	6.353	3.654	1.946	1.336

Table C.7 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.352	43.867	0.022	955.609	0.000	0.000	0.019	0.000	0.000	0.000
500.0	0.350	43.389	0.112	955.400	0.002	0.000	0.188	0.001	0.000	0.000
1000.0	0.348	42.800	0.222	955.138	0.003	0.000	0.405	0.006	0.000	0.000
1500.0	0.346	42.219	0.330	954.877	0.006	0.000	0.612	0.015	0.001	0.000
2000.0	0.344	41.646	0.437	954.616	0.008	0.000	0.809	0.026	0.002	0.000
2500.0	0.342	41.081	0.542	954.356	0.010	0.000	0.996	0.039	0.004	0.000
3000.0	0.340	40.522	0.645	954.096	0.013	0.000	1.174	0.055	0.007	0.000
3500.0	0.338	39.971	0.747	953.837	0.016	0.000	1.343	0.073	0.011	0.000
4000.0	0.336	39.426	0.847	953.578	0.019	0.001	1.505	0.092	0.016	0.000
4500.0	0.335	38.887	0.946	953.319	0.022	0.001	1.658	0.113	0.021	0.001
5000.0	0.333	38.354	1.043	953.062	0.025	0.001	1.804	0.135	0.027	0.001
5500.0	0.331	37.827	1.139	952.804	0.029	0.001	1.943	0.158	0.035	0.001
6000.0	0.329	37.305	1.233	952.548	0.032	0.002	2.074	0.182	0.043	0.002
6500.0	0.327	36.789	1.327	952.292	0.036	0.002	2.200	0.207	0.052	0.002
7000.0	0.325	36.278	1.419	952.036	0.040	0.002	2.318	0.233	0.062	0.003
7500.0	0.324	35.772	1.509	951.781	0.044	0.003	2.431	0.260	0.072	0.004
8000.0	0.322	35.271	1.599	951.526	0.048	0.003	2.539	0.287	0.083	0.005
8500.0	0.320	34.775	1.687	951.272	0.052	0.004	2.640	0.315	0.095	0.006
9000.0	0.318	34.283	1.775	951.018	0.056	0.004	2.737	0.343	0.108	0.007
9500.0	0.316	33.796	1.861	950.764	0.061	0.005	2.829	0.372	0.120	0.008
10000.0	0.315	33.313	1.946	950.511	0.065	0.005	2.915	0.402	0.134	0.010
10500.0	0.313	32.834	2.030	950.258	0.070	0.006	2.997	0.431	0.148	0.012
11000.0	0.311	32.360	2.113	950.005	0.074	0.006	3.075	0.461	0.162	0.013
11500.0	0.310	31.889	2.195	949.753	0.079	0.007	3.148	0.492	0.177	0.015
12000.0	0.308	31.423	2.276	949.501	0.084	0.008	3.217	0.523	0.192	0.018
12500.0	0.306	30.961	2.356	949.250	0.089	0.009	3.282	0.553	0.207	0.020
13000.0	0.304	30.502	2.435	948.998	0.094	0.010	3.343	0.585	0.222	0.022
13500.0	0.303	30.047	2.513	948.748	0.099	0.010	3.401	0.616	0.238	0.025
14000.0	0.301	29.595	2.590	948.498	0.104	0.011	3.454	0.648	0.253	0.028
14500.0	0.299	29.147	2.666	948.248	0.109	0.012	3.504	0.679	0.269	0.031
15000.0	0.298	28.702	2.741	947.999	0.114	0.013	3.551	0.711	0.285	0.034
15500.0	0.296	28.260	2.816	947.750	0.119	0.014	3.594	0.743	0.301	0.038
16000.0	0.295	27.822	2.890	947.502	0.125	0.015	3.634	0.775	0.317	0.041
16500.0	0.293	27.386	2.962	947.255	0.130	0.017	3.671	0.807	0.332	0.045
17000.0	0.291	26.954	3.034	947.008	0.135	0.018	3.705	0.839	0.348	0.049
17500.0	0.290	26.525	3.106	946.760	0.140	0.019	3.737	0.871	0.364	0.053
18000.0	0.288	26.098	3.176	946.513	0.146	0.020	3.766	0.903	0.380	0.057
18500.0	0.287	25.675	3.246	946.266	0.151	0.021	3.793	0.935	0.395	0.062
19000.0	0.285	25.255	3.315	946.018	0.157	0.023	3.818	0.966	0.411	0.067
19500.0	0.283	24.838	3.383	945.769	0.162	0.024	3.841	0.998	0.426	0.072
20000.0	0.282	24.424	3.450	945.520	0.168	0.026	3.862	1.030	0.442	0.077
20500.0	0.280	24.013	3.517	945.269	0.173	0.027	3.882	1.061	0.457	0.082
21000.0	0.279	23.606	3.583	945.017	0.179	0.029	3.900	1.092	0.473	0.088
22500.0	0.274	22.403	3.776	944.253	0.196	0.034	3.948	1.185	0.520	0.106
25000.0	0.266	20.464	4.083	942.950	0.226	0.044	4.004	1.336	0.596	0.140
27500.0	0.258	18.608	4.370	941.608	0.257	0.055	4.034	1.481	0.670	0.180
30000.0	0.250	16.837	4.639	940.224	0.289	0.068	4.041	1.620	0.739	0.226
32500.0	0.242	15.153	4.888	938.795	0.322	0.082	4.030	1.753	0.802	0.278
35000.0	0.234	13.555	5.117	937.318	0.355	0.098	4.003	1.879	0.859	0.335
37500.0	0.226	12.048	5.327	935.788	0.389	0.116	3.963	1.998	0.910	0.399
40000.0	0.218	10.632	5.517	934.204	0.423	0.135	3.913	2.108	0.954	0.468
42500.0	0.210	9.311	5.687	932.561	0.456	0.156	3.855	2.209	0.991	0.543
45000.0	0.202	8.087	5.835	930.854	0.489	0.178	3.792	2.302	1.021	0.623
47500.0	0.194	6.961	5.963	929.081	0.521	0.201	3.726	2.385	1.045	0.709
50000.0	0.185	5.936	6.070	927.239	0.552	0.224	3.659	2.458	1.064	0.799
52500.0	0.177	5.013	6.157	925.324	0.582	0.248	3.592	2.522	1.077	0.894
55000.0	0.169	4.190	6.222	923.336	0.610	0.271	3.527	2.576	1.085	0.992
57500.0	0.161	3.466	6.268	921.274	0.636	0.294	3.466	2.621	1.089	1.094
60000.0	0.154	2.839	6.294	919.139	0.660	0.317	3.409	2.658	1.090	1.198
62500.0	0.146	2.302	6.303	916.935	0.682	0.338	3.357	2.687	1.089	1.304
65000.0	0.138	1.849	6.295	914.665	0.701	0.358	3.310	2.708	1.085	1.411
67500.0	0.131	1.474	6.272	912.335	0.718	0.376	3.269	2.724	1.081	1.518
70000.0	0.124	1.166	6.236	909.952	0.733	0.393	3.233	2.734	1.075	1.625

Table C.8 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.351	43.867	0.023	955.602	0.000	0.000	0.021	0.000	0.000	0.000
500.0	0.350	43.391	0.116	955.363	0.002	0.000	0.213	0.002	0.000	0.000
1000.0	0.347	42.805	0.231	955.065	0.004	0.000	0.459	0.007	0.000	0.000
1500.0	0.345	42.228	0.343	954.768	0.007	0.000	0.693	0.017	0.001	0.000
2000.0	0.343	41.660	0.453	954.471	0.009	0.000	0.916	0.029	0.003	0.000
2500.0	0.341	41.100	0.561	954.175	0.012	0.000	1.128	0.044	0.006	0.000
3000.0	0.339	40.548	0.667	953.880	0.015	0.000	1.329	0.061	0.009	0.000
3500.0	0.337	40.004	0.772	953.586	0.019	0.001	1.521	0.080	0.014	0.000
4000.0	0.335	39.467	0.875	953.292	0.022	0.001	1.704	0.101	0.020	0.001
4500.0	0.332	38.937	0.976	952.999	0.026	0.001	1.878	0.124	0.027	0.001
5000.0	0.330	38.413	1.075	952.706	0.030	0.001	2.043	0.148	0.035	0.001
5500.0	0.328	37.896	1.173	952.415	0.034	0.002	2.200	0.173	0.044	0.002
6000.0	0.326	37.385	1.269	952.124	0.038	0.002	2.350	0.199	0.054	0.002
6500.0	0.324	36.879	1.364	951.833	0.043	0.003	2.493	0.225	0.065	0.003
7000.0	0.322	36.380	1.458	951.544	0.047	0.003	2.628	0.253	0.077	0.004
7500.0	0.320	35.886	1.550	951.255	0.052	0.004	2.757	0.281	0.090	0.005
8000.0	0.318	35.397	1.640	950.967	0.057	0.004	2.880	0.311	0.103	0.006
8500.0	0.316	34.913	1.730	950.680	0.062	0.005	2.996	0.340	0.117	0.007
9000.0	0.314	34.435	1.818	950.393	0.067	0.005	3.107	0.370	0.132	0.008
9500.0	0.313	33.961	1.905	950.106	0.072	0.006	3.212	0.401	0.148	0.010
10000.0	0.311	33.492	1.990	949.821	0.077	0.007	3.312	0.432	0.164	0.012
10500.0	0.309	33.028	2.075	949.535	0.083	0.008	3.407	0.463	0.180	0.014
11000.0	0.307	32.568	2.158	949.250	0.088	0.008	3.497	0.495	0.197	0.016
11500.0	0.305	32.112	2.240	948.966	0.094	0.009	3.582	0.527	0.214	0.018
12000.0	0.303	31.661	2.321	948.682	0.100	0.010	3.663	0.560	0.232	0.020
12500.0	0.301	31.214	2.401	948.399	0.105	0.011	3.739	0.592	0.250	0.023
13000.0	0.299	30.771	2.480	948.116	0.111	0.012	3.811	0.625	0.268	0.026
13500.0	0.298	30.331	2.557	947.834	0.117	0.014	3.878	0.658	0.286	0.029
14000.0	0.296	29.896	2.634	947.552	0.123	0.015	3.942	0.692	0.304	0.032
14500.0	0.294	29.465	2.710	947.271	0.129	0.016	4.001	0.725	0.323	0.035
15000.0	0.292	29.036	2.785	946.991	0.135	0.017	4.057	0.759	0.341	0.039
15500.0	0.290	28.612	2.859	946.712	0.141	0.019	4.110	0.792	0.359	0.043
16000.0	0.289	28.191	2.932	946.433	0.147	0.020	4.158	0.826	0.378	0.047
16500.0	0.287	27.773	3.004	946.154	0.153	0.022	4.204	0.860	0.396	0.051
17000.0	0.285	27.358	3.075	945.877	0.159	0.023	4.246	0.893	0.415	0.055
17500.0	0.284	26.947	3.146	945.599	0.165	0.025	4.285	0.927	0.433	0.060
18000.0	0.282	26.538	3.215	945.322	0.172	0.026	4.322	0.961	0.451	0.065
18500.0	0.280	26.133	3.284	945.045	0.178	0.028	4.356	0.995	0.469	0.070
19000.0	0.278	25.731	3.352	944.768	0.184	0.030	4.387	1.028	0.487	0.075
19500.0	0.277	25.332	3.419	944.490	0.190	0.032	4.417	1.062	0.505	0.080
20000.0	0.275	24.936	3.485	944.212	0.197	0.033	4.444	1.095	0.523	0.086
20500.0	0.273	24.544	3.551	943.933	0.203	0.035	4.470	1.128	0.540	0.091
21000.0	0.272	24.155	3.615	943.654	0.210	0.037	4.494	1.161	0.558	0.097
21500.0	0.270	23.768	3.679	943.373	0.216	0.039	4.516	1.194	0.576	0.104
22000.0	0.268	23.386	3.742	943.091	0.223	0.041	4.537	1.226	0.594	0.110
22500.0	0.267	23.006	3.804	942.808	0.229	0.044	4.557	1.259	0.611	0.116
25000.0	0.259	21.157	4.104	941.371	0.263	0.056	4.637	1.417	0.698	0.152
27500.0	0.250	19.391	4.384	939.899	0.298	0.070	4.689	1.570	0.783	0.194
30000.0	0.242	17.707	4.644	938.389	0.334	0.086	4.717	1.717	0.862	0.241
32500.0	0.234	16.106	4.885	936.841	0.371	0.103	4.724	1.857	0.936	0.293
35000.0	0.226	14.588	5.106	935.251	0.407	0.123	4.714	1.989	1.003	0.350
37500.0	0.218	13.153	5.309	933.618	0.444	0.144	4.689	2.115	1.064	0.412
40000.0	0.210	11.801	5.492	931.939	0.480	0.168	4.652	2.232	1.118	0.479
42500.0	0.202	10.533	5.657	930.213	0.516	0.192	4.604	2.340	1.165	0.551
45000.0	0.195	9.350	5.803	928.438	0.551	0.218	4.549	2.441	1.205	0.627
47500.0	0.187	8.251	5.929	926.611	0.585	0.246	4.489	2.532	1.239	0.707
50000.0	0.180	7.237	6.037	924.731	0.618	0.273	4.425	2.614	1.266	0.791
52500.0	0.172	6.307	6.127	922.797	0.650	0.302	4.359	2.687	1.287	0.878
55000.0	0.165	5.461	6.199	920.807	0.680	0.330	4.292	2.751	1.304	0.968
57500.0	0.158	4.697	6.253	918.762	0.708	0.359	4.226	2.807	1.315	1.061
60000.0	0.151	4.013	6.290	916.662	0.735	0.387	4.162	2.854	1.322	1.155
62500.0	0.144	3.406	6.311	914.507	0.759	0.414	4.101	2.894	1.326	1.252
65000.0	0.137	2.872	6.316	912.301	0.781	0.440	4.044	2.926	1.326	1.349
67500.0	0.131	2.407	6.308	910.044	0.801	0.464	3.991	2.951	1.325	1.447
70000.0	0.125	2.006	6.287	907.741	0.819	0.487	3.942	2.970	1.321	1.544

Table C.9 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.351	43.868	0.025	955.592	0.000	0.000	0.024	0.000	0.000	0.000
500.0	0.349	43.395	0.122	955.315	0.002	0.000	0.245	0.002	0.000	0.000
1000.0	0.347	42.813	0.242	954.969	0.005	0.000	0.528	0.008	0.001	0.000
1500.0	0.344	42.242	0.359	954.625	0.008	0.000	0.797	0.019	0.002	0.000
2000.0	0.342	41.681	0.474	954.281	0.011	0.000	1.054	0.032	0.004	0.000
2500.0	0.339	41.129	0.587	953.938	0.015	0.000	1.298	0.049	0.007	0.000
3000.0	0.337	40.586	0.698	953.596	0.019	0.001	1.530	0.067	0.012	0.000
3500.0	0.334	40.051	0.806	953.255	0.023	0.001	1.752	0.088	0.018	0.000
4000.0	0.332	39.524	0.913	952.916	0.027	0.001	1.963	0.111	0.026	0.001
4500.0	0.329	39.005	1.017	952.576	0.032	0.002	2.165	0.135	0.035	0.001
5000.0	0.327	38.492	1.120	952.238	0.037	0.002	2.358	0.160	0.045	0.001
5500.0	0.325	37.987	1.221	951.901	0.042	0.002	2.541	0.186	0.056	0.002
6000.0	0.322	37.488	1.320	951.565	0.047	0.003	2.717	0.214	0.069	0.003
6500.0	0.320	36.996	1.418	951.230	0.052	0.004	2.884	0.242	0.082	0.003
7000.0	0.318	36.510	1.514	950.895	0.058	0.004	3.044	0.271	0.097	0.004
7500.0	0.316	36.030	1.608	950.561	0.064	0.005	3.197	0.301	0.112	0.005
8000.0	0.313	35.556	1.701	950.229	0.070	0.006	3.342	0.332	0.128	0.007
8500.0	0.311	35.087	1.792	949.897	0.076	0.007	3.481	0.363	0.145	0.008
9000.0	0.309	34.623	1.882	949.565	0.082	0.008	3.614	0.395	0.163	0.010
9500.0	0.307	34.165	1.970	949.235	0.089	0.009	3.741	0.428	0.181	0.011
10000.0	0.305	33.712	2.058	948.905	0.095	0.010	3.861	0.460	0.200	0.013
10500.0	0.303	33.264	2.143	948.576	0.102	0.011	3.977	0.494	0.219	0.016
11000.0	0.300	32.821	2.228	948.247	0.108	0.012	4.086	0.527	0.239	0.018
11500.0	0.298	32.382	2.311	947.919	0.115	0.013	4.191	0.561	0.259	0.020
12000.0	0.296	31.948	2.393	947.592	0.122	0.015	4.291	0.596	0.279	0.023
12500.0	0.294	31.519	2.474	947.265	0.129	0.016	4.385	0.631	0.300	0.026
13000.0	0.292	31.094	2.554	946.939	0.136	0.018	4.475	0.666	0.321	0.029
13500.0	0.290	30.673	2.632	946.614	0.143	0.019	4.561	0.701	0.342	0.032
14000.0	0.288	30.256	2.710	946.289	0.150	0.021	4.642	0.736	0.363	0.036
14500.0	0.286	29.843	2.786	945.965	0.157	0.023	4.719	0.772	0.384	0.039
15000.0	0.284	29.434	2.861	945.642	0.164	0.024	4.791	0.808	0.405	0.043
15500.0	0.282	29.030	2.935	945.319	0.172	0.026	4.860	0.844	0.427	0.047
16000.0	0.280	28.628	3.008	944.997	0.179	0.028	4.925	0.880	0.448	0.051
16500.0	0.278	28.231	3.080	944.676	0.186	0.030	4.986	0.916	0.469	0.056
17000.0	0.277	27.836	3.151	944.356	0.194	0.032	5.044	0.953	0.491	0.060
17500.0	0.275	27.446	3.221	944.036	0.201	0.034	5.098	0.989	0.512	0.065
18000.0	0.273	27.058	3.290	943.716	0.208	0.037	5.149	1.025	0.533	0.070
18500.0	0.271	26.674	3.358	943.397	0.216	0.039	5.197	1.062	0.554	0.075
19000.0	0.269	26.294	3.426	943.078	0.223	0.041	5.242	1.098	0.574	0.080
19500.0	0.267	25.916	3.492	942.760	0.231	0.044	5.285	1.135	0.595	0.086
20000.0	0.266	25.542	3.557	942.441	0.238	0.047	5.325	1.171	0.616	0.091
20500.0	0.264	25.171	3.622	942.122	0.246	0.049	5.362	1.207	0.636	0.097
21000.0	0.262	24.803	3.686	941.802	0.253	0.052	5.398	1.243	0.656	0.103
21500.0	0.260	24.439	3.749	941.482	0.261	0.055	5.432	1.279	0.677	0.110
22000.0	0.258	24.077	3.811	941.161	0.268	0.058	5.464	1.315	0.697	0.116
22500.0	0.257	23.719	3.872	940.840	0.276	0.061	5.494	1.350	0.717	0.123
23000.0	0.255	23.365	3.932	940.517	0.284	0.064	5.523	1.385	0.737	0.129
23500.0	0.253	23.013	3.992	940.194	0.291	0.067	5.550	1.421	0.757	0.136
24000.0	0.251	22.664	4.051	939.869	0.299	0.070	5.576	1.456	0.777	0.144
24500.0	0.250	22.319	4.109	939.543	0.307	0.073	5.601	1.490	0.797	0.151
25000.0	0.248	21.978	4.166	939.216	0.315	0.077	5.624	1.525	0.816	0.159
27500.0	0.239	20.316	4.440	937.561	0.354	0.095	5.723	1.694	0.913	0.199
30000.0	0.231	18.736	4.693	935.872	0.393	0.116	5.796	1.857	1.007	0.244
32500.0	0.223	17.233	4.928	934.150	0.433	0.139	5.845	2.013	1.095	0.293
35000.0	0.215	15.809	5.143	932.394	0.473	0.164	5.874	2.163	1.178	0.347
37500.0	0.207	14.462	5.339	930.602	0.513	0.191	5.886	2.305	1.255	0.404
40000.0	0.199	13.191	5.517	928.773	0.553	0.220	5.883	2.440	1.325	0.466
42500.0	0.191	11.995	5.676	926.908	0.591	0.251	5.866	2.568	1.389	0.531
45000.0	0.184	10.873	5.817	925.005	0.629	0.284	5.839	2.687	1.447	0.599
47500.0	0.177	9.823	5.941	923.064	0.666	0.318	5.803	2.798	1.499	0.671
50000.0	0.170	8.845	6.048	921.085	0.702	0.354	5.761	2.901	1.544	0.745
52500.0	0.163	7.937	6.139	919.067	0.736	0.390	5.712	2.996	1.584	0.822
55000.0	0.157	7.098	6.213	917.011	0.768	0.426	5.660	3.082	1.618	0.902
57500.0	0.150	6.325	6.272	914.916	0.799	0.463	5.605	3.161	1.647	0.983
60000.0	0.144	5.616	6.316	912.784	0.827	0.500	5.549	3.231	1.671	1.067
62500.0	0.138	4.969	6.347	910.615	0.854	0.537	5.492	3.294	1.690	1.152
65000.0	0.133	4.381	6.364	908.410	0.879	0.573	5.436	3.349	1.706	1.238
67500.0	0.127	3.850	6.368	906.171	0.902	0.607	5.381	3.398	1.718	1.326
70000.0	0.122	3.373	6.361	903.898	0.922	0.641	5.327	3.440	1.726	1.415

Table C.10 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.352	43.867	0.022	955.611	0.000	0.000	0.018	0.000	0.000	0.000
500.0	0.350	43.390	0.111	955.411	0.002	0.000	0.180	0.001	0.000	0.000
1000.0	0.348	42.800	0.220	955.160	0.003	0.000	0.388	0.006	0.000	0.000
1500.0	0.346	42.218	0.327	954.910	0.005	0.000	0.586	0.014	0.001	0.000
2000.0	0.344	41.645	0.433	954.659	0.008	0.000	0.775	0.025	0.002	0.000
2500.0	0.342	41.078	0.537	954.410	0.010	0.000	0.954	0.038	0.004	0.000
3000.0	0.341	40.518	0.639	954.160	0.013	0.000	1.125	0.053	0.007	0.000
3500.0	0.339	39.964	0.740	953.912	0.015	0.000	1.287	0.070	0.010	0.000
4000.0	0.337	39.417	0.840	953.663	0.018	0.001	1.442	0.088	0.014	0.000
4500.0	0.335	38.876	0.938	953.415	0.021	0.001	1.589	0.108	0.019	0.001
5000.0	0.333	38.340	1.034	953.168	0.024	0.001	1.728	0.130	0.025	0.001
5500.0	0.332	37.810	1.130	952.921	0.028	0.001	1.861	0.152	0.032	0.001
6000.0	0.330	37.286	1.224	952.675	0.031	0.002	1.987	0.176	0.040	0.002
6500.0	0.328	36.766	1.317	952.429	0.035	0.002	2.107	0.200	0.048	0.002
7000.0	0.326	36.252	1.408	952.184	0.038	0.002	2.221	0.225	0.057	0.003
7500.0	0.324	35.742	1.499	951.940	0.042	0.003	2.329	0.251	0.067	0.004
8000.0	0.323	35.237	1.588	951.695	0.046	0.003	2.431	0.278	0.077	0.004
8500.0	0.321	34.737	1.676	951.451	0.050	0.003	2.528	0.305	0.088	0.005
9000.0	0.319	34.241	1.764	951.208	0.054	0.004	2.620	0.333	0.100	0.007
9500.0	0.318	33.750	1.850	950.964	0.058	0.004	2.708	0.361	0.112	0.008
10000.0	0.316	33.263	1.935	950.721	0.062	0.005	2.790	0.390	0.125	0.009
10500.0	0.314	32.779	2.019	950.479	0.067	0.005	2.869	0.419	0.138	0.011
11000.0	0.313	32.300	2.102	950.236	0.071	0.006	2.942	0.448	0.151	0.013
11500.0	0.311	31.825	2.184	949.994	0.076	0.007	3.012	0.478	0.165	0.014
12000.0	0.309	31.354	2.265	949.752	0.080	0.007	3.078	0.508	0.179	0.016
12500.0	0.308	30.886	2.345	949.511	0.085	0.008	3.139	0.538	0.193	0.019
13000.0	0.306	30.422	2.424	949.270	0.090	0.009	3.197	0.569	0.207	0.021
13500.0	0.304	29.962	2.502	949.029	0.094	0.010	3.251	0.600	0.222	0.024
14000.0	0.303	29.505	2.579	948.789	0.099	0.011	3.302	0.631	0.237	0.026
14500.0	0.301	29.051	2.656	948.549	0.104	0.011	3.349	0.662	0.252	0.029
15000.0	0.299	28.601	2.731	948.310	0.109	0.012	3.392	0.693	0.267	0.032
15500.0	0.298	28.153	2.806	948.072	0.114	0.013	3.433	0.724	0.282	0.035
16000.0	0.296	27.709	2.880	947.834	0.119	0.014	3.470	0.756	0.296	0.039
16500.0	0.295	27.268	2.953	947.596	0.124	0.015	3.505	0.787	0.311	0.042
17000.0	0.293	26.829	3.026	947.359	0.129	0.016	3.536	0.819	0.326	0.046
17500.0	0.291	26.394	3.098	947.122	0.134	0.018	3.566	0.850	0.341	0.050
18000.0	0.290	25.962	3.168	946.884	0.139	0.019	3.593	0.881	0.356	0.054
18500.0	0.288	25.532	3.239	946.646	0.144	0.020	3.618	0.913	0.371	0.059
19000.0	0.287	25.106	3.308	946.408	0.149	0.021	3.641	0.944	0.385	0.063
19500.0	0.285	24.682	3.377	946.168	0.155	0.023	3.662	0.975	0.400	0.068
20000.0	0.284	24.262	3.445	945.928	0.160	0.024	3.682	1.006	0.415	0.073
20500.0	0.282	23.845	3.512	945.686	0.165	0.025	3.700	1.037	0.430	0.078
22500.0	0.276	22.210	3.773	944.705	0.187	0.031	3.761	1.159	0.489	0.101
25000.0	0.268	20.239	4.083	943.445	0.216	0.040	3.811	1.307	0.562	0.134
27500.0	0.260	18.353	4.374	942.144	0.246	0.051	3.835	1.451	0.632	0.173
30000.0	0.252	16.551	4.646	940.801	0.277	0.063	3.837	1.589	0.697	0.218
32500.0	0.244	14.836	4.899	939.410	0.309	0.077	3.822	1.722	0.757	0.269
35000.0	0.236	13.210	5.132	937.969	0.341	0.092	3.791	1.847	0.811	0.326
37500.0	0.228	11.677	5.345	936.473	0.373	0.109	3.748	1.964	0.858	0.389
40000.0	0.220	10.238	5.537	934.918	0.406	0.127	3.695	2.074	0.899	0.458
42500.0	0.212	8.899	5.709	933.300	0.438	0.146	3.635	2.175	0.933	0.533
45000.0	0.203	7.661	5.860	931.612	0.470	0.167	3.571	2.266	0.961	0.614
47500.0	0.195	6.528	5.988	929.853	0.501	0.189	3.504	2.348	0.983	0.701
50000.0	0.187	5.502	6.095	928.017	0.531	0.211	3.437	2.421	0.999	0.793
52500.0	0.178	4.584	6.180	926.101	0.560	0.233	3.371	2.483	1.010	0.890
55000.0	0.170	3.775	6.243	924.105	0.587	0.255	3.308	2.536	1.017	0.991
57500.0	0.162	3.072	6.285	922.028	0.612	0.277	3.249	2.579	1.020	1.095
60000.0	0.154	2.471	6.307	919.871	0.635	0.298	3.196	2.613	1.020	1.203
62500.0	0.146	1.965	6.311	917.640	0.656	0.317	3.148	2.640	1.018	1.312
65000.0	0.138	1.548	6.298	915.339	0.674	0.335	3.106	2.659	1.014	1.423
67500.0	0.130	1.208	6.270	912.976	0.690	0.351	3.069	2.673	1.009	1.533
70000.0	0.123	0.936	6.229	910.560	0.704	0.366	3.038	2.682	1.004	1.643

