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ADJUDICATIONS STAFF

Exhibit 2

Relevant Portions of Duke Energy's Response to NRC Request for Additional Information Submitted to the NRC, November 3, 2003

NUCLEAR REGULATORY COMMISSION

Docket No. 50-413/414-OLA Official Exh. No. 2
In the matter of Duke Catawba
Staff ______ IDENTIFIED 7/14/04
Applicant ✓ RECEIVED 7/14/04
Intervenor ______ REJECTED ______
Cont'g Off'r ______
Contractor ______ DATE ______
Other ______ Witness ______
Reporter [signature]

Template = SECY-028

SECY-02

12. Provide the appropriate regulatory criteria to be satisfied by the information in section 3.7, i.e., how this section meets the general design criteria specified in the Standard Review Plan.

Response (Previously submitted October 3, 2003)
Section 3.7 contains the safety analysis of three distinct subject areas; loss of coolant accidents (LOCA), non-LOCA accidents, and radiological consequences. The appropriate regulatory criteria for each of these topics are summarized in Tables Q12-1 through Q12-3.

LOCA Criteria
The LOCA acceptance criteria of 10CFR 50.46 (b) were established for light water reactors fueled with UO_2 pellets within cylindrical Zircaloy cladding. The MOX fuel lead assemblies have M5™ cladding and mixed oxide fuel pellets. The applicability of the 10CFR 50.46 criteria to the MOX fuel lead assemblies is established in Table Q12-1.

Non-LOCA Criteria
The criteria used to evaluate the non-LOCA transients/accidents in the Updated Final Safety Analysis Report are summarized in Table Q12-2 and except for rod ejection accident criteria are the same criteria used for analysis of non-LOCA transients/accidents in LEU fuel cores.

Provisional Rod Ejection Accident Criteria
The current acceptance criteria for a rod ejection accident (REA) at Catawba are described in Section 4.1.2 of Reference Q12-1. These criteria are based on Section 15.4.8 of the *Standard Review Plan* (Reference Q12-2), and are summarized below.

1. The radially averaged fuel pellet enthalpy shall not exceed 280 cal/gm at any location.
2. Doses must be "well within" the 10 CFR 100 dose limits of 25 rem whole-body and 300 rem to the thyroid, where "well within" is interpreted as less than 25% of those values.
3. The peak Reactor Coolant System pressure must be within Service Limit C as defined by the ASME Code, which is 3000 psia (120% of the 2500 psia design pressure).

With the exception of the enthalpy limit of 280 cal/gm, those criteria are equally valid for mixed oxide (MOX) fuel as for low enriched uranium (LEU) fuel during a REA. The dose acceptance criteria relate to the radiological consequences to the public, not the fuel type. The primary system pressure acceptance criterion relates to the integrity of the pressure boundary, not the fuel type.

The enthalpy limit was established to ensure coolability of the core after a REA and to preclude the energetic dispersal of fuel particles into the coolant (Reference Q12-3). The current pressurized water reactor regulatory acceptance criterion of 280 cal/gm is based primarily on experiments such as SPERT that were conducted by the Atomic Energy Commission. More recent REA experiments conducted at the Cabri facility in France, among others, suggest that a lower enthalpy limit may be appropriate, particularly for high burnup irradiated fuel. The Electric Power Research Institute (EPRI) has used the more recent experimental data, coupled with cladding failure predictions using the Critical Strain Energy Density (CSED) approach, to develop proposed REA enthalpy limits as a function of burnup. The work is documented in EPRI's Topical Report on Reactivity Initiated Accident: Bases for RIA Fuel and Core Coolability Criteria" (Reference Q12-4), which as been submitted to the Nuclear Regulatory Commission (NRC) and is currently under review.

Four MOX fuel rods have been tested under simulated REA conditions as part of the Cabri test program. Of those tests, three experienced no cladding failure with peak enthalpies of 138, 203, and 90 cal/gm. However, the Rep Na-7 test saw a cladding failure with fuel dispersal at an enthalpy of 120 cal/gm. The Rep Na-7 rod had a burnup of 55 GWd/MThm and a cladding oxidation layer of 50 microns (Reference Q12-4, Table 2-1). Based on the results of that test, it has been postulated that differences in fuel pellet microstructure between MOX and LEU fuel may make MOX fuel more susceptible to disruptive cladding failure at lower fuel pellet enthalpy values.

Accordingly, for the MOX fuel lead assemblies, Duke proposes to use a radial average peak fuel enthalpy limit that is substantially more conservative than the current NUREG-0800 acceptance criterion for LEU fuel. Duke proposes to use a value of 100 cal/gm at all burnups as the acceptance criterion for MOX fuel rods experiencing a power excursion from hot zero power (HZP). This criterion is considered to be appropriate and conservative, for the reasons provided below.

1. The value is significantly lower than enthalpies at which disruptive failure has been experienced in any MOX fuel REA tests.
2. The value is significantly lower than the Fuel Rod Failure Threshold curve for LEU fuel as proposed by EPRI (Reference Q12-4, Figure S-1).
3. MOX fuel rods will be clad in M5™. Fuel rod corrosion is considered to be a contributing factor to cladding failure under REA conditions. M5™ has demonstrated extremely low corrosion relative to Zircaloy-4, the cladding material that was used in all MOX fuel REA tests (see Figure 6.1 of Reference Q12-5).
4. MOX fuel lead assembly rod burnup will be limited to less than 60 GWd/MThm.
5. Applying the criterion only to accidents from HZP excludes accidents initiating from hot full power with a high initial enthalpy (reflective of full power) but no rapid energy deposition in the fuel pellet.

Duke will use the SIMULATE-3K MOX computer code to perform three-dimensional reactor kinetics calculations of licensing basis REAs for all cores containing MOX fuel lead assemblies. Duke will verify that the peak enthalpy in all MOX fuel lead assembly rods remains below the 100 cal/gm acceptance criterion during postulated REAs. SIMULATE-3K MOX, described in Section 2.4 of Reference Q12-6, is an extension of SIMULATE-3K. Application of SIMULATE-3K for REAs at Catawba has been reviewed and approved by the NRC (Reference Q12-7) for cores containing LEU fuel. Analyses of representative cores containing MOX fuel lead assemblies are summarized in

Section 3.7.2.4 of Reference Q12-8 and further detail will be provided in the response to Reactor Systems RAI Question 33.

The above criteria are conservative provisional criteria for the MOX fuel lead assembly program. To support the batch use of MOX fuel, Duke intends to propose alternative REA acceptance criteria. Duke plans to document the batch use MOX fuel REA acceptance criteria and REA analytical methodology in a MOX fuel safety analysis topical report and submit the report to the NRC for review in 2004.

References

Q12-1. DPC-NE-3001-PA, *Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology*, Duke Power Company, December 2000.

Q12-2. NUREG-0800, U. S. Nuclear Regulatory Commission *Standard Review Plan*, Revision 2, July 1981.

Q12-3. Meyer, R. O., McCardell, R. K., Chung, H. M. Diamond, D. J. and Scott, H. H., *A Regulatory Assessment of Test Data for Reactivity-Insertion Accidents*, Nuclear Safety, Volume 37, No. 4, October-December 1996.

Q12-4. EPRI Technical Report 1002865, *Topical Report on Reactivity Initiated Accident: Bases for RIA Fuel and Core Coolability Criteria*, June 2002 (currently under NRC review).

Q12-5. BAW-10238(P), Revision 1, *MOX Fuel Design Report*, Framatome ANP, May 2003 (currently under NRC review).

Q12-6. DPC-NE-1005P, *Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX*, August 2001 (currently under NRC review).

Q12-7. DPC-NE-2009-P-A, Revision 2, *Duke Power Company Westinghouse Fuel Transition Report*, December 2002.

Q12-8. Tuckman, M. S., February 27, 2003 Letter to U.S. Nuclear Regulatory Commission, Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50.

Radiological Dose Criteria

General radiological criteria are provided in 10CFR 20, 10CFR 50 Appendix A, 10CFR 50.67 and 10CFR 100. These are not published as uranium specific criteria, but have been consistently applied to reactor applications by the nuclear industry. Some of these regulations also apply to other applications, such as nuclear medicine. The applicable acceptance criteria in 10CFR are determined by the purpose or scenario for which the consequences must be calculated, rather than by the source term or specific isotopes involved.

The purpose of modeling the event and projecting consequences is to protect the health and safety of the public. To that end, there must be a standard for comparison to draw a definitive conclusion as to the impact upon the public. In order to compare the biological effects from the various isotopes which are produced in nuclear applications and industries, the concept of dose equivalent (or committed dose equivalent) was adopted. Usually expressed in Rems or Sieverts, these units provide a comparison of biological effects by accounting for the energy deposition and the relative biological effectiveness from radiation emitted by isotopes.