Table C.11 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.351	43.867	0.023	955.604	0.000	0.000	0.020	0.000	0.000	0.000
500.0	0.350	43.391	0.115	955.373	0.002	0.000	0.206	0.002	0.000	0.000
1000.0	0.348	42.805	0.228	955.085	0.004	0.000	0.444	0.007	0.000	0.000
1500.0	0.345	42.228	0.340	954.797	0.006	0.000	0.670	0.016	0.001	0.000
2000.0	0.343	41.659	0.449	954.510	0.009	0.000	0.885	0.028	0.003	0.000
2500.0	0.341	41.098	0.556	954.223	0.012	0.000	1.090	0.042	0.005	0.000
3000.0	0.339	40.545	0.662	953.938	0.015	0.000	1.285	0.059	0.009	0.000
3500.0	0.337	39.999	0.765	953.652	0.018	0.001	1.470	0.078	0.013	0.000
4000.0	0.335	39.460	0.867	953.368	0.022	0.001	1.646	0.098	0.019	0.000
4500.0	0.333	38.928	0.968	953.084	0.025	0.001	1.814	0.120	0.025	0.001
5000.0	0.331	38.402	1.067	952.802	0.029	0.001	1.974	0.144	0.033	0.001
5500.0	0.329	37.883	1.164	952.520	0.033	0.002	2.125	0.168	0.041	0.002
6000.0	0.327	37.369	1.260	952.239	0.037	0.002	2.270	0.193	0.051	0.002
6500.0	0.325	36.861	1.355	951.958	0.042	0.002	2.407	0.220	0.061	0.003
7000.0	0.323	36.359	1.448	951.678	0.046	0.003	2.537	0.247	0.073	0.003
7500.0	0.321	35.862	1.539	951.399	0.050	0.003	2.661	0.275	0.085	0.004
8000.0	0.319	35.370	1.630	951.121	0.055	0.004	2.779	0.303	0.098	0.005
8500.0	0.317	34.883	1.719	950.843	0.060	0.004	2.891	0.332	0.111	0.007
9000.0	0.315	34.401	1.807	950.565	0.065	0.005	2.997	0.362	0.125	0.008
9500.0	0.313	33.924	1.893	950.288	0.070	0.006	3.098	0.392	0.140	0.009
10000.0	0.312	33.451	1.979	950.012	0.075	0.006	3.194	0.422	0.155	0.011
10500.0	0.310	32.983	2.063	949.736	0.080	0.007	3.284	0.453	0.171	0.013
11000.0	0.308	32.519	2.146	949.461	0.085	0.008	3.370	0.485	0.187	0.015
11500.0	0.306	32.060	2.228	949.186	0.091	0.009	3.451	0.516	0.203	0.017
12000.0	0.304	31.604	2.309	948.912	0.096	0.010	3.528	0.548	0.220	0.019
12500.0	0.302	31.153	2.389	948.638	0.102	0.011	3.601	0.580	0.237	0.022
13000.0	0.301	30.706	2.468	948.365	0.108	0.012	3.669	0.612	0.254	0.025
13500.0	0.299	30.262	2.546	948.092	0.113	0.013	3.733	0.645	0.272	0.028
14000.0	0.297	29.822	2.623	947.820	0.119	0.014	3.793	0.678	0.289	0.031
14500.0	0.295	29.386	2.699	947.549	0.125	0.015	3.849	0.711	0.307	0.034
15000.0	0.294	28.953	2.774	947.278	0.130	0.016	3.902	0.744	0.324	0.037
15500.0	0.292	28.524	2.848	947.008	0.136	0.018	3.951	0.777	0.342	0.041
16000.0	0.290	28.098	2.921	946.738	0.142	0.019	3.996	0.810	0.360	0.045
16500.0	0.288	27.675	2.994	946.470	0.148	0.020	4.038	0.843	0.377	0.049
17000.0	0.287	27.255	3.065	946.202	0.154	0.022	4.078	0.876	0.395	0.053
17500.0	0.285	26.839	3.136	945.934	0.160	0.023	4.114	0.910	0.412	0.058
18000.0	0.283	26.425	3.206	945.666	0.166	0.025	4.148	0.943	0.429	0.062
18500.0	0.282	26.014	3.275	945.398	0.172	0.026	4.179	0.976	0.447	0.067
19000.0	0.280	25.607	3.343	945.130	0.178	0.028	4.208	1.009	0.464	0.072
19500.0	0.278	25.203	3.410	944.862	0.184	0.030	4.236	1.042	0.481	0.077
20000.0	0.277	24.802	3.477	944.593	0.190	0.031	4.261	1.075	0.498	0.083
20500.0	0.275	24.404	3.543	944.322	0.196	0.033	4.284	1.107	0.515	0.088
21000.0	0.274	24.009	3.608	944.051	0.202	0.035	4.306	1.140	0.532	0.094
21500.0	0.272	23.617	3.672	943.779	0.209	0.037	4.327	1.172	0.549	0.100
22500.0	0.269	22.844	3.798	943.231	0.221	0.041	4.364	1.236	0.583	0.113
25000.0	0.260	20.968	4.100	941.836	0.254	0.053	4.436	1.392	0.667	0.148
27500.0	0.252	19.175	4.383	940.404	0.288	0.066	4.480	1.542	0.747	0.189
30000.0	0.244	17.465	4.646	938.934	0.323	0.081	4.500	1.687	0.823	0.235
32500.0	0.236	15.838	4.889	937.423	0.359	0.098	4.501	1.825	0.893	0.287
35000.0	0.228	14.295	5.114	935.870	0.394	0.116	4.484	1.956	0.957	0.344
37500.0	0.220	12.836	5.319	934.271	0.430	0.137	4.453	2.079	1.015	0.406
40000.0	0.212	11.463	5.505	932.625	0.466	0.159	4.410	2.194	1.065	0.473
42500.0	0.204	10.177	5.671	930.928	0.501	0.183	4.358	2.301	1.109	0.545
45000.0	0.197	8.978	5.819	929.179	0.535	0.207	4.299	2.399	1.145	0.622
47500.0	0.189	7.867	5.946	927.374	0.568	0.233	4.235	2.488	1.176	0.703
50000.0	0.181	6.845	6.055	925.512	0.601	0.260	4.168	2.568	1.200	0.789
52500.0	0.174	5.912	6.145	923.590	0.632	0.287	4.100	2.638	1.218	0.878
55000.0	0.166	5.068	6.215	921.609	0.661	0.314	4.032	2.700	1.231	0.970
57500.0	0.159	4.311	6.268	919.566	0.688	0.341	3.966	2.752	1.240	1.065
60000.0	0.152	3.638	6.302	917.464	0.714	0.367	3.902	2.796	1.244	1.162
62500.0	0.144	3.047	6.320	915.302	0.737	0.392	3.842	2.832	1.246	1.260
65000.0	0.138	2.534	6.322	913.083	0.759	0.416	3.787	2.861	1.244	1.360
67500.0	0.131	2.092	6.310	910.811	0.778	0.439	3.736	2.883	1.241	1.461
70000.0	0.124	1.716	6.285	908.490	0.794	0.459	3.690	2.899	1.236	1.561

Table C.12 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.351	43.868	0.024	955.593	0.000	0.000	0.024	0.000	0.000	0.000
500.0	0.349	43.395	0.122	955.320	0.002	0.000	0.241	0.002	0.000	0.000
1000.0	0.347	42.814	0.241	954.980	0.005	0.000	0.519	0.008	0.000	0.000
1500.0	0.344	42.243	0.357	954.641	0.008	0.000	0.783	0.018	0.002	0.000
2000.0	0.342	41.682	0.472	954.302	0.011	0.000	1.035	0.032	0.004	0.000
2500.0	0.339	41.130	0.584	953.965	0.015	0.000	1.275	0.048	0.007	0.000
3000.0	0.337	40.587	0.694	953.629	0.019	0.001	1.503	0.066	0.012	0.000
3500.0	0.334	40.051	0.802	953.293	0.023	0.001	1.721	0.087	0.018	0.000
4000.0	0.332	39.524	0.908	952.959	0.027	0.001	1.928	0.109	0.025	0.001
4500.0	0.330	39.004	1.012	952.625	0.032	0.002	2.126	0.133	0.034	0.001
5000.0	0.327	38.491	1.114	952.293	0.036	0.002	2.315	0.158	0.044	0.001
5500.0	0.325	37.985	1.215	951.961	0.041	0.002	2.495	0.184	0.055	0.002
6000.0	0.323	37.485	1.314	951.630	0.047	0.003	2.666	0.211	0.067	0.003
6500.0	0.321	36.992	1.411	951.301	0.052	0.004	2.830	0.239	0.080	0.003
7000.0	0.318	36.505	1.506	950.972	0.058	0.004	2.986	0.268	0.094	0.004
7500.0	0.316	36.023	1.600	950.644	0.063	0.005	3.136	0.298	0.109	0.005
8000.0	0.314	35.548	1.693	950.317	0.069	0.006	3.278	0.328	0.125	0.007
8500.0	0.312	35.078	1.784	949.990	0.075	0.006	3.413	0.359	0.141	0.008
9000.0	0.310	34.613	1.873	949.665	0.081	0.007	3.543	0.391	0.159	0.010
9500.0	0.307	34.153	1.962	949.340	0.088	0.008	3.666	0.423	0.177	0.011
10000.0	0.305	33.699	2.049	949.016	0.094	0.009	3.784	0.456	0.195	0.013
10500.0	0.303	33.249	2.134	948.692	0.100	0.011	3.896	0.489	0.214	0.015
11000.0	0.301	32.804	2.219	948.369	0.107	0.012	4.003	0.522	0.233	0.017
11500.0	0.299	32.364	2.302	948.047	0.114	0.013	4.104	0.556	0.253	0.020
12000.0	0.297	31.928	2.384	947.726	0.120	0.014	4.201	0.590	0.273	0.023
12500.0	0.295	31.497	2.465	947.405	0.127	0.016	4.293	0.624	0.293	0.025
13000.0	0.293	31.069	2.544	947.085	0.134	0.017	4.380	0.659	0.313	0.028
13500.0	0.291	30.647	2.623	946.765	0.141	0.019	4.462	0.694	0.334	0.032
14000.0	0.289	30.228	2.700	946.447	0.148	0.020	4.541	0.729	0.355	0.035
14500.0	0.287	29.813	2.776	946.128	0.155	0.022	4.615	0.764	0.376	0.038
15000.0	0.285	29.402	2.851	945.811	0.162	0.024	4.684	0.800	0.397	0.042
15500.0	0.283	28.995	2.925	945.495	0.169	0.026	4.750	0.835	0.417	0.046
16000.0	0.281	28.591	2.998	945.179	0.176	0.027	4.813	0.871	0.438	0.050
16500.0	0.279	28.191	3.070	944.864	0.184	0.029	4.871	0.907	0.459	0.055
17000.0	0.278	27.794	3.141	944.549	0.191	0.031	4.926	0.943	0.480	0.059
17500.0	0.276	27.401	3.212	944.235	0.198	0.034	4.978	0.979	0.500	0.064
18000.0	0.274	27.011	3.281	943.922	0.205	0.036	5.026	1.015	0.521	0.069
18500.0	0.272	26.625	3.349	943.609	0.213	0.038	5.072	1.051	0.541	0.074
19000.0	0.270	26.241	3.416	943.296	0.220	0.040	5.115	1.087	0.562	0.079
19500.0	0.268	25.861	3.483	942.984	0.227	0.043	5.155	1.123	0.582	0.085
20000.0	0.267	25.484	3.548	942.671	0.235	0.045	5.193	1.159	0.602	0.090
20500.0	0.265	25.110	3.613	942.357	0.242	0.048	5.228	1.195	0.622	0.096
21000.0	0.263	24.740	3.677	942.044	0.250	0.050	5.262	1.230	0.642	0.102
21500.0	0.261	24.373	3.740	941.730	0.257	0.053	5.294	1.266	0.662	0.108
22000.0	0.260	24.008	3.802	941.414	0.264	0.056	5.324	1.301	0.682	0.115
22500.0	0.258	23.648	3.863	941.098	0.272	0.059	5.353	1.336	0.701	0.121
23000.0	0.256	23.290	3.924	940.781	0.280	0.062	5.380	1.371	0.721	0.128
23500.0	0.254	22.936	3.984	940.463	0.287	0.065	5.405	1.406	0.740	0.135
24000.0	0.253	22.584	4.043	940.144	0.295	0.068	5.429	1.440	0.760	0.142
24500.0	0.251	22.236	4.101	939.823	0.302	0.071	5.452	1.474	0.779	0.149
25000.0	0.249	21.892	4.158	939.501	0.310	0.074	5.474	1.509	0.798	0.157
27500.0	0.241	20.217	4.433	937.872	0.349	0.092	5.565	1.675	0.894	0.197
30000.0	0.232	18.622	4.688	936.210	0.388	0.112	5.630	1.836	0.985	0.242
32500.0	0.224	17.106	4.923	934.513	0.428	0.135	5.673	1.990	1.071	0.292
35000.0	0.216	15.669	5.140	932.781	0.467	0.159	5.695	2.137	1.151	0.345
37500.0	0.208	14.309	5.337	931.013	0.507	0.186	5.700	2.277	1.226	0.403
40000.0	0.200	13.026	5.516	929.208	0.546	0.214	5.691	2.410	1.294	0.465
42500.0	0.193	11.819	5.676	927.365	0.584	0.245	5.668	2.535	1.356	0.531
45000.0	0.185	10.687	5.818	925.483	0.622	0.277	5.636	2.652	1.411	0.600
47500.0	0.178	9.629	5.943	923.561	0.658	0.310	5.595	2.760	1.460	0.672
50000.0	0.171	8.644	6.050	921.600	0.694	0.344	5.548	2.860	1.502	0.747
52500.0	0.164	7.730	6.141	919.599	0.727	0.380	5.495	2.952	1.539	0.825
55000.0	0.158	6.886	6.215	917.557	0.760	0.415	5.439	3.035	1.570	0.906
57500.0	0.151	6.111	6.274	915.475	0.790	0.451	5.381	3.111	1.596	0.988
60000.0	0.145	5.402	6.318	913.354	0.818	0.487	5.322	3.178	1.617	1.073
62500.0	0.139	4.756	6.348	911.194	0.845	0.522	5.262	3.237	1.634	1.159
65000.0	0.133	4.172	6.364	908.996	0.869	0.557	5.204	3.289	1.647	1.247
67500.0	0.128	3.645	6.367	906.761	0.892	0.590	5.147	3.334	1.656	1.336
70000.0	0.122	3.174	6.358	904.492	0.912	0.622	5.092	3.372	1.663	1.426

Table C.13 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.352	43.867	0.022	955.613	0.000	0.000	0.017	0.000	0.000	0.000
500.0	0.350	43.389	0.110	955.418	0.002	0.000	0.175	0.001	0.000	0.000
1000.0	0.348	42.799	0.218	955.174	0.003	0.000	0.378	0.006	0.000	0.000
1500.0	0.346	42.217	0.325	954.929	0.005	0.000	0.571	0.014	0.001	0.000
2000.0	0.344	41.642	0.430	954.685	0.007	0.000	0.756	0.024	0.002	0.000
2500.0	0.343	41.074	0.534	954.441	0.010	0.000	0.931	0.036	0.004	0.000
3000.0	0.341	40.513	0.636	954.196	0.012	0.000	1.099	0.051	0.006	0.000
3500.0	0.339	39.958	0.737	953.952	0.015	0.000	1.259	0.068	0.009	0.000
4000.0	0.337	39.409	0.836	953.707	0.018	0.001	1.412	0.086	0.014	0.000
4500.0	0.335	38.866	0.934	953.463	0.021	0.001	1.558	0.106	0.018	0.001
5000.0	0.334	38.329	1.031	953.218	0.024	0.001	1.697	0.126	0.024	0.001
5500.0	0.332	37.798	1.126	952.974	0.027	0.001	1.829	0.149	0.031	0.001
6000.0	0.330	37.272	1.220	952.729	0.030	0.001	1.956	0.172	0.038	0.002
6500.0	0.328	36.751	1.313	952.485	0.034	0.002	2.076	0.196	0.046	0.002
7000.0	0.327	36.236	1.405	952.241	0.038	0.002	2.190	0.221	0.055	0.003
7500.0	0.325	35.725	1.496	951.996	0.041	0.002	2.299	0.247	0.065	0.003
8000.0	0.323	35.219	1.585	951.752	0.045	0.003	2.402	0.273	0.075	0.004
8500.0	0.321	34.718	1.674	951.509	0.049	0.003	2.501	0.300	0.086	0.005
9000.0	0.320	34.221	1.761	951.265	0.053	0.004	2.594	0.328	0.098	0.006
9500.0	0.318	33.729	1.847	951.021	0.057	0.004	2.682	0.356	0.110	0.008
10000.0	0.316	33.241	1.932	950.778	0.062	0.005	2.766	0.384	0.122	0.009
10500.0	0.314	32.757	2.016	950.535	0.066	0.005	2.845	0.413	0.135	0.010
11000.0	0.313	32.277	2.099	950.292	0.070	0.006	2.920	0.442	0.148	0.012
11500.0	0.311	31.801	2.181	950.050	0.075	0.007	2.991	0.472	0.162	0.014
12000.0	0.309	31.329	2.263	949.808	0.080	0.007	3.058	0.502	0.176	0.016
12500.0	0.308	30.861	2.343	949.566	0.084	0.008	3.120	0.532	0.190	0.018
13000.0	0.306	30.397	2.422	949.324	0.089	0.009	3.179	0.563	0.205	0.021
13500.0	0.305	29.936	2.500	949.083	0.094	0.010	3.234	0.594	0.219	0.023
14000.0	0.303	29.478	2.578	948.843	0.099	0.010	3.285	0.625	0.234	0.026
14500.0	0.301	29.024	2.654	948.603	0.103	0.011	3.333	0.656	0.249	0.029
15000.0	0.300	28.573	2.730	948.363	0.108	0.012	3.378	0.687	0.264	0.032
15500.0	0.298	28.126	2.805	948.124	0.113	0.013	3.419	0.718	0.279	0.035
16000.0	0.296	27.681	2.879	947.886	0.118	0.014	3.457	0.750	0.294	0.038
16500.0	0.295	27.239	2.952	947.647	0.123	0.015	3.492	0.781	0.308	0.042
17000.0	0.293	26.801	3.025	947.409	0.128	0.016	3.525	0.812	0.323	0.046
17500.0	0.292	26.365	3.096	947.172	0.133	0.017	3.554	0.844	0.338	0.050
18000.0	0.290	25.933	3.167	946.934	0.139	0.019	3.582	0.875	0.353	0.054
18500.0	0.289	25.503	3.237	946.696	0.144	0.020	3.608	0.907	0.368	0.058
19000.0	0.287	25.076	3.307	946.457	0.149	0.021	3.631	0.938	0.383	0.062
19500.0	0.285	24.653	3.376	946.217	0.154	0.022	3.653	0.969	0.397	0.067
20000.0	0.284	24.233	3.444	945.976	0.159	0.024	3.673	1.000	0.412	0.072
20500.0	0.282	23.815	3.511	945.734	0.165	0.025	3.692	1.031	0.427	0.077
21000.0	0.281	23.401	3.577	945.491	0.170	0.027	3.709	1.062	0.442	0.083
22500.0	0.276	22.179	3.772	944.753	0.187	0.031	3.753	1.154	0.486	0.100
25000.0	0.268	20.208	4.082	943.492	0.215	0.040	3.804	1.303	0.559	0.133
27500.0	0.260	18.320	4.373	942.191	0.245	0.051	3.829	1.447	0.629	0.172
30000.0	0.252	16.518	4.645	940.846	0.276	0.063	3.833	1.585	0.695	0.217
32500.0	0.244	14.803	4.898	939.455	0.308	0.076	3.817	1.718	0.755	0.267
35000.0	0.236	13.178	5.131	938.012	0.340	0.092	3.787	1.843	0.809	0.324
37500.0	0.228	11.645	5.344	936.515	0.373	0.108	3.744	1.962	0.856	0.388
40000.0	0.220	10.208	5.536	934.959	0.406	0.127	3.692	2.071	0.897	0.457
42500.0	0.212	8.870	5.708	933.338	0.438	0.146	3.632	2.172	0.932	0.532
45000.0	0.203	7.634	5.858	931.649	0.470	0.167	3.568	2.264	0.960	0.613
47500.0	0.195	6.502	5.986	929.887	0.501	0.188	3.501	2.347	0.982	0.700
50000.0	0.187	5.478	6.092	928.049	0.531	0.211	3.434	2.419	0.998	0.792
52500.0	0.178	4.563	6.177	926.132	0.560	0.233	3.369	2.482	1.009	0.889
55000.0	0.170	3.756	6.239	924.133	0.587	0.255	3.306	2.534	1.016	0.990
57500.0	0.162	3.055	6.281	922.053	0.612	0.277	3.248	2.578	1.019	1.095
60000.0	0.154	2.456	6.303	919.895	0.635	0.297	3.194	2.612	1.019	1.202
62500.0	0.146	1.953	6.306	917.661	0.656	0.317	3.146	2.639	1.017	1.312
65000.0	0.138	1.537	6.292	915.359	0.674	0.335	3.105	2.659	1.013	1.422
67500.0	0.130	1.200	6.264	912.994	0.690	0.351	3.068	2.672	1.009	1.533
70000.0	0.123	0.929	6.223	910.575	0.704	0.366	3.037	2.681	1.004	1.643

Table C.14 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.351	43.867	0.023	955.605	0.000	0.000	0.020	0.000	0.000	0.000
500.0	0.350	43.391	0.114	955.381	0.002	0.000	0.200	0.002	0.000	0.000
1000.0	0.348	42.804	0.227	955.100	0.004	0.000	0.432	0.007	0.000	0.000
1500.0	0.346	42.226	0.337	954.819	0.006	0.000	0.653	0.015	0.001	0.000
2000.0	0.343	41.656	0.446	954.539	0.009	0.000	0.863	0.027	0.003	0.000
2500.0	0.341	41.094	0.553	954.259	0.012	0.000	1.064	0.041	0.005	0.000
3000.0	0.339	40.540	0.658	953.978	0.015	0.000	1.255	0.058	0.008	0.000
3500.0	0.337	39.992	0.762	953.698	0.018	0.001	1.438	0.076	0.012	0.000
4000.0	0.335	39.452	0.864	953.418	0.021	0.001	1.612	0.096	0.018	0.000
4500.0	0.333	38.918	0.964	953.139	0.025	0.001	1.778	0.117	0.024	0.001
5000.0	0.331	38.391	1.063	952.859	0.028	0.001	1.937	0.140	0.031	0.001
5500.0	0.329	37.870	1.160	952.580	0.032	0.002	2.088	0.164	0.040	0.001
6000.0	0.327	37.355	1.256	952.301	0.036	0.002	2.232	0.189	0.049	0.002
6500.0	0.325	36.845	1.350	952.022	0.041	0.002	2.370	0.215	0.059	0.003
7000.0	0.323	36.341	1.444	951.743	0.045	0.003	2.501	0.242	0.070	0.003
7500.0	0.321	35.843	1.536	951.465	0.049	0.003	2.626	0.270	0.082	0.004
8000.0	0.320	35.350	1.626	951.187	0.054	0.004	2.744	0.298	0.095	0.005
8500.0	0.318	34.862	1.715	950.910	0.059	0.004	2.857	0.327	0.108	0.006
9000.0	0.316	34.379	1.803	950.632	0.064	0.005	2.965	0.356	0.122	0.008
9500.0	0.314	33.901	1.890	950.355	0.069	0.006	3.067	0.386	0.137	0.009
10000.0	0.312	33.427	1.976	950.079	0.074	0.006	3.164	0.416	0.152	0.011
10500.0	0.310	32.958	2.060	949.803	0.079	0.007	3.255	0.447	0.168	0.013
11000.0	0.308	32.493	2.143	949.527	0.085	0.008	3.342	0.478	0.184	0.015
11500.0	0.306	32.033	2.226	949.252	0.090	0.009	3.425	0.510	0.200	0.017
12000.0	0.305	31.577	2.307	948.977	0.095	0.010	3.503	0.542	0.217	0.019
12500.0	0.303	31.125	2.387	948.703	0.101	0.011	3.576	0.574	0.234	0.021
13000.0	0.301	30.677	2.466	948.429	0.107	0.012	3.645	0.606	0.251	0.024
13500.0	0.299	30.233	2.544	948.157	0.112	0.013	3.710	0.638	0.268	0.027
14000.0	0.297	29.792	2.621	947.884	0.118	0.014	3.771	0.671	0.286	0.030
14500.0	0.296	29.355	2.697	947.612	0.124	0.015	3.828	0.704	0.303	0.033
15000.0	0.294	28.922	2.772	947.341	0.130	0.016	3.882	0.737	0.321	0.037
15500.0	0.292	28.492	2.846	947.071	0.135	0.017	3.932	0.770	0.338	0.040
16000.0	0.291	28.066	2.920	946.801	0.141	0.019	3.978	0.803	0.356	0.044
16500.0	0.289	27.642	2.992	946.532	0.147	0.020	4.021	0.836	0.374	0.048
17000.0	0.287	27.222	3.064	946.263	0.153	0.022	4.061	0.870	0.391	0.052
17500.0	0.285	26.805	3.134	945.995	0.159	0.023	4.098	0.903	0.408	0.057
18000.0	0.284	26.391	3.204	945.727	0.165	0.025	4.133	0.936	0.426	0.061
18500.0	0.282	25.981	3.273	945.459	0.171	0.026	4.165	0.969	0.443	0.066
19000.0	0.280	25.573	3.342	945.190	0.177	0.028	4.195	1.002	0.460	0.071
19500.0	0.279	25.168	3.409	944.921	0.183	0.029	4.222	1.035	0.477	0.076
20000.0	0.277	24.767	3.476	944.652	0.189	0.031	4.248	1.068	0.494	0.082
20500.0	0.275	24.369	3.542	944.382	0.195	0.033	4.272	1.101	0.512	0.087
21000.0	0.274	23.974	3.607	944.110	0.202	0.035	4.294	1.133	0.529	0.093
21500.0	0.272	23.582	3.671	943.838	0.208	0.037	4.315	1.166	0.546	0.099
22500.0	0.269	22.808	3.797	943.289	0.221	0.041	4.353	1.230	0.579	0.112
25000.0	0.261	20.932	4.099	941.893	0.254	0.052	4.426	1.386	0.663	0.147
27500.0	0.253	19.138	4.382	940.461	0.288	0.065	4.471	1.537	0.744	0.187
30000.0	0.244	17.427	4.645	938.989	0.323	0.081	4.493	1.682	0.820	0.234
32500.0	0.236	15.800	4.888	937.478	0.358	0.097	4.494	1.820	0.890	0.285
35000.0	0.228	14.257	5.112	935.923	0.394	0.116	4.478	1.952	0.955	0.342
37500.0	0.220	12.799	5.317	934.324	0.430	0.137	4.447	2.075	1.012	0.404
40000.0	0.212	11.427	5.503	932.676	0.465	0.159	4.405	2.191	1.063	0.472
42500.0	0.204	10.142	5.669	930.977	0.500	0.182	4.353	2.298	1.106	0.544
45000.0	0.197	8.944	5.816	929.226	0.535	0.207	4.294	2.396	1.143	0.621
47500.0	0.189	7.835	5.944	927.419	0.568	0.233	4.231	2.486	1.174	0.702
50000.0	0.181	6.815	6.052	925.555	0.600	0.260	4.164	2.566	1.198	0.787
52500.0	0.174	5.884	6.141	923.632	0.631	0.287	4.096	2.636	1.216	0.876
55000.0	0.166	5.042	6.211	921.648	0.661	0.314	4.028	2.698	1.230	0.969
57500.0	0.159	4.287	6.263	919.603	0.688	0.341	3.962	2.750	1.238	1.064
60000.0	0.152	3.617	6.297	917.498	0.714	0.367	3.899	2.795	1.243	1.161
62500.0	0.144	3.028	6.315	915.334	0.737	0.392	3.839	2.831	1.244	1.260
65000.0	0.138	2.516	6.317	913.113	0.758	0.416	3.784	2.859	1.243	1.360
67500.0	0.131	2.077	6.304	910.839	0.777	0.438	3.733	2.881	1.240	1.460
70000.0	0.124	1.703	6.278	908.515	0.794	0.459	3.688	2.897	1.235	1.561

Table C.15 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.351	43.868	0.024	955.595	0.000	0.000	0.023	0.000	0.000	0.000
500.0	0.349	43.395	0.121	955.329	0.002	0.000	0.234	0.002	0.000	0.000
1000.0	0.347	42.813	0.239	954.997	0.005	0.000	0.506	0.008	0.000	0.000
1500.0	0.344	42.242	0.355	954.666	0.008	0.000	0.764	0.018	0.002	0.000
2000.0	0.342	41.680	0.468	954.335	0.011	0.000	1.010	0.031	0.004	0.000
2500.0	0.340	41.126	0.580	954.004	0.014	0.000	1.245	0.047	0.007	0.000
3000.0	0.337	40.581	0.690	953.674	0.018	0.001	1.470	0.065	0.011	0.000
3500.0	0.335	40.045	0.797	953.344	0.022	0.001	1.684	0.085	0.017	0.000
4000.0	0.332	39.516	0.903	953.015	0.026	0.001	1.889	0.107	0.024	0.001
4500.0	0.330	38.994	1.007	952.686	0.031	0.001	2.085	0.130	0.032	0.001
5000.0	0.328	38.479	1.109	952.357	0.036	0.002	2.272	0.155	0.042	0.001
5500.0	0.326	37.972	1.210	952.029	0.041	0.002	2.451	0.180	0.053	0.002
6000.0	0.323	37.470	1.308	951.701	0.046	0.003	2.622	0.207	0.064	0.002
6500.0	0.321	36.976	1.406	951.374	0.051	0.003	2.786	0.235	0.077	0.003
7000.0	0.319	36.487	1.501	951.047	0.057	0.004	2.942	0.264	0.091	0.004
7500.0	0.317	36.004	1.595	950.720	0.062	0.005	3.091	0.293	0.106	0.005
8000.0	0.314	35.527	1.688	950.394	0.068	0.005	3.234	0.323	0.122	0.006
8500.0	0.312	35.055	1.779	950.069	0.074	0.006	3.371	0.354	0.138	0.008
9000.0	0.310	34.589	1.869	949.744	0.080	0.007	3.501	0.385	0.155	0.009
9500.0	0.308	34.128	1.957	949.419	0.087	0.008	3.626	0.417	0.173	0.011
10000.0	0.306	33.673	2.045	949.095	0.093	0.009	3.744	0.450	0.191	0.013
10500.0	0.304	33.222	2.130	948.772	0.099	0.010	3.857	0.482	0.210	0.015
11000.0	0.302	32.776	2.215	948.449	0.106	0.011	3.965	0.516	0.229	0.017
11500.0	0.299	32.335	2.298	948.127	0.112	0.013	4.068	0.549	0.249	0.019
12000.0	0.297	31.898	2.380	947.806	0.119	0.014	4.166	0.583	0.268	0.022
12500.0	0.295	31.466	2.461	947.485	0.126	0.015	4.259	0.617	0.289	0.025
13000.0	0.293	31.038	2.541	947.164	0.133	0.017	4.347	0.652	0.309	0.028
13500.0	0.291	30.614	2.620	946.844	0.140	0.018	4.430	0.687	0.330	0.031
14000.0	0.289	30.195	2.697	946.525	0.147	0.020	4.510	0.722	0.350	0.034
14500.0	0.287	29.779	2.773	946.207	0.154	0.022	4.585	0.757	0.371	0.038
15000.0	0.286	29.368	2.849	945.889	0.161	0.023	4.656	0.793	0.392	0.042
15500.0	0.284	28.960	2.923	945.572	0.168	0.025	4.723	0.828	0.413	0.045
16000.0	0.282	28.556	2.996	945.256	0.175	0.027	4.786	0.864	0.434	0.050
16500.0	0.280	28.155	3.068	944.941	0.183	0.029	4.845	0.900	0.455	0.054
17000.0	0.278	27.758	3.139	944.626	0.190	0.031	4.901	0.936	0.475	0.058
17500.0	0.276	27.364	3.209	944.312	0.197	0.033	4.953	0.972	0.496	0.063
18000.0	0.274	26.974	3.278	943.998	0.204	0.035	5.002	1.008	0.516	0.068
18500.0	0.272	26.587	3.347	943.685	0.212	0.038	5.049	1.044	0.537	0.073
19000.0	0.271	26.203	3.414	943.372	0.219	0.040	5.092	1.080	0.557	0.078
19500.0	0.269	25.822	3.481	943.059	0.226	0.042	5.133	1.116	0.577	0.084
20000.0	0.267	25.445	3.546	942.746	0.234	0.045	5.172	1.152	0.598	0.089
20500.0	0.265	25.071	3.611	942.432	0.241	0.047	5.208	1.187	0.618	0.095
21000.0	0.263	24.700	3.675	942.118	0.249	0.050	5.242	1.223	0.638	0.101
21500.0	0.262	24.332	3.738	941.804	0.256	0.053	5.274	1.258	0.657	0.107
22000.0	0.260	23.968	3.800	941.488	0.264	0.055	5.305	1.294	0.677	0.114
22500.0	0.258	23.607	3.862	941.172	0.271	0.058	5.334	1.329	0.697	0.120
23000.0	0.256	23.249	3.922	940.855	0.279	0.061	5.361	1.364	0.716	0.127
23500.0	0.255	22.894	3.982	940.536	0.286	0.064	5.387	1.399	0.736	0.134
24000.0	0.253	22.543	4.041	940.217	0.294	0.067	5.412	1.433	0.755	0.141
24500.0	0.251	22.195	4.099	939.896	0.302	0.071	5.435	1.468	0.775	0.148
25000.0	0.249	21.850	4.157	939.574	0.309	0.074	5.457	1.502	0.794	0.156
27500.0	0.241	20.174	4.431	937.944	0.348	0.092	5.550	1.669	0.889	0.196
30000.0	0.233	18.578	4.687	936.280	0.387	0.112	5.617	1.829	0.981	0.241
32500.0	0.224	17.063	4.922	934.582	0.427	0.134	5.660	1.984	1.067	0.290
35000.0	0.216	15.625	5.138	932.849	0.467	0.159	5.684	2.132	1.148	0.344
37500.0	0.208	14.266	5.335	931.080	0.506	0.185	5.689	2.272	1.222	0.402
40000.0	0.200	12.983	5.514	929.273	0.545	0.214	5.680	2.405	1.290	0.464
42500.0	0.193	11.777	5.674	927.428	0.584	0.244	5.659	2.530	1.352	0.529
45000.0	0.185	10.646	5.816	925.545	0.621	0.276	5.627	2.648	1.407	0.598
47500.0	0.178	9.589	5.940	923.622	0.658	0.309	5.586	2.756	1.456	0.670
50000.0	0.171	8.605	6.047	921.658	0.693	0.344	5.539	2.857	1.499	0.746
52500.0	0.164	7.693	6.137	919.655	0.727	0.379	5.487	2.949	1.536	0.824
55000.0	0.158	6.851	6.211	917.611	0.759	0.415	5.431	3.033	1.567	0.904
57500.0	0.151	6.078	6.270	915.528	0.790	0.451	5.373	3.108	1.593	0.987
60000.0	0.145	5.371	6.313	913.404	0.818	0.486	5.314	3.175	1.615	1.072
62500.0	0.139	4.727	6.342	911.242	0.845	0.522	5.255	3.235	1.632	1.158
65000.0	0.133	4.145	6.358	909.042	0.869	0.556	5.197	3.287	1.645	1.246
67500.0	0.128	3.621	6.360	906.805	0.892	0.590	5.141	3.332	1.654	1.335
70000.0	0.122	3.152	6.351	904.533	0.912	0.622	5.086	3.370	1.660	1.425

Table C.16 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.352	43.867	0.022	955.613	0.000	0.000	0.017	0.000	0.000	0.000
500.0	0.350	43.389	0.110	955.423	0.001	0.000	0.171	0.001	0.000	0.000
1000.0	0.348	42.798	0.218	955.186	0.003	0.000	0.368	0.006	0.000	0.000
1500.0	0.346	42.215	0.324	954.950	0.005	0.000	0.555	0.013	0.001	0.000
2000.0	0.345	41.639	0.429	954.714	0.007	0.000	0.733	0.024	0.002	0.000
2500.0	0.343	41.069	0.532	954.480	0.009	0.000	0.901	0.036	0.004	0.000
3000.0	0.341	40.506	0.634	954.248	0.012	0.000	1.060	0.050	0.006	0.000
3500.0	0.339	39.949	0.734	954.016	0.014	0.000	1.211	0.067	0.009	0.000
4000.0	0.338	39.398	0.833	953.786	0.017	0.001	1.354	0.084	0.013	0.000
4500.0	0.336	38.852	0.930	953.557	0.019	0.001	1.489	0.103	0.017	0.000
5000.0	0.334	38.311	1.026	953.330	0.022	0.001	1.616	0.124	0.022	0.001
5500.0	0.333	37.775	1.121	953.104	0.025	0.001	1.737	0.145	0.028	0.001
6000.0	0.331	37.244	1.214	952.880	0.028	0.001	1.851	0.168	0.035	0.001
6500.0	0.329	36.717	1.307	952.657	0.031	0.002	1.958	0.191	0.042	0.002
7000.0	0.328	36.194	1.398	952.436	0.034	0.002	2.060	0.215	0.050	0.002
7500.0	0.326	35.676	1.488	952.216	0.038	0.002	2.156	0.239	0.058	0.003
8000.0	0.324	35.161	1.577	951.997	0.041	0.002	2.247	0.264	0.067	0.004
8500.0	0.323	34.651	1.665	951.778	0.044	0.003	2.333	0.290	0.076	0.004
9000.0	0.321	34.144	1.752	951.559	0.048	0.003	2.415	0.316	0.086	0.005
9500.0	0.320	33.641	1.838	951.341	0.051	0.004	2.493	0.342	0.096	0.006
10000.0	0.318	33.142	1.923	951.122	0.055	0.004	2.567	0.369	0.107	0.008
10500.0	0.317	32.647	2.007	950.902	0.059	0.004	2.638	0.397	0.118	0.009
11000.0	0.315	32.156	2.091	950.682	0.063	0.005	2.705	0.424	0.129	0.010
11500.0	0.313	31.668	2.173	950.460	0.067	0.006	2.770	0.452	0.141	0.012
12000.0	0.312	31.184	2.255	950.238	0.071	0.006	2.832	0.480	0.154	0.013
12500.0	0.310	30.704	2.336	950.015	0.075	0.007	2.891	0.509	0.166	0.015
15000.0	0.302	28.360	2.728	948.880	0.098	0.010	3.149	0.654	0.234	0.027
17500.0	0.294	26.106	3.099	947.715	0.122	0.015	3.351	0.803	0.308	0.043
20000.0	0.286	23.942	3.450	946.517	0.149	0.021	3.504	0.953	0.384	0.064
22500.0	0.278	21.867	3.782	945.283	0.177	0.028	3.616	1.103	0.459	0.090
25000.0	0.270	19.880	4.093	944.012	0.207	0.037	3.692	1.251	0.533	0.122
27500.0	0.262	17.982	4.385	942.699	0.237	0.047	3.737	1.396	0.604	0.160
30000.0	0.254	16.174	4.658	941.342	0.269	0.059	3.757	1.537	0.670	0.204
32500.0	0.246	14.457	4.910	939.937	0.301	0.073	3.754	1.672	0.732	0.255
35000.0	0.238	12.834	5.141	938.480	0.334	0.088	3.734	1.802	0.787	0.311
37500.0	0.229	11.307	5.352	936.966	0.368	0.105	3.699	1.924	0.836	0.374
40000.0	0.221	9.879	5.542	935.391	0.401	0.123	3.652	2.037	0.878	0.443
42500.0	0.213	8.554	5.710	933.751	0.434	0.143	3.597	2.143	0.913	0.518
45000.0	0.204	7.334	5.857	932.040	0.467	0.164	3.537	2.238	0.943	0.600
47500.0	0.196	6.222	5.981	930.255	0.498	0.186	3.473	2.324	0.966	0.687
50000.0	0.187	5.219	6.083	928.391	0.528	0.208	3.408	2.400	0.983	0.780
52500.0	0.179	4.328	6.162	926.447	0.557	0.231	3.345	2.465	0.996	0.877
55000.0	0.170	3.546	6.220	924.422	0.585	0.253	3.285	2.520	1.004	0.979
57500.0	0.162	2.872	6.257	922.315	0.610	0.275	3.229	2.566	1.008	1.085
60000.0	0.153	2.299	6.273	920.129	0.633	0.296	3.178	2.602	1.009	1.193
62500.0	0.145	1.820	6.272	917.870	0.653	0.315	3.132	2.630	1.008	1.304
65000.0	0.137	1.428	6.254	915.543	0.672	0.333	3.093	2.651	1.006	1.415
67500.0	0.130	1.110	6.222	913.157	0.687	0.350	3.059	2.666	1.002	1.526
70000.0	0.122	0.858	6.178	910.720	0.701	0.364	3.030	2.676	0.998	1.636

Table C.17 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.351	43.867	0.023	955.607	0.000	0.000	0.020	0.000	0.000	0.000
500.0	0.350	43.391	0.114	955.389	0.002	0.000	0.195	0.001	0.000	0.000
1000.0	0.348	42.803	0.226	955.117	0.004	0.000	0.419	0.007	0.000	0.000
1500.0	0.346	42.224	0.336	954.847	0.006	0.000	0.631	0.015	0.001	0.000
2000.0	0.344	41.652	0.444	954.578	0.008	0.000	0.832	0.027	0.002	0.000
2500.0	0.342	41.089	0.550	954.311	0.011	0.000	1.023	0.041	0.005	0.000
3000.0	0.340	40.532	0.655	954.045	0.014	0.000	1.203	0.057	0.008	0.000
3500.0	0.338	39.983	0.757	953.781	0.017	0.001	1.374	0.075	0.012	0.000
4000.0	0.336	39.439	0.858	953.518	0.020	0.001	1.536	0.094	0.017	0.000
4500.0	0.334	38.902	0.958	953.257	0.023	0.001	1.689	0.116	0.022	0.001
5000.0	0.332	38.371	1.056	952.998	0.027	0.001	1.833	0.138	0.029	0.001
5500.0	0.330	37.845	1.152	952.741	0.030	0.001	1.970	0.161	0.037	0.001
6000.0	0.328	37.324	1.247	952.485	0.034	0.002	2.100	0.185	0.045	0.002
6500.0	0.327	36.809	1.341	952.231	0.037	0.002	2.222	0.211	0.054	0.002
7000.0	0.325	36.298	1.433	951.979	0.041	0.002	2.337	0.236	0.064	0.003
7500.0	0.323	35.791	1.524	951.728	0.045	0.003	2.447	0.263	0.074	0.004
8000.0	0.321	35.289	1.614	951.478	0.049	0.003	2.550	0.290	0.085	0.005
8500.0	0.319	34.792	1.703	951.228	0.053	0.004	2.649	0.318	0.096	0.006
9000.0	0.318	34.298	1.790	950.980	0.058	0.004	2.742	0.346	0.108	0.007
9500.0	0.316	33.809	1.877	950.731	0.062	0.005	2.831	0.374	0.121	0.008
10000.0	0.314	33.324	1.962	950.483	0.066	0.005	2.916	0.403	0.134	0.009
10500.0	0.312	32.843	2.047	950.234	0.071	0.006	2.997	0.432	0.147	0.011
11000.0	0.311	32.366	2.130	949.984	0.076	0.007	3.075	0.461	0.161	0.012
11500.0	0.309	31.893	2.213	949.733	0.080	0.007	3.149	0.491	0.175	0.014
12000.0	0.307	31.424	2.294	949.482	0.085	0.008	3.220	0.521	0.190	0.016
12500.0	0.306	30.959	2.375	949.229	0.090	0.009	3.289	0.551	0.205	0.018
13000.0	0.304	30.498	2.455	948.976	0.096	0.010	3.354	0.581	0.221	0.021
15000.0	0.297	28.694	2.764	947.951	0.117	0.014	3.589	0.705	0.285	0.032
17500.0	0.289	26.525	3.132	946.642	0.146	0.020	3.828	0.861	0.371	0.050
20000.0	0.280	24.448	3.478	945.301	0.177	0.027	4.016	1.018	0.458	0.073
22500.0	0.272	22.463	3.803	943.928	0.209	0.037	4.159	1.174	0.545	0.102
25000.0	0.263	20.567	4.108	942.519	0.243	0.048	4.263	1.328	0.631	0.136
27500.0	0.255	18.759	4.391	941.074	0.278	0.061	4.334	1.479	0.712	0.175
30000.0	0.247	17.040	4.655	939.589	0.314	0.076	4.377	1.625	0.789	0.221
32500.0	0.238	15.409	4.898	938.062	0.350	0.092	4.395	1.766	0.860	0.272
35000.0	0.230	13.866	5.121	936.491	0.387	0.111	4.392	1.900	0.925	0.328
37500.0	0.222	12.413	5.324	934.874	0.423	0.132	4.373	2.028	0.983	0.390
40000.0	0.214	11.048	5.507	933.207	0.459	0.154	4.339	2.147	1.035	0.457
42500.0	0.206	9.773	5.670	931.489	0.495	0.178	4.294	2.258	1.080	0.529
45000.0	0.198	8.590	5.814	929.715	0.530	0.203	4.241	2.360	1.118	0.607
47500.0	0.190	7.498	5.937	927.886	0.564	0.229	4.183	2.453	1.150	0.688
50000.0	0.182	6.497	6.041	925.997	0.596	0.256	4.120	2.537	1.175	0.774
52500.0	0.174	5.588	6.125	924.048	0.627	0.283	4.055	2.611	1.195	0.863
55000.0	0.166	4.769	6.191	922.038	0.657	0.310	3.991	2.675	1.210	0.956
57500.0	0.159	4.039	6.238	919.966	0.684	0.337	3.928	2.730	1.220	1.052
60000.0	0.152	3.393	6.267	917.834	0.710	0.364	3.868	2.777	1.226	1.150
62500.0	0.144	2.830	6.279	915.643	0.733	0.389	3.811	2.815	1.229	1.249
65000.0	0.137	2.343	6.276	913.396	0.755	0.413	3.759	2.845	1.229	1.350
67500.0	0.131	1.926	6.259	911.096	0.773	0.435	3.711	2.869	1.227	1.451
70000.0	0.124	1.574	6.229	908.749	0.790	0.456	3.668	2.886	1.223	1.552

Table C.18 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.352	43.987	0.000	955.661	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.351	43.868	0.024	955.597	0.000	0.000	0.023	0.000	0.000	0.000
500.0	0.349	43.394	0.120	955.340	0.002	0.000	0.227	0.002	0.000	0.000
1000.0	0.347	42.812	0.237	955.021	0.005	0.000	0.487	0.008	0.000	0.000
1500.0	0.345	42.239	0.352	954.704	0.007	0.000	0.734	0.018	0.002	0.000
2000.0	0.342	41.676	0.465	954.388	0.010	0.000	0.968	0.031	0.004	0.000
2500.0	0.340	41.121	0.576	954.074	0.014	0.000	1.189	0.046	0.007	0.000
3000.0	0.338	40.574	0.684	953.762	0.017	0.001	1.399	0.064	0.011	0.000
3500.0	0.335	40.035	0.791	953.451	0.021	0.001	1.598	0.084	0.016	0.000
4000.0	0.333	39.503	0.895	953.143	0.025	0.001	1.787	0.106	0.023	0.001
4500.0	0.331	38.978	0.998	952.836	0.029	0.001	1.967	0.129	0.030	0.001
5000.0	0.329	38.459	1.099	952.531	0.033	0.002	2.137	0.153	0.039	0.001
5500.0	0.327	37.947	1.198	952.228	0.038	0.002	2.298	0.179	0.049	0.002
6000.0	0.324	37.441	1.296	951.926	0.042	0.003	2.451	0.205	0.060	0.002
6500.0	0.322	36.940	1.392	951.627	0.047	0.003	2.596	0.232	0.071	0.003
7000.0	0.320	36.445	1.486	951.329	0.052	0.004	2.734	0.260	0.084	0.004
7500.0	0.318	35.954	1.579	951.033	0.057	0.004	2.865	0.289	0.097	0.005
8000.0	0.316	35.469	1.671	950.738	0.062	0.005	2.989	0.318	0.110	0.006
8500.0	0.314	34.988	1.761	950.444	0.068	0.006	3.108	0.348	0.125	0.007
9000.0	0.312	34.512	1.850	950.151	0.073	0.006	3.221	0.378	0.140	0.008
9500.0	0.310	34.041	1.938	949.859	0.079	0.007	3.329	0.409	0.155	0.010
10000.0	0.308	33.574	2.025	949.566	0.084	0.008	3.432	0.440	0.171	0.011
10500.0	0.306	33.112	2.110	949.274	0.090	0.009	3.531	0.471	0.187	0.013
11000.0	0.304	32.653	2.194	948.981	0.096	0.010	3.625	0.503	0.204	0.015
11500.0	0.303	32.200	2.278	948.688	0.102	0.011	3.716	0.535	0.221	0.017
12000.0	0.301	31.750	2.360	948.394	0.108	0.012	3.804	0.567	0.239	0.020
12500.0	0.299	31.305	2.441	948.099	0.114	0.013	3.888	0.599	0.257	0.022
13000.0	0.297	30.864	2.521	947.803	0.120	0.014	3.969	0.632	0.275	0.025
13500.0	0.295	30.427	2.600	947.507	0.126	0.016	4.046	0.665	0.293	0.027
14000.0	0.293	29.994	2.678	947.209	0.133	0.017	4.121	0.698	0.312	0.030
14500.0	0.291	29.566	2.755	946.910	0.139	0.018	4.193	0.731	0.331	0.034
15000.0	0.289	29.141	2.831	946.610	0.146	0.020	4.263	0.765	0.351	0.037
15500.0	0.287	28.721	2.907	946.309	0.153	0.022	4.330	0.798	0.370	0.041
17500.0	0.280	27.078	3.197	945.093	0.181	0.029	4.573	0.933	0.450	0.057
20000.0	0.271	25.112	3.540	943.544	0.218	0.039	4.827	1.102	0.552	0.081
22500.0	0.262	23.239	3.860	941.965	0.256	0.052	5.032	1.271	0.653	0.111
25000.0	0.253	21.457	4.158	940.353	0.295	0.068	5.196	1.438	0.752	0.145
27500.0	0.244	19.763	4.434	938.707	0.335	0.085	5.323	1.602	0.848	0.185
30000.0	0.235	18.155	4.690	937.028	0.375	0.105	5.418	1.762	0.940	0.229
32500.0	0.227	16.632	4.926	935.314	0.416	0.127	5.485	1.917	1.027	0.278
35000.0	0.218	15.191	5.141	933.563	0.456	0.151	5.528	2.067	1.108	0.331
37500.0	0.210	13.832	5.337	931.776	0.497	0.178	5.551	2.210	1.183	0.388
40000.0	0.202	12.554	5.513	929.949	0.537	0.206	5.556	2.346	1.251	0.450
42500.0	0.194	11.355	5.671	928.084	0.576	0.237	5.546	2.474	1.314	0.516
45000.0	0.187	10.234	5.809	926.180	0.614	0.269	5.524	2.595	1.370	0.585
47500.0	0.179	9.190	5.930	924.234	0.651	0.302	5.492	2.707	1.420	0.657
50000.0	0.172	8.222	6.032	922.248	0.687	0.337	5.452	2.811	1.464	0.732
52500.0	0.165	7.327	6.118	920.221	0.721	0.373	5.406	2.906	1.502	0.811
55000.0	0.158	6.504	6.188	918.153	0.754	0.408	5.356	2.993	1.534	0.892
57500.0	0.152	5.751	6.241	916.044	0.784	0.445	5.303	3.071	1.562	0.975
60000.0	0.145	5.065	6.280	913.895	0.813	0.480	5.249	3.141	1.584	1.060
62500.0	0.139	4.443	6.304	911.707	0.839	0.516	5.194	3.203	1.602	1.147
65000.0	0.133	3.883	6.314	909.482	0.864	0.551	5.140	3.258	1.617	1.235
67500.0	0.128	3.381	6.312	907.220	0.886	0.584	5.088	3.305	1.627	1.324
70000.0	0.122	2.934	6.298	904.924	0.906	0.616	5.038	3.345	1.635	1.415

Table C.19 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.360	44.948	0.000	954.692	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.359	44.828	0.023	954.640	0.000	0.000	0.021	0.000	0.000	0.000
500.0	0.358	44.351	0.113	954.430	0.002	0.000	0.193	0.001	0.000	0.000
1000.0	0.356	43.762	0.224	954.168	0.003	0.000	0.411	0.006	0.000	0.000
1500.0	0.354	43.181	0.333	953.905	0.006	0.000	0.618	0.015	0.001	0.000
2000.0	0.352	42.607	0.440	953.642	0.008	0.000	0.817	0.025	0.002	0.000
2500.0	0.350	42.041	0.546	953.379	0.010	0.000	1.006	0.039	0.004	0.000
3000.0	0.348	41.483	0.650	953.116	0.013	0.000	1.187	0.054	0.007	0.000
3500.0	0.346	40.930	0.753	952.853	0.016	0.000	1.360	0.072	0.011	0.000
4000.0	0.344	40.385	0.854	952.589	0.019	0.001	1.525	0.091	0.015	0.000
4500.0	0.342	39.846	0.954	952.326	0.023	0.001	1.683	0.111	0.021	0.001
5000.0	0.340	39.312	1.053	952.062	0.026	0.001	1.833	0.133	0.028	0.001
5500.0	0.338	38.785	1.150	951.799	0.030	0.001	1.977	0.156	0.035	0.001
6000.0	0.336	38.263	1.245	951.535	0.033	0.002	2.115	0.180	0.043	0.002
6500.0	0.334	37.747	1.340	951.271	0.037	0.002	2.247	0.205	0.053	0.002
7000.0	0.332	37.236	1.433	951.008	0.041	0.002	2.372	0.231	0.063	0.003
7500.0	0.330	36.731	1.525	950.744	0.045	0.003	2.492	0.257	0.073	0.004
8000.0	0.328	36.230	1.616	950.480	0.050	0.003	2.606	0.284	0.085	0.005
8500.0	0.327	35.735	1.705	950.216	0.054	0.004	2.715	0.312	0.097	0.006
9000.0	0.325	35.244	1.794	949.952	0.059	0.004	2.819	0.340	0.110	0.007
9500.0	0.323	34.758	1.881	949.688	0.063	0.005	2.918	0.369	0.123	0.008
10000.0	0.321	34.276	1.967	949.425	0.068	0.005	3.012	0.398	0.137	0.010
10500.0	0.319	33.799	2.052	949.161	0.073	0.006	3.102	0.428	0.151	0.011
11000.0	0.317	33.326	2.136	948.897	0.078	0.007	3.187	0.458	0.166	0.013
11500.0	0.316	32.858	2.219	948.634	0.083	0.008	3.268	0.488	0.182	0.015
12000.0	0.314	32.393	2.301	948.370	0.088	0.008	3.344	0.519	0.197	0.017
12500.0	0.312	31.933	2.382	948.107	0.093	0.009	3.417	0.550	0.213	0.020
13000.0	0.310	31.476	2.462	947.844	0.099	0.010	3.486	0.581	0.229	0.022
13500.0	0.308	31.024	2.541	947.581	0.104	0.011	3.551	0.612	0.246	0.025
14000.0	0.307	30.575	2.619	947.318	0.110	0.012	3.612	0.644	0.262	0.028
14500.0	0.305	30.130	2.696	947.056	0.115	0.013	3.670	0.676	0.279	0.031
15000.0	0.303	29.688	2.772	946.793	0.121	0.014	3.724	0.707	0.296	0.034
15500.0	0.302	29.250	2.847	946.532	0.126	0.015	3.775	0.739	0.312	0.038
16000.0	0.300	28.815	2.921	946.270	0.132	0.017	3.822	0.771	0.329	0.041
16500.0	0.298	28.384	2.995	946.008	0.138	0.018	3.867	0.803	0.346	0.045
17000.0	0.296	27.956	3.067	945.747	0.144	0.019	3.909	0.836	0.363	0.049
17500.0	0.295	27.531	3.139	945.486	0.149	0.021	3.948	0.868	0.380	0.053
18000.0	0.293	27.109	3.210	945.224	0.155	0.022	3.984	0.900	0.397	0.058
18500.0	0.291	26.690	3.280	944.962	0.161	0.023	4.018	0.932	0.413	0.062
19000.0	0.290	26.275	3.349	944.700	0.167	0.025	4.050	0.964	0.430	0.067
19500.0	0.288	25.863	3.418	944.437	0.173	0.027	4.080	0.996	0.447	0.072
20000.0	0.286	25.454	3.485	944.174	0.179	0.028	4.108	1.027	0.464	0.077
20500.0	0.285	25.048	3.552	943.910	0.185	0.030	4.134	1.059	0.480	0.082
21000.0	0.283	24.646	3.618	943.644	0.191	0.032	4.159	1.091	0.497	0.088
21500.0	0.281	24.247	3.684	943.377	0.197	0.034	4.182	1.122	0.514	0.094
22000.0	0.280	23.851	3.748	943.110	0.204	0.035	4.204	1.153	0.530	0.100
22500.0	0.278	23.458	3.812	942.840	0.210	0.037	4.224	1.185	0.547	0.106
25000.0	0.270	21.546	4.119	941.473	0.242	0.048	4.307	1.337	0.629	0.140
27500.0	0.262	19.716	4.407	940.068	0.276	0.061	4.361	1.484	0.708	0.180
30000.0	0.253	17.971	4.675	938.624	0.311	0.076	4.390	1.626	0.783	0.225
32500.0	0.245	16.309	4.924	937.140	0.346	0.092	4.398	1.762	0.853	0.276
35000.0	0.237	14.732	5.154	935.612	0.382	0.110	4.388	1.891	0.916	0.333
37500.0	0.229	13.240	5.364	934.039	0.418	0.130	4.363	2.013	0.973	0.394
40000.0	0.221	11.835	5.555	932.417	0.453	0.152	4.325	2.127	1.024	0.461
42500.0	0.213	10.517	5.727	930.743	0.489	0.175	4.277	2.233	1.068	0.533
45000.0	0.205	9.287	5.879	929.017	0.524	0.200	4.222	2.330	1.105	0.610
47500.0	0.197	8.147	6.012	927.234	0.558	0.225	4.161	2.419	1.136	0.691
50000.0	0.189	7.096	6.125	925.393	0.591	0.252	4.097	2.499	1.161	0.776
52500.0	0.181	6.136	6.219	923.492	0.623	0.279	4.031	2.570	1.180	0.865
55000.0	0.174	5.266	6.294	921.530	0.653	0.306	3.965	2.632	1.194	0.957
57500.0	0.166	4.484	6.350	919.505	0.681	0.333	3.900	2.685	1.203	1.053
60000.0	0.159	3.788	6.389	917.420	0.708	0.359	3.837	2.730	1.209	1.150
62500.0	0.152	3.176	6.410	915.273	0.732	0.384	3.778	2.767	1.211	1.250
65000.0	0.145	2.643	6.416	913.069	0.754	0.408	3.723	2.796	1.210	1.350
67500.0	0.138	2.183	6.406	910.809	0.774	0.431	3.673	2.819	1.208	1.451
70000.0	0.131	1.792	6.384	908.498	0.792	0.452	3.627	2.837	1.203	1.553

Table C.20 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.360	44.948	0.000	954.692	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.359	44.829	0.024	954.633	0.000	0.000	0.023	0.000	0.000	0.000
500.0	0.357	44.353	0.117	954.394	0.002	0.000	0.218	0.002	0.000	0.000
1000.0	0.355	43.767	0.232	954.095	0.004	0.000	0.465	0.007	0.000	0.000
1500.0	0.353	43.190	0.345	953.796	0.007	0.000	0.699	0.016	0.001	0.000
2000.0	0.351	42.622	0.456	953.497	0.009	0.000	0.923	0.029	0.003	0.000
2500.0	0.348	42.062	0.565	953.198	0.012	0.000	1.137	0.043	0.006	0.000
3000.0	0.346	41.510	0.673	952.899	0.016	0.000	1.342	0.060	0.009	0.000
3500.0	0.344	40.966	0.778	952.600	0.019	0.001	1.537	0.079	0.014	0.000
4000.0	0.342	40.429	0.882	952.302	0.023	0.001	1.724	0.100	0.020	0.001
4500.0	0.340	39.898	0.984	952.003	0.027	0.001	1.902	0.122	0.027	0.001
5000.0	0.337	39.374	1.085	951.704	0.031	0.001	2.073	0.146	0.035	0.001
5500.0	0.335	38.857	1.184	951.405	0.035	0.002	2.236	0.170	0.044	0.002
6000.0	0.333	38.346	1.282	951.107	0.040	0.002	2.392	0.196	0.054	0.002
6500.0	0.331	37.841	1.378	950.808	0.044	0.003	2.542	0.223	0.065	0.003
7000.0	0.329	37.342	1.473	950.510	0.049	0.003	2.684	0.250	0.078	0.004
7500.0	0.327	36.848	1.566	950.211	0.054	0.004	2.821	0.278	0.090	0.005
8000.0	0.325	36.360	1.658	949.913	0.059	0.004	2.952	0.307	0.104	0.006
8500.0	0.323	35.877	1.749	949.615	0.064	0.005	3.077	0.337	0.119	0.007
9000.0	0.321	35.399	1.838	949.317	0.070	0.006	3.196	0.367	0.134	0.008
9500.0	0.319	34.927	1.927	949.019	0.075	0.006	3.310	0.397	0.150	0.010
10000.0	0.317	34.459	2.013	948.721	0.081	0.007	3.418	0.428	0.166	0.012
10500.0	0.315	33.996	2.099	948.423	0.087	0.008	3.522	0.459	0.183	0.013
11000.0	0.313	33.538	2.183	948.126	0.092	0.009	3.621	0.491	0.201	0.015
11500.0	0.311	33.084	2.267	947.829	0.098	0.010	3.715	0.523	0.218	0.018
12000.0	0.309	32.635	2.349	947.532	0.105	0.011	3.805	0.555	0.237	0.020
12500.0	0.307	32.190	2.430	947.235	0.111	0.012	3.890	0.588	0.255	0.023
13000.0	0.305	31.749	2.510	946.939	0.117	0.013	3.971	0.621	0.274	0.026
13500.0	0.303	31.312	2.589	946.642	0.123	0.015	4.048	0.654	0.293	0.029
14000.0	0.301	30.879	2.666	946.347	0.130	0.016	4.121	0.687	0.312	0.032
14500.0	0.299	30.451	2.743	946.051	0.136	0.017	4.190	0.720	0.331	0.035
15000.0	0.297	30.026	2.819	945.756	0.143	0.019	4.256	0.754	0.350	0.039
15500.0	0.296	29.604	2.894	945.461	0.149	0.020	4.317	0.787	0.370	0.042
16000.0	0.294	29.187	2.968	945.166	0.156	0.022	4.376	0.821	0.389	0.046
16500.0	0.292	28.773	3.041	944.872	0.162	0.023	4.431	0.855	0.409	0.051
17000.0	0.290	28.362	3.113	944.578	0.169	0.025	4.483	0.889	0.428	0.055
17500.0	0.288	27.955	3.184	944.284	0.176	0.027	4.532	0.923	0.447	0.060
18000.0	0.286	27.551	3.254	943.990	0.182	0.029	4.578	0.956	0.467	0.064
18500.0	0.285	27.151	3.323	943.696	0.189	0.031	4.621	0.990	0.486	0.069
19000.0	0.283	26.754	3.392	943.402	0.196	0.033	4.662	1.024	0.505	0.074
19500.0	0.281	26.360	3.459	943.107	0.203	0.035	4.701	1.058	0.524	0.080
20000.0	0.279	25.969	3.526	942.812	0.210	0.037	4.737	1.091	0.544	0.085
20500.0	0.278	25.582	3.592	942.516	0.217	0.039	4.772	1.125	0.563	0.091
21000.0	0.276	25.198	3.657	942.220	0.224	0.041	4.804	1.158	0.582	0.097
21500.0	0.274	24.817	3.721	941.923	0.231	0.043	4.835	1.191	0.601	0.103
22000.0	0.272	24.440	3.784	941.624	0.238	0.046	4.864	1.224	0.619	0.109
22500.0	0.271	24.066	3.847	941.325	0.245	0.048	4.892	1.257	0.638	0.116
23000.0	0.269	23.695	3.909	941.025	0.253	0.051	4.918	1.290	0.657	0.123
23500.0	0.267	23.327	3.970	940.723	0.260	0.054	4.943	1.322	0.676	0.130
25000.0	0.262	22.245	4.147	939.810	0.282	0.062	5.009	1.419	0.731	0.152
27500.0	0.253	20.506	4.428	938.261	0.320	0.078	5.095	1.575	0.822	0.192
30000.0	0.245	18.850	4.689	936.677	0.358	0.096	5.155	1.725	0.908	0.238
32500.0	0.237	17.275	4.930	935.057	0.397	0.116	5.192	1.869	0.989	0.289
35000.0	0.229	15.781	5.152	933.399	0.436	0.138	5.208	2.006	1.064	0.344
37500.0	0.220	14.367	5.356	931.702	0.475	0.162	5.207	2.137	1.133	0.404
40000.0	0.213	13.032	5.540	929.965	0.514	0.189	5.192	2.260	1.196	0.468
42500.0	0.205	11.776	5.706	928.187	0.552	0.216	5.164	2.375	1.252	0.536
45000.0	0.197	10.599	5.854	926.366	0.589	0.246	5.127	2.482	1.301	0.608
47500.0	0.190	9.499	5.984	924.502	0.625	0.277	5.081	2.582	1.344	0.684
50000.0	0.182	8.476	6.097	922.593	0.660	0.308	5.029	2.673	1.381	0.763
52500.0	0.175	7.530	6.192	920.640	0.694	0.341	4.973	2.755	1.412	0.845
55000.0	0.168	6.659	6.271	918.640	0.726	0.374	4.914	2.830	1.437	0.929
57500.0	0.161	5.860	6.333	916.595	0.757	0.407	4.853	2.896	1.457	1.015
60000.0	0.155	5.134	6.379	914.505	0.786	0.440	4.792	2.955	1.473	1.103
62500.0	0.148	4.476	6.410	912.369	0.812	0.472	4.732	3.006	1.485	1.193
65000.0	0.142	3.884	6.427	910.191	0.837	0.503	4.673	3.050	1.493	1.284
67500.0	0.136	3.355	6.431	907.970	0.860	0.533	4.616	3.087	1.497	1.375
70000.0	0.130	2.886	6.422	905.709	0.880	0.562	4.563	3.118	1.499	1.467