Since dose is a measure of the cumulative biological effect of the emitted particles and rays regardless of the isotope of their origin, there is no need to specify specific dose acceptance criteria for a reactor using MOX fuel. Furthermore, the criteria which are currently in regulations for the protection of the health and safety of the public and control room operators can be applied for the same purpose and application that they currently are being applied within a plant's licensing basis. The dose acceptance criteria in 10 CFR can be applied in the same manner as applied for LEU fuel. Standard Review Plan guidance can continue to be applied in accordance with a plant's licensing basis as it has been for LEU fuel. The specific regulatory dose criteria used to analyze MOX fuel events are summarized in Table Q12-3.

Table Q12-1
Applicability of 10CFR 50.46 Criteria
to MOX Fuel Lead Assemblies

10CFR 50.46 (b) Criteria	Applicability to MOX Fuel Lead Assemblies
Peak Clad Temperature < 2200 °F	This criterion concerns the performance of the fuel pin cladding material during LOCA and is, therefore, primarily related to cladding properties. The MOX lead assembly fuel rods will be constructed using Framatome ANP's M5™ cladding. The 2200 °F criterion has been approved by the NRC as applicable to M5™ cladding in granting the licensing of replacement fuel for several light water reactors over the last few years. The basis for approval is experimental evidence that M5™ behavior during LOCA conditions is equivalent to or superior to Zircaloy and is documented in BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5™) in PWR Reactor Fuel," February 2000." This temperature criterion has no dependence on the fuel pellet design or makeup and is equally applicable for use with either UO_2 or MOX fuel pellets. This criterion is fully applicable to the MOX fuel lead assemblies.
17% Local Oxidation	This criterion concerns the performance of the fuel pin cladding material during LOCA and is, therefore, primarily related to cladding properties. The MOX lead assembly fuel rods will be constructed using Framatome ANP's M5™ cladding. The 17 percent criterion has been approved by the NRC as applicable to M5™ cladding in granting the licensing of replacement fuel for several light water reactors over the last few years. The basis for approval is experimental evidence that M5™ behavior during LOCA conditions is equivalent to or superior to Zircaloy and is documented in BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5™) in PWR Reactor Fuel," February 2000." The oxidation limit criterion controls the amount of hydrogen available to develop zirconium hydrides which increase the brittleness of the cladding in the post-accident environment. The criterion is not affected by the type of fuel pellet. This criterion is fully applicable to the MOX fuel lead assemblies.
1% Core-wide Oxidation	This criterion assures acceptable conditions within the reactor building and is unrelated to the core fuel and cladding so long as the hydrogen produced per percent cladding reacted is unchanged. Because the reaction for both M5™ and Zircaloy is between zirconium and oxygen, the hydrogen produced per reaction percent is the same for both materials. The criterion is unaffected by the use of M5™ cladding and is fully applicable to the MOX fuel lead assemblies.
Core Amenable to Cooling	This criterion controls the geometry of the core following a LOCA. As a criterion, it achieves its purpose regardless of the cladding material or the fuel pellet makeup. It is fully applicable to the MOX fuel lead assemblies.

Long-term Core Cooling	This criterion controls the availability of long-term cooling systems and core conditions. As a criterion, it achieves its purpose regardless of the cladding material or the fuel pellet makeup. It is fully applicable to the MOX fuel lead assemblies.

Table Q12-2
Acceptance Criteria for Non-LOCA Transients/Accidents
with MOX Fuel Lead Assemblies

Transient/Accident Description	Acceptance Criteria
6.2.1.3 LOCA Mass and Energy Release and Containment Pressure/Temperature Response	• Containment design margin is maintained. • Environmental qualification of the safety related equipment inside containment is not compromised.
6.2.1.4 Secondary System Pipe Ruptures and Containment Pressure/Temperature Response	• Containment design margin is maintained. • Environmental qualification of the safety related equipment inside containment is not compromised.
15.1.1 Feedwater System Malfunctions that Result in a Reduction in Feedwater Temperature	• Bounded by excessive increase in secondary steam flow analysis in Section 15.1.2 and same criteria apply.
15.1.2 Feedwater System Malfunction Causing an Increase in Feedwater Flow	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.1.3 Excessive Increase in Secondary Steam Flow	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.1.5 Steam System Piping Failure	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit based on an acceptable DNBR correlation. If the DNBR falls below these values, fuel failure must be assumed for all rods that do not meet these criteria. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. • Offsite doses calculated shall not exceed the guidelines of 10CFR100.

Table Q12-2
Acceptance Criteria for Non-LOCA Transients/Accidents
with MOX Fuel Lead Assemblies

Transient/Accident Description	Acceptance Criteria
15.2.1 Steam Pressure Regulator Malfunction or Failure That Results In Decreasing Steam Flow	• Not applicable, there are no pressure regulators in the McGuire or Catawba plants whose failure or malfunction could cause a steam flow transient.
15.2.2 Loss of External Load	• Bounded by turbine trip analysis in Section 15.2.3 and same criteria apply.
15.2.3 Turbine Trip	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.2.4 Inadvertent Closure of Main Steam Isolation Valves	• Bounded by turbine trip analysis in Section 15.2.3 and same criteria apply.
15.2.5 Loss of Condenser Vacuum and Other Events Causing a Turbine Trip	• Bounded by turbine trip analysis in Section 15.2.3 and same criteria apply.
15.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.2.7 Loss of Normal Feedwater Flow	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.2.8 Feedwater System Pipe Break	• Peak RCS pressure remains below 110 % of the design limit (<2750 psia) for low probability events. • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation. • No hot leg boiling occurs.
15.3.1 Partial Loss of Forced Reactor Coolant Flow	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.

Table Q12-2
Acceptance Criteria for Non-LOCA Transients/Accidents
with MOX Fuel Lead Assemblies

Transient/Accident Description	Acceptance Criteria
15.3.2 Complete Loss of Forced Reactor Coolant Flow	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. • Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10CFR100 guidelines.
15.3.4 Reactor Coolant Pump Shaft Break	• Bounded by reactor coolant pump shaft seizure analysis in Section 15.3.3 and same criteria apply.
15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low Power Startup Condition .	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation. • Fuel centerline temperatures do not exceed the melting point
15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation. • Fuel centerline temperatures do not exceed the melting point.
15.4.3 Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error) - Rod Drop	• Peak RCS pressure remains below 110% of the design limit (< 2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation. • Fuel centerline temperatures do not exceed the melting point
15.4.3 Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error) - Single Rod Withdrawal	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation. • Fuel centerline temperatures do not exceed the melting point • Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10CFR100 guidelines.

Table Q12-2
Acceptance Criteria for Non-LOCA Transients/Accidents
with MOX Fuel Lead Assemblies

Transient/Accident Description	Acceptance Criteria
15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	• Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10CFR100 guidelines.
15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents	• Peak RCS pressure remains below 120% of design for very low probability events (< 3000 psia). • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation. • Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. • The fission product release to the environment is well within the established dose acceptance criteria of 10CFR100. • See provisional cal/gm acceptance criteria attached.
15.5.1 Inadvertent Operation of Emergency Core Cooling System During Power Operation	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	• Bounded by inadvertent operation of emergency core cooling system during power operation analysis in Section 15.5.1 and same criteria apply.
15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve	• Peak RCS pressure remains below 110% of the design limit (<2750 psia) • Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation.
15.6.2 Break In Instrument Line or Other Lines From Reactor Coolant Pressure Boundary That Penetrate Containment	• Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10CFR100 guidelines.

Table Q12-2
Acceptance Criteria for Non-LOCA Transients/Accidents
with MOX Fuel Lead Assemblies

Transient/Accident Description	Acceptance Criteria
15.6.3 Steam Generator Tube Failure	• Fuel cladding integrity shall be maintained by ensuring that the calculated DNB ratio remains above the 95/95 DNBR limit based on an acceptable DNBR correlation. • Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10CFR100 guidelines.