Table C.21 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.360	44.948	0.000	954.692	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.359	44.829	0.025	954.623	0.000	0.000	0.027	0.000	0.000	0.000
500.0	0.357	44.357	0.123	954.346	0.002	0.000	0.251	0.002	0.000	0.000
1000.0	0.354	43.776	0.243	954.000	0.005	0.000	0.533	0.008	0.001	0.000
1500.0	0.352	43.206	0.361	953.654	0.008	0.000	0.802	0.018	0.002	0.000
2000.0	0.349	42.646	0.477	953.308	0.011	0.000	1.059	0.032	0.004	0.000
2500.0	0.347	42.094	0.590	952.962	0.015	0.000	1.305	0.048	0.007	0.000
3000.0	0.344	41.551	0.702	952.617	0.019	0.001	1.540	0.066	0.012	0.000
3500.0	0.342	41.017	0.811	952.271	0.023	0.001	1.765	0.087	0.018	0.000
4000.0	0.339	40.490	0.919	951.926	0.028	0.001	1.981	0.109	0.026	0.001
4500.0	0.337	39.970	1.025	951.581	0.033	0.002	2.188	0.133	0.034	0.001
5000.0	0.334	39.458	1.129	951.237	0.038	0.002	2.386	0.157	0.044	0.001
5500.0	0.332	38.953	1.231	950.892	0.043	0.003	2.576	0.184	0.056	0.002
6000.0	0.329	38.455	1.331	950.547	0.048	0.003	2.758	0.211	0.068	0.003
6500.0	0.327	37.963	1.430	950.203	0.054	0.004	2.933	0.239	0.081	0.003
7000.0	0.325	37.477	1.527	949.859	0.060	0.004	3.101	0.268	0.096	0.004
7500.0	0.322	36.997	1.623	949.515	0.066	0.005	3.263	0.297	0.111	0.005
8000.0	0.320	36.523	1.717	949.171	0.072	0.006	3.418	0.328	0.127	0.007
8500.0	0.318	36.055	1.810	948.828	0.079	0.007	3.566	0.359	0.144	0.008
9000.0	0.315	35.592	1.901	948.484	0.085	0.008	3.709	0.390	0.162	0.009
9500.0	0.313	35.135	1.991	948.141	0.092	0.009	3.846	0.422	0.180	0.011
10000.0	0.311	34.683	2.080	947.798	0.099	0.010	3.977	0.455	0.199	0.013
10500.0	0.309	34.236	2.167	947.455	0.106	0.011	4.103	0.488	0.219	0.015
11000.0	0.306	33.794	2.253	947.113	0.113	0.013	4.223	0.521	0.239	0.017
11500.0	0.304	33.357	2.337	946.771	0.120	0.014	4.339	0.555	0.259	0.020
12000.0	0.302	32.924	2.421	946.429	0.127	0.015	4.450	0.590	0.280	0.022
12500.0	0.300	32.496	2.503	946.087	0.134	0.017	4.556	0.624	0.301	0.025
13000.0	0.298	32.073	2.584	945.746	0.142	0.019	4.658	0.659	0.322	0.028
13500.0	0.296	31.654	2.664	945.405	0.149	0.020	4.755	0.694	0.344	0.031
14000.0	0.293	31.239	2.742	945.064	0.157	0.022	4.848	0.729	0.365	0.035
14500.0	0.291	30.829	2.820	944.724	0.165	0.024	4.937	0.765	0.387	0.038
15000.0	0.289	30.422	2.896	944.384	0.172	0.026	5.022	0.801	0.409	0.042
15500.0	0.287	30.020	2.971	944.044	0.180	0.028	5.103	0.837	0.431	0.046
16000.0	0.285	29.621	3.045	943.705	0.188	0.030	5.180	0.873	0.453	0.050
16500.0	0.283	29.226	3.118	943.366	0.196	0.032	5.254	0.909	0.475	0.054
17000.0	0.281	28.835	3.190	943.027	0.204	0.035	5.324	0.945	0.497	0.059
17500.0	0.279	28.448	3.262	942.688	0.212	0.037	5.391	0.982	0.519	0.063
18000.0	0.277	28.064	3.332	942.350	0.220	0.040	5.455	1.018	0.541	0.068
18500.0	0.275	27.683	3.401	942.011	0.228	0.042	5.516	1.055	0.563	0.073
19000.0	0.273	27.307	3.469	941.673	0.236	0.045	5.574	1.091	0.585	0.078
19500.0	0.271	26.933	3.536	941.335	0.244	0.048	5.629	1.128	0.607	0.084
20000.0	0.270	26.563	3.602	940.996	0.252	0.051	5.682	1.164	0.628	0.089
20500.0	0.268	26.197	3.668	940.657	0.260	0.054	5.732	1.201	0.650	0.095
21000.0	0.266	25.834	3.732	940.318	0.268	0.057	5.780	1.237	0.671	0.101
21500.0	0.264	25.474	3.796	939.978	0.276	0.060	5.826	1.274	0.693	0.107
22000.0	0.262	25.117	3.859	939.637	0.284	0.063	5.870	1.310	0.714	0.113
22500.0	0.260	24.764	3.920	939.296	0.292	0.066	5.912	1.346	0.736	0.120
23000.0	0.258	24.414	3.981	938.954	0.301	0.070	5.952	1.382	0.757	0.126
23500.0	0.257	24.067	4.042	938.611	0.309	0.073	5.991	1.418	0.778	0.133
24000.0	0.255	23.724	4.101	938.267	0.317	0.077	6.028	1.453	0.799	0.140
24500.0	0.253	23.384	4.159	937.922	0.325	0.081	6.063	1.489	0.820	0.147
25000.0	0.251	23.047	4.217	937.576	0.334	0.084	6.097	1.524	0.841	0.155
25500.0	0.249	22.713	4.274	937.229	0.342	0.088	6.130	1.559	0.861	0.162
26000.0	0.248	22.383	4.330	936.880	0.350	0.092	6.161	1.594	0.882	0.170
26500.0	0.246	22.056	4.385	936.531	0.359	0.096	6.191	1.629	0.903	0.178
27000.0	0.244	21.732	4.440	936.180	0.367	0.101	6.220	1.663	0.923	0.186
27500.0	0.242	21.411	4.493	935.828	0.375	0.105	6.248	1.698	0.943	0.194
30000.0	0.233	19.855	4.749	934.049	0.418	0.128	6.369	1.866	1.043	0.237
32500.0	0.225	18.377	4.985	932.237	0.460	0.154	6.464	2.028	1.139	0.284
35000.0	0.217	16.976	5.202	930.394	0.502	0.182	6.536	2.185	1.230	0.335
37500.0	0.209	15.650	5.400	928.519	0.545	0.212	6.589	2.335	1.316	0.390
40000.0	0.201	14.397	5.580	926.610	0.586	0.245	6.624	2.479	1.396	0.448
42500.0	0.193	13.216	5.741	924.669	0.627	0.280	6.644	2.616	1.470	0.509
45000.0	0.186	12.105	5.885	922.694	0.667	0.317	6.650	2.746	1.539	0.573
47500.0	0.179	11.063	6.012	920.686	0.706	0.355	6.646	2.868	1.602	0.639
50000.0	0.172	10.087	6.123	918.646	0.743	0.395	6.632	2.984	1.659	0.709
52500.0	0.166	9.177	6.218	916.572	0.779	0.436	6.610	3.091	1.710	0.780
55000.0	0.159	8.329	6.298	914.467	0.814	0.478	6.582	3.192	1.757	0.854
57500.0	0.153	7.542	6.363	912.329	0.847	0.520	6.549	3.285	1.798	0.929
60000.0	0.147	6.814	6.414	910.161	0.877	0.563	6.512	3.371	1.835	1.007
62500.0	0.142	6.142	6.452	907.962	0.906	0.605	6.472	3.450	1.867	1.086
65000.0	0.136	5.523	6.477	905.734	0.933	0.647	6.430	3.522	1.895	1.166
67500.0	0.131	4.957	6.491	903.478	0.958	0.689	6.388	3.588	1.920	1.247
70000.0	0.127	4.439	6.493	901.195	0.981	0.729	6.344	3.647	1.940	1.330

Table C.22 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.354	44.251	0.000	955.395	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.131	0.022	955.345	0.000	0.000	0.018	0.000	0.000	0.000
500.0	0.352	43.653	0.111	955.142	0.002	0.000	0.182	0.001	0.000	0.000
1000.0	0.350	43.063	0.221	954.889	0.003	0.000	0.393	0.006	0.000	0.000
1500.0	0.348	42.481	0.329	954.636	0.005	0.000	0.594	0.014	0.001	0.000
2000.0	0.346	41.906	0.435	954.382	0.008	0.000	0.785	0.025	0.002	0.000
2500.0	0.344	41.339	0.540	954.129	0.010	0.000	0.968	0.037	0.004	0.000
3000.0	0.343	40.778	0.643	953.876	0.013	0.000	1.142	0.053	0.007	0.000
3500.0	0.341	40.224	0.745	953.623	0.015	0.000	1.309	0.070	0.010	0.000
4000.0	0.339	39.677	0.845	953.370	0.018	0.001	1.467	0.088	0.014	0.000
4500.0	0.337	39.135	0.944	953.117	0.021	0.001	1.618	0.108	0.020	0.001
5000.0	0.335	38.600	1.041	952.865	0.025	0.001	1.763	0.130	0.026	0.001
5500.0	0.333	38.070	1.137	952.612	0.028	0.001	1.900	0.152	0.033	0.001
6000.0	0.331	37.545	1.232	952.360	0.032	0.002	2.031	0.176	0.041	0.002
6500.0	0.330	37.026	1.325	952.108	0.035	0.002	2.156	0.200	0.049	0.002
7000.0	0.328	36.512	1.418	951.856	0.039	0.002	2.274	0.226	0.059	0.003
7500.0	0.326	36.004	1.509	951.604	0.043	0.003	2.387	0.252	0.069	0.004
8000.0	0.324	35.500	1.599	951.353	0.047	0.003	2.495	0.279	0.080	0.004
8500.0	0.322	35.000	1.688	951.101	0.051	0.003	2.596	0.306	0.091	0.005
9000.0	0.321	34.506	1.775	950.850	0.055	0.004	2.693	0.334	0.103	0.007
9500.0	0.319	34.016	1.862	950.600	0.059	0.004	2.785	0.363	0.116	0.008
10000.0	0.317	33.530	1.947	950.350	0.064	0.005	2.872	0.392	0.129	0.009
10500.0	0.315	33.049	2.032	950.100	0.068	0.006	2.955	0.421	0.143	0.011
11000.0	0.314	32.571	2.115	949.850	0.073	0.006	3.032	0.451	0.157	0.012
11500.0	0.312	32.098	2.197	949.601	0.078	0.007	3.106	0.481	0.171	0.014
12000.0	0.310	31.628	2.279	949.352	0.082	0.008	3.175	0.511	0.185	0.016
12500.0	0.309	31.163	2.359	949.103	0.087	0.008	3.240	0.542	0.200	0.019
13000.0	0.307	30.701	2.439	948.856	0.092	0.009	3.302	0.573	0.215	0.021
13500.0	0.305	30.243	2.517	948.608	0.097	0.010	3.359	0.604	0.231	0.024
14000.0	0.304	29.788	2.595	948.361	0.102	0.011	3.413	0.635	0.246	0.026
14500.0	0.302	29.336	2.671	948.115	0.107	0.012	3.463	0.666	0.262	0.029
15000.0	0.300	28.888	2.747	947.869	0.112	0.013	3.510	0.698	0.277	0.032
15500.0	0.299	28.443	2.822	947.624	0.117	0.014	3.553	0.730	0.293	0.036
16000.0	0.297	28.001	2.896	947.379	0.123	0.015	3.593	0.761	0.308	0.039
16500.0	0.295	27.563	2.969	947.135	0.128	0.016	3.631	0.793	0.324	0.043
17000.0	0.294	27.127	3.042	946.890	0.133	0.017	3.666	0.825	0.340	0.047
17500.0	0.292	26.695	3.114	946.646	0.138	0.018	3.698	0.856	0.355	0.051
18000.0	0.291	26.265	3.184	946.400	0.144	0.020	3.728	0.888	0.371	0.055
18500.0	0.289	25.839	3.255	946.155	0.149	0.021	3.756	0.920	0.386	0.059
19000.0	0.287	25.416	3.324	945.908	0.155	0.022	3.782	0.951	0.402	0.064
19500.0	0.286	24.997	3.393	945.661	0.160	0.024	3.807	0.983	0.417	0.069
20000.0	0.284	24.580	3.460	945.412	0.166	0.025	3.830	1.014	0.433	0.074
20500.0	0.283	24.167	3.527	945.163	0.171	0.026	3.851	1.045	0.449	0.079
22500.0	0.276	22.547	3.788	944.150	0.194	0.033	3.923	1.168	0.511	0.102
25000.0	0.268	20.596	4.097	942.850	0.224	0.043	3.985	1.318	0.589	0.135
27500.0	0.260	18.730	4.386	941.512	0.256	0.054	4.020	1.463	0.663	0.175
30000.0	0.252	16.949	4.656	940.132	0.288	0.066	4.032	1.603	0.732	0.220
32500.0	0.244	15.255	4.907	938.707	0.321	0.081	4.024	1.737	0.796	0.271
35000.0	0.236	13.648	5.138	937.234	0.355	0.097	4.000	1.864	0.853	0.328
37500.0	0.228	12.132	5.350	935.710	0.388	0.115	3.962	1.983	0.904	0.391
40000.0	0.220	10.707	5.541	934.130	0.422	0.134	3.914	2.094	0.949	0.460
42500.0	0.212	9.378	5.712	932.491	0.456	0.155	3.857	2.197	0.986	0.534
45000.0	0.204	8.145	5.862	930.789	0.489	0.177	3.795	2.290	1.017	0.614
47500.0	0.195	7.012	5.991	929.021	0.521	0.200	3.729	2.375	1.042	0.699
50000.0	0.187	5.980	6.099	927.183	0.552	0.223	3.662	2.449	1.061	0.789
52500.0	0.179	5.050	6.186	925.272	0.582	0.247	3.595	2.514	1.074	0.883
55000.0	0.171	4.221	6.253	923.288	0.610	0.271	3.530	2.570	1.083	0.982
57500.0	0.163	3.491	6.299	921.230	0.637	0.294	3.469	2.616	1.088	1.083
60000.0	0.155	2.859	6.326	919.099	0.661	0.317	3.411	2.653	1.089	1.187
62500.0	0.147	2.318	6.335	916.897	0.683	0.338	3.359	2.683	1.088	1.293
65000.0	0.140	1.862	6.327	914.630	0.703	0.358	3.312	2.705	1.085	1.401
67500.0	0.132	1.483	6.305	912.302	0.720	0.377	3.271	2.721	1.080	1.508
70000.0	0.125	1.173	6.269	909.920	0.735	0.393	3.235	2.732	1.075	1.615

Table C.23 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.354	44.251	0.000	955.395	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.131	0.023	955.337	0.000	0.000	0.021	0.000	0.000	0.000
500.0	0.352	43.655	0.116	955.106	0.002	0.000	0.207	0.002	0.000	0.000
1000.0	0.350	43.068	0.229	954.817	0.004	0.000	0.446	0.007	0.000	0.000
1500.0	0.347	42.490	0.341	954.528	0.006	0.000	0.674	0.016	0.001	0.000
2000.0	0.345	41.920	0.451	954.239	0.009	0.000	0.891	0.028	0.003	0.000
2500.0	0.343	41.358	0.559	953.951	0.012	0.000	1.098	0.042	0.005	0.000
3000.0	0.341	40.804	0.665	953.663	0.015	0.000	1.295	0.059	0.009	0.000
3500.0	0.339	40.257	0.769	953.375	0.018	0.001	1.483	0.077	0.013	0.000
4000.0	0.337	39.718	0.872	953.088	0.022	0.001	1.662	0.098	0.019	0.000
4500.0	0.335	39.185	0.973	952.801	0.025	0.001	1.834	0.120	0.025	0.001
5000.0	0.333	38.659	1.073	952.514	0.029	0.001	1.997	0.143	0.033	0.001
5500.0	0.331	38.139	1.171	952.228	0.033	0.002	2.153	0.167	0.042	0.001
6000.0	0.329	37.625	1.267	951.942	0.037	0.002	2.301	0.193	0.051	0.002
6500.0	0.327	37.117	1.363	951.656	0.042	0.002	2.443	0.219	0.062	0.003
7000.0	0.325	36.614	1.456	951.371	0.046	0.003	2.578	0.246	0.074	0.003
7500.0	0.323	36.117	1.549	951.086	0.051	0.003	2.706	0.274	0.086	0.004
8000.0	0.321	35.625	1.640	950.802	0.055	0.004	2.828	0.303	0.099	0.005
8500.0	0.319	35.139	1.729	950.518	0.060	0.005	2.945	0.332	0.113	0.006
9000.0	0.317	34.657	1.818	950.235	0.065	0.005	3.055	0.362	0.127	0.008
9500.0	0.315	34.181	1.905	949.952	0.071	0.006	3.161	0.392	0.143	0.009
10000.0	0.313	33.708	1.991	949.669	0.076	0.007	3.261	0.422	0.158	0.011
10500.0	0.311	33.241	2.076	949.387	0.081	0.007	3.355	0.454	0.174	0.013
11000.0	0.309	32.778	2.160	949.106	0.087	0.008	3.445	0.485	0.191	0.015
11500.0	0.307	32.320	2.242	948.825	0.092	0.009	3.530	0.517	0.208	0.017
12000.0	0.306	31.865	2.323	948.545	0.098	0.010	3.611	0.549	0.225	0.019
12500.0	0.304	31.415	2.404	948.265	0.104	0.011	3.687	0.581	0.242	0.022
13000.0	0.302	30.969	2.483	947.986	0.109	0.012	3.758	0.614	0.260	0.024
13500.0	0.300	30.526	2.561	947.707	0.115	0.013	3.826	0.647	0.278	0.027
14000.0	0.298	30.088	2.639	947.430	0.121	0.014	3.889	0.680	0.296	0.030
14500.0	0.297	29.653	2.715	947.153	0.127	0.016	3.949	0.713	0.314	0.034
15000.0	0.295	29.221	2.790	946.876	0.133	0.017	4.004	0.746	0.332	0.037
15500.0	0.293	28.793	2.865	946.601	0.139	0.018	4.057	0.779	0.350	0.041
16000.0	0.291	28.368	2.938	946.325	0.145	0.019	4.105	0.813	0.368	0.045
16500.0	0.290	27.947	3.011	946.050	0.151	0.021	4.151	0.846	0.387	0.049
17000.0	0.288	27.529	3.082	945.776	0.157	0.022	4.194	0.880	0.405	0.053
17500.0	0.286	27.114	3.153	945.502	0.163	0.024	4.233	0.913	0.423	0.057
18000.0	0.284	26.702	3.223	945.227	0.169	0.025	4.271	0.947	0.441	0.062
18500.0	0.283	26.294	3.292	944.953	0.175	0.027	4.306	0.980	0.458	0.067
19000.0	0.281	25.889	3.361	944.678	0.182	0.029	4.339	1.013	0.476	0.072
19500.0	0.279	25.487	3.428	944.402	0.188	0.031	4.369	1.046	0.494	0.077
20000.0	0.278	25.088	3.495	944.125	0.194	0.032	4.398	1.079	0.512	0.082
20500.0	0.276	24.693	3.561	943.847	0.201	0.034	4.425	1.112	0.530	0.088
21000.0	0.274	24.300	3.626	943.569	0.207	0.036	4.451	1.145	0.548	0.094
21500.0	0.273	23.912	3.690	943.289	0.214	0.038	4.475	1.177	0.565	0.100
22500.0	0.269	23.144	3.817	942.725	0.227	0.043	4.519	1.242	0.601	0.112
25000.0	0.261	21.283	4.118	941.292	0.261	0.055	4.607	1.400	0.689	0.148
27500.0	0.253	19.506	4.400	939.822	0.297	0.068	4.666	1.552	0.773	0.189
30000.0	0.245	17.812	4.662	938.316	0.333	0.084	4.699	1.698	0.853	0.235
32500.0	0.236	16.201	4.905	936.770	0.369	0.102	4.711	1.839	0.927	0.286
35000.0	0.228	14.674	5.128	935.184	0.406	0.122	4.705	1.972	0.995	0.343
37500.0	0.220	13.230	5.332	933.554	0.443	0.143	4.683	2.098	1.057	0.405
40000.0	0.212	11.870	5.517	931.879	0.480	0.166	4.648	2.216	1.111	0.471
42500.0	0.204	10.594	5.682	930.156	0.516	0.191	4.602	2.326	1.159	0.542
45000.0	0.197	9.403	5.829	928.384	0.551	0.217	4.549	2.427	1.199	0.618
47500.0	0.189	8.298	5.957	926.560	0.586	0.244	4.490	2.520	1.234	0.698
50000.0	0.181	7.277	6.066	924.684	0.619	0.272	4.426	2.603	1.262	0.781
52500.0	0.174	6.342	6.157	922.752	0.651	0.301	4.361	2.678	1.284	0.868
55000.0	0.167	5.490	6.229	920.765	0.681	0.330	4.294	2.743	1.300	0.958
57500.0	0.159	4.722	6.284	918.723	0.709	0.358	4.229	2.800	1.312	1.051
60000.0	0.152	4.033	6.321	916.625	0.736	0.386	4.165	2.848	1.320	1.145
62500.0	0.145	3.422	6.343	914.473	0.761	0.414	4.104	2.888	1.324	1.241
65000.0	0.139	2.885	6.349	912.267	0.783	0.440	4.046	2.921	1.325	1.339
67500.0	0.132	2.417	6.340	910.012	0.803	0.465	3.993	2.947	1.323	1.437
70000.0	0.126	2.014	6.319	907.710	0.821	0.488	3.944	2.967	1.320	1.535

Table C.24 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.354	44.251	0.000	955.395	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.353	44.132	0.024	955.328	0.000	0.000	0.024	0.000	0.000	0.000
500.0	0.351	43.658	0.122	955.059	0.002	0.000	0.238	0.002	0.000	0.000
1000.0	0.349	43.076	0.241	954.722	0.005	0.000	0.514	0.008	0.000	0.000
1500.0	0.346	42.504	0.358	954.386	0.008	0.000	0.776	0.018	0.002	0.000
2000.0	0.344	41.941	0.472	954.051	0.011	0.000	1.026	0.031	0.004	0.000
2500.0	0.341	41.387	0.584	953.716	0.014	0.000	1.265	0.047	0.007	0.000
3000.0	0.339	40.842	0.695	953.382	0.018	0.001	1.492	0.065	0.012	0.000
3500.0	0.337	40.305	0.803	953.049	0.022	0.001	1.710	0.085	0.017	0.000
4000.0	0.334	39.776	0.910	952.716	0.027	0.001	1.917	0.107	0.025	0.001
4500.0	0.332	39.254	1.014	952.384	0.031	0.001	2.116	0.131	0.033	0.001
5000.0	0.330	38.739	1.117	952.052	0.036	0.002	2.306	0.156	0.043	0.001
5500.0	0.327	38.231	1.218	951.721	0.041	0.002	2.487	0.182	0.054	0.002
6000.0	0.325	37.729	1.318	951.390	0.046	0.003	2.660	0.209	0.066	0.002
6500.0	0.323	37.234	1.415	951.060	0.051	0.003	2.826	0.236	0.079	0.003
7000.0	0.320	36.745	1.511	950.730	0.057	0.004	2.985	0.265	0.093	0.004
7500.0	0.318	36.262	1.606	950.401	0.063	0.005	3.136	0.295	0.108	0.005
8000.0	0.316	35.785	1.699	950.072	0.068	0.006	3.281	0.325	0.124	0.006
8500.0	0.314	35.313	1.791	949.745	0.075	0.006	3.419	0.356	0.140	0.008
9000.0	0.311	34.847	1.881	949.418	0.081	0.007	3.551	0.387	0.158	0.009
9500.0	0.309	34.386	1.970	949.091	0.087	0.008	3.677	0.419	0.175	0.011
10000.0	0.307	33.930	2.058	948.765	0.093	0.009	3.797	0.452	0.194	0.013
10500.0	0.305	33.479	2.144	948.440	0.100	0.010	3.912	0.485	0.213	0.015
11000.0	0.303	33.033	2.229	948.115	0.106	0.012	4.021	0.518	0.232	0.017
11500.0	0.301	32.591	2.313	947.791	0.113	0.013	4.125	0.552	0.252	0.019
12000.0	0.299	32.154	2.395	947.468	0.120	0.014	4.224	0.586	0.272	0.022
12500.0	0.297	31.722	2.476	947.145	0.127	0.016	4.318	0.620	0.292	0.025
13000.0	0.295	31.293	2.556	946.823	0.134	0.017	4.408	0.655	0.313	0.028
13500.0	0.293	30.869	2.635	946.502	0.141	0.019	4.492	0.690	0.333	0.031
14000.0	0.291	30.449	2.713	946.182	0.148	0.020	4.573	0.725	0.354	0.034
14500.0	0.289	30.033	2.790	945.862	0.155	0.022	4.649	0.761	0.375	0.038
15000.0	0.287	29.621	2.865	945.543	0.162	0.024	4.721	0.796	0.396	0.041
15500.0	0.285	29.213	2.940	945.225	0.169	0.025	4.789	0.832	0.417	0.045
16000.0	0.283	28.808	3.013	944.908	0.176	0.027	4.854	0.868	0.438	0.049
16500.0	0.281	28.407	3.086	944.591	0.184	0.029	4.915	0.904	0.459	0.053
17000.0	0.279	28.009	3.157	944.274	0.191	0.031	4.972	0.940	0.480	0.058
17500.0	0.277	27.615	3.228	943.958	0.198	0.033	5.026	0.976	0.500	0.063
18000.0	0.275	27.224	3.298	943.642	0.206	0.036	5.078	1.012	0.521	0.067
18500.0	0.274	26.836	3.366	943.326	0.213	0.038	5.126	1.048	0.542	0.072
19000.0	0.272	26.452	3.434	943.011	0.220	0.040	5.172	1.084	0.562	0.078
19500.0	0.270	26.071	3.501	942.695	0.228	0.043	5.215	1.120	0.583	0.083
20000.0	0.268	25.694	3.567	942.378	0.235	0.045	5.257	1.156	0.603	0.088
20500.0	0.266	25.320	3.632	942.061	0.243	0.048	5.296	1.192	0.624	0.094
21000.0	0.264	24.949	3.696	941.743	0.250	0.051	5.333	1.227	0.644	0.100
21500.0	0.263	24.581	3.760	941.424	0.258	0.053	5.368	1.263	0.664	0.106
22000.0	0.261	24.217	3.822	941.105	0.266	0.056	5.402	1.298	0.684	0.113
22500.0	0.259	23.856	3.884	940.784	0.273	0.059	5.434	1.333	0.705	0.119
23000.0	0.257	23.498	3.945	940.463	0.281	0.062	5.464	1.369	0.725	0.126
23500.0	0.256	23.144	4.005	940.140	0.289	0.065	5.493	1.403	0.745	0.133
25000.0	0.250	22.101	4.180	939.164	0.312	0.075	5.573	1.507	0.804	0.154
27500.0	0.242	20.428	4.456	937.511	0.351	0.093	5.680	1.675	0.902	0.194
30000.0	0.233	18.837	4.712	935.825	0.391	0.114	5.760	1.837	0.996	0.239
32500.0	0.225	17.324	4.948	934.105	0.432	0.137	5.815	1.993	1.084	0.288
35000.0	0.217	15.891	5.165	932.351	0.472	0.162	5.850	2.143	1.168	0.341
37500.0	0.209	14.535	5.363	930.561	0.512	0.189	5.866	2.286	1.245	0.398
40000.0	0.201	13.256	5.542	928.735	0.552	0.218	5.866	2.422	1.316	0.459
42500.0	0.193	12.052	5.702	926.872	0.591	0.249	5.853	2.550	1.380	0.524
45000.0	0.186	10.923	5.845	924.971	0.629	0.282	5.828	2.671	1.439	0.592
47500.0	0.179	9.867	5.969	923.032	0.666	0.316	5.795	2.783	1.491	0.663
50000.0	0.172	8.883	6.077	921.055	0.702	0.352	5.754	2.887	1.537	0.737
52500.0	0.165	7.970	6.168	919.039	0.736	0.388	5.707	2.983	1.577	0.814
55000.0	0.158	7.125	6.243	916.984	0.769	0.425	5.656	3.070	1.611	0.893
57500.0	0.152	6.348	6.303	914.891	0.800	0.462	5.602	3.150	1.641	0.974
60000.0	0.146	5.635	6.348	912.760	0.829	0.499	5.546	3.221	1.665	1.058
62500.0	0.140	4.984	6.378	910.592	0.856	0.536	5.490	3.285	1.685	1.143
65000.0	0.134	4.394	6.395	908.388	0.881	0.572	5.434	3.342	1.701	1.229
67500.0	0.128	3.860	6.400	906.149	0.904	0.607	5.379	3.391	1.713	1.317
70000.0	0.123	3.380	6.393	903.877	0.924	0.641	5.326	3.434	1.723	1.405

Table C.25 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.354	44.251	0.000	955.395	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.131	0.022	955.347	0.000	0.000	0.017	0.000	0.000	0.000
500.0	0.352	43.653	0.110	955.153	0.001	0.000	0.174	0.001	0.000	0.000
1000.0	0.350	43.063	0.219	954.911	0.003	0.000	0.375	0.006	0.000	0.000
1500.0	0.348	42.480	0.326	954.668	0.005	0.000	0.567	0.013	0.001	0.000
2000.0	0.347	41.904	0.431	954.426	0.007	0.000	0.751	0.023	0.002	0.000
2500.0	0.345	41.335	0.535	954.184	0.010	0.000	0.926	0.036	0.004	0.000
3000.0	0.343	40.773	0.637	953.941	0.012	0.000	1.093	0.050	0.006	0.000
3500.0	0.341	40.217	0.738	953.699	0.015	0.000	1.252	0.067	0.009	0.000
4000.0	0.339	39.667	0.837	953.457	0.018	0.001	1.404	0.085	0.013	0.000
4500.0	0.338	39.124	0.935	953.214	0.021	0.001	1.548	0.104	0.018	0.001
5000.0	0.336	38.585	1.032	952.972	0.024	0.001	1.686	0.125	0.024	0.001
5500.0	0.334	38.053	1.128	952.730	0.027	0.001	1.818	0.146	0.030	0.001
6000.0	0.332	37.525	1.222	952.488	0.030	0.001	1.943	0.169	0.037	0.001
6500.0	0.330	37.003	1.315	952.246	0.034	0.002	2.063	0.193	0.045	0.002
7000.0	0.329	36.486	1.407	952.005	0.037	0.002	2.176	0.218	0.054	0.003
7500.0	0.327	35.973	1.498	951.763	0.041	0.002	2.284	0.243	0.064	0.003
8000.0	0.325	35.466	1.588	951.522	0.045	0.003	2.387	0.269	0.074	0.004
8500.0	0.323	34.962	1.676	951.281	0.049	0.003	2.485	0.296	0.084	0.005
9000.0	0.322	34.464	1.764	951.040	0.053	0.004	2.577	0.323	0.096	0.006
9500.0	0.320	33.970	1.850	950.800	0.057	0.004	2.665	0.351	0.107	0.007
10000.0	0.318	33.479	1.936	950.560	0.061	0.005	2.748	0.379	0.120	0.009
10500.0	0.317	32.993	2.020	950.320	0.065	0.005	2.826	0.408	0.133	0.010
11000.0	0.315	32.511	2.104	950.080	0.070	0.006	2.901	0.437	0.146	0.012
11500.0	0.313	32.033	2.186	949.841	0.074	0.006	2.971	0.467	0.159	0.013
12000.0	0.312	31.559	2.267	949.602	0.079	0.007	3.037	0.496	0.173	0.015
12500.0	0.310	31.089	2.348	949.363	0.083	0.008	3.099	0.526	0.187	0.017
13000.0	0.308	30.622	2.427	949.125	0.088	0.009	3.157	0.557	0.201	0.020
13500.0	0.307	30.158	2.506	948.887	0.093	0.009	3.212	0.587	0.215	0.022
14000.0	0.305	29.698	2.584	948.650	0.098	0.010	3.262	0.618	0.230	0.025
14500.0	0.303	29.241	2.661	948.413	0.102	0.011	3.310	0.649	0.245	0.027
15000.0	0.302	28.788	2.737	948.177	0.107	0.012	3.354	0.679	0.259	0.030
15500.0	0.300	28.337	2.812	947.941	0.112	0.013	3.395	0.711	0.274	0.034
16000.0	0.299	27.890	2.887	947.706	0.117	0.014	3.433	0.742	0.289	0.037
16500.0	0.297	27.446	2.960	947.471	0.122	0.015	3.468	0.773	0.303	0.040
17000.0	0.296	27.004	3.033	947.236	0.127	0.016	3.501	0.804	0.318	0.044
17500.0	0.294	26.566	3.105	947.000	0.132	0.017	3.531	0.835	0.333	0.048
18000.0	0.292	26.131	3.176	946.765	0.137	0.018	3.559	0.866	0.347	0.052
18500.0	0.291	25.699	3.247	946.528	0.142	0.019	3.585	0.897	0.362	0.056
19000.0	0.289	25.270	3.317	946.291	0.148	0.021	3.610	0.928	0.377	0.060
19500.0	0.288	24.844	3.386	946.052	0.153	0.022	3.632	0.959	0.392	0.065
20000.0	0.286	24.422	3.454	945.813	0.158	0.023	3.654	0.990	0.407	0.070
22500.0	0.278	22.358	3.785	944.593	0.185	0.031	3.740	1.142	0.481	0.097
25000.0	0.270	20.377	4.096	943.337	0.214	0.040	3.795	1.290	0.555	0.130
27500.0	0.262	18.480	4.389	942.040	0.244	0.050	3.824	1.434	0.625	0.168
30000.0	0.254	16.668	4.663	940.701	0.276	0.062	3.830	1.573	0.690	0.212
32500.0	0.246	14.944	4.917	939.314	0.308	0.076	3.818	1.705	0.751	0.262
35000.0	0.238	13.309	5.152	937.878	0.340	0.091	3.789	1.831	0.805	0.319
37500.0	0.230	11.767	5.366	936.387	0.373	0.108	3.748	1.950	0.853	0.381
40000.0	0.222	10.320	5.560	934.837	0.406	0.126	3.697	2.061	0.895	0.450
42500.0	0.214	8.971	5.734	933.223	0.438	0.145	3.638	2.163	0.929	0.525
45000.0	0.205	7.725	5.885	931.540	0.470	0.166	3.574	2.255	0.958	0.605
47500.0	0.197	6.584	6.015	929.786	0.502	0.188	3.508	2.339	0.980	0.691
50000.0	0.189	5.551	6.123	927.955	0.532	0.210	3.440	2.412	0.997	0.783
52500.0	0.180	4.626	6.209	926.044	0.561	0.232	3.375	2.476	1.008	0.879
55000.0	0.172	3.810	6.273	924.052	0.588	0.255	3.312	2.529	1.015	0.980
57500.0	0.163	3.101	6.316	921.979	0.613	0.277	3.253	2.574	1.019	1.085
60000.0	0.155	2.494	6.339	919.827	0.636	0.297	3.199	2.609	1.019	1.192
62500.0	0.147	1.984	6.343	917.599	0.657	0.317	3.151	2.637	1.017	1.302
65000.0	0.139	1.563	6.330	915.301	0.676	0.335	3.108	2.657	1.014	1.412
67500.0	0.132	1.220	6.302	912.941	0.692	0.352	3.072	2.671	1.009	1.523
70000.0	0.124	0.945	6.262	910.526	0.706	0.367	3.040	2.680	1.004	1.633

Table C.26 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.354	44.251	0.000	955.395	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.131	0.023	955.339	0.000	0.000	0.020	0.000	0.000	0.000
500.0	0.352	43.655	0.115	955.116	0.002	0.000	0.199	0.001	0.000	0.000
1000.0	0.350	43.068	0.227	954.837	0.004	0.000	0.430	0.007	0.000	0.000
1500.0	0.348	42.489	0.338	954.558	0.006	0.000	0.650	0.015	0.001	0.000
2000.0	0.346	41.918	0.447	954.279	0.009	0.000	0.859	0.027	0.003	0.000
2500.0	0.344	41.356	0.554	954.000	0.012	0.000	1.059	0.041	0.005	0.000
3000.0	0.341	40.800	0.659	953.722	0.015	0.000	1.249	0.057	0.008	0.000
3500.0	0.339	40.252	0.763	953.444	0.018	0.001	1.431	0.075	0.012	0.000
4000.0	0.337	39.711	0.865	953.166	0.021	0.001	1.604	0.095	0.017	0.000
4500.0	0.335	39.176	0.965	952.889	0.025	0.001	1.769	0.116	0.024	0.001
5000.0	0.333	38.647	1.064	952.612	0.028	0.001	1.927	0.139	0.031	0.001
5500.0	0.331	38.125	1.162	952.335	0.032	0.002	2.077	0.162	0.039	0.001
6000.0	0.329	37.608	1.258	952.059	0.036	0.002	2.220	0.187	0.048	0.002
6500.0	0.327	37.098	1.353	951.782	0.040	0.002	2.356	0.213	0.058	0.002
7000.0	0.326	36.592	1.446	951.507	0.045	0.003	2.486	0.240	0.069	0.003
7500.0	0.324	36.092	1.538	951.231	0.049	0.003	2.610	0.267	0.081	0.004
8000.0	0.322	35.597	1.629	950.956	0.054	0.004	2.727	0.295	0.093	0.005
8500.0	0.320	35.108	1.718	950.682	0.058	0.004	2.839	0.324	0.107	0.006
9000.0	0.318	34.623	1.807	950.408	0.063	0.005	2.946	0.353	0.120	0.007
9500.0	0.316	34.143	1.894	950.134	0.068	0.005	3.047	0.382	0.135	0.009
10000.0	0.314	33.667	1.980	949.861	0.073	0.006	3.142	0.412	0.150	0.010
10500.0	0.312	33.196	2.064	949.589	0.078	0.007	3.233	0.443	0.165	0.012
11000.0	0.310	32.729	2.148	949.317	0.084	0.008	3.319	0.474	0.181	0.014
11500.0	0.309	32.267	2.230	949.045	0.089	0.009	3.400	0.505	0.197	0.016
12000.0	0.307	31.808	2.312	948.774	0.095	0.009	3.477	0.537	0.213	0.018
12500.0	0.305	31.354	2.392	948.504	0.100	0.010	3.550	0.569	0.230	0.021
13000.0	0.303	30.903	2.471	948.234	0.106	0.011	3.618	0.601	0.247	0.023
13500.0	0.301	30.456	2.550	947.965	0.111	0.012	3.682	0.633	0.264	0.026
14000.0	0.300	30.013	2.627	947.696	0.117	0.013	3.742	0.665	0.281	0.029
14500.0	0.298	29.574	2.703	947.428	0.123	0.015	3.799	0.698	0.299	0.032
15000.0	0.296	29.138	2.779	947.161	0.128	0.016	3.851	0.731	0.316	0.036
15500.0	0.294	28.705	2.854	946.894	0.134	0.017	3.900	0.763	0.333	0.039
16000.0	0.293	28.275	2.927	946.629	0.140	0.018	3.946	0.796	0.350	0.043
16500.0	0.291	27.849	3.000	946.363	0.146	0.020	3.989	0.829	0.368	0.047
17000.0	0.289	27.426	3.072	946.098	0.151	0.021	4.029	0.862	0.385	0.051
17500.0	0.288	27.006	3.143	945.832	0.157	0.022	4.066	0.895	0.402	0.055
18000.0	0.286	26.589	3.213	945.567	0.163	0.024	4.101	0.928	0.419	0.060
18500.0	0.284	26.176	3.283	945.301	0.169	0.026	4.133	0.961	0.437	0.064
19000.0	0.283	25.765	3.352	945.035	0.175	0.027	4.164	0.993	0.454	0.069
19500.0	0.281	25.358	3.420	944.768	0.181	0.029	4.193	1.026	0.471	0.074
20000.0	0.279	24.954	3.487	944.499	0.188	0.031	4.219	1.059	0.488	0.079
20500.0	0.278	24.554	3.553	944.230	0.194	0.032	4.244	1.091	0.505	0.085
21000.0	0.276	24.156	3.619	943.960	0.200	0.034	4.268	1.123	0.522	0.091
22500.0	0.271	22.984	3.810	943.141	0.219	0.040	4.331	1.219	0.573	0.109
25000.0	0.263	21.097	4.114	941.749	0.252	0.051	4.409	1.374	0.658	0.144
27500.0	0.255	19.294	4.398	940.320	0.287	0.065	4.460	1.524	0.739	0.184
30000.0	0.247	17.573	4.663	938.854	0.322	0.080	4.485	1.669	0.815	0.229
32500.0	0.238	15.937	4.908	937.346	0.358	0.096	4.490	1.807	0.886	0.280
35000.0	0.230	14.384	5.134	935.797	0.393	0.115	4.477	1.939	0.950	0.337
37500.0	0.222	12.917	5.341	934.201	0.429	0.136	4.448	2.063	1.008	0.399
40000.0	0.214	11.536	5.528	932.559	0.465	0.158	4.408	2.179	1.059	0.465
42500.0	0.206	10.242	5.696	930.865	0.500	0.181	4.358	2.287	1.103	0.537
45000.0	0.198	9.035	5.845	929.119	0.535	0.206	4.300	2.387	1.140	0.613
47500.0	0.191	7.918	5.974	927.318	0.569	0.232	4.237	2.477	1.171	0.694
50000.0	0.183	6.889	6.083	925.459	0.601	0.259	4.170	2.558	1.196	0.779
52500.0	0.175	5.951	6.174	923.541	0.632	0.286	4.103	2.629	1.215	0.868
55000.0	0.168	5.101	6.245	921.563	0.662	0.313	4.035	2.692	1.228	0.960
57500.0	0.160	4.338	6.298	919.523	0.689	0.340	3.969	2.745	1.238	1.054
60000.0	0.153	3.661	6.333	917.423	0.715	0.367	3.905	2.790	1.243	1.151
62500.0	0.146	3.066	6.352	915.264	0.739	0.392	3.845	2.827	1.244	1.250
65000.0	0.139	2.549	6.354	913.047	0.760	0.416	3.790	2.857	1.243	1.350
67500.0	0.132	2.104	6.342	910.777	0.780	0.439	3.738	2.879	1.240	1.451
70000.0	0.126	1.725	6.317	908.457	0.796	0.460	3.692	2.896	1.236	1.551

Table C.27 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.354	44.251	0.000	955.395	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.132	0.024	955.329	0.000	0.000	0.023	0.000	0.000	0.000
500.0	0.351	43.659	0.121	955.064	0.002	0.000	0.234	0.002	0.000	0.000
1000.0	0.349	43.077	0.239	954.733	0.005	0.000	0.504	0.008	0.000	0.000
1500.0	0.347	42.505	0.355	954.403	0.008	0.000	0.762	0.018	0.002	0.000
2000.0	0.344	41.942	0.469	954.074	0.011	0.000	1.007	0.031	0.004	0.000
2500.0	0.342	41.388	0.581	953.745	0.014	0.000	1.241	0.046	0.007	0.000
3000.0	0.339	40.842	0.691	953.416	0.018	0.001	1.464	0.064	0.011	0.000
3500.0	0.337	40.305	0.799	953.088	0.022	0.001	1.678	0.084	0.017	0.000
4000.0	0.335	39.775	0.905	952.761	0.026	0.001	1.881	0.106	0.024	0.001
4500.0	0.332	39.252	1.009	952.434	0.031	0.001	2.076	0.129	0.032	0.001
5000.0	0.330	38.737	1.111	952.108	0.035	0.002	2.262	0.154	0.041	0.001
5500.0	0.328	38.228	1.212	951.782	0.040	0.002	2.439	0.179	0.052	0.002
6000.0	0.325	37.726	1.311	951.457	0.045	0.003	2.609	0.206	0.064	0.002
6500.0	0.323	37.229	1.408	951.132	0.051	0.003	2.771	0.234	0.077	0.003
7000.0	0.321	36.739	1.504	950.808	0.056	0.004	2.926	0.262	0.090	0.004
7500.0	0.319	36.255	1.598	950.485	0.062	0.005	3.074	0.291	0.105	0.005
8000.0	0.316	35.777	1.691	950.162	0.068	0.005	3.216	0.321	0.120	0.006
8500.0	0.314	35.304	1.783	949.840	0.074	0.006	3.351	0.352	0.137	0.007
9000.0	0.312	34.836	1.873	949.519	0.080	0.007	3.479	0.383	0.153	0.009
9500.0	0.310	34.373	1.961	949.198	0.086	0.008	3.602	0.415	0.171	0.011
10000.0	0.308	33.916	2.049	948.877	0.092	0.009	3.719	0.447	0.189	0.012
10500.0	0.306	33.463	2.135	948.558	0.099	0.010	3.831	0.479	0.207	0.014
11000.0	0.304	33.015	2.220	948.239	0.105	0.011	3.937	0.512	0.226	0.017
11500.0	0.302	32.572	2.303	947.921	0.112	0.012	4.038	0.546	0.246	0.019
12000.0	0.300	32.133	2.386	947.603	0.118	0.014	4.135	0.580	0.265	0.021
12500.0	0.298	31.698	2.467	947.286	0.125	0.015	4.226	0.614	0.285	0.024
13000.0	0.296	31.268	2.547	946.970	0.132	0.017	4.312	0.648	0.305	0.027
13500.0	0.294	30.842	2.626	946.655	0.139	0.018	4.395	0.683	0.326	0.030
14000.0	0.292	30.420	2.703	946.340	0.146	0.020	4.472	0.718	0.346	0.033
14500.0	0.290	30.001	2.780	946.026	0.153	0.021	4.546	0.753	0.366	0.037
15000.0	0.288	29.587	2.855	945.713	0.160	0.023	4.615	0.788	0.387	0.041
15500.0	0.286	29.176	2.930	945.401	0.167	0.025	4.681	0.823	0.407	0.044
16000.0	0.284	28.769	3.004	945.089	0.174	0.027	4.743	0.859	0.428	0.048
16500.0	0.282	28.366	3.076	944.778	0.181	0.028	4.801	0.894	0.448	0.053
17000.0	0.280	27.965	3.148	944.468	0.188	0.030	4.856	0.930	0.469	0.057
17500.0	0.278	27.569	3.218	944.157	0.195	0.033	4.908	0.966	0.489	0.062
18000.0	0.276	27.175	3.288	943.848	0.203	0.035	4.957	1.001	0.509	0.066
18500.0	0.275	26.785	3.356	943.538	0.210	0.037	5.003	1.037	0.530	0.071
19000.0	0.273	26.398	3.424	943.228	0.217	0.039	5.047	1.073	0.550	0.076
19500.0	0.271	26.015	3.491	942.917	0.224	0.042	5.088	1.108	0.570	0.082
20000.0	0.269	25.635	3.557	942.606	0.232	0.044	5.127	1.144	0.590	0.087
20500.0	0.267	25.258	3.623	942.295	0.239	0.046	5.164	1.179	0.610	0.093
21000.0	0.266	24.885	3.687	941.982	0.247	0.049	5.200	1.214	0.630	0.099
21500.0	0.264	24.514	3.750	941.669	0.254	0.052	5.233	1.249	0.649	0.105
22000.0	0.262	24.147	3.813	941.355	0.262	0.055	5.265	1.284	0.669	0.111
22500.0	0.260	23.784	3.875	941.040	0.269	0.057	5.295	1.319	0.689	0.118
23000.0	0.259	23.424	3.936	940.724	0.277	0.060	5.324	1.354	0.709	0.124
25000.0	0.252	22.015	4.172	939.446	0.307	0.073	5.426	1.490	0.787	0.153
27500.0	0.243	20.329	4.449	937.819	0.347	0.091	5.525	1.656	0.883	0.193
30000.0	0.235	18.724	4.706	936.159	0.386	0.111	5.597	1.816	0.974	0.237
32500.0	0.226	17.198	4.943	934.464	0.426	0.133	5.645	1.970	1.061	0.286
35000.0	0.218	15.751	5.161	932.735	0.466	0.157	5.673	2.118	1.141	0.339
37500.0	0.210	14.383	5.360	930.969	0.506	0.184	5.682	2.258	1.216	0.397
40000.0	0.202	13.092	5.540	929.166	0.545	0.212	5.676	2.392	1.285	0.458
42500.0	0.195	11.877	5.702	927.326	0.584	0.243	5.657	2.518	1.347	0.523
45000.0	0.187	10.738	5.845	925.446	0.622	0.275	5.627	2.635	1.402	0.592
47500.0	0.180	9.674	5.970	923.526	0.658	0.308	5.588	2.745	1.452	0.664
50000.0	0.173	8.683	6.079	921.567	0.694	0.343	5.542	2.846	1.495	0.739
52500.0	0.166	7.764	6.170	919.568	0.728	0.378	5.491	2.939	1.532	0.817
55000.0	0.159	6.915	6.245	917.528	0.761	0.414	5.436	3.024	1.564	0.897
57500.0	0.153	6.135	6.305	915.448	0.791	0.450	5.378	3.100	1.590	0.979
60000.0	0.146	5.422	6.349	913.328	0.820	0.486	5.320	3.168	1.612	1.064
62500.0	0.140	4.773	6.379	911.169	0.847	0.521	5.261	3.229	1.629	1.150
65000.0	0.134	4.185	6.395	908.972	0.871	0.556	5.203	3.282	1.643	1.237
67500.0	0.129	3.656	6.398	906.738	0.894	0.590	5.146	3.327	1.653	1.326
70000.0	0.123	3.183	6.389	904.469	0.914	0.622	5.092	3.367	1.659	1.416

Table C.28 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.354	44.250	0.000	955.396	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.130	0.022	955.348	0.000	0.000	0.017	0.000	0.000	0.000
500.0	0.352	43.652	0.110	955.156	0.001	0.000	0.173	0.001	0.000	0.000
1000.0	0.350	43.061	0.219	954.915	0.003	0.000	0.373	0.006	0.000	0.000
1500.0	0.348	42.478	0.325	954.675	0.005	0.000	0.563	0.013	0.001	0.000
2000.0	0.347	41.902	0.431	954.435	0.007	0.000	0.745	0.023	0.002	0.000
2500.0	0.345	41.333	0.534	954.195	0.009	0.000	0.918	0.036	0.004	0.000
3000.0	0.343	40.771	0.636	953.955	0.012	0.000	1.083	0.050	0.006	0.000
3500.0	0.341	40.214	0.737	953.716	0.015	0.000	1.239	0.067	0.009	0.000
4000.0	0.339	39.664	0.836	953.477	0.017	0.001	1.389	0.084	0.013	0.000
4500.0	0.338	39.119	0.934	953.238	0.020	0.001	1.531	0.104	0.018	0.001
5000.0	0.336	38.580	1.031	953.000	0.023	0.001	1.666	0.124	0.023	0.001
5500.0	0.334	38.047	1.127	952.763	0.026	0.001	1.794	0.146	0.030	0.001
6000.0	0.332	37.518	1.221	952.525	0.030	0.001	1.917	0.169	0.037	0.001
6500.0	0.331	36.995	1.314	952.289	0.033	0.002	2.033	0.193	0.044	0.002
7000.0	0.329	36.476	1.406	952.053	0.037	0.002	2.143	0.217	0.053	0.002
7500.0	0.327	35.962	1.496	951.817	0.040	0.002	2.247	0.242	0.062	0.003
8000.0	0.326	35.453	1.586	951.582	0.044	0.003	2.346	0.268	0.072	0.004
8500.0	0.324	34.948	1.674	951.347	0.048	0.003	2.440	0.295	0.082	0.005
9000.0	0.322	34.447	1.762	951.114	0.051	0.004	2.529	0.322	0.093	0.006
9500.0	0.320	33.950	1.848	950.880	0.055	0.004	2.613	0.350	0.104	0.007
10000.0	0.319	33.457	1.934	950.648	0.059	0.004	2.692	0.378	0.116	0.008
10500.0	0.317	32.968	2.018	950.416	0.063	0.005	2.766	0.406	0.128	0.010
11000.0	0.316	32.483	2.101	950.185	0.068	0.006	2.836	0.435	0.141	0.011
11500.0	0.314	32.001	2.184	949.955	0.072	0.006	2.902	0.464	0.153	0.013
12000.0	0.312	31.523	2.265	949.726	0.076	0.007	2.964	0.493	0.166	0.015
12500.0	0.311	31.048	2.346	949.497	0.080	0.007	3.022	0.523	0.180	0.017
13000.0	0.309	30.576	2.425	949.269	0.085	0.008	3.076	0.552	0.193	0.019
13500.0	0.308	30.108	2.504	949.040	0.089	0.009	3.128	0.582	0.207	0.021
14000.0	0.306	29.643	2.582	948.812	0.094	0.010	3.176	0.612	0.220	0.024
14500.0	0.304	29.181	2.659	948.584	0.099	0.010	3.221	0.643	0.234	0.026
15000.0	0.303	28.723	2.736	948.355	0.103	0.011	3.264	0.673	0.248	0.029
15500.0	0.301	28.267	2.811	948.126	0.108	0.012	3.305	0.703	0.262	0.032
16000.0	0.300	27.815	2.886	947.895	0.113	0.013	3.343	0.733	0.277	0.035
16500.0	0.298	27.367	2.960	947.664	0.118	0.014	3.379	0.764	0.291	0.038
17000.0	0.297	26.922	3.033	947.431	0.123	0.015	3.414	0.794	0.306	0.042
17500.0	0.295	26.480	3.106	947.198	0.128	0.016	3.446	0.824	0.320	0.046
20000.0	0.287	24.323	3.457	946.009	0.154	0.022	3.583	0.976	0.396	0.067
22500.0	0.279	22.250	3.789	944.786	0.182	0.030	3.682	1.125	0.471	0.094
25000.0	0.271	20.263	4.101	943.525	0.211	0.038	3.748	1.273	0.545	0.126
27500.0	0.263	18.361	4.394	942.224	0.241	0.049	3.786	1.416	0.616	0.164
30000.0	0.255	16.547	4.668	940.879	0.273	0.061	3.799	1.555	0.682	0.208
32500.0	0.247	14.822	4.922	939.488	0.305	0.074	3.792	1.689	0.742	0.258
35000.0	0.239	13.188	5.156	938.046	0.337	0.089	3.768	1.816	0.797	0.314
37500.0	0.231	11.647	5.370	936.548	0.371	0.106	3.730	1.936	0.845	0.377
40000.0	0.222	10.203	5.563	934.991	0.404	0.124	3.681	2.048	0.887	0.445
42500.0	0.214	8.860	5.735	933.370	0.436	0.144	3.624	2.151	0.923	0.520
45000.0	0.206	7.619	5.886	931.680	0.469	0.165	3.562	2.245	0.951	0.601
47500.0	0.197	6.485	6.014	929.916	0.500	0.187	3.497	2.330	0.974	0.687
50000.0	0.189	5.459	6.121	928.076	0.531	0.209	3.431	2.405	0.991	0.779
52500.0	0.180	4.543	6.205	926.156	0.560	0.231	3.366	2.469	1.003	0.876
55000.0	0.172	3.736	6.267	924.154	0.587	0.254	3.304	2.524	1.011	0.977
57500.0	0.163	3.036	6.308	922.072	0.612	0.276	3.246	2.569	1.014	1.081
60000.0	0.155	2.438	6.329	919.910	0.636	0.297	3.193	2.605	1.015	1.189
62500.0	0.147	1.937	6.332	917.673	0.656	0.317	3.146	2.633	1.014	1.299
65000.0	0.139	1.524	6.317	915.366	0.675	0.335	3.104	2.654	1.011	1.409
67500.0	0.131	1.188	6.288	912.998	0.691	0.351	3.068	2.669	1.007	1.520
70000.0	0.124	0.920	6.247	910.577	0.705	0.366	3.037	2.678	1.002	1.630

Table C.29 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.354	44.250	0.000	955.396	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.130	0.023	955.341	0.000	0.000	0.020	0.000	0.000	0.000
500.0	0.352	43.654	0.114	955.119	0.002	0.000	0.198	0.001	0.000	0.000
1000.0	0.350	43.066	0.227	954.842	0.004	0.000	0.426	0.007	0.000	0.000
1500.0	0.348	42.487	0.338	954.566	0.006	0.000	0.644	0.015	0.001	0.000
2000.0	0.346	41.916	0.446	954.290	0.009	0.000	0.851	0.027	0.003	0.000
2500.0	0.344	41.353	0.553	954.015	0.011	0.000	1.048	0.041	0.005	0.000
3000.0	0.342	40.798	0.658	953.741	0.014	0.000	1.235	0.057	0.008	0.000
3500.0	0.340	40.249	0.762	953.467	0.017	0.001	1.414	0.075	0.012	0.000
4000.0	0.338	39.707	0.864	953.193	0.021	0.001	1.583	0.095	0.017	0.000
4500.0	0.336	39.171	0.964	952.921	0.024	0.001	1.745	0.116	0.023	0.001
5000.0	0.334	38.642	1.063	952.649	0.028	0.001	1.899	0.138	0.030	0.001
5500.0	0.332	38.118	1.160	952.377	0.032	0.002	2.045	0.162	0.038	0.001
6000.0	0.330	37.601	1.256	952.106	0.036	0.002	2.184	0.187	0.047	0.002
6500.0	0.328	37.089	1.350	951.837	0.040	0.002	2.316	0.212	0.057	0.002
7000.0	0.326	36.582	1.444	951.567	0.044	0.003	2.442	0.239	0.068	0.003
7500.0	0.324	36.080	1.535	951.299	0.048	0.003	2.561	0.266	0.079	0.004
8000.0	0.322	35.583	1.626	951.031	0.052	0.004	2.674	0.294	0.091	0.005
8500.0	0.320	35.091	1.715	950.764	0.057	0.004	2.782	0.322	0.104	0.006
9000.0	0.318	34.604	1.803	950.498	0.062	0.005	2.883	0.351	0.117	0.007
9500.0	0.317	34.121	1.890	950.233	0.066	0.005	2.979	0.381	0.131	0.009
10000.0	0.315	33.643	1.976	949.969	0.071	0.006	3.070	0.411	0.145	0.010
10500.0	0.313	33.169	2.061	949.706	0.076	0.007	3.156	0.441	0.160	0.012
11000.0	0.311	32.698	2.144	949.444	0.081	0.007	3.237	0.472	0.175	0.014
11500.0	0.309	32.232	2.227	949.182	0.086	0.008	3.314	0.503	0.190	0.016
12000.0	0.308	31.770	2.308	948.922	0.092	0.009	3.386	0.534	0.206	0.018
12500.0	0.306	31.311	2.388	948.662	0.097	0.010	3.454	0.565	0.222	0.020
13000.0	0.304	30.855	2.468	948.403	0.102	0.011	3.517	0.597	0.238	0.022
13500.0	0.302	30.404	2.546	948.144	0.107	0.012	3.577	0.629	0.254	0.025
14000.0	0.301	29.955	2.624	947.886	0.113	0.013	3.634	0.661	0.270	0.028
14500.0	0.299	29.510	2.700	947.628	0.118	0.014	3.688	0.692	0.286	0.031
15000.0	0.297	29.069	2.776	947.369	0.124	0.015	3.739	0.724	0.303	0.034
15500.0	0.296	28.631	2.851	947.110	0.129	0.016	3.787	0.757	0.319	0.038
16000.0	0.294	28.197	2.925	946.851	0.135	0.017	3.832	0.789	0.336	0.041
16500.0	0.292	27.766	2.998	946.590	0.141	0.019	3.876	0.821	0.353	0.045
17000.0	0.291	27.338	3.071	946.329	0.146	0.020	3.917	0.853	0.370	0.049
17500.0	0.289	26.914	3.142	946.067	0.152	0.021	3.956	0.885	0.387	0.053
18000.0	0.287	26.494	3.213	945.803	0.158	0.023	3.993	0.917	0.404	0.057
20000.0	0.280	24.846	3.488	944.737	0.182	0.029	4.124	1.044	0.474	0.077
22500.0	0.272	22.866	3.813	943.374	0.214	0.039	4.251	1.201	0.561	0.105
25000.0	0.264	20.971	4.118	941.976	0.248	0.050	4.343	1.354	0.646	0.140
27500.0	0.256	19.161	4.403	940.543	0.283	0.063	4.404	1.504	0.727	0.180
30000.0	0.247	17.438	4.667	939.070	0.318	0.078	4.438	1.649	0.804	0.225
32500.0	0.239	15.799	4.913	937.558	0.354	0.095	4.450	1.788	0.874	0.276
35000.0	0.231	14.247	5.138	936.001	0.390	0.113	4.442	1.920	0.939	0.332
37500.0	0.223	12.781	5.344	934.400	0.427	0.134	4.419	2.046	0.997	0.394
40000.0	0.215	11.402	5.531	932.750	0.463	0.156	4.382	2.163	1.049	0.461
42500.0	0.207	10.111	5.697	931.049	0.498	0.179	4.335	2.272	1.093	0.532
45000.0	0.199	8.910	5.845	929.295	0.533	0.204	4.279	2.373	1.131	0.609
47500.0	0.191	7.798	5.972	927.486	0.567	0.230	4.218	2.464	1.162	0.690
50000.0	0.183	6.777	6.080	925.618	0.599	0.257	4.154	2.546	1.187	0.775
52500.0	0.175	5.846	6.169	923.690	0.631	0.284	4.087	2.619	1.207	0.863
55000.0	0.168	5.004	6.239	921.702	0.660	0.312	4.021	2.683	1.221	0.955
57500.0	0.160	4.250	6.290	919.653	0.688	0.339	3.956	2.738	1.231	1.050
60000.0	0.153	3.582	6.323	917.543	0.714	0.365	3.894	2.783	1.236	1.148
62500.0	0.146	2.996	6.340	915.374	0.737	0.391	3.835	2.821	1.238	1.247
65000.0	0.139	2.487	6.341	913.148	0.759	0.415	3.780	2.851	1.238	1.347
67500.0	0.132	2.050	6.327	910.869	0.778	0.438	3.730	2.875	1.235	1.448
70000.0	0.126	1.680	6.300	908.540	0.795	0.459	3.685	2.892	1.231	1.548