Table Q12-3
Regulatory Dose Criteria
For Accidents with MOX Fuel Lead Assemblies

Accident	Classic Source Term	Reference	Alternative Source Term	Reference
Offsite Doses (EAB and LPZ)				
LOCA	300 Rem Thyroid 25 Rem WB[1]	RG[1] 1.195 10CFR100.11 SRP[1] 15.6.5 App. A	25 Rem TEDE	RG 1.183 10CFR50.67
Steam Generator Tube Rupture with fuel failure or pre-incident Iodine spike	300 Rem Thyroid 25 Rem WB	RG 1.195 10CFR100.11/ SRP 15.6.3	25 Rem TEDE	RG 1.183
Steam Generator Tube Rupture with concurrent Iodine spike	30 Rem Thyroid 2.5 Rem WB	RG 1.195 10CFR100.11/ SRP 15.6.3	2.5 Rem TEDE	RG 1.183 10CFR50.67
Main Steam Line Break with fuel failure or pre-incident Iodine spike	300 Rem Thyroid 25 Rem WB	RG 1.195 10CFR100.11/ SRP 15.1.5 App. A	25 Rem TEDE	RG 1.183 10CFR50.67
Main Steam Line Break with concurrent Iodine spike	30 Rem Thyroid 2.5 Rem WB	RG 1.195 10CFR100.11/ SRP 15.1.5 App. A	2.5 Rem TEDE	RG 1.183
Locked Rotor Accident	30 Rem Thyroid 2.5 Rem WB	RG 1.195 SRP 15.3.3	2.5 Rem TEDE	RG 1.183
Rod Ejection Accident	75 Rem Thyroid 6.3 Rem WB[2]	RG 1.195 SRP 15.4.8 App A	6.3 Rem TEDE	RG 1.183
Fuel Handling Accident	75 Rem Thyroid 6.3 Rem WB[2]	RG 1.195 SRP 15.7.4	6.3 Rem TEDE	RG 1.183
Control Room Doses				
All	50 Rem Thyroid[3] 5 Rem WB[3] 50 Rem skin	RG 1.195 10CFR50/ Appendix A/ GDC 19	5 Rem TEDE	RG 1.183 10CFR50.67

1 WB= Whole body, RG=Regulatory Guide, SRP= Standard Review Plan
2 Where a conflict exists between SRP and RG 1.195 on the whole body dose limit for a particular accident, the more current guidance is shown.
3 RG 1.195 specifically states that this criterion may be used in lieu of the one in the SRP.

13. To allow the NRC staff to perform confirmatory analysis, please provide both the McGuire and Catawba loss-of-coolant accident (LOCA) input decks for the low enriched uranium (LEU) as well as the MOX fuel rods. Provide the decks in an electronic format, including nodalization diagrams.

Response (Previously submitted October 3, 2003)
The accompanying compact disc includes two RELAP5/MOD2-B&W input decks in UNIX format as follows:

r5moxnrc.in - Input deck for MOX fuel pins, power peaked at 10.3 ft.
r5uo2nrc.in - Input deck for LEU fuel pins, power peaked at 10.3 ft.

These are blowdown input decks used in the deterministic evaluations of MOX and LEU fuel pins reported in the license amendment request. The deterministic MOX fuel calculations comprise the licensing basis for the MOX fuel lead assemblies. Deterministic LEU fuel calculations were included to address the relative LOCA performance between MOX and LEU fuel.

Figures Q13-1 and Q13-2 are node diagrams for the decks. Figure Q13-1 shows the loop node arrangement while Figure Q13-2 shows the reactor vessel node arrangement. Figure 3-5 of Attachment 3 to Reference Q13-1 provides some additional detail specific to the core region.

RELAP5/MOD2-B&W is a derivative of the INEL code RELAP5/MOD2. Many changes were made to the INEL code to create the approved Framatome ANP deterministic LOCA code. Because the input for these changes may not be recognizable by other versions of RELAP5, the following list of related input card images is provided to assist the NRC staff.

Card 190: EM Choking Model Specification Card
(Activates Framatome ANP specific choked flow break modeling.)

Card 192: EM Critical Flow Transition Data
(Activates Framatome ANP specific critical flow break modeling.)

Card 195: Interface Heat Transfer Weighting
(Activates Framatome ANP specific interface heat transfer weighting.)

Cards 10000020-10000029: Heat Structure Cards
(Activate Framatome ANP specific filtered flow model - 10CFR50.46 Appendix K requirement.)

Cards 10000S80-10000S99: Reflood Grid and Wall Heat Transfer Factor Data
(Activate Framatome ANP specific grid model for droplet breakup and convective heat transfer due to grids.)

Cards 1CCCG801-1CCCG899: Left Boundary Heat Structure Cards
Cards 1CCCG901-1CCCG999: Right Boundary Heat Structure Cards

(Activate the Framatome ANP specific EM heat transfer package.)

Cards 19997000-19999999: EM Pin Model Specification
(Activate Framatome ANP specific EM core package providing for dynamic fuel-clad gap conductance and fuel rod swell and rupture. Also provide the M5™ cladding properties.)

Reference

Q13-1. Tuckman, M. S., February 27, 2003 Letter to U.S. Nuclear Regulatory Commission, Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50.

Figure Q13-1
Loop Noding Diagram

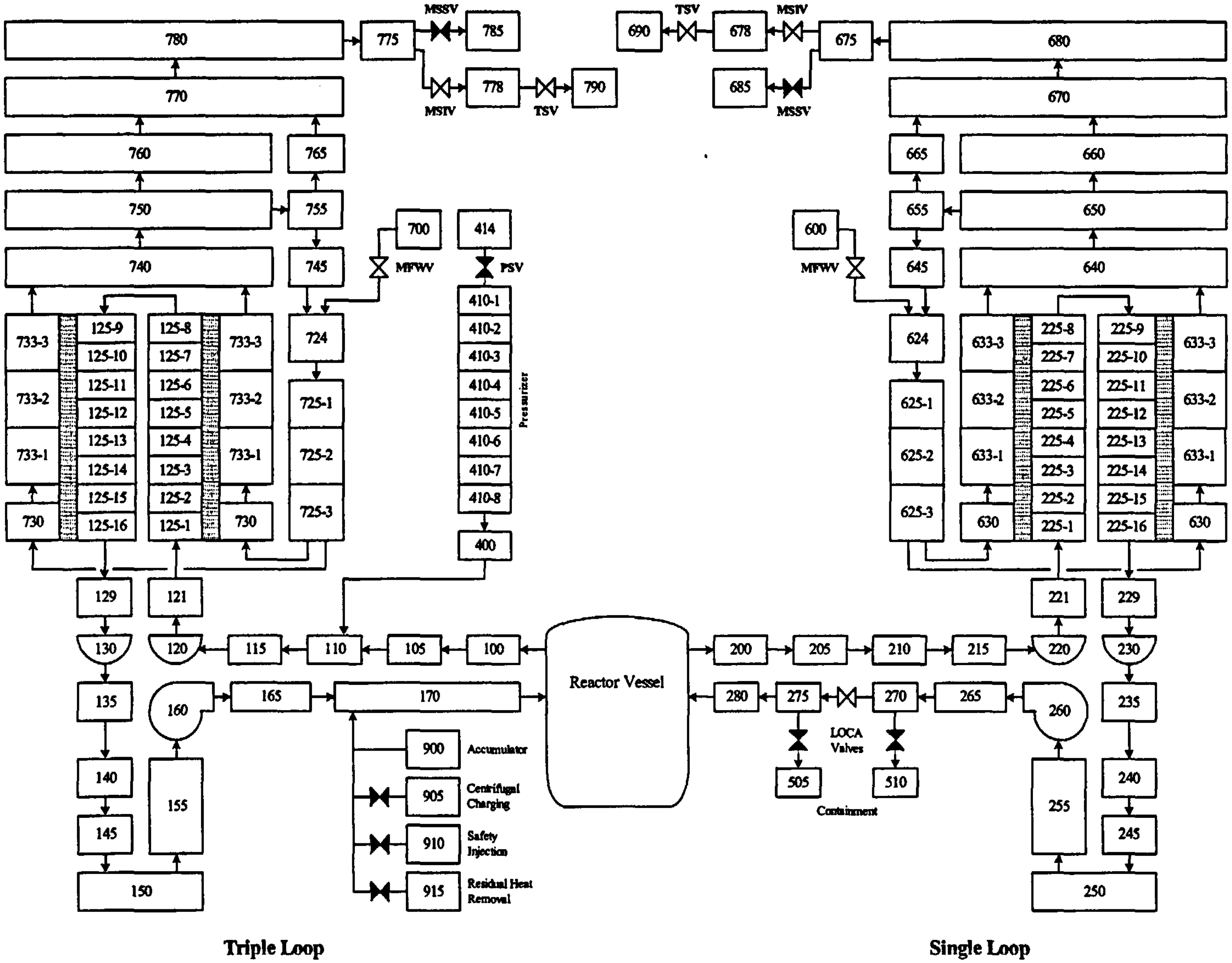
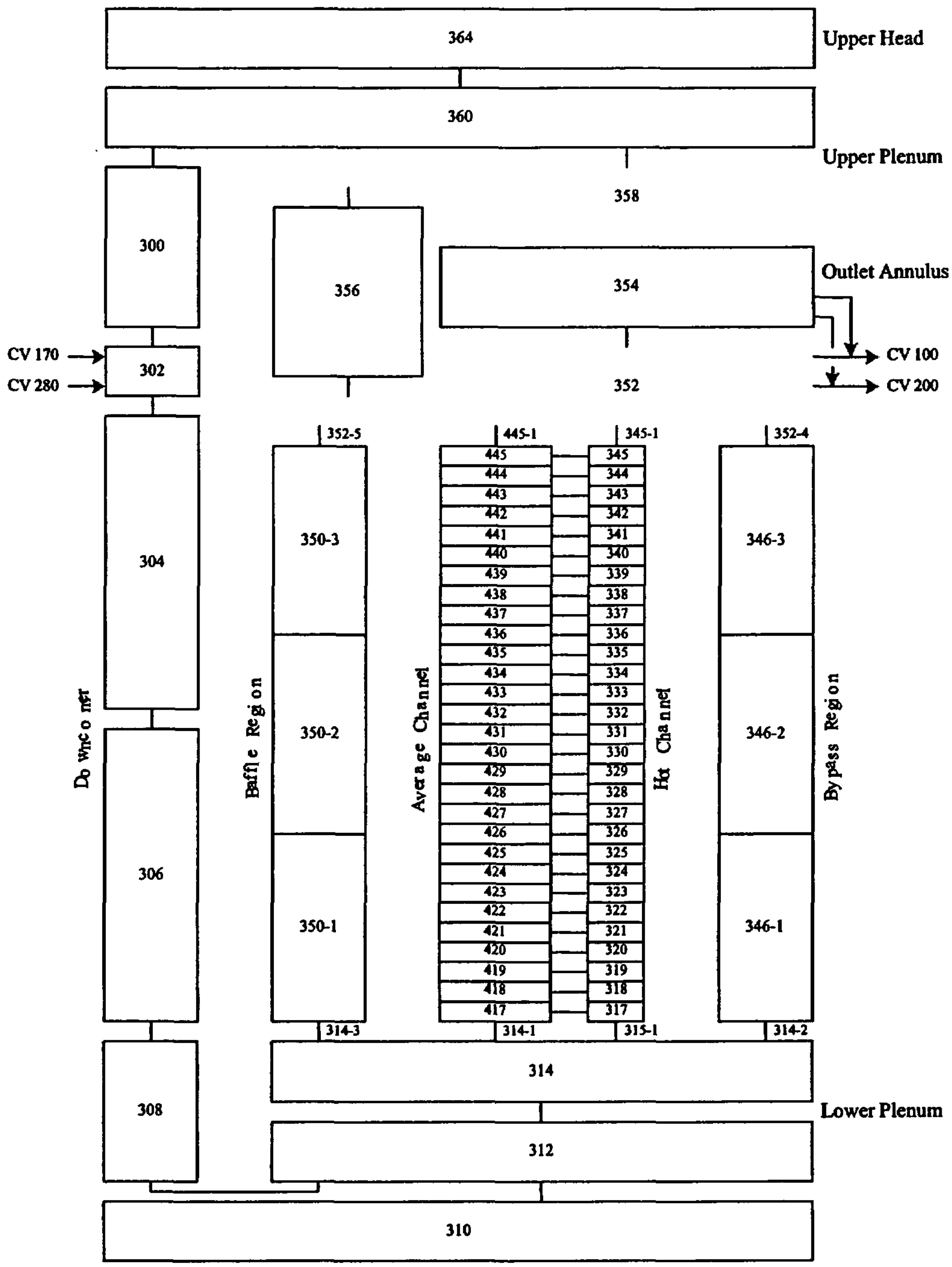


Figure Q13-2
Core Noding Diagram



14. Provide the reference to the best estimate LOCA model noted in section 3.7.1.7.

Response (Previously submitted October 3, 2003)
Based on RAI Questions 14, 15, and 16 it appears that some clarification is needed with respect to the LOCA analysis performed for the MOX fuel lead assemblies and how this analysis is used to support the lead assembly cores. In summary, the licensing basis for the resident Westinghouse RFA fuel remains the best estimate large break LOCA analysis performed by Westinghouse. Framatome ANP Appendix K analyses demonstrate that changing the fuel pellet material to MOX has no significant impact on peak cladding temperature following a large break LOCA. Framatome ANP Appendix K analyses provide peaking limits that ensure the peak cladding temperature for MOX fuel rods following a large break LOCA remain within the regulatory limit. The following discussion provides a further description of the analysis performed for the resident fuel assemblies as well as the MOX fuel lead assemblies.

Resident Fuel
The resident fuel in MOX fuel lead assembly cores will be the robust fuel assembly (RFA) design that is supplied by Westinghouse. The large break LOCA analysis that supports this fuel design is the Westinghouse best estimate method described in Reference Q14-1. The analysis is based on the WCOBRA/TRAC method and includes detailed treatment of the uncertainties associated with the computer models and the inputs related with plant operation. As part of the analysis, Westinghouse performed sensitivity studies to address transition or mixed core effects. This was necessary because the RFA fuel was initially introduced into cores containing Framatome ANP Mark-BW design fuel. The conclusion of the mixed core sensitivities was that the presence of the Mark-BW fuel assemblies had an insignificant impact on the calculated results. Westinghouse also performed small break LOCA calculations for McGuire and Catawba using the NOTRUMP methodology as described in Reference Q14-2. A mixed core penalty of 10°F was assessed and applied to the small break LOCA results to accommodate the presence of the Mark-BW fuel assemblies. Given that the MOX fuel lead assemblies are more similar hydraulically to the RFA fuel than the Mark-BW design fuel, the mixed core penalty developed for the Mark-BW fuel assemblies bounds the MOX fuel lead assemblies. Therefore, the Westinghouse LOCA analyses for the resident RFA fuel remain valid in the presence of four MOX fuel lead assemblies.

MOX Fuel Lead Assemblies
To address the MOX fuel lead assemblies, Framatome ANP performed deterministic large break LOCA calculations consistent with the requirements of 10 CFR 50 Appendix K. In order to model accurately the effect of changing the fuel pellet material to MOX, Framatome ANP made modifications to their deterministic large break LOCA method as described in Reference Q14-3. These modifications are described in Section 3.7.1.2 of Attachment 3 to Reference Q14-4. Next, Framatome ANP performed large break LOCA calculations for a MOX fuel lead assembly as well as a Framatome ANP LEU fuel assembly, with both analyses assuming the hydraulic characteristics of the Advanced Mark-BW fuel assembly design. This sensitivity study was performed to assess the impact of the change in fuel rod parameters (MOX vs. LEU) on the calculated results. As discussed in Section 3.7.1.3 of Attachment 3 to Reference Q14-4, this sensitivity study showed that there is essentially no difference between the LOCA results for the MOX

fuel and the LEU fuel (ΔPCT of 37°F). The Framatome ANP MOX fuel lead assembly results were also compared to the Westinghouse best estimate results to illustrate the similarity of the results. Given the differences in the two analytical methods, a direct comparison of the results is not completely valid. However, the comparison illustrates that the MOX fuel lead assembly with the lower peaking assumptions yields lower peak cladding temperature results (ΔPCT of -38°F).

Following submittal of the MOX fuel license amendment request, Framatome ANP completed additional cases to investigate the impact of steam generator type, time in life, and axial power shape. Two different steam generator designs were examined: Westinghouse Model D5 steam generators (Catawba Unit 2), with a 10% tube plugging assumption; and BWI steam generators (Catawba Unit 1), with 5% tube plugging. The study concluded that the Model D5 steam generators with the 10% tube plugging assumption are limiting with respect to the Framatome ANP deterministic large break LOCA analysis. The D5 case provided the base case input for the other sensitivities cases.

Framatome ANP performed time in life sensitivities to assess the large break LOCA results as the stored energy in the fuel rod varies with cycle burnup. At burnups greater than 30 GWd/MThm, a K_{BU} factor is applied to limit the PCT for these cases. The K_{BU} factor reduces the F_Q (total peaking factor) as well as the $F\Delta h$ (enthalpy rise factor or radial peaking factor).

Furthermore, using the limiting burnup case which uses a K_{BU} of 1.0, i.e., the 30 GWd/MThm case, Framatome ANP evaluated power peaks at different elevations. The purpose of these sensitivities was to establish LOCA limits as a function of core height. At elevations above the 8 foot elevation a K_Z factor was applied. The K_Z factor reduces the F_Q as well as the axial peaking factor (F_Z).

A summary of the sensitivity cases is provided in Table Q14-1. The resulting LOCA peaking requirements for the MOX fuel lead assemblies are shown in Figure Q14-1. These peaking requirements will assure that the MOX fuel will comply with the regulatory limits for LOCA as provided in the response to Reactor System RAI Question 12.