Table C.30 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.354	44.250	0.000	955.396	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.131	0.024	955.331	0.000	0.000	0.023	0.000	0.000	0.000
500.0	0.351	43.658	0.121	955.069	0.002	0.000	0.232	0.002	0.000	0.000
1000.0	0.349	43.075	0.239	954.741	0.005	0.000	0.499	0.008	0.000	0.000
1500.0	0.347	42.503	0.355	954.415	0.008	0.000	0.753	0.018	0.002	0.000
2000.0	0.344	41.940	0.468	954.090	0.011	0.000	0.995	0.031	0.004	0.000
2500.0	0.342	41.385	0.580	953.765	0.014	0.000	1.225	0.046	0.007	0.000
3000.0	0.339	40.839	0.689	953.442	0.018	0.001	1.445	0.064	0.011	0.000
3500.0	0.337	40.301	0.797	953.119	0.022	0.001	1.654	0.084	0.017	0.000
4000.0	0.335	39.771	0.903	952.797	0.026	0.001	1.853	0.106	0.024	0.001
4500.0	0.332	39.247	1.006	952.476	0.030	0.001	2.043	0.129	0.032	0.001
5000.0	0.330	38.731	1.108	952.156	0.035	0.002	2.224	0.153	0.041	0.001
5500.0	0.328	38.221	1.209	951.837	0.040	0.002	2.396	0.179	0.051	0.002
6000.0	0.326	37.718	1.308	951.519	0.045	0.003	2.561	0.206	0.063	0.002
6500.0	0.323	37.220	1.405	951.201	0.050	0.003	2.718	0.233	0.075	0.003
7000.0	0.321	36.729	1.500	950.885	0.055	0.004	2.868	0.262	0.089	0.004
7500.0	0.319	36.243	1.594	950.570	0.061	0.005	3.010	0.291	0.103	0.005
8000.0	0.317	35.763	1.687	950.255	0.066	0.005	3.146	0.321	0.118	0.006
8500.0	0.315	35.287	1.778	949.942	0.072	0.006	3.275	0.351	0.134	0.007
9000.0	0.313	34.818	1.868	949.629	0.078	0.007	3.398	0.382	0.150	0.009
9500.0	0.311	34.353	1.956	949.318	0.084	0.008	3.515	0.414	0.167	0.010
10000.0	0.309	33.892	2.043	949.007	0.090	0.009	3.626	0.446	0.184	0.012
10500.0	0.306	33.437	2.129	948.698	0.096	0.010	3.732	0.479	0.202	0.014
11000.0	0.304	32.985	2.214	948.390	0.102	0.011	3.832	0.512	0.220	0.016
11500.0	0.302	32.539	2.297	948.083	0.108	0.012	3.927	0.545	0.238	0.018
12000.0	0.300	32.096	2.379	947.776	0.115	0.013	4.018	0.578	0.257	0.021
12500.0	0.298	31.657	2.460	947.471	0.121	0.014	4.103	0.612	0.276	0.024
13000.0	0.297	31.223	2.540	947.167	0.128	0.016	4.184	0.646	0.295	0.026
13500.0	0.295	30.792	2.619	946.863	0.134	0.017	4.261	0.680	0.314	0.029
14000.0	0.293	30.365	2.697	946.560	0.141	0.019	4.335	0.715	0.334	0.032
14500.0	0.291	29.942	2.774	946.257	0.148	0.020	4.404	0.749	0.353	0.036
15000.0	0.289	29.522	2.850	945.954	0.154	0.022	4.470	0.784	0.373	0.039
15500.0	0.287	29.106	2.924	945.651	0.161	0.023	4.533	0.818	0.392	0.043
16000.0	0.285	28.694	2.998	945.348	0.168	0.025	4.593	0.853	0.412	0.047
16500.0	0.283	28.285	3.071	945.045	0.175	0.027	4.650	0.888	0.432	0.051
17000.0	0.281	27.880	3.143	944.741	0.182	0.029	4.705	0.923	0.452	0.055
17500.0	0.280	27.478	3.214	944.436	0.189	0.031	4.757	0.957	0.471	0.060
18000.0	0.278	27.080	3.284	944.130	0.196	0.033	4.807	0.992	0.491	0.064
18500.0	0.276	26.686	3.353	943.824	0.203	0.035	4.855	1.027	0.511	0.069
19000.0	0.274	26.295	3.421	943.516	0.211	0.037	4.901	1.061	0.531	0.074
19500.0	0.272	25.908	3.489	943.207	0.218	0.040	4.945	1.096	0.552	0.079
20000.0	0.271	25.524	3.555	942.898	0.225	0.042	4.987	1.131	0.572	0.085
22500.0	0.262	23.659	3.875	941.329	0.263	0.055	5.173	1.301	0.673	0.115
25000.0	0.253	21.880	4.173	939.729	0.302	0.070	5.320	1.469	0.772	0.149
27500.0	0.244	20.187	4.451	938.097	0.341	0.088	5.433	1.633	0.868	0.189
30000.0	0.236	18.576	4.708	936.430	0.381	0.108	5.517	1.792	0.960	0.233
32500.0	0.227	17.047	4.946	934.730	0.421	0.130	5.575	1.946	1.046	0.282
35000.0	0.219	15.598	5.163	932.994	0.462	0.155	5.611	2.094	1.127	0.335
37500.0	0.211	14.230	5.362	931.222	0.502	0.181	5.627	2.235	1.202	0.392
40000.0	0.203	12.940	5.541	929.412	0.541	0.210	5.627	2.369	1.270	0.454
42500.0	0.195	11.727	5.702	927.563	0.581	0.240	5.612	2.496	1.333	0.519
45000.0	0.188	10.592	5.844	925.676	0.619	0.272	5.586	2.615	1.388	0.587
47500.0	0.180	9.532	5.968	923.748	0.656	0.305	5.551	2.726	1.438	0.659
50000.0	0.173	8.546	6.075	921.781	0.692	0.340	5.508	2.828	1.482	0.734
52500.0	0.166	7.633	6.165	919.772	0.726	0.375	5.460	2.923	1.519	0.812
55000.0	0.159	6.791	6.238	917.724	0.758	0.411	5.407	3.008	1.552	0.892
57500.0	0.153	6.018	6.296	915.634	0.789	0.448	5.352	3.086	1.579	0.975
60000.0	0.146	5.312	6.338	913.505	0.818	0.484	5.295	3.155	1.601	1.059
62500.0	0.140	4.671	6.366	911.337	0.845	0.519	5.238	3.216	1.618	1.146
65000.0	0.134	4.091	6.381	909.131	0.869	0.554	5.182	3.270	1.632	1.234
67500.0	0.129	3.570	6.382	906.888	0.892	0.588	5.127	3.317	1.643	1.323
70000.0	0.123	3.104	6.372	904.610	0.912	0.620	5.074	3.357	1.650	1.412

Table C.31 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.362	45.231	0.000	954.407	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.361	45.111	0.023	954.356	0.000	0.000	0.020	0.000	0.000	0.000
500.0	0.360	44.633	0.112	954.151	0.002	0.000	0.189	0.001	0.000	0.000
1000.0	0.358	44.044	0.223	953.894	0.003	0.000	0.402	0.006	0.000	0.000
1500.0	0.356	43.462	0.332	953.636	0.005	0.000	0.606	0.014	0.001	0.000
2000.0	0.354	42.888	0.439	953.379	0.008	0.000	0.800	0.025	0.002	0.000
2500.0	0.352	42.320	0.545	953.121	0.010	0.000	0.986	0.038	0.004	0.000
3000.0	0.350	41.760	0.649	952.863	0.013	0.000	1.164	0.053	0.007	0.000
3500.0	0.348	41.207	0.752	952.604	0.016	0.000	1.334	0.070	0.010	0.000
4000.0	0.346	40.660	0.853	952.345	0.019	0.001	1.497	0.088	0.015	0.000
4500.0	0.344	40.119	0.953	952.086	0.022	0.001	1.653	0.109	0.020	0.001
5000.0	0.342	39.583	1.051	951.827	0.025	0.001	1.802	0.130	0.026	0.001
5500.0	0.340	39.054	1.148	951.568	0.029	0.001	1.944	0.152	0.034	0.001
6000.0	0.338	38.531	1.244	951.308	0.033	0.002	2.081	0.176	0.042	0.002
6500.0	0.337	38.013	1.339	951.048	0.036	0.002	2.211	0.200	0.050	0.002
7000.0	0.335	37.500	1.432	950.787	0.040	0.002	2.336	0.226	0.060	0.003
7500.0	0.333	36.992	1.524	950.527	0.045	0.003	2.455	0.252	0.071	0.004
8000.0	0.331	36.489	1.615	950.266	0.049	0.003	2.569	0.279	0.082	0.004
8500.0	0.329	35.991	1.705	950.005	0.053	0.004	2.678	0.306	0.094	0.006
9000.0	0.327	35.498	1.794	949.744	0.058	0.004	2.781	0.334	0.106	0.007
9500.0	0.325	35.010	1.882	949.483	0.062	0.005	2.881	0.362	0.119	0.008
10000.0	0.324	34.526	1.968	949.222	0.067	0.005	2.975	0.391	0.133	0.009
10500.0	0.322	34.046	2.053	948.960	0.072	0.006	3.065	0.420	0.147	0.011
11000.0	0.320	33.571	2.138	948.698	0.077	0.007	3.151	0.450	0.162	0.013
11500.0	0.318	33.100	2.221	948.437	0.082	0.007	3.233	0.480	0.177	0.015
12000.0	0.316	32.633	2.303	948.175	0.087	0.008	3.310	0.510	0.192	0.017
12500.0	0.315	32.171	2.385	947.913	0.092	0.009	3.384	0.541	0.208	0.019
13000.0	0.313	31.712	2.465	947.651	0.098	0.010	3.454	0.572	0.224	0.021
13500.0	0.311	31.257	2.544	947.389	0.103	0.011	3.520	0.603	0.240	0.024
14000.0	0.309	30.806	2.623	947.127	0.108	0.012	3.583	0.634	0.256	0.027
14500.0	0.307	30.359	2.700	946.865	0.114	0.013	3.642	0.665	0.273	0.030
15000.0	0.306	29.916	2.776	946.603	0.120	0.014	3.698	0.697	0.290	0.033
15500.0	0.304	29.476	2.852	946.341	0.125	0.015	3.751	0.728	0.306	0.036
16000.0	0.302	29.039	2.927	946.079	0.131	0.016	3.800	0.760	0.323	0.040
16500.0	0.301	28.606	3.000	945.818	0.137	0.018	3.846	0.792	0.340	0.044
17000.0	0.299	28.176	3.073	945.556	0.143	0.019	3.890	0.824	0.357	0.048
17500.0	0.297	27.750	3.145	945.295	0.148	0.020	3.931	0.856	0.374	0.052
18000.0	0.295	27.326	3.217	945.033	0.154	0.022	3.968	0.888	0.391	0.056
18500.0	0.294	26.906	3.287	944.771	0.160	0.023	4.004	0.920	0.408	0.061
19000.0	0.292	26.489	3.357	944.509	0.166	0.025	4.037	0.951	0.424	0.065
19500.0	0.290	26.076	3.426	944.247	0.172	0.026	4.068	0.983	0.441	0.070
20000.0	0.289	25.665	3.494	943.984	0.178	0.028	4.097	1.015	0.458	0.075
20500.0	0.287	25.258	3.561	943.720	0.184	0.030	4.124	1.047	0.474	0.081
21000.0	0.285	24.854	3.627	943.455	0.190	0.031	4.150	1.078	0.491	0.086
21500.0	0.284	24.453	3.693	943.189	0.197	0.033	4.174	1.109	0.508	0.092
22000.0	0.282	24.055	3.758	942.922	0.203	0.035	4.196	1.141	0.524	0.098
22500.0	0.280	23.661	3.822	942.653	0.209	0.037	4.217	1.172	0.541	0.104
25000.0	0.272	21.740	4.131	941.290	0.242	0.048	4.303	1.324	0.623	0.138
27500.0	0.264	19.902	4.420	939.890	0.275	0.060	4.360	1.471	0.703	0.177
30000.0	0.256	18.148	4.690	938.451	0.310	0.075	4.391	1.614	0.778	0.222
32500.0	0.247	16.477	4.941	936.971	0.346	0.091	4.401	1.750	0.848	0.272
35000.0	0.239	14.891	5.172	935.449	0.381	0.109	4.393	1.879	0.912	0.328
37500.0	0.231	13.390	5.384	933.881	0.417	0.129	4.369	2.002	0.969	0.389
40000.0	0.223	11.976	5.577	932.265	0.453	0.151	4.332	2.117	1.020	0.455
42500.0	0.215	10.648	5.750	930.598	0.489	0.174	4.286	2.223	1.065	0.526
45000.0	0.207	9.409	5.904	928.877	0.524	0.199	4.231	2.322	1.102	0.602
47500.0	0.199	8.260	6.038	927.101	0.558	0.225	4.170	2.411	1.134	0.683
50000.0	0.191	7.200	6.153	925.267	0.592	0.251	4.106	2.492	1.159	0.768
52500.0	0.183	6.231	6.249	923.372	0.623	0.278	4.040	2.564	1.179	0.856
55000.0	0.175	5.351	6.325	921.417	0.654	0.305	3.974	2.627	1.193	0.948
57500.0	0.168	4.560	6.383	919.400	0.682	0.332	3.908	2.681	1.203	1.043
60000.0	0.160	3.856	6.423	917.321	0.709	0.359	3.845	2.726	1.209	1.141
62500.0	0.153	3.235	6.446	915.181	0.734	0.384	3.786	2.764	1.211	1.240
65000.0	0.146	2.694	6.453	912.983	0.756	0.409	3.730	2.794	1.211	1.340
67500.0	0.139	2.227	6.444	910.729	0.776	0.431	3.679	2.818	1.209	1.442
70000.0	0.132	1.829	6.422	908.424	0.794	0.452	3.633	2.836	1.204	1.543

Table C.32 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.362	45.231	0.000	954.407	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.361	45.112	0.023	954.349	0.000	0.000	0.023	0.000	0.000	0.000
500.0	0.360	44.636	0.117	954.115	0.002	0.000	0.214	0.002	0.000	0.000
1000.0	0.357	44.049	0.231	953.823	0.004	0.000	0.454	0.007	0.000	0.000
1500.0	0.355	43.471	0.344	953.530	0.006	0.000	0.684	0.016	0.001	0.000
2000.0	0.353	42.902	0.455	953.237	0.009	0.000	0.904	0.028	0.003	0.000
2500.0	0.351	42.341	0.564	952.944	0.012	0.000	1.114	0.042	0.005	0.000
3000.0	0.349	41.787	0.671	952.651	0.015	0.000	1.315	0.059	0.009	0.000
3500.0	0.346	41.241	0.776	952.358	0.019	0.001	1.507	0.077	0.013	0.000
4000.0	0.344	40.702	0.880	952.065	0.022	0.001	1.691	0.098	0.019	0.000
4500.0	0.342	40.170	0.982	951.771	0.026	0.001	1.867	0.119	0.026	0.001
5000.0	0.340	39.644	1.083	951.477	0.030	0.001	2.035	0.142	0.033	0.001
5500.0	0.338	39.125	1.182	951.183	0.034	0.002	2.197	0.167	0.042	0.002
6000.0	0.336	38.611	1.280	950.889	0.039	0.002	2.351	0.192	0.052	0.002
6500.0	0.334	38.104	1.377	950.594	0.043	0.003	2.500	0.218	0.063	0.003
7000.0	0.332	37.602	1.472	950.300	0.048	0.003	2.642	0.245	0.075	0.003
7500.0	0.329	37.106	1.565	950.005	0.053	0.004	2.777	0.273	0.087	0.004
8000.0	0.327	36.616	1.658	949.710	0.058	0.004	2.908	0.301	0.101	0.005
8500.0	0.325	36.130	1.749	949.416	0.063	0.005	3.032	0.330	0.115	0.007
9000.0	0.323	35.650	1.838	949.121	0.068	0.005	3.151	0.360	0.130	0.008
9500.0	0.321	35.175	1.927	948.825	0.074	0.006	3.265	0.390	0.145	0.009
10000.0	0.319	34.705	2.014	948.530	0.079	0.007	3.374	0.420	0.161	0.011
10500.0	0.317	34.239	2.100	948.235	0.085	0.008	3.478	0.451	0.178	0.013
11000.0	0.315	33.778	2.185	947.940	0.091	0.009	3.578	0.482	0.195	0.015
11500.0	0.313	33.322	2.268	947.645	0.097	0.010	3.673	0.514	0.212	0.017
12000.0	0.311	32.870	2.351	947.350	0.103	0.011	3.764	0.546	0.230	0.019
12500.0	0.309	32.423	2.432	947.054	0.109	0.012	3.850	0.578	0.249	0.022
13000.0	0.308	31.980	2.513	946.759	0.116	0.013	3.932	0.611	0.267	0.025
13500.0	0.306	31.541	2.592	946.464	0.122	0.014	4.011	0.643	0.286	0.028
14000.0	0.304	31.106	2.670	946.169	0.128	0.016	4.085	0.676	0.305	0.031
14500.0	0.302	30.675	2.748	945.874	0.135	0.017	4.156	0.709	0.324	0.034
15000.0	0.300	30.247	2.824	945.579	0.141	0.018	4.223	0.742	0.343	0.038
15500.0	0.298	29.824	2.899	945.284	0.148	0.020	4.287	0.776	0.363	0.041
16000.0	0.296	29.404	2.973	944.990	0.154	0.021	4.347	0.809	0.382	0.045
16500.0	0.294	28.989	3.047	944.695	0.161	0.023	4.404	0.843	0.402	0.049
17000.0	0.292	28.576	3.119	944.401	0.168	0.025	4.458	0.876	0.421	0.054
17500.0	0.291	28.167	3.191	944.107	0.174	0.026	4.509	0.910	0.440	0.058
18000.0	0.289	27.762	3.261	943.813	0.181	0.028	4.557	0.943	0.460	0.063
18500.0	0.287	27.359	3.331	943.519	0.188	0.030	4.602	0.977	0.479	0.068
19000.0	0.285	26.961	3.400	943.224	0.195	0.032	4.644	1.011	0.498	0.073
19500.0	0.283	26.565	3.468	942.930	0.202	0.034	4.684	1.044	0.518	0.078
20000.0	0.282	26.173	3.535	942.635	0.209	0.036	4.722	1.078	0.537	0.083
20500.0	0.280	25.784	3.601	942.340	0.216	0.038	4.757	1.111	0.556	0.089
21000.0	0.278	25.398	3.666	942.044	0.223	0.041	4.791	1.144	0.575	0.095
21500.0	0.276	25.015	3.731	941.748	0.230	0.043	4.823	1.178	0.594	0.101
22000.0	0.275	24.636	3.795	941.450	0.237	0.045	4.853	1.211	0.613	0.107
22500.0	0.273	24.260	3.858	941.151	0.244	0.048	4.881	1.243	0.631	0.113
23000.0	0.271	23.888	3.920	940.852	0.252	0.050	4.908	1.276	0.650	0.120
23500.0	0.269	23.518	3.981	940.551	0.259	0.053	4.934	1.309	0.669	0.127
24000.0	0.268	23.152	4.041	940.248	0.266	0.056	4.958	1.341	0.687	0.134
25000.0	0.264	22.431	4.160	939.640	0.281	0.061	5.002	1.405	0.724	0.149
27500.0	0.256	20.683	4.442	938.095	0.319	0.077	5.092	1.561	0.815	0.189
30000.0	0.247	19.018	4.705	936.516	0.358	0.095	5.154	1.711	0.901	0.234
32500.0	0.239	17.435	4.948	934.900	0.397	0.115	5.193	1.856	0.983	0.284
35000.0	0.230	15.932	5.172	933.246	0.436	0.137	5.212	1.994	1.059	0.339
37500.0	0.222	14.509	5.377	931.554	0.475	0.161	5.212	2.125	1.128	0.398
40000.0	0.214	13.166	5.563	929.822	0.514	0.188	5.198	2.249	1.191	0.462
42500.0	0.207	11.902	5.731	928.049	0.552	0.216	5.172	2.365	1.248	0.530
45000.0	0.199	10.716	5.880	926.234	0.589	0.245	5.135	2.473	1.297	0.602
47500.0	0.191	9.608	6.012	924.375	0.626	0.276	5.090	2.573	1.341	0.677
50000.0	0.184	8.577	6.126	922.471	0.661	0.308	5.038	2.665	1.378	0.755
52500.0	0.177	7.623	6.222	920.522	0.695	0.340	4.982	2.748	1.410	0.837
55000.0	0.170	6.744	6.302	918.528	0.727	0.373	4.923	2.824	1.436	0.921
57500.0	0.163	5.938	6.365	916.488	0.758	0.406	4.863	2.891	1.457	1.007
60000.0	0.156	5.204	6.413	914.403	0.787	0.439	4.801	2.950	1.473	1.095
62500.0	0.150	4.539	6.445	912.273	0.814	0.472	4.741	3.002	1.485	1.184
65000.0	0.144	3.941	6.463	910.099	0.839	0.503	4.682	3.047	1.493	1.275
67500.0	0.137	3.406	6.468	907.883	0.862	0.533	4.625	3.084	1.498	1.367
70000.0	0.132	2.931	6.459	905.627	0.883	0.562	4.571	3.116	1.500	1.459

Table C.33 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.362	45.231	0.000	954.407	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.361	45.112	0.025	954.340	0.000	0.000	0.026	0.000	0.000	0.000
500.0	0.359	44.640	0.122	954.069	0.002	0.000	0.245	0.002	0.000	0.000
1000.0	0.357	44.058	0.242	953.730	0.005	0.000	0.521	0.008	0.000	0.000
1500.0	0.354	43.487	0.360	953.391	0.008	0.000	0.784	0.018	0.002	0.000
2000.0	0.352	42.925	0.475	953.052	0.011	0.000	1.037	0.031	0.004	0.000
2500.0	0.349	42.372	0.588	952.713	0.015	0.000	1.278	0.047	0.007	0.000
3000.0	0.347	41.828	0.699	952.374	0.019	0.001	1.509	0.065	0.012	0.000
3500.0	0.344	41.292	0.809	952.035	0.023	0.001	1.730	0.085	0.017	0.000
4000.0	0.342	40.763	0.916	951.696	0.027	0.001	1.942	0.107	0.025	0.001
4500.0	0.339	40.241	1.022	951.357	0.032	0.002	2.146	0.130	0.033	0.001
5000.0	0.337	39.727	1.126	951.018	0.037	0.002	2.342	0.154	0.043	0.001
5500.0	0.334	39.220	1.228	950.679	0.042	0.002	2.530	0.180	0.053	0.002
6000.0	0.332	38.719	1.329	950.339	0.047	0.003	2.710	0.206	0.065	0.002
6500.0	0.330	38.224	1.428	950.000	0.053	0.004	2.884	0.234	0.078	0.003
7000.0	0.327	37.736	1.525	949.660	0.059	0.004	3.051	0.262	0.092	0.004
7500.0	0.325	37.254	1.621	949.320	0.065	0.005	3.211	0.292	0.107	0.005
8000.0	0.323	36.777	1.716	948.980	0.071	0.006	3.365	0.322	0.123	0.006
8500.0	0.320	36.306	1.809	948.641	0.077	0.007	3.513	0.352	0.140	0.008
9000.0	0.318	35.841	1.900	948.301	0.084	0.008	3.655	0.383	0.157	0.009
9500.0	0.316	35.381	1.991	947.961	0.091	0.009	3.792	0.415	0.175	0.011
10000.0	0.313	34.926	2.080	947.621	0.097	0.010	3.923	0.447	0.194	0.013
10500.0	0.311	34.476	2.167	947.281	0.104	0.011	4.049	0.480	0.213	0.015
11000.0	0.309	34.032	2.254	946.941	0.111	0.012	4.170	0.513	0.232	0.017
11500.0	0.307	33.592	2.339	946.602	0.118	0.014	4.287	0.546	0.252	0.019
12000.0	0.305	33.157	2.422	946.262	0.125	0.015	4.398	0.580	0.273	0.022
12500.0	0.302	32.726	2.505	945.922	0.133	0.017	4.506	0.614	0.293	0.024
13000.0	0.300	32.300	2.587	945.583	0.140	0.018	4.608	0.649	0.315	0.027
13500.0	0.298	31.878	2.667	945.243	0.148	0.020	4.707	0.683	0.336	0.030
14000.0	0.296	31.461	2.746	944.904	0.155	0.022	4.801	0.718	0.357	0.034
14500.0	0.294	31.048	2.824	944.564	0.163	0.024	4.892	0.754	0.379	0.037
15000.0	0.292	30.639	2.901	944.225	0.171	0.025	4.978	0.789	0.401	0.041
15500.0	0.290	30.234	2.976	943.886	0.178	0.028	5.061	0.825	0.423	0.045
16000.0	0.288	29.833	3.051	943.547	0.186	0.030	5.140	0.860	0.445	0.049
16500.0	0.286	29.436	3.124	943.208	0.194	0.032	5.216	0.896	0.467	0.053
17000.0	0.284	29.043	3.197	942.869	0.202	0.034	5.288	0.932	0.489	0.057
17500.0	0.282	28.654	3.269	942.531	0.210	0.037	5.357	0.969	0.511	0.062
18000.0	0.280	28.268	3.339	942.193	0.218	0.039	5.423	1.005	0.533	0.067
18500.0	0.278	27.886	3.409	941.854	0.226	0.042	5.485	1.041	0.555	0.071
19000.0	0.276	27.507	3.477	941.516	0.234	0.044	5.545	1.077	0.577	0.077
19500.0	0.274	27.132	3.545	941.178	0.242	0.047	5.602	1.114	0.598	0.082
20000.0	0.272	26.760	3.612	940.839	0.251	0.050	5.656	1.150	0.620	0.087
20500.0	0.270	26.391	3.677	940.500	0.259	0.053	5.708	1.186	0.642	0.093
21000.0	0.268	26.026	3.742	940.161	0.267	0.056	5.757	1.223	0.663	0.099
21500.0	0.266	25.664	3.806	939.822	0.275	0.059	5.804	1.259	0.685	0.105
22000.0	0.264	25.306	3.869	939.482	0.283	0.062	5.849	1.295	0.706	0.111
22500.0	0.262	24.951	3.932	939.141	0.291	0.065	5.893	1.331	0.727	0.117
23000.0	0.261	24.599	3.993	938.800	0.300	0.069	5.934	1.367	0.749	0.124
23500.0	0.259	24.250	4.054	938.457	0.308	0.072	5.974	1.402	0.770	0.131
24000.0	0.257	23.905	4.113	938.114	0.316	0.076	6.011	1.438	0.791	0.138
24500.0	0.255	23.563	4.172	937.770	0.324	0.080	6.048	1.474	0.812	0.145
25000.0	0.253	23.224	4.230	937.424	0.333	0.083	6.083	1.509	0.833	0.152
25500.0	0.251	22.889	4.288	937.078	0.341	0.087	6.116	1.544	0.853	0.159
26000.0	0.250	22.557	4.344	936.730	0.349	0.091	6.148	1.579	0.874	0.167
26500.0	0.248	22.228	4.400	936.382	0.358	0.095	6.179	1.614	0.895	0.174
27000.0	0.246	21.902	4.454	936.032	0.366	0.100	6.208	1.648	0.915	0.182
27500.0	0.244	21.579	4.508	935.680	0.375	0.104	6.237	1.683	0.935	0.190
30000.0	0.236	20.014	4.766	933.905	0.417	0.127	6.361	1.851	1.035	0.233
32500.0	0.227	18.528	5.004	932.098	0.460	0.153	6.459	2.014	1.132	0.280
35000.0	0.219	17.118	5.223	930.258	0.502	0.181	6.534	2.171	1.223	0.330
37500.0	0.211	15.784	5.422	928.387	0.545	0.211	6.589	2.322	1.309	0.385
40000.0	0.203	14.523	5.603	926.482	0.587	0.244	6.626	2.466	1.389	0.442
42500.0	0.195	13.334	5.766	924.545	0.628	0.279	6.647	2.604	1.464	0.503
45000.0	0.188	12.215	5.912	922.574	0.668	0.316	6.656	2.734	1.533	0.567
47500.0	0.181	11.166	6.040	920.570	0.707	0.354	6.652	2.858	1.596	0.633
50000.0	0.174	10.183	6.152	918.534	0.745	0.394	6.639	2.974	1.654	0.702
52500.0	0.167	9.265	6.248	916.464	0.781	0.435	6.618	3.082	1.706	0.773
55000.0	0.161	8.411	6.329	914.362	0.816	0.477	6.591	3.184	1.753	0.847
57500.0	0.154	7.618	6.395	912.229	0.849	0.519	6.558	3.278	1.795	0.922
60000.0	0.149	6.883	6.447	910.064	0.880	0.562	6.521	3.365	1.833	0.999
62500.0	0.143	6.206	6.485	907.869	0.909	0.605	6.482	3.444	1.865	1.078
65000.0	0.138	5.582	6.512	905.644	0.936	0.647	6.440	3.517	1.894	1.158
67500.0	0.133	5.010	6.526	903.392	0.961	0.689	6.397	3.583	1.918	1.240
70000.0	0.128	4.487	6.529	901.112	0.984	0.730	6.354	3.643	1.940	1.322

Table C.34 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.355	44.360	0.000	955.285	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.355	44.240	0.022	955.236	0.000	0.000	0.018	0.000	0.000	0.000
500.0	0.353	43.762	0.111	955.038	0.001	0.000	0.178	0.001	0.000	0.000
1000.0	0.351	43.171	0.220	954.791	0.003	0.000	0.384	0.006	0.000	0.000
1500.0	0.349	42.588	0.328	954.543	0.005	0.000	0.580	0.014	0.001	0.000
2000.0	0.347	42.013	0.433	954.295	0.007	0.000	0.768	0.024	0.002	0.000
2500.0	0.345	41.444	0.538	954.047	0.010	0.000	0.947	0.037	0.004	0.000
3000.0	0.344	40.883	0.641	953.799	0.012	0.000	1.119	0.051	0.006	0.000
3500.0	0.342	40.327	0.742	953.551	0.015	0.000	1.282	0.068	0.010	0.000
4000.0	0.340	39.778	0.842	953.302	0.018	0.001	1.438	0.086	0.014	0.000
4500.0	0.338	39.235	0.941	953.054	0.021	0.001	1.587	0.106	0.019	0.001
5000.0	0.336	38.698	1.038	952.805	0.024	0.001	1.730	0.127	0.025	0.001
5500.0	0.334	38.167	1.134	952.557	0.027	0.001	1.866	0.149	0.031	0.001
6000.0	0.333	37.640	1.229	952.308	0.031	0.001	1.995	0.172	0.039	0.002
6500.0	0.331	37.119	1.323	952.059	0.034	0.002	2.119	0.196	0.047	0.002
7000.0	0.329	36.604	1.415	951.811	0.038	0.002	2.237	0.221	0.056	0.003
7500.0	0.327	36.093	1.506	951.562	0.042	0.002	2.349	0.247	0.066	0.003
8000.0	0.325	35.587	1.596	951.313	0.046	0.003	2.456	0.274	0.077	0.004
8500.0	0.324	35.086	1.685	951.065	0.050	0.003	2.558	0.301	0.088	0.005
9000.0	0.322	34.590	1.773	950.816	0.054	0.004	2.655	0.328	0.100	0.006
9500.0	0.320	34.098	1.860	950.568	0.058	0.004	2.747	0.356	0.112	0.007
10000.0	0.318	33.610	1.946	950.319	0.063	0.005	2.835	0.385	0.125	0.009
10500.0	0.317	33.127	2.030	950.071	0.067	0.005	2.917	0.414	0.138	0.010
11000.0	0.315	32.648	2.114	949.823	0.072	0.006	2.996	0.443	0.152	0.012
11500.0	0.313	32.172	2.196	949.575	0.076	0.007	3.071	0.473	0.166	0.014
12000.0	0.312	31.701	2.278	949.326	0.081	0.007	3.141	0.503	0.181	0.016
12500.0	0.310	31.234	2.359	949.079	0.086	0.008	3.207	0.534	0.196	0.018
13000.0	0.308	30.771	2.438	948.831	0.091	0.009	3.270	0.564	0.211	0.020
13500.0	0.307	30.311	2.517	948.584	0.096	0.010	3.329	0.595	0.226	0.023
14000.0	0.305	29.855	2.595	948.337	0.101	0.011	3.384	0.626	0.241	0.026
14500.0	0.303	29.402	2.672	948.090	0.106	0.012	3.436	0.657	0.257	0.028
15000.0	0.302	28.953	2.748	947.843	0.111	0.013	3.484	0.689	0.272	0.032
15500.0	0.300	28.507	2.823	947.597	0.116	0.014	3.530	0.720	0.288	0.035
16000.0	0.298	28.064	2.897	947.351	0.122	0.015	3.572	0.752	0.304	0.038
16500.0	0.297	27.625	2.971	947.106	0.127	0.016	3.611	0.783	0.319	0.042
17000.0	0.295	27.189	3.043	946.860	0.132	0.017	3.647	0.815	0.335	0.046
17500.0	0.293	26.755	3.115	946.615	0.137	0.018	3.681	0.846	0.350	0.049
18000.0	0.292	26.325	3.186	946.369	0.143	0.019	3.712	0.878	0.366	0.054
18500.0	0.290	25.898	3.257	946.123	0.148	0.021	3.741	0.910	0.382	0.058
19000.0	0.289	25.475	3.326	945.877	0.154	0.022	3.768	0.941	0.397	0.063
19500.0	0.287	25.054	3.395	945.629	0.159	0.023	3.794	0.972	0.413	0.067
20000.0	0.285	24.636	3.463	945.381	0.165	0.025	3.817	1.004	0.428	0.072
20500.0	0.284	24.222	3.530	945.131	0.170	0.026	3.839	1.035	0.444	0.077
21000.0	0.282	23.812	3.597	944.880	0.176	0.028	3.860	1.066	0.460	0.083
22500.0	0.277	22.599	3.791	944.119	0.193	0.033	3.913	1.158	0.506	0.100
25000.0	0.269	20.645	4.101	942.821	0.224	0.042	3.978	1.309	0.584	0.134
27500.0	0.261	18.775	4.391	941.483	0.255	0.053	4.015	1.454	0.658	0.173
30000.0	0.253	16.990	4.662	940.104	0.287	0.066	4.028	1.594	0.727	0.217
32500.0	0.245	15.292	4.913	938.680	0.321	0.081	4.022	1.729	0.791	0.268
35000.0	0.237	13.682	5.145	937.208	0.354	0.097	3.999	1.856	0.849	0.325
37500.0	0.229	12.163	5.357	935.685	0.388	0.114	3.962	1.976	0.901	0.388
40000.0	0.221	10.736	5.548	934.106	0.422	0.134	3.914	2.088	0.946	0.456
42500.0	0.213	9.403	5.720	932.469	0.456	0.154	3.858	2.191	0.984	0.530
45000.0	0.204	8.168	5.870	930.768	0.489	0.176	3.796	2.285	1.015	0.610
47500.0	0.196	7.033	6.000	929.001	0.521	0.199	3.730	2.370	1.040	0.695
50000.0	0.188	5.998	6.108	927.164	0.553	0.223	3.663	2.445	1.059	0.785
52500.0	0.180	5.066	6.195	925.255	0.583	0.247	3.596	2.511	1.073	0.879
55000.0	0.172	4.235	6.262	923.272	0.611	0.270	3.531	2.567	1.082	0.977
57500.0	0.164	3.504	6.308	921.215	0.637	0.294	3.470	2.613	1.087	1.079
60000.0	0.156	2.869	6.336	919.084	0.662	0.317	3.412	2.651	1.088	1.183
62500.0	0.148	2.327	6.345	916.884	0.684	0.338	3.360	2.681	1.087	1.289
65000.0	0.140	1.869	6.337	914.618	0.704	0.358	3.313	2.704	1.084	1.396
67500.0	0.133	1.489	6.314	912.291	0.721	0.377	3.272	2.720	1.080	1.504
70000.0	0.126	1.178	6.279	909.910	0.736	0.393	3.236	2.731	1.075	1.611

Table C.35 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.355	44.360	0.000	955.285	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.240	0.023	955.229	0.000	0.000	0.020	0.000	0.000	0.000
500.0	0.353	43.764	0.115	955.003	0.002	0.000	0.202	0.001	0.000	0.000
1000.0	0.351	43.176	0.228	954.721	0.004	0.000	0.435	0.007	0.000	0.000
1500.0	0.348	42.597	0.339	954.439	0.006	0.000	0.658	0.015	0.001	0.000
2000.0	0.346	42.026	0.449	954.156	0.009	0.000	0.870	0.027	0.003	0.000
2500.0	0.344	41.463	0.556	953.874	0.011	0.000	1.073	0.041	0.005	0.000
3000.0	0.342	40.908	0.662	953.592	0.014	0.000	1.266	0.057	0.008	0.000
3500.0	0.340	40.360	0.766	953.310	0.018	0.001	1.451	0.076	0.012	0.000
4000.0	0.338	39.819	0.869	953.028	0.021	0.001	1.628	0.096	0.018	0.000
4500.0	0.336	39.284	0.970	952.746	0.025	0.001	1.797	0.117	0.024	0.001
5000.0	0.334	38.756	1.069	952.464	0.028	0.001	1.958	0.140	0.032	0.001
5500.0	0.332	38.234	1.167	952.182	0.032	0.002	2.112	0.164	0.040	0.001
6000.0	0.330	37.718	1.264	951.901	0.036	0.002	2.259	0.189	0.049	0.002
6500.0	0.328	37.208	1.359	951.619	0.041	0.002	2.399	0.215	0.060	0.003
7000.0	0.326	36.703	1.453	951.337	0.045	0.003	2.533	0.241	0.071	0.003
7500.0	0.324	36.204	1.545	951.056	0.050	0.003	2.661	0.269	0.083	0.004
8000.0	0.322	35.710	1.637	950.775	0.054	0.004	2.783	0.297	0.096	0.005
8500.0	0.320	35.222	1.726	950.494	0.059	0.004	2.899	0.326	0.109	0.006
9000.0	0.318	34.738	1.815	950.213	0.064	0.005	3.009	0.355	0.123	0.008
9500.0	0.316	34.259	1.903	949.933	0.069	0.006	3.115	0.385	0.138	0.009
10000.0	0.314	33.785	1.989	949.652	0.074	0.006	3.215	0.415	0.153	0.011
10500.0	0.313	33.316	2.074	949.372	0.080	0.007	3.311	0.446	0.169	0.012
11000.0	0.311	32.851	2.158	949.092	0.085	0.008	3.401	0.477	0.186	0.014
11500.0	0.309	32.390	2.240	948.812	0.091	0.009	3.487	0.509	0.202	0.016
12000.0	0.307	31.934	2.322	948.533	0.096	0.010	3.569	0.540	0.219	0.019
12500.0	0.305	31.482	2.403	948.254	0.102	0.011	3.646	0.572	0.237	0.021
13000.0	0.303	31.034	2.482	947.976	0.108	0.012	3.719	0.605	0.254	0.024
13500.0	0.301	30.590	2.561	947.697	0.114	0.013	3.789	0.637	0.272	0.027
14000.0	0.300	30.149	2.638	947.419	0.120	0.014	3.854	0.670	0.290	0.030
14500.0	0.298	29.713	2.715	947.142	0.126	0.015	3.915	0.703	0.308	0.033
15000.0	0.296	29.280	2.790	946.865	0.132	0.016	3.973	0.736	0.326	0.036
15500.0	0.294	28.851	2.865	946.588	0.138	0.018	4.027	0.769	0.344	0.040
16000.0	0.292	28.425	2.939	946.312	0.144	0.019	4.078	0.802	0.362	0.044
16500.0	0.291	28.003	3.012	946.036	0.150	0.021	4.125	0.835	0.381	0.048
17000.0	0.289	27.584	3.084	945.761	0.156	0.022	4.169	0.869	0.399	0.052
17500.0	0.287	27.168	3.155	945.486	0.162	0.024	4.211	0.902	0.417	0.056
18000.0	0.286	26.756	3.225	945.211	0.168	0.025	4.250	0.935	0.435	0.061
18500.0	0.284	26.346	3.294	944.936	0.174	0.027	4.286	0.969	0.453	0.065
19000.0	0.282	25.940	3.363	944.660	0.181	0.028	4.320	1.002	0.471	0.070
19500.0	0.280	25.537	3.431	944.384	0.187	0.030	4.352	1.035	0.489	0.076
20000.0	0.279	25.138	3.498	944.107	0.193	0.032	4.382	1.068	0.507	0.081
20500.0	0.277	24.741	3.564	943.829	0.200	0.034	4.410	1.101	0.524	0.086
21000.0	0.275	24.348	3.629	943.551	0.206	0.036	4.436	1.134	0.542	0.092
21500.0	0.274	23.958	3.693	943.271	0.213	0.038	4.461	1.166	0.560	0.098
22000.0	0.272	23.572	3.757	942.990	0.219	0.040	4.484	1.199	0.578	0.104
22500.0	0.270	23.189	3.820	942.707	0.226	0.042	4.506	1.231	0.595	0.111
25000.0	0.262	21.324	4.122	941.275	0.260	0.054	4.597	1.389	0.683	0.146
27500.0	0.254	19.543	4.405	939.806	0.296	0.068	4.658	1.542	0.768	0.186
30000.0	0.245	17.845	4.667	938.301	0.332	0.084	4.693	1.689	0.848	0.232
32500.0	0.237	16.231	4.910	936.756	0.369	0.101	4.706	1.830	0.922	0.283
35000.0	0.229	14.700	5.134	935.170	0.406	0.121	4.701	1.964	0.991	0.340
37500.0	0.221	13.253	5.339	933.541	0.443	0.142	4.680	2.090	1.052	0.401
40000.0	0.213	11.891	5.524	931.867	0.480	0.166	4.646	2.209	1.107	0.468
42500.0	0.205	10.613	5.690	930.145	0.516	0.190	4.601	2.319	1.155	0.539
45000.0	0.197	9.420	5.837	928.373	0.551	0.216	4.549	2.421	1.196	0.614
47500.0	0.190	8.312	5.965	926.550	0.586	0.244	4.490	2.514	1.231	0.694
50000.0	0.182	7.290	6.075	924.674	0.619	0.272	4.426	2.598	1.259	0.777
52500.0	0.175	6.352	6.165	922.743	0.651	0.300	4.361	2.673	1.282	0.864
55000.0	0.167	5.500	6.238	920.757	0.681	0.329	4.295	2.739	1.299	0.954
57500.0	0.160	4.729	6.292	918.715	0.710	0.358	4.229	2.796	1.311	1.046
60000.0	0.153	4.040	6.330	916.617	0.737	0.386	4.165	2.845	1.319	1.141
62500.0	0.146	3.428	6.351	914.465	0.761	0.413	4.104	2.886	1.323	1.237
65000.0	0.139	2.890	6.357	912.260	0.784	0.440	4.047	2.919	1.324	1.334
67500.0	0.133	2.421	6.349	910.005	0.804	0.465	3.994	2.946	1.323	1.432
70000.0	0.126	2.017	6.328	907.704	0.822	0.488	3.944	2.966	1.320	1.531

Table C.36 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.355	44.360	0.000	955.285	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.241	0.024	955.220	0.000	0.000	0.023	0.000	0.000	0.000
500.0	0.352	43.767	0.121	954.957	0.002	0.000	0.232	0.002	0.000	0.000
1000.0	0.350	43.184	0.239	954.629	0.005	0.000	0.501	0.008	0.000	0.000
1500.0	0.347	42.611	0.355	954.301	0.007	0.000	0.757	0.017	0.002	0.000
2000.0	0.345	42.047	0.469	953.973	0.011	0.000	1.002	0.030	0.004	0.000
2500.0	0.343	41.492	0.581	953.645	0.014	0.000	1.236	0.046	0.007	0.000
3000.0	0.340	40.946	0.691	953.318	0.018	0.001	1.459	0.064	0.011	0.000
3500.0	0.338	40.407	0.799	952.991	0.022	0.001	1.672	0.084	0.017	0.000
4000.0	0.335	39.876	0.906	952.664	0.026	0.001	1.877	0.105	0.024	0.001
4500.0	0.333	39.352	1.010	952.338	0.030	0.001	2.072	0.128	0.032	0.001
5000.0	0.331	38.835	1.113	952.012	0.035	0.002	2.259	0.152	0.041	0.001
5500.0	0.328	38.325	1.214	951.685	0.040	0.002	2.438	0.178	0.052	0.002
6000.0	0.326	37.821	1.313	951.360	0.045	0.003	2.610	0.204	0.063	0.002
6500.0	0.324	37.324	1.411	951.034	0.050	0.003	2.774	0.232	0.076	0.003
7000.0	0.322	36.833	1.507	950.709	0.056	0.004	2.931	0.260	0.090	0.004
7500.0	0.319	36.348	1.602	950.384	0.061	0.005	3.081	0.289	0.104	0.005
8000.0	0.317	35.868	1.695	950.059	0.067	0.005	3.225	0.319	0.120	0.006
8500.0	0.315	35.394	1.787	949.734	0.073	0.006	3.363	0.350	0.136	0.007
9000.0	0.313	34.926	1.877	949.410	0.079	0.007	3.495	0.381	0.153	0.009
9500.0	0.311	34.462	1.967	949.087	0.085	0.008	3.621	0.412	0.170	0.010
10000.0	0.309	34.004	2.054	948.763	0.092	0.009	3.741	0.445	0.188	0.012
10500.0	0.306	33.551	2.141	948.440	0.098	0.010	3.856	0.477	0.207	0.014
11000.0	0.304	33.103	2.226	948.117	0.105	0.011	3.966	0.510	0.226	0.016
11500.0	0.302	32.659	2.310	947.795	0.111	0.012	4.071	0.543	0.245	0.019
12000.0	0.300	32.220	2.393	947.473	0.118	0.014	4.171	0.577	0.265	0.021
12500.0	0.298	31.785	2.475	947.152	0.125	0.015	4.267	0.611	0.285	0.024
13000.0	0.296	31.355	2.555	946.831	0.132	0.017	4.357	0.646	0.306	0.027
13500.0	0.294	30.929	2.634	946.510	0.139	0.018	4.444	0.680	0.326	0.030
14000.0	0.292	30.507	2.712	946.190	0.146	0.020	4.526	0.715	0.347	0.033
14500.0	0.290	30.089	2.789	945.870	0.153	0.021	4.604	0.750	0.368	0.037
15000.0	0.288	29.675	2.865	945.551	0.160	0.023	4.678	0.786	0.389	0.040
15500.0	0.286	29.265	2.940	945.232	0.167	0.025	4.748	0.821	0.410	0.044
16000.0	0.284	28.859	3.014	944.914	0.175	0.027	4.815	0.857	0.431	0.048
16500.0	0.282	28.457	3.087	944.596	0.182	0.029	4.878	0.892	0.452	0.052
17000.0	0.280	28.058	3.158	944.279	0.189	0.031	4.937	0.928	0.473	0.057
17500.0	0.279	27.662	3.229	943.962	0.197	0.033	4.993	0.964	0.493	0.061
18000.0	0.277	27.270	3.299	943.645	0.204	0.035	5.046	1.000	0.514	0.066
18500.0	0.275	26.882	3.368	943.329	0.212	0.037	5.096	1.036	0.535	0.071
19000.0	0.273	26.497	3.436	943.013	0.219	0.040	5.143	1.072	0.555	0.076
19500.0	0.271	26.115	3.503	942.696	0.227	0.042	5.188	1.108	0.576	0.081
20000.0	0.269	25.736	3.569	942.379	0.234	0.045	5.231	1.144	0.596	0.087
20500.0	0.267	25.361	3.634	942.062	0.242	0.047	5.271	1.179	0.617	0.093
21000.0	0.266	24.989	3.699	941.744	0.249	0.050	5.309	1.215	0.637	0.099
21500.0	0.264	24.621	3.762	941.426	0.257	0.053	5.345	1.250	0.657	0.105
22000.0	0.262	24.255	3.825	941.106	0.264	0.056	5.380	1.286	0.678	0.111
22500.0	0.260	23.893	3.887	940.785	0.272	0.058	5.413	1.321	0.698	0.117
23000.0	0.258	23.535	3.948	940.464	0.280	0.061	5.444	1.356	0.718	0.124
23500.0	0.257	23.180	4.008	940.141	0.288	0.065	5.474	1.391	0.738	0.131
25000.0	0.251	22.134	4.184	939.165	0.311	0.074	5.555	1.495	0.798	0.152
27500.0	0.243	20.457	4.461	937.513	0.351	0.093	5.666	1.663	0.895	0.192
30000.0	0.234	18.861	4.717	935.827	0.391	0.113	5.748	1.825	0.989	0.236
32500.0	0.226	17.345	4.954	934.108	0.431	0.136	5.805	1.982	1.078	0.285
35000.0	0.218	15.908	5.172	932.354	0.472	0.161	5.841	2.133	1.162	0.338
37500.0	0.210	14.549	5.370	930.565	0.512	0.188	5.859	2.276	1.239	0.395
40000.0	0.202	13.267	5.549	928.739	0.552	0.218	5.860	2.413	1.310	0.456
42500.0	0.194	12.062	5.710	926.876	0.591	0.249	5.848	2.542	1.375	0.520
45000.0	0.187	10.930	5.852	924.976	0.629	0.282	5.825	2.663	1.434	0.588
47500.0	0.179	9.873	5.977	923.037	0.666	0.316	5.792	2.776	1.486	0.659
50000.0	0.172	8.887	6.085	921.060	0.702	0.351	5.751	2.880	1.533	0.733
52500.0	0.165	7.973	6.176	919.043	0.737	0.388	5.705	2.977	1.573	0.810
55000.0	0.159	7.127	6.251	916.989	0.770	0.424	5.654	3.065	1.608	0.889
57500.0	0.152	6.349	6.311	914.895	0.801	0.462	5.600	3.145	1.638	0.970
60000.0	0.146	5.635	6.356	912.764	0.830	0.499	5.545	3.217	1.662	1.054
62500.0	0.140	4.984	6.386	910.596	0.857	0.536	5.489	3.281	1.683	1.139
65000.0	0.134	4.393	6.403	908.391	0.882	0.572	5.433	3.338	1.699	1.225
67500.0	0.129	3.859	6.407	906.152	0.905	0.607	5.379	3.388	1.711	1.313
70000.0	0.123	3.379	6.400	903.880	0.926	0.641	5.326	3.431	1.721	1.402

Table C.37 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.355	44.360	0.000	955.285	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.355	44.240	0.022	955.238	0.000	0.000	0.017	0.000	0.000	0.000
500.0	0.353	43.762	0.110	955.049	0.001	0.000	0.170	0.001	0.000	0.000
1000.0	0.351	43.171	0.218	954.812	0.003	0.000	0.367	0.006	0.000	0.000
1500.0	0.349	42.587	0.324	954.575	0.005	0.000	0.555	0.013	0.001	0.000
2000.0	0.348	42.011	0.429	954.337	0.007	0.000	0.735	0.023	0.002	0.000
2500.0	0.346	41.441	0.533	954.100	0.009	0.000	0.907	0.035	0.003	0.000
3000.0	0.344	40.878	0.635	953.862	0.012	0.000	1.070	0.049	0.006	0.000
3500.0	0.342	40.320	0.735	953.624	0.014	0.000	1.227	0.065	0.009	0.000
4000.0	0.340	39.769	0.835	953.386	0.017	0.001	1.377	0.083	0.013	0.000
4500.0	0.339	39.224	0.933	953.148	0.020	0.001	1.519	0.102	0.017	0.000
5000.0	0.337	38.684	1.030	952.910	0.023	0.001	1.656	0.122	0.023	0.001
5500.0	0.335	38.150	1.125	952.672	0.026	0.001	1.786	0.143	0.029	0.001
6000.0	0.333	37.621	1.220	952.433	0.029	0.001	1.910	0.166	0.036	0.001
6500.0	0.332	37.097	1.313	952.195	0.033	0.002	2.029	0.189	0.044	0.002
7000.0	0.330	36.578	1.405	951.956	0.036	0.002	2.141	0.214	0.052	0.002
7500.0	0.328	36.063	1.496	951.718	0.040	0.002	2.249	0.239	0.061	0.003
8000.0	0.326	35.554	1.586	951.479	0.044	0.003	2.351	0.264	0.071	0.004
8500.0	0.325	35.049	1.674	951.241	0.048	0.003	2.449	0.291	0.082	0.005
9000.0	0.323	34.548	1.762	951.002	0.052	0.004	2.541	0.318	0.093	0.006
9500.0	0.321	34.052	1.849	950.764	0.056	0.004	2.629	0.345	0.104	0.007
10000.0	0.320	33.560	1.934	950.525	0.060	0.004	2.713	0.373	0.116	0.008
10500.0	0.318	33.073	2.019	950.287	0.064	0.005	2.792	0.402	0.129	0.010
11000.0	0.316	32.589	2.103	950.048	0.068	0.006	2.867	0.430	0.142	0.011
11500.0	0.315	32.109	2.185	949.810	0.073	0.006	2.938	0.459	0.155	0.013
12000.0	0.313	31.633	2.267	949.572	0.077	0.007	3.005	0.489	0.168	0.015
12500.0	0.311	31.161	2.348	949.334	0.082	0.008	3.068	0.519	0.182	0.017
13000.0	0.310	30.693	2.427	949.096	0.087	0.008	3.127	0.548	0.196	0.019
13500.0	0.308	30.228	2.506	948.859	0.092	0.009	3.183	0.579	0.211	0.021
14000.0	0.306	29.767	2.584	948.621	0.096	0.010	3.236	0.609	0.225	0.024
14500.0	0.305	29.309	2.661	948.384	0.101	0.011	3.285	0.640	0.240	0.027
15000.0	0.303	28.854	2.738	948.147	0.106	0.012	3.330	0.670	0.255	0.030
15500.0	0.301	28.403	2.813	947.911	0.111	0.013	3.373	0.701	0.269	0.033
16000.0	0.300	27.954	2.888	947.675	0.116	0.014	3.412	0.732	0.284	0.036
16500.0	0.298	27.509	2.962	947.439	0.121	0.015	3.449	0.763	0.299	0.039
17000.0	0.297	27.067	3.035	947.203	0.126	0.016	3.483	0.794	0.314	0.043
17500.0	0.295	26.628	3.107	946.966	0.131	0.017	3.515	0.825	0.328	0.047
18000.0	0.293	26.192	3.178	946.730	0.136	0.018	3.544	0.856	0.343	0.051
18500.0	0.292	25.759	3.249	946.493	0.141	0.019	3.572	0.887	0.358	0.055
19000.0	0.290	25.330	3.319	946.255	0.147	0.020	3.597	0.918	0.373	0.059
19500.0	0.289	24.903	3.388	946.017	0.152	0.022	3.620	0.949	0.387	0.064
20000.0	0.287	24.480	3.457	945.777	0.157	0.023	3.642	0.980	0.402	0.069
20500.0	0.286	24.060	3.525	945.537	0.163	0.024	3.663	1.011	0.417	0.074
22500.0	0.279	22.412	3.788	944.559	0.185	0.030	3.731	1.132	0.477	0.095
25000.0	0.271	20.427	4.101	943.304	0.214	0.039	3.789	1.281	0.550	0.128
27500.0	0.263	18.526	4.394	942.008	0.244	0.050	3.819	1.425	0.620	0.166
30000.0	0.255	16.711	4.668	940.670	0.275	0.062	3.827	1.564	0.686	0.210
32500.0	0.247	14.983	4.923	939.285	0.307	0.075	3.816	1.698	0.747	0.260
35000.0	0.239	13.345	5.158	937.849	0.340	0.090	3.788	1.824	0.802	0.316
37500.0	0.231	11.799	5.373	936.359	0.372	0.107	3.748	1.943	0.850	0.378
40000.0	0.223	10.350	5.568	934.810	0.405	0.125	3.697	2.055	0.892	0.447
42500.0	0.214	8.998	5.741	933.198	0.438	0.145	3.638	2.157	0.927	0.521
45000.0	0.206	7.750	5.894	931.517	0.470	0.166	3.575	2.251	0.956	0.601
47500.0	0.198	6.606	6.024	929.764	0.502	0.187	3.509	2.334	0.978	0.687
50000.0	0.189	5.570	6.132	927.934	0.532	0.210	3.442	2.408	0.995	0.779
52500.0	0.181	4.642	6.218	926.025	0.561	0.232	3.376	2.472	1.007	0.875
55000.0	0.172	3.824	6.283	924.035	0.588	0.254	3.313	2.527	1.014	0.976
57500.0	0.164	3.113	6.326	921.963	0.614	0.276	3.254	2.571	1.018	1.080
60000.0	0.156	2.504	6.349	919.812	0.637	0.297	3.200	2.607	1.018	1.188
62500.0	0.148	1.993	6.353	917.585	0.658	0.317	3.151	2.635	1.017	1.297
65000.0	0.140	1.569	6.340	915.289	0.677	0.335	3.109	2.656	1.013	1.408
67500.0	0.132	1.225	6.313	912.929	0.693	0.352	3.072	2.670	1.009	1.518
70000.0	0.125	0.949	6.272	910.516	0.707	0.367	3.041	2.679	1.004	1.629

Table C.38 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.355	44.360	0.000	955.285	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.240	0.023	955.231	0.000	0.000	0.019	0.000	0.000	0.000
500.0	0.353	43.764	0.114	955.013	0.002	0.000	0.194	0.001	0.000	0.000
1000.0	0.351	43.176	0.226	954.740	0.004	0.000	0.420	0.006	0.000	0.000
1500.0	0.349	42.596	0.336	954.468	0.006	0.000	0.635	0.015	0.001	0.000
2000.0	0.347	42.025	0.445	954.195	0.008	0.000	0.840	0.026	0.002	0.000
2500.0	0.345	41.461	0.551	953.922	0.011	0.000	1.035	0.040	0.005	0.000
3000.0	0.343	40.904	0.656	953.650	0.014	0.000	1.222	0.055	0.008	0.000
3500.0	0.341	40.354	0.760	953.377	0.017	0.001	1.401	0.073	0.012	0.000
4000.0	0.339	39.812	0.862	953.104	0.020	0.001	1.571	0.093	0.017	0.000
4500.0	0.337	39.275	0.962	952.832	0.024	0.001	1.734	0.113	0.023	0.001
5000.0	0.335	38.745	1.061	952.559	0.028	0.001	1.889	0.136	0.030	0.001
5500.0	0.333	38.220	1.159	952.287	0.031	0.002	2.038	0.159	0.037	0.001
6000.0	0.331	37.702	1.255	952.015	0.035	0.002	2.179	0.183	0.046	0.002
6500.0	0.329	37.189	1.350	951.742	0.039	0.002	2.315	0.209	0.056	0.002
7000.0	0.327	36.682	1.443	951.470	0.044	0.003	2.444	0.235	0.067	0.003
7500.0	0.325	36.180	1.535	951.198	0.048	0.003	2.567	0.262	0.078	0.004
8000.0	0.323	35.683	1.626	950.927	0.053	0.004	2.684	0.289	0.090	0.005
8500.0	0.321	35.191	1.716	950.655	0.057	0.004	2.796	0.318	0.103	0.006
9000.0	0.319	34.704	1.804	950.383	0.062	0.005	2.902	0.346	0.117	0.007
9500.0	0.317	34.222	1.891	950.112	0.067	0.005	3.003	0.376	0.131	0.008
10000.0	0.315	33.744	1.977	949.841	0.072	0.006	3.099	0.406	0.145	0.010
10500.0	0.314	33.271	2.062	949.570	0.077	0.007	3.191	0.436	0.160	0.012
11000.0	0.312	32.802	2.146	949.300	0.082	0.007	3.278	0.466	0.176	0.014
11500.0	0.310	32.338	2.229	949.029	0.088	0.008	3.360	0.497	0.192	0.016
12000.0	0.308	31.878	2.310	948.759	0.093	0.009	3.438	0.528	0.208	0.018
12500.0	0.306	31.421	2.391	948.490	0.099	0.010	3.512	0.560	0.225	0.020
13000.0	0.304	30.969	2.471	948.220	0.104	0.011	3.581	0.592	0.241	0.023
13500.0	0.303	30.521	2.549	947.951	0.110	0.012	3.647	0.624	0.258	0.025
14000.0	0.301	30.076	2.627	947.682	0.116	0.013	3.709	0.656	0.275	0.028
14500.0	0.299	29.635	2.704	947.414	0.121	0.014	3.767	0.688	0.293	0.031
15000.0	0.297	29.198	2.779	947.146	0.127	0.015	3.821	0.721	0.310	0.035
15500.0	0.296	28.764	2.854	946.879	0.133	0.017	3.873	0.753	0.327	0.038
16000.0	0.294	28.333	2.928	946.612	0.139	0.018	3.920	0.786	0.345	0.042
16500.0	0.292	27.906	3.001	946.345	0.144	0.019	3.965	0.819	0.362	0.046
17000.0	0.291	27.482	3.073	946.079	0.150	0.021	4.006	0.852	0.379	0.050
17500.0	0.289	27.061	3.145	945.813	0.156	0.022	4.045	0.884	0.397	0.054
18000.0	0.287	26.644	3.215	945.547	0.162	0.024	4.081	0.917	0.414	0.058
18500.0	0.285	26.229	3.285	945.281	0.168	0.025	4.115	0.950	0.431	0.063
19000.0	0.284	25.818	3.354	945.014	0.174	0.027	4.147	0.983	0.448	0.068
19500.0	0.282	25.410	3.422	944.747	0.180	0.028	4.176	1.015	0.465	0.073
20000.0	0.280	25.005	3.489	944.478	0.187	0.030	4.204	1.048	0.483	0.078
20500.0	0.279	24.604	3.556	944.209	0.193	0.032	4.230	1.080	0.500	0.083
21000.0	0.277	24.206	3.621	943.939	0.199	0.034	4.254	1.112	0.517	0.089
21500.0	0.275	23.811	3.686	943.668	0.205	0.036	4.277	1.144	0.534	0.095
22500.0	0.272	23.031	3.814	943.121	0.218	0.040	4.319	1.208	0.568	0.107
25000.0	0.264	21.139	4.118	941.729	0.252	0.051	4.400	1.364	0.652	0.142
27500.0	0.256	19.332	4.403	940.302	0.286	0.064	4.452	1.514	0.733	0.181
30000.0	0.247	17.608	4.669	938.836	0.321	0.079	4.480	1.659	0.810	0.227
32500.0	0.239	15.968	4.914	937.329	0.357	0.096	4.486	1.798	0.881	0.278
35000.0	0.231	14.412	5.141	935.780	0.393	0.115	4.474	1.931	0.946	0.334
37500.0	0.223	12.942	5.348	934.186	0.429	0.135	4.447	2.056	1.004	0.395
40000.0	0.215	11.558	5.536	932.544	0.465	0.157	4.407	2.172	1.055	0.462
42500.0	0.207	10.262	5.704	930.852	0.500	0.181	4.357	2.281	1.100	0.533
45000.0	0.199	9.053	5.853	929.106	0.535	0.206	4.300	2.381	1.137	0.610
47500.0	0.191	7.933	5.982	927.306	0.569	0.232	4.237	2.472	1.169	0.690
50000.0	0.184	6.903	6.092	925.447	0.601	0.258	4.171	2.553	1.194	0.775
52500.0	0.176	5.962	6.182	923.530	0.632	0.285	4.103	2.625	1.213	0.863
55000.0	0.168	5.111	6.254	921.552	0.662	0.313	4.036	2.688	1.227	0.955
57500.0	0.161	4.347	6.307	919.513	0.690	0.340	3.969	2.743	1.236	1.050
60000.0	0.154	3.669	6.342	917.414	0.716	0.366	3.906	2.788	1.242	1.147
62500.0	0.146	3.073	6.361	915.255	0.739	0.392	3.846	2.825	1.243	1.246
65000.0	0.139	2.554	6.363	913.039	0.761	0.416	3.790	2.855	1.243	1.346
67500.0	0.133	2.108	6.351	910.769	0.780	0.439	3.739	2.878	1.240	1.447
70000.0	0.126	1.729	6.326	908.449	0.797	0.460	3.693	2.895	1.235	1.547