MOX Fuel Lead Assembly Licensing Basis

The licensing of the MOX lead assemblies will be based on analysis to determine the relative accident performance between the MOX and resident LEU assemblies because of the different fission source materials. As presented in the license amendment request, large break LOCA calculations, using the Framatome ANP deterministic LOCA evaluation model, have been performed for both LEU and MOX assemblies. The LEU calculations applied the evaluation model as approved by NRC. The MOX calculations applied the evaluation model with specified alterations, described in the LAR, necessary to simulate MOX fuel. The comparison of these two calculations demonstrated the expected result: that there is essentially no difference in the large break LOCA performance between fuel, of comparable design, using MOX pellets and fuel using LEU pellets. An evaluation of the small break LOCA, provided in the LAR, also determined that there would be no differences in the calculated results between the MOX and LEU fuel assemblies. Therefore, the assessment of the Catawba LOCA performance for the cores with four MOX lead assemblies is that LOCA performance is not altered. This result, in combination with a reduction in the allowed peaking factor for the MOX fuel pins, provides the licensing basis for the MOX fuel lead assemblies assuring that all of the criteria of 10CFR50.46 are met.

References

Q14-1. WCAP-12945P-A, Volume 1 Revision 2 and Volumes 2-5 Revision 1, *Code Qualification Document for Best-Estimate Loss of Coolant Analysis*, March 1998.

Q14-2. WCAP-100564P-A, *Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code*, August 1985.

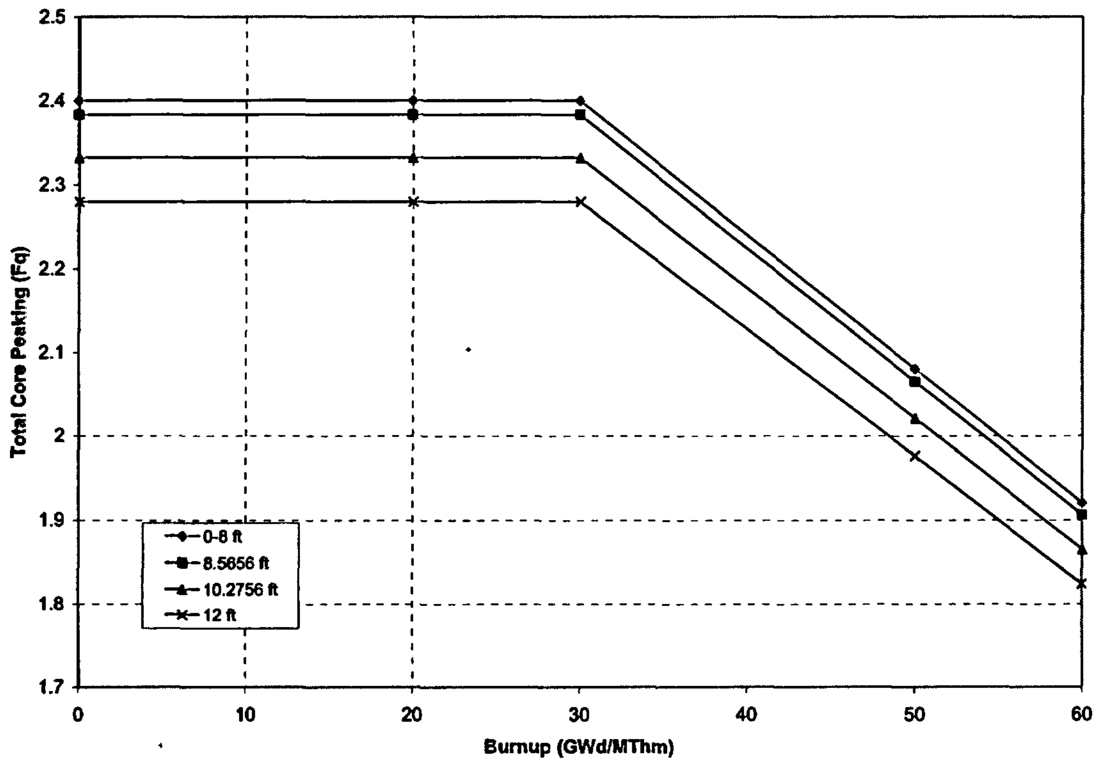
Q14-3. BAW-10168P-A, Revision 3, *RSG LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants*, December 1996.

Q14-4. Tuckman, M. S., February 27, 2003 Letter to U.S. Nuclear Regulatory Commission, Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50.

Table Q14-1
Summary of MOX Fuel Lead Assembly
Large Break LOCA Sensitivity Cases
Model D5 SGs with 10% Tube Plugging

TIL (GWd/MThm)	Elevation (ft)	K_{BU}	K_Z	$F_{\Delta h}$	F_Z	Fq	PCT (°F)
BOL	6.8556	1.0	1.0	1.6	1.500	2.4	1919.2
20	6.8556	1.0	1.0	1.6	1.500	2.4	1943.6
30	6.8556	1.0	1.0	1.6	1.500	2.4	1948.8
50	6.8556	0.867	1.0	1.387	1.500	2.08	1824.4
60	6.8556	0.8	1.0	1.280	1.500	1.92	1787.6
30	4.7001	1.0	1.0	1.6	1.500	2.4	1815.0
30	8.5656	1.0	0.993	1.6	1.490	2.383	1964.0
30	10.2756	1.0	0.972	1.6	1.458	2.332	2019.5

Figure Q14-1
MOX Fuel Lead Assembly Total Core Peaking Factor



15. Provide the uncertainty analysis that was performed for the LEU and MOX LTA demonstrating that the 95/95 peak cladding temperature has been calculated for the core. The response is expected to include a complete discussion of the statistical methodology used.

    Response (Previously submitted October 3, 2003)
    The MOX fuel and LEU fuel LOCA analyses that support the use of the MOX fuel lead assemblies are deterministic calculations and therefore no uncertainty analysis was performed. See the response to Reactor Systems RAI Question 14 for additional explanation.

16. Section 3.7.1 states that the LOCA model used for the LEU fuel is a best estimate model. Provide the Phenomena Identification and Ranking Table for the LOCA analyses performed with the best estimate model and reference the best estimate model used for the analysis.

    Response (Previously submitted October 3, 2003)
    The Phenomena Identification and Ranking Table (PIRT) used in the Westinghouse best-estimate LBLOCA analysis is contained in Reference Q16-1. Since this method was not used to directly support the MOX fuel lead assemblies, this PIRT is not applicable to the MOX fuel lead assembly analysis. See the response to Reactor Systems RAI Question 14 for additional explanation.

    Reference
    Q16-1. WCAP-12945P-A, Volume 1 Revision 2 and Volumes 2-5 Revision 1, *Code Qualification Document for Best-Estimate Loss of Coolant Analysis,* March 1998.

17. Provide the experimental data base used to assess the biases and to determine the uncertainties in the fuel rod behavior for the MOX LTA.

    Response (Previously submitted October 3, 2003)
    The database is provided in Chapter 3 of the COPERNIC topical report (Reference Q17-1). Additionally, at the NRC's request, several MOX fuel rods from the Halden experiments were analyzed with COPERNIC to end-of-life burnups in the range of 50 to 64 GWd/MThm.

    Reference
    Q17-1. BAW-10231P Revision 2, *COPERNIC Fuel Rod Design Computer Code*, July 2000.

18. In sub-section 3.7.1.1.1, nothing is mentioned about the MOX/LEU interface behavior. Provide a qualitative and quantitative discussion regarding the neutron flux behavior at the interface of the MOX and LEU fuel assemblies.

    Response (Previously submitted October 3, 2003)
    Duke used the CASMO-4 computer code to model pin cell neutron flux and power at the intersection of four quarter-assembly lattices. These "colorsets" provide detailed two dimensional neutronic calculations that account for interface effects between dissimilar fuel assemblies. MOX fuel assemblies and LEU fuel assemblies of equivalent lifetime

21. How does the lower fuel conductivity of the MOX fuel impact the maximum pellet centerline temperature during a LOCA as compared to LEU fuel? Please provide a qualitative and quantitative discussion of the differences.

Response

There is only a slight difference in the fuel pellet conductivity between MOX fuel of the lead assembly design and plutonium concentration and comparable LEU fuel. Figure Q21-1 compares the thermal conductivity for MOX fuel pellets of the lead assembly design to comparable LEU fuel pellets for both un-irradiated fuel and fuel irradiated to 40 GWd/MThm. The thermal conductivity values shown in Figure Q21-1 are from the fuel performance code COPERNIC (Reference Q21-1). COPERNIC has been approved by NRC for use with LEU fuel and is under review for MOX fuel applications. Although thermal conductivity values in Figure Q21-1 change with burnup for both MOX fuel and LEU fuel, the offset, approximately two percent, is constant.

The analyses presented in Section 3.7.1 of Attachment 3 to the license amendment request (Reference Q21-2) directly compare the effect of the MOX to LEU offset in conductivity in conjunction with the other differences in the fuel pin designs. Figures Q21-2 and Q21-3 provide a fuel pin temperature profile comparison between MOX and LEU fuel pellets at the accident initial conditions and at the approximate time of peak cladding temperature. As expected, there is little difference in the temperature distributions between the two fuel types. Figure Q21-4 provides the evolution of the centerline fuel temperatures with time for the MOX and LEU fuel at the location of peak cladding temperature. The two fuel temperatures differ slightly during the course of the transient. The variation is attributed to fuel pellet thermal conductivity and to other differences in the fuel pin design. As an example, the LEU fuel pin has a higher pre-fill pressure than the MOX pin. The higher pressure increases the hoop stress resulting in a slightly lower calculated rupture temperature and earlier calculated rupture time. Combined with all of the models interacting to determine the cladding and pellet temperatures the LEU fuel centerline temperature is 40°F cooler at the time of peak cladding temperature.