Table C.39 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.355	44.360	0.000	955.285	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.241	0.024	955.221	0.000	0.000	0.023	0.000	0.000	0.000
500.0	0.352	43.768	0.120	954.963	0.002	0.000	0.228	0.002	0.000	0.000
1000.0	0.350	43.185	0.238	954.640	0.005	0.000	0.492	0.008	0.000	0.000
1500.0	0.348	42.612	0.353	954.317	0.007	0.000	0.743	0.017	0.001	0.000
2000.0	0.345	42.048	0.467	953.995	0.011	0.000	0.983	0.030	0.003	0.000
2500.0	0.343	41.493	0.578	953.673	0.014	0.000	1.213	0.045	0.006	0.000
3000.0	0.340	40.946	0.687	953.351	0.018	0.001	1.432	0.063	0.011	0.000
3500.0	0.338	40.407	0.795	953.030	0.022	0.001	1.641	0.082	0.016	0.000
4000.0	0.336	39.875	0.901	952.709	0.026	0.001	1.841	0.104	0.023	0.001
4500.0	0.333	39.350	1.005	952.388	0.030	0.001	2.033	0.126	0.031	0.001
5000.0	0.331	38.833	1.107	952.067	0.035	0.002	2.216	0.150	0.040	0.001
5500.0	0.329	38.322	1.208	951.746	0.039	0.002	2.391	0.176	0.050	0.002
6000.0	0.327	37.817	1.307	951.426	0.044	0.003	2.559	0.202	0.061	0.002
6500.0	0.324	37.319	1.404	951.106	0.050	0.003	2.720	0.229	0.074	0.003
7000.0	0.322	36.827	1.500	950.786	0.055	0.004	2.874	0.257	0.087	0.004
7500.0	0.320	36.340	1.594	950.467	0.061	0.004	3.021	0.286	0.101	0.005
8000.0	0.318	35.859	1.687	950.147	0.066	0.005	3.161	0.315	0.116	0.006
8500.0	0.316	35.384	1.779	949.829	0.072	0.006	3.296	0.346	0.132	0.007
9000.0	0.313	34.914	1.869	949.510	0.078	0.007	3.425	0.376	0.149	0.009
9500.0	0.311	34.449	1.958	949.192	0.084	0.008	3.547	0.408	0.166	0.010
10000.0	0.309	33.989	2.046	948.874	0.091	0.009	3.665	0.440	0.183	0.012
10500.0	0.307	33.535	2.132	948.557	0.097	0.010	3.777	0.472	0.202	0.014
11000.0	0.305	33.084	2.217	948.240	0.103	0.011	3.884	0.504	0.220	0.016
11500.0	0.303	32.639	2.301	947.923	0.110	0.012	3.986	0.538	0.239	0.018
12000.0	0.301	32.198	2.384	947.607	0.117	0.013	4.083	0.571	0.259	0.021
12500.0	0.299	31.761	2.465	947.291	0.123	0.015	4.176	0.605	0.278	0.023
13000.0	0.297	31.329	2.545	946.976	0.130	0.016	4.264	0.639	0.298	0.026
13500.0	0.295	30.901	2.624	946.661	0.137	0.018	4.347	0.673	0.319	0.029
14000.0	0.293	30.477	2.703	946.347	0.144	0.019	4.427	0.708	0.339	0.033
14500.0	0.291	30.057	2.779	946.033	0.151	0.021	4.502	0.742	0.359	0.036
15000.0	0.289	29.641	2.855	945.719	0.158	0.022	4.574	0.777	0.380	0.040
15500.0	0.287	29.229	2.930	945.406	0.165	0.024	4.641	0.812	0.400	0.043
16000.0	0.285	28.821	3.004	945.094	0.172	0.026	4.705	0.848	0.421	0.047
16500.0	0.283	28.416	3.077	944.782	0.179	0.028	4.766	0.883	0.441	0.051
17000.0	0.281	28.014	3.149	944.471	0.187	0.030	4.822	0.918	0.462	0.056
17500.0	0.280	27.617	3.219	944.160	0.194	0.032	4.876	0.954	0.482	0.060
18000.0	0.278	27.222	3.289	943.849	0.201	0.034	4.927	0.990	0.503	0.065
18500.0	0.276	26.831	3.358	943.538	0.209	0.036	4.975	1.025	0.523	0.070
19000.0	0.274	26.443	3.426	943.228	0.216	0.039	5.020	1.061	0.543	0.075
19500.0	0.272	26.059	3.493	942.917	0.223	0.041	5.062	1.096	0.563	0.080
20000.0	0.270	25.678	3.560	942.606	0.231	0.043	5.103	1.131	0.583	0.086
20500.0	0.269	25.300	3.625	942.294	0.238	0.046	5.141	1.167	0.603	0.091
21000.0	0.267	24.925	3.690	941.982	0.246	0.048	5.177	1.202	0.623	0.097
21500.0	0.265	24.554	3.753	941.669	0.253	0.051	5.211	1.237	0.643	0.103
22000.0	0.263	24.186	3.816	941.355	0.261	0.054	5.244	1.272	0.663	0.109
22500.0	0.261	23.821	3.878	941.040	0.268	0.057	5.275	1.307	0.682	0.116
23000.0	0.260	23.460	3.940	940.723	0.276	0.060	5.305	1.341	0.702	0.122
23500.0	0.258	23.102	4.000	940.406	0.283	0.063	5.333	1.376	0.722	0.129
25000.0	0.253	22.048	4.176	939.446	0.307	0.072	5.409	1.478	0.780	0.151
27500.0	0.244	20.357	4.454	937.820	0.346	0.090	5.511	1.644	0.876	0.190
30000.0	0.236	18.748	4.712	936.160	0.385	0.110	5.586	1.805	0.968	0.234
32500.0	0.227	17.219	4.950	934.466	0.425	0.132	5.636	1.959	1.054	0.283
35000.0	0.219	15.769	5.168	932.737	0.465	0.157	5.665	2.108	1.136	0.336
37500.0	0.211	14.398	5.367	930.972	0.505	0.183	5.676	2.249	1.211	0.393
40000.0	0.203	13.104	5.548	929.169	0.545	0.212	5.671	2.383	1.279	0.455
42500.0	0.195	11.887	5.709	927.328	0.584	0.242	5.653	2.509	1.342	0.520
45000.0	0.188	10.746	5.853	925.449	0.622	0.274	5.623	2.628	1.398	0.588
47500.0	0.181	9.680	5.979	923.530	0.659	0.308	5.585	2.738	1.448	0.660
50000.0	0.173	8.687	6.087	921.570	0.694	0.342	5.540	2.840	1.491	0.735
52500.0	0.166	7.767	6.178	919.571	0.729	0.377	5.489	2.934	1.529	0.813
55000.0	0.160	6.917	6.254	917.531	0.761	0.413	5.434	3.019	1.561	0.893
57500.0	0.153	6.136	6.313	915.451	0.792	0.450	5.377	3.096	1.588	0.975
60000.0	0.147	5.422	6.357	913.330	0.821	0.486	5.319	3.164	1.610	1.060
62500.0	0.141	4.773	6.387	911.171	0.848	0.521	5.260	3.225	1.627	1.146
65000.0	0.135	4.185	6.403	908.974	0.872	0.556	5.202	3.279	1.641	1.234
67500.0	0.129	3.656	6.406	906.740	0.895	0.590	5.146	3.325	1.651	1.323
70000.0	0.124	3.182	6.397	904.471	0.915	0.622	5.091	3.364	1.658	1.413

Table C.40 [ ] Exposure-Dependent 0% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.355	44.359	0.000	955.286	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.355	44.239	0.022	955.240	0.000	0.000	0.017	0.000	0.000	0.000
500.0	0.353	43.760	0.110	955.052	0.001	0.000	0.169	0.001	0.000	0.000
1000.0	0.351	43.169	0.218	954.816	0.003	0.000	0.365	0.006	0.000	0.000
1500.0	0.349	42.585	0.324	954.581	0.005	0.000	0.552	0.013	0.001	0.000
2000.0	0.348	42.009	0.429	954.345	0.007	0.000	0.730	0.023	0.002	0.000
2500.0	0.346	41.439	0.532	954.110	0.009	0.000	0.900	0.035	0.003	0.000
3000.0	0.344	40.875	0.634	953.874	0.012	0.000	1.062	0.049	0.006	0.000
3500.0	0.342	40.317	0.735	953.639	0.014	0.000	1.217	0.065	0.009	0.000
4000.0	0.340	39.766	0.834	953.404	0.017	0.001	1.364	0.082	0.012	0.000
4500.0	0.339	39.220	0.932	953.169	0.020	0.001	1.505	0.101	0.017	0.000
5000.0	0.337	38.679	1.029	952.934	0.023	0.001	1.638	0.122	0.022	0.001
5500.0	0.335	38.144	1.124	952.700	0.026	0.001	1.766	0.143	0.028	0.001
6000.0	0.333	37.614	1.219	952.465	0.029	0.001	1.887	0.165	0.035	0.001
6500.0	0.332	37.089	1.312	952.231	0.032	0.002	2.003	0.189	0.043	0.002
7000.0	0.330	36.569	1.404	951.997	0.036	0.002	2.113	0.213	0.051	0.002
7500.0	0.328	36.054	1.494	951.763	0.039	0.002	2.217	0.238	0.060	0.003
8000.0	0.327	35.542	1.584	951.530	0.043	0.003	2.317	0.264	0.070	0.004
8500.0	0.325	35.036	1.673	951.297	0.047	0.003	2.411	0.290	0.080	0.005
9000.0	0.323	34.534	1.760	951.065	0.051	0.003	2.500	0.317	0.090	0.006
9500.0	0.322	34.035	1.847	950.832	0.054	0.004	2.584	0.344	0.102	0.007
10000.0	0.320	33.541	1.933	950.601	0.058	0.004	2.664	0.372	0.113	0.008
10500.0	0.318	33.051	2.017	950.369	0.063	0.005	2.740	0.400	0.125	0.009
11000.0	0.317	32.564	2.101	950.138	0.067	0.005	2.811	0.428	0.138	0.011
11500.0	0.315	32.081	2.183	949.908	0.071	0.006	2.879	0.457	0.150	0.013
12000.0	0.313	31.602	2.265	949.678	0.075	0.007	2.942	0.486	0.163	0.014
12500.0	0.312	31.126	2.346	949.449	0.080	0.007	3.001	0.516	0.176	0.016
13000.0	0.310	30.654	2.426	949.220	0.084	0.008	3.057	0.545	0.190	0.018
13500.0	0.309	30.185	2.505	948.991	0.089	0.009	3.110	0.575	0.203	0.021
14000.0	0.307	29.719	2.583	948.763	0.093	0.009	3.159	0.605	0.217	0.023
14500.0	0.305	29.256	2.660	948.534	0.098	0.010	3.206	0.635	0.231	0.026
15000.0	0.304	28.797	2.737	948.305	0.103	0.011	3.250	0.665	0.245	0.028
15500.0	0.302	28.341	2.813	948.075	0.107	0.012	3.291	0.695	0.259	0.031
16000.0	0.301	27.888	2.888	947.845	0.112	0.013	3.330	0.726	0.273	0.034
16500.0	0.299	27.439	2.962	947.614	0.117	0.014	3.367	0.756	0.288	0.038
17000.0	0.298	26.993	3.035	947.381	0.122	0.015	3.402	0.786	0.302	0.041
17500.0	0.296	26.550	3.108	947.148	0.127	0.016	3.435	0.817	0.317	0.045
20000.0	0.288	24.389	3.460	945.960	0.153	0.022	3.575	0.968	0.392	0.066
22500.0	0.280	22.312	3.792	944.738	0.181	0.029	3.676	1.118	0.468	0.092
25000.0	0.272	20.321	4.105	943.479	0.210	0.038	3.744	1.265	0.542	0.124
27500.0	0.264	18.415	4.399	942.180	0.241	0.048	3.783	1.409	0.612	0.162
30000.0	0.256	16.597	4.673	940.837	0.272	0.060	3.797	1.548	0.678	0.206
32500.0	0.248	14.868	4.928	939.447	0.304	0.074	3.791	1.682	0.739	0.256
35000.0	0.240	13.231	5.163	938.007	0.337	0.089	3.768	1.810	0.794	0.312
37500.0	0.231	11.687	5.377	936.511	0.370	0.106	3.730	1.930	0.843	0.374
40000.0	0.223	10.240	5.571	934.956	0.403	0.124	3.682	2.043	0.885	0.442
42500.0	0.215	8.893	5.744	933.337	0.436	0.144	3.626	2.146	0.921	0.517
45000.0	0.206	7.649	5.895	931.649	0.469	0.165	3.564	2.241	0.950	0.597
47500.0	0.198	6.511	6.024	929.887	0.500	0.186	3.499	2.326	0.973	0.683
50000.0	0.189	5.482	6.130	928.049	0.531	0.209	3.433	2.401	0.990	0.775
52500.0	0.181	4.563	6.215	926.131	0.560	0.231	3.368	2.466	1.002	0.871
55000.0	0.172	3.753	6.278	924.132	0.587	0.254	3.305	2.522	1.010	0.972
57500.0	0.164	3.050	6.319	922.051	0.613	0.276	3.247	2.567	1.014	1.077
60000.0	0.156	2.450	6.340	919.891	0.636	0.297	3.194	2.604	1.015	1.185
62500.0	0.148	1.947	6.343	917.655	0.657	0.317	3.147	2.632	1.014	1.294
65000.0	0.140	1.532	6.329	915.351	0.676	0.335	3.105	2.653	1.011	1.405
67500.0	0.132	1.194	6.300	912.984	0.692	0.352	3.069	2.668	1.007	1.516
70000.0	0.124	0.924	6.258	910.564	0.706	0.366	3.038	2.678	1.002	1.626

Table C.41 [ ] Exposure-Dependent 40% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.355	44.359	0.000	955.286	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.239	0.023	955.233	0.000	0.000	0.019	0.000	0.000	0.000
500.0	0.353	43.762	0.114	955.016	0.002	0.000	0.193	0.001	0.000	0.000
1000.0	0.351	43.174	0.226	954.746	0.004	0.000	0.417	0.006	0.000	0.000
1500.0	0.349	42.594	0.336	954.475	0.006	0.000	0.630	0.015	0.001	0.000
2000.0	0.347	42.022	0.444	954.205	0.008	0.000	0.833	0.026	0.002	0.000
2500.0	0.345	41.458	0.551	953.935	0.011	0.000	1.026	0.040	0.005	0.000
3000.0	0.343	40.901	0.656	953.666	0.014	0.000	1.210	0.055	0.008	0.000
3500.0	0.341	40.351	0.759	953.397	0.017	0.001	1.386	0.073	0.012	0.000
4000.0	0.339	39.808	0.861	953.128	0.020	0.001	1.554	0.092	0.016	0.000
4500.0	0.337	39.271	0.961	952.859	0.024	0.001	1.713	0.113	0.022	0.001
5000.0	0.335	38.740	1.060	952.591	0.027	0.001	1.866	0.135	0.029	0.001
5500.0	0.333	38.214	1.157	952.323	0.031	0.001	2.011	0.159	0.037	0.001
6000.0	0.331	37.695	1.253	952.056	0.035	0.002	2.149	0.183	0.046	0.002
6500.0	0.329	37.181	1.348	951.788	0.039	0.002	2.281	0.208	0.055	0.002
7000.0	0.327	36.672	1.441	951.522	0.043	0.003	2.406	0.234	0.066	0.003
7500.0	0.325	36.169	1.533	951.256	0.047	0.003	2.525	0.261	0.077	0.004
8000.0	0.323	35.671	1.624	950.990	0.051	0.003	2.639	0.289	0.088	0.005
8500.0	0.321	35.177	1.713	950.725	0.056	0.004	2.746	0.317	0.101	0.006
9000.0	0.320	34.688	1.802	950.460	0.061	0.005	2.849	0.345	0.114	0.007
9500.0	0.318	34.203	1.889	950.196	0.065	0.005	2.946	0.375	0.128	0.008
10000.0	0.316	33.723	1.975	949.933	0.070	0.006	3.038	0.404	0.142	0.010
10500.0	0.314	33.248	2.059	949.670	0.075	0.006	3.125	0.434	0.156	0.011
11000.0	0.312	32.776	2.143	949.408	0.080	0.007	3.207	0.465	0.171	0.013
11500.0	0.311	32.308	2.226	949.146	0.085	0.008	3.285	0.495	0.186	0.015
12000.0	0.309	31.845	2.307	948.885	0.091	0.009	3.359	0.526	0.202	0.017
12500.0	0.307	31.385	2.388	948.625	0.096	0.010	3.428	0.558	0.218	0.019
13000.0	0.305	30.928	2.468	948.365	0.101	0.011	3.494	0.589	0.233	0.022
13500.0	0.304	30.476	2.546	948.106	0.107	0.012	3.555	0.621	0.250	0.025
14000.0	0.302	30.026	2.624	947.847	0.112	0.013	3.613	0.652	0.266	0.027
14500.0	0.300	29.580	2.701	947.588	0.117	0.014	3.668	0.684	0.282	0.030
15000.0	0.298	29.138	2.777	947.330	0.123	0.015	3.720	0.716	0.299	0.033
15500.0	0.297	28.699	2.852	947.070	0.128	0.016	3.769	0.748	0.315	0.037
16000.0	0.295	28.264	2.926	946.811	0.134	0.017	3.816	0.780	0.332	0.040
16500.0	0.293	27.832	3.000	946.550	0.140	0.018	3.860	0.812	0.349	0.044
17000.0	0.292	27.404	3.072	946.289	0.146	0.020	3.902	0.844	0.366	0.048
17500.0	0.290	26.979	3.144	946.027	0.151	0.021	3.942	0.876	0.383	0.052
18000.0	0.288	26.557	3.215	945.764	0.157	0.023	3.980	0.908	0.400	0.056
18500.0	0.287	26.139	3.285	945.499	0.163	0.024	4.016	0.940	0.417	0.061
20000.0	0.281	24.906	3.491	944.698	0.182	0.029	4.113	1.035	0.469	0.075
22500.0	0.273	22.921	3.816	943.337	0.214	0.038	4.243	1.192	0.556	0.104
25000.0	0.265	21.022	4.122	941.940	0.247	0.049	4.336	1.346	0.641	0.138
27500.0	0.256	19.209	4.407	940.508	0.282	0.062	4.399	1.496	0.723	0.178
30000.0	0.248	17.481	4.673	939.037	0.318	0.077	4.435	1.641	0.799	0.223
32500.0	0.240	15.839	4.919	937.526	0.354	0.094	4.448	1.780	0.871	0.273
35000.0	0.232	14.283	5.145	935.971	0.390	0.113	4.441	1.913	0.936	0.329
37500.0	0.224	12.814	5.352	934.371	0.427	0.133	4.419	2.039	0.994	0.391
40000.0	0.216	11.432	5.538	932.722	0.463	0.155	4.382	2.157	1.046	0.457
42500.0	0.208	10.139	5.706	931.023	0.498	0.179	4.335	2.267	1.091	0.529
45000.0	0.200	8.935	5.853	929.271	0.533	0.204	4.281	2.368	1.129	0.605
47500.0	0.192	7.820	5.981	927.462	0.567	0.230	4.220	2.460	1.160	0.686
50000.0	0.184	6.796	6.090	925.596	0.600	0.257	4.155	2.543	1.186	0.771
52500.0	0.176	5.863	6.179	923.670	0.631	0.284	4.089	2.616	1.206	0.859
55000.0	0.168	5.019	6.249	921.683	0.661	0.311	4.023	2.680	1.220	0.951
57500.0	0.161	4.263	6.300	919.635	0.688	0.339	3.958	2.735	1.230	1.046
60000.0	0.154	3.593	6.334	917.526	0.714	0.365	3.895	2.781	1.236	1.143
62500.0	0.146	3.005	6.350	915.359	0.738	0.391	3.836	2.819	1.238	1.243
65000.0	0.139	2.495	6.351	913.134	0.760	0.415	3.782	2.850	1.238	1.343
67500.0	0.132	2.057	6.338	910.855	0.779	0.438	3.732	2.873	1.235	1.444
70000.0	0.126	1.685	6.311	908.527	0.796	0.459	3.686	2.891	1.231	1.544

Table C.42 [ ] Exposure-Dependent 80% Void Isotopics (kg/MTU Initial)

Exposure MWd/MTU	U-234	U-235	U-236	U-238	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242
0.0	0.355	44.359	0.000	955.286	0.000	0.000	0.000	0.000	0.000	0.000
100.0	0.354	44.240	0.024	955.223	0.000	0.000	0.023	0.000	0.000	0.000
500.0	0.352	43.766	0.120	954.967	0.002	0.000	0.226	0.002	0.000	0.000
1000.0	0.350	43.183	0.237	954.647	0.005	0.000	0.487	0.008	0.000	0.000
1500.0	0.348	42.610	0.353	954.328	0.007	0.000	0.736	0.017	0.001	0.000
2000.0	0.345	42.046	0.466	954.009	0.010	0.000	0.973	0.030	0.003	0.000
2500.0	0.343	41.490	0.577	953.691	0.014	0.000	1.199	0.045	0.006	0.000
3000.0	0.340	40.943	0.686	953.373	0.017	0.001	1.414	0.063	0.011	0.000
3500.0	0.338	40.403	0.794	953.056	0.021	0.001	1.620	0.082	0.016	0.000
4000.0	0.336	39.871	0.899	952.740	0.025	0.001	1.817	0.104	0.023	0.001
4500.0	0.334	39.346	1.003	952.424	0.030	0.001	2.004	0.126	0.030	0.001
5000.0	0.331	38.827	1.105	952.108	0.034	0.002	2.183	0.150	0.039	0.001
5500.0	0.329	38.316	1.205	951.793	0.039	0.002	2.355	0.176	0.049	0.002
6000.0	0.327	37.810	1.304	951.479	0.044	0.003	2.518	0.202	0.061	0.002
6500.0	0.325	37.311	1.401	951.165	0.049	0.003	2.674	0.229	0.073	0.003
7000.0	0.322	36.817	1.497	950.852	0.054	0.004	2.823	0.257	0.086	0.004
7500.0	0.320	36.330	1.591	950.539	0.059	0.004	2.966	0.286	0.100	0.005
8000.0	0.318	35.847	1.684	950.227	0.065	0.005	3.101	0.315	0.114	0.006
8500.0	0.316	35.370	1.775	949.916	0.071	0.006	3.231	0.345	0.130	0.007
9000.0	0.314	34.898	1.865	949.605	0.077	0.007	3.355	0.376	0.146	0.008
9500.0	0.312	34.431	1.954	949.295	0.083	0.008	3.472	0.407	0.162	0.010
10000.0	0.310	33.969	2.041	948.986	0.089	0.009	3.585	0.439	0.180	0.012
10500.0	0.308	33.512	2.127	948.677	0.095	0.010	3.691	0.471	0.197	0.014
11000.0	0.306	33.059	2.212	948.369	0.101	0.011	3.793	0.504	0.215	0.016
11500.0	0.304	32.611	2.296	948.062	0.107	0.012	3.890	0.537	0.233	0.018
12000.0	0.302	32.167	2.378	947.756	0.114	0.013	3.982	0.570	0.252	0.020
12500.0	0.300	31.727	2.460	947.450	0.120	0.014	4.069	0.604	0.271	0.023
13000.0	0.298	31.291	2.540	947.145	0.127	0.016	4.152	0.637	0.290	0.026
13500.0	0.296	30.859	2.619	946.841	0.133	0.017	4.231	0.671	0.309	0.029
14000.0	0.294	30.430	2.697	946.537	0.140	0.018	4.305	0.706	0.329	0.032
14500.0	0.292	30.006	2.774	946.234	0.147	0.020	4.376	0.740	0.348	0.035
15000.0	0.290	29.585	2.850	945.931	0.153	0.021	4.444	0.774	0.367	0.038
15500.0	0.288	29.168	2.925	945.628	0.160	0.023	4.508	0.809	0.387	0.042
16000.0	0.286	28.755	2.999	945.324	0.167	0.025	4.569	0.844	0.407	0.046
16500.0	0.284	28.345	3.072	945.021	0.174	0.027	4.628	0.878	0.427	0.050
17000.0	0.283	27.939	3.144	944.717	0.181	0.029	4.683	0.913	0.446	0.054
17500.0	0.281	27.536	3.216	944.412	0.188	0.031	4.737	0.948	0.466	0.059
18000.0	0.279	27.137	3.286	944.106	0.195	0.033	4.788	0.982	0.486	0.063
18500.0	0.277	26.742	3.355	943.800	0.203	0.035	4.836	1.017	0.506	0.068
19000.0	0.275	26.350	3.424	943.492	0.210	0.037	4.883	1.052	0.526	0.073
19500.0	0.273	25.962	3.491	943.184	0.217	0.039	4.928	1.086	0.546	0.078
20000.0	0.272	25.577	3.558	942.874	0.225	0.042	4.971	1.121	0.566	0.083
22500.0	0.263	23.707	3.878	941.307	0.263	0.055	5.160	1.291	0.667	0.113
25000.0	0.254	21.924	4.177	939.708	0.301	0.070	5.309	1.459	0.767	0.147
27500.0	0.245	20.226	4.456	938.076	0.341	0.088	5.425	1.623	0.863	0.187
30000.0	0.236	18.612	4.714	936.411	0.381	0.107	5.510	1.783	0.955	0.231
32500.0	0.228	17.079	4.952	934.712	0.421	0.130	5.570	1.937	1.041	0.279
35000.0	0.220	15.628	5.170	932.977	0.462	0.154	5.607	2.086	1.122	0.332
37500.0	0.212	14.256	5.369	931.205	0.502	0.181	5.624	2.227	1.198	0.389
40000.0	0.204	12.963	5.549	929.396	0.541	0.209	5.625	2.362	1.266	0.451
42500.0	0.196	11.748	5.710	927.549	0.581	0.239	5.611	2.490	1.329	0.515
45000.0	0.188	10.610	5.852	925.662	0.619	0.272	5.586	2.609	1.385	0.584
47500.0	0.181	9.548	5.977	923.735	0.656	0.305	5.551	2.720	1.435	0.656
50000.0	0.174	8.560	6.084	921.768	0.692	0.340	5.508	2.823	1.479	0.731
52500.0	0.167	7.645	6.174	919.760	0.726	0.375	5.460	2.918	1.517	0.808
55000.0	0.160	6.801	6.247	917.712	0.759	0.411	5.408	3.004	1.549	0.889
57500.0	0.153	6.027	6.305	915.624	0.790	0.447	5.352	3.082	1.577	0.971
60000.0	0.147	5.320	6.348	913.495	0.819	0.483	5.296	3.152	1.599	1.056
62500.0	0.141	4.677	6.376	911.327	0.846	0.519	5.239	3.214	1.617	1.142
65000.0	0.135	4.097	6.390	909.121	0.871	0.554	5.182	3.268	1.631	1.230
67500.0	0.129	3.575	6.392	906.879	0.893	0.588	5.128	3.315	1.642	1.319
70000.0	0.124	3.108	6.381	904.601	0.913	0.620	5.075	3.355	1.649	1.409

Appendix D Lattice Enrichment Distribution Maps

[

]

Figure D.1 [ ] Enrichment Distribution

[

]

Figure D.2 [ ] Enrichment Distribution

[

]

Figure D.3 [ ] Enrichment Distribution

[

]

Figure D.4 [ ] Enrichment Distribution

[

]

Figure D.5 [ ] Enrichment Distribution

[

]

Figure D.6 [ ] Enrichment Distribution

[

]

Figure D.7 [ ] Enrichment Distribution

[

]

Figure D.8 [ ] Enrichment Distribution

[

]

Figure D.9 [ ] Enrichment Distribution

[

]

Figure D.10 [ ] Enrichment Distribution

[

]

Figure D.11 [ ] Enrichment Distribution

[

]

Figure D.12 [ ] Enrichment Distribution

[

]

Figure D.13 [ ] Enrichment Distribution

[

]

Figure D.14 [ ] Enrichment Distribution

[

]

Figure D.15 [ ] Enrichment Distribution

[

]

Figure D.16 [ ] Enrichment Distribution

[

]

Figure D.17 [ ] Enrichment Distribution

[

]

Figure D.18 [ ] Enrichment Distribution

[

]

Figure D.19 [ ] Enrichment Distribution

[

]

Figure D.20 [ ] Enrichment Distribution

[

]

Figure D.21 [ ] Enrichment Distribution

[

]

Figure D.22 [ ] Enrichment Distribution

[

]

Figure D.23 [ ] Enrichment Distribution

[

]

Figure D.24 [ ] Enrichment Distribution

[

]

Figure D.25 [ ] Enrichment Distribution

[

]

Figure D.26 [ ] Enrichment Distribution

[

]

Figure D.27 [ ] Enrichment Distribution