The difference in thermal conductivity between MOX fuel of the lead assembly design and comparable LEU fuel is small. The effect of this difference on LOCA calculational results is nil and not distinguishable from the effects of normal fuel design variations.

References

Q21-1. BAW-10231P Revision 2, *COPERNIC Fuel Rod Design Computer Code*, July 2000.

Q21-2. Tuckman, M. S., February 27, 2003, Letter to U.S. Nuclear Regulatory Commission, Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50.

Figure Q21-1
Thermal Conductivity Comparison for MOX and LEU Fuel
(Fuel porosity of 0.0479)

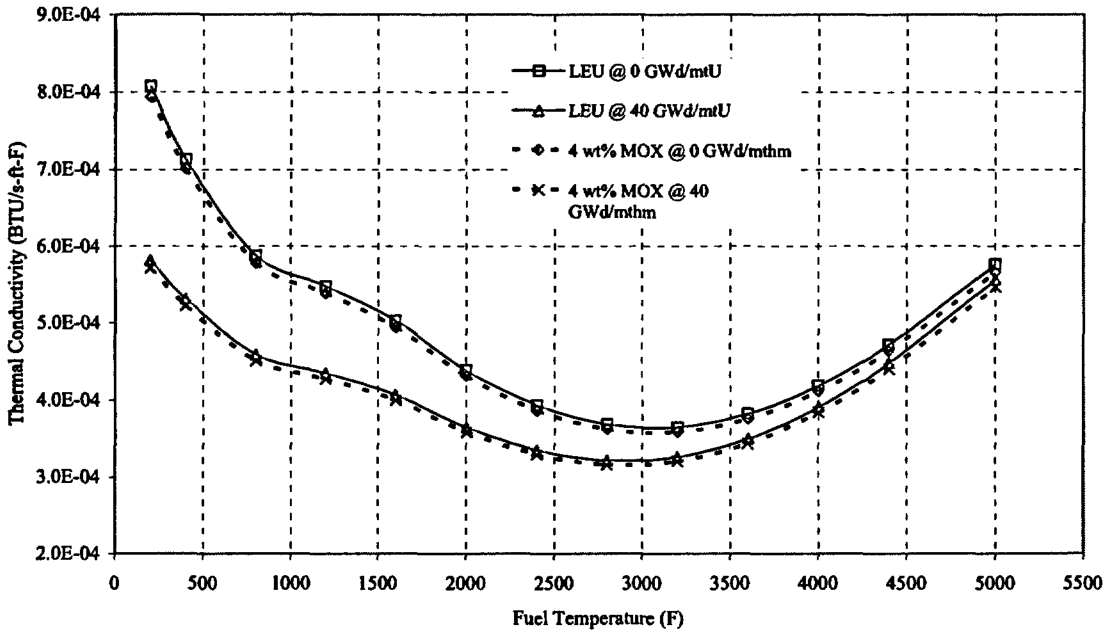


Figure Q21-2
MOX and LEU Fuel Pin Temperature Profile Comparison
at Loss of Coolant Accident Initiation

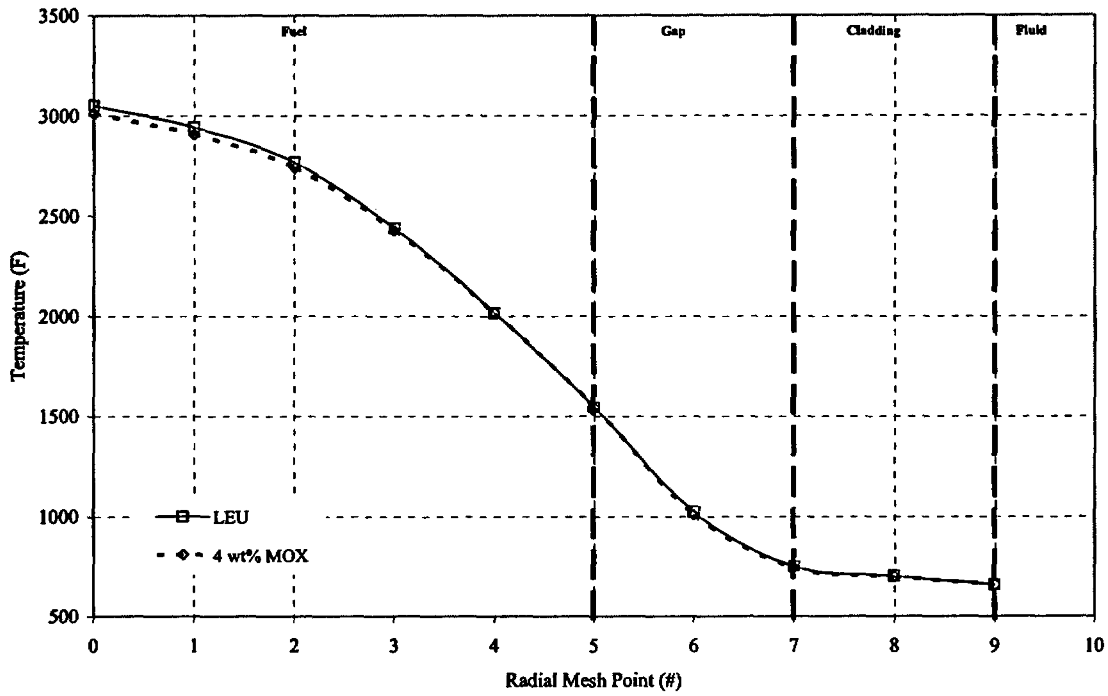


Figure Q21-3
MOX and LEU Fuel Pin Temperature Profile Comparison
at Time of Peak Cladding Temperature

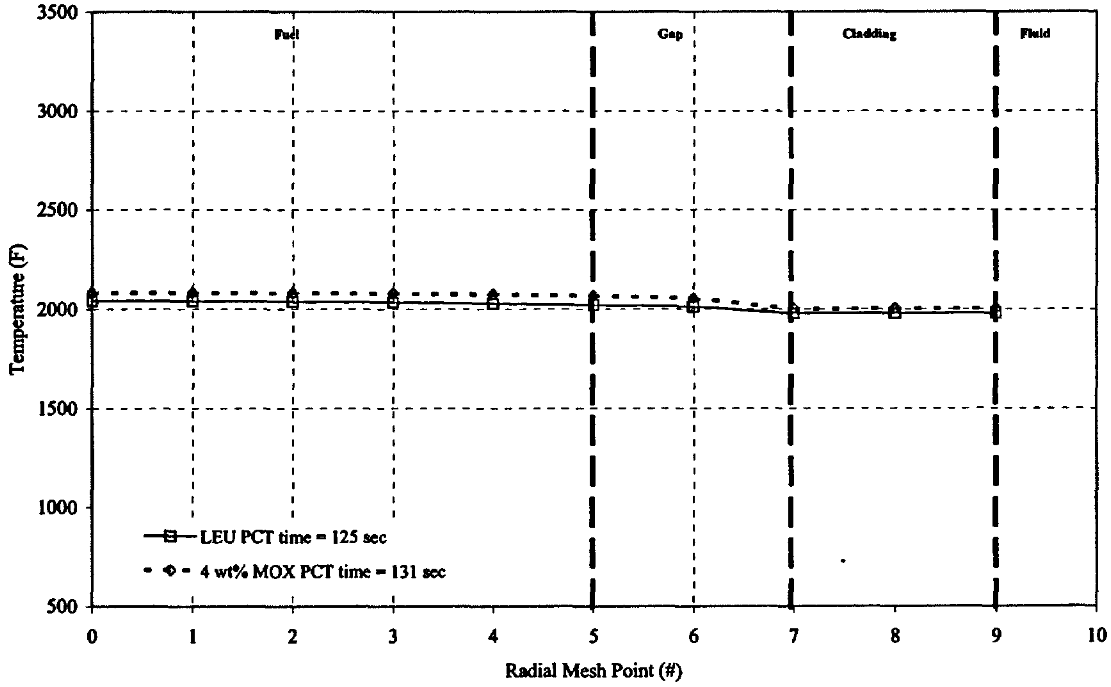
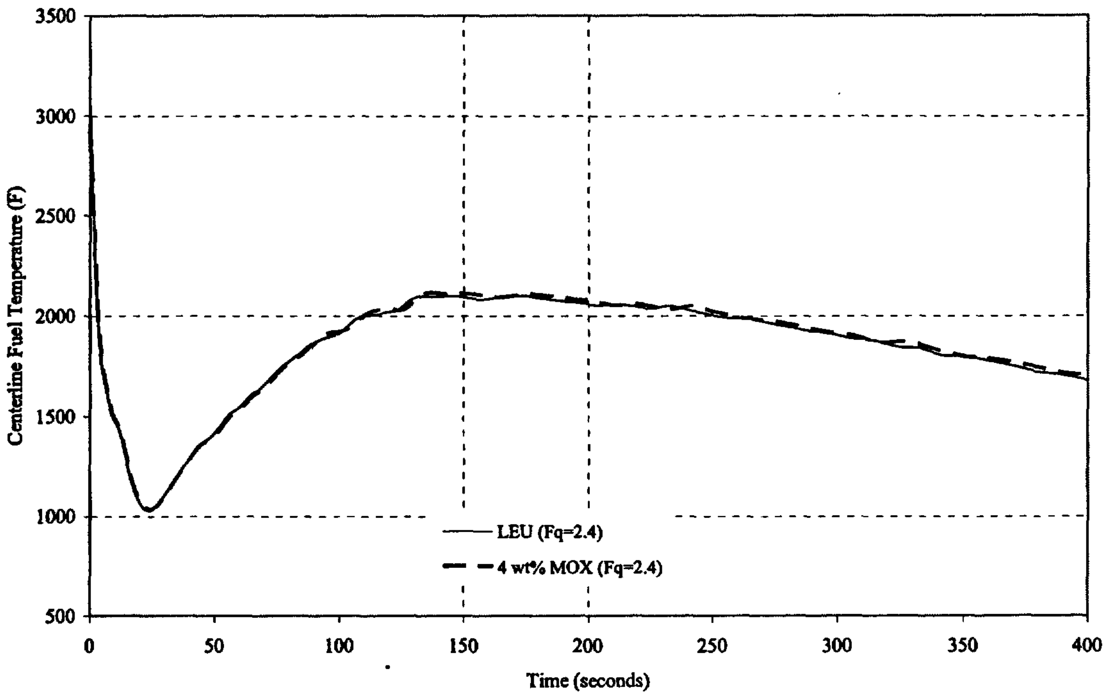


Figure Q21-4
MOX and LEU Fuel Centerline Temperature Comparison for Loss of Coolant Accident



22. The first paragraph of section 3.7.1.1.2 states that "The result, including appropriate uncertainties, is that .." Please state the uncertainties that are being referred to in this section along with what is considered to be appropriate.

Response
References Q22-1 and Q22-2 are industry standard tools for calculating decay heat for low-enriched uranium (LEU) cores. Analysis of highly burned LEU fuel shows that it produces the majority of its energy from the fission of plutonium isotopes. Therefore, these standard tools are appropriate for calculating decay heat in cores containing MOX fuel and for determining the uncertainties to be applied.

The uncertainties included in the MOX fuel decay heat analysis include:

(1) ANSI/ANS-5.1-1994 standard uncertainties for infinite irradiation by isotope,
(2) ANSI/ANS-5.1-1994 "ISO standard" for energy released from fission (the "Q" value),
(3) ANSI/ANS-5.1-1994 standard for absorption burnup correction factors, G_{max} (t), and
(4) actinide decay uncertainties.

Many of these values are a function of time after shutdown. Table Q22-1 shows the effect of time after shutdown on each of these uncertainties.

To obtain a reasonable statistical (95/95) tolerance/confidence factor to apply to the one sigma uncertainty, the Appendix K requirement and the standards were examined. As explained in the ANSI/ANS-5.1-1994 standard, the 1.2 uncertainty factor was based on work reported in the Bettis Technical Review by K. Shure. Shure's work stated that a relative uncertainty of 20% would bound all positive deviations in decay periods less than 10^7 seconds. The measured data indicate that the one sigma uncertainty is about 10%. Thus, there is a factor of two in the Appendix K requirements between the sigma and the bounding value. This implies that a tolerance/confidence factor of two is acceptable to use as a 95/95 percent level of confidence in the determination of conservative decay heat calculations. The MOX fuel decay heat model uses a tolerance/confidence factor of two applied to the uncertainties.

The 95/95 actinide decay heat fraction and the 95/95 fission product decay heat fraction are calculated and summed to produce the MOX fuel decay heat model. Comparing the results of the 95/95 MOX model with the standard Appendix K decay heat model for LEU fuel shows that the LEU model produces higher values of decay heat than MOX fuel. This is shown in Figure 3-3 of Attachment 3 of Reference Q22-3 for LOCA-typical decay times.

References
Q22-1. American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1994, American Nuclear Society, 1994.
Q22-2. O.W. Hermann, R.M. Westfall, *ORIGEN-S: Scale System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms*, NUREG/CR-0200, September 1998.

Q22-3. Tuckman, M.S., February 27, 2003, Letter to U.S. Nuclear Regulatory Commission, Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50.

Table Q22-1
Effect of Time after Shutdown on Decay Heat Uncertainty Factors

Uncertainty (From Text of Q22)	Parameter	Uncertainty Value (1.0 Second after Shutdown)	Uncertainty Range (10^0 to 10^7 Seconds after Shutdown)	Reference
(1)	One sigma uncertainty for ^{235}U fission product decay heat	2.8%	1.7 - 2.8%	ANSI / ANS-5.1-1994, Page 14
(1)	One sigma uncertainty for ^{238}U fission product decay heat	9.0%	3.8 – 9.0%	ANSI / ANS-5.1-1994, Page 18
(1)	One sigma uncertainty for ^{239}Pu fission product decay heat	4.5%	3.6 – 5.3%	ANSI / ANS-5.1-1994, Page 16
(1)	One sigma uncertainty for ^{241}Pu fission product decay heat	5.4%	4.4 – 10.0%	ANSI / ANS-5.1-1994, Page 20
(2)	Q- sigma for ^{235}U (MeV per Fission)	$\pm$0.5	NA	ANSI / ANS-5.1-1994, Page 38
(2)	Q- sigma for ^{238}U (MeV per Fission)	$\pm$1.0	NA	ANSI / ANS-5.1-1994 , Page 38
(2)	Q- sigma for ^{239}Pu (MeV per Fission)	$\pm$0.7	NA	ANSI / ANS-5.1-1994, Page 38
(2)	Q- sigma for ^{241}Pu (MeV per Fission)	$\pm$0.7	NA	ANSI / ANS-5.1-1994, Page 38
(3)	G_{max} (t) (Note 1)	2%	2.0 – 18.1%	ANSI / ANS-5.1-1994 , Page 26
(4)	^{239}U decay heat one sigma uncertainty	10%	NA	Note 2
(4)	^{239}Np decay heat one sigma uncertainty	15%	NA	Note 2
(4)	Decay heat for all other actinides one sigma uncertainty	20%	NA	Note 2

NA – Not applicable because there is no apparent time dependence of this parameter.

Note 1: $G_{max}(t)$ is the maximum correction relative to the nominal value of G(t).
Note 2: The actinide decay heat uncertainties are estimated based on the accuracy of ORIGEN-S and measured data.

24. Section 3.7.1.1.4 discusses the LOCA transient initialization and the changes made to accommodate using the COPERNIC code instead of the TACO3 code, including the adjustments made to some of the parameters. Provide additional information on the adjustments made, how the adjustments were developed and include any data used to develop the adjustment. Additionally, since these values are used in RELAP5 initialization, please show that throughout the fuel lifetime, the TACO3 and COPERNIC codes predict consistent values for the different fuel parameters used as input for the LOCA analysis.

Response

The discussion in Section 3.7.1.1.4 of Attachment 3 to Reference Q24-1 concerns alterations in the approach used to determine the fuel-to-clad gap conductance and in the values used for the initial fuel temperatures in the three core heat structures of the LOCA simulation. The approach to the fuel-to-clad gap conductance is described in detail in the response to Reactor Systems Question 25. The following discussion presents additional detail regarding the determination of the initial fuel temperatures for the core heat structures.

Because COPERNIC is NRC-approved for LOCA application to LEU fuel and includes modeling for MOX fuel properties, it was selected for the prediction of initial fuel temperatures for the MOX simulations and for the LEU comparison case. COPERNIC is an advanced fuel performance code relative to TACO3 and predictive consistency between COPERNIC and TACO3 should not be expected.

The Framatome ANP deterministic LOCA evaluation model, used to evaluate the MOX fuel lead assemblies, incorporates a two coolant channel, three heat structure core model to assure that the coolant and pin conditions for the hot spot are appropriate. The two coolant channels represent flow in the average core and flow in the hot fuel assembly respectively. The three heat structures represent the average core, the hot bundle, and the hot pin. Both the hot bundle and the hot pin couple thermal-hydraulically with the hot fuel assembly fluid channel. Figure 3-5 of Attachment 3 to Reference Q24-1 illustrates the arrangement. The NRC approved this core representation in, Reference Q24-2.

LOCA calculations include provision for appropriate uncertainties in both transient and initial conditions. One of those uncertainties is the initial fuel temperature or initial stored energy used in the core simulation. To determine the initial fuel temperatures, an NRC-approved fuel performance code, such as COPERNIC or TACO3, is run in accordance with the plant boundary conditions and core power distributions to be simulated. These codes produce best estimate predictions of the core temperature distributions that are transferred, after adding appropriate prediction uncertainties, to RELAP5/MOD2-B&W for the LOCA calculations. The uncertainties are determined from the benchmarks of the fuel performance codes and the make-up of the core region being modeled in RELAP5/MOD2-B&W.

For the hot pin, the LOCA calculation resolves a conservative representation of a single region of fuel pellets in a single rod. The appropriate level of uncertainty to add to the hot pin initial temperature prediction is a temperature increment that gives a 95/95 confidence that the resultant temperature is not under predicted. For COPERNIC, the fuel performance code used for MOX simulations, this would comprise an addition of [ ] to the prediction of the fuel temperatures along the entire hot pin. For a TACO3-based evaluation, 11.5 percent of the predicted fuel temperature would be added.

For the average core, the LOCA calculation resolves a representation of a large group of fuel pellets in many rods. The appropriate level of uncertainty to add to the initial temperature predictions includes the integration of individual pellet uncertainties over this entire group and a determination of the 95/95 confidence band for the entire group. With the size of the group involved, the aggregate uncertainty is near zero and it is appropriate to initialize this group, the average core, at the fuel performance code prediction without adjustment. With this selection, the COPERNIC [ ] the benchmark temperatures is conservatively ignored.

The more interesting initialization is that for the hot bundle representation. The purpose of the hot bundle is to provide the coolant conditions with which to cool the hot pin. As such, the hot bundle configuration is selected to represent the aggregate of the eight fuel pins immediately surrounding the hot pin. For TACO3, the appropriate 95/95 confidence level for the aggregate initial temperature or stored energy of a group of eight pins requires that the TACO3 prediction be increased by about 2.5 percent. The modeling approved by the NRC in Reference Q24-2 stipulated that the temperature prediction be increased by 3.0 percent to provide a small additional conservatism. The determination of the increase is dependent on the distribution of the uncertainty and bias for the fuel performance code. The TACO3 uncertainty distribution is a Gaussian or normal distribution and the difference in a temperature adjustment to achieve 95/95 confidence between a single member set and the set representing the eight fuel pins surrounding the hot pin is significant, 11.5 percent for the hot pin and about 2.5 percent for the surrounding pins. If the uncertainty distribution for COPERNIC is close to Gaussian, there will be little difference in the relative temperature adjustment. That is, the appropriate adjustment will be the same fraction of the 95/95 adjustment factor for both codes. In determining the uncertainty adjustment for COPERNIC applications, it was assumed that the COPERNIC uncertainty distribution was sufficiently close to Gaussian to employ this logic.

The justification of this argument only requires that the distribution of uncertainty for COPERNIC be reasonably normal and that the temperature adjustment providing a 95/95 confidence for a single member set be known. That the COPERNIC uncertainty is reasonably normal can be observed in a comparison of the TACO3 uncertainty distribution to a histogram of the COPERNIC benchmarks. This comparison is presented in Figure Q24-1 as normalized predicted minus measured data. By observation, the uncertainty distribution for COPERNIC, if correlated, would not differ markedly from that of TACO3 except for a slightly different bias. Thus, for the LOCA evaluation of the MOX lead assemblies, the COPERNIC prediction of the hot bundle initial temperature was increased by the ratio of hot bundle to hot pin adjustment for TACO3 times the hot pin adjustment for COPERNIC, 3.0/11.5 times [ ].

Reference

Q24-1. Tuckman, M. S., February 27, 2003, Letter to U.S. Nuclear Regulatory Commission, Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50

Q24-2. Letter, U.S. Nuclear Regulatory Commission to Framatome ANP, *Safety Evaluation of Framatome Technologies Topical Report BAW-10164P Revision 4, RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis,* April 9, 2002.

Figure Q24-1
TACO Uncertainty Distribution Compared to
COPERNIC Benchmark Histogram

25. Section 3.7.1.1.4 discusses RELAP5 initialization, stating that the core model will not be in steady state at transient initialization. Since a false declared steady state can lead to errors from an imbalance, please provide justification for why the RELAP5 model will not be in steady state at transient initiation and how steady state conditions for initialization are assured.

Response
The RELAP5/MOD2-B&W code includes a fuel pin model that represents the fuel rod in accordance with the requirements of 10CFR50 Appendix K. This model explicitly considers the fuel pellet, fuel-to-clad gap and clad-to-coolant heat transfer. It allows for specification of material conductivities for the pellet, gap, and cladding. The gap conductance term accounts for gaseous conductance, fuel pellet-to-cladding contact and radiation.

The initial fuel thermal conditions for LOCA are determined by an NRC-approved steady-state fuel performance code. For the analysis of the MOX lead assemblies, COPERNIC is used. The following input from COPERNIC is transferred to RELAP5/MOD2-B&W:

- Fuel rod temperatures after adjustment for uncertainties (29 axial and 10 radial nodes),
- Fuel pellet and cladding radial geometry,
- Fuel-to-cladding contact pressure,
- Initial internal fuel pin pressure,
- Fuel thermal conductivity, and
- Gas composition.

COPERNIC provides a best-estimate calculation of the initial fuel temperature distributions. To provide suitable inputs for RELAP5/MOD2-B&W, appropriate uncertainties are added to the predicted temperatures when they are transferred. This increase in temperature combined with the fact that COPERNIC and RELAP5/MOD2-B&W have slightly different gap models means that the steady-state initial fuel temperature predictions for the two codes will differ. Previous LBLOCA analyses, based on the TACO3 fuel performance code, accounted for the differences by the application of a gap gaseous conductance multiplier. The multiplier, which was held constant throughout the transient, forces the initial fuel temperature prediction of RELAP5/MOD2-B&W to match the fuel performance code prediction plus uncertainty.

An evaluation of the RELAP5/MOD2-B&W and COPERNIC gap conductance models was performed to understand the differences between the models and to determine whether the application of a constant gap gaseous conductance multiplier (determined at steady-state) remained the appropriate method for accounting for the differences between the models and for the uncertainty adjustment of the initial fuel temperatures. Figure Q25-1 illustrates the differences between the RELAP5/MOD2-B&W and COPERNIC gap gaseous conductance models. The figure shows the multiplier on the RELAP5/MOD2-B&W term that would be necessary for it to match the COPERNIC prediction as a function of steady-state gap thickness. The gap thickness effectively translates to the inverse of time-in-life, where open gap conditions exist at BOL and the gap closes and contact pressures develop with increasing burnup.

The results of the evaluation determined that RELAP5/MOD2-B&W and COPERNIC provide similar gap conductance results when the gap thickness is relatively large. However, there was a noticeable difference in the gap conductance when the gap is small. The accounting of gaseous conductance for gas space between rough surfaces in contact differs between the two codes. Although a gaseous conductance multiplier would allow RELAP5/MOD2-B&W to generate an initialization that matched the uncertainty-adjusted COPERNIC fuel temperatures, the multiplier value would be large for small gaps and applicable only so long as the gap remains small.

Figure Q25-2 demonstrates the transient gap thickness for LBLOCAs initialized at BOL and 45,000 MWd/MtU. For BOL, the gap is initially open and the increased transient gap does not significantly alter the gaseous conductance. A multiplier of between one and two could be applied without significantly affecting the transient simulation. However, for exposed fuel, the initialization multiplier based on the gaseous conductance model may be as high as six and would only be reduced to between two and three by application of the COPERNIC fuel pellet temperature uncertainties. Such a multiplier would quickly become inappropriate as the gap opens during the transient. Because RELAP5/MOD2-B&W does not have the ability to modify the gap gaseous conductance multiplier during the transient, and it is apparent that the multiplier should be less than two after about five seconds, the gaseous conductance multiplier approach was deemed inappropriate for COPERNIC-based LOCA calculations.

RELAP5/MOD2-B&W does have the capability to directly specify the initial fuel rod temperatures independent of the gap conductance. It is, therefore, possible to force the initial heat structure temperatures to the correct values, albeit by giving up a strict steady-state configuration. To determine the effects of starting the core in a non-steady-state condition, a study of several fuel pins with differing gap coefficients was performed. LOCA simulations with multiple hot fuel pins, each with the same initial fuel temperature distribution (input specified), but with gaseous conductance multipliers varying from 0.5 to 2.0 were run. The results, Figure Q25-3, demonstrated timing differences in cladding heating and cooling rates, particularly in the first few seconds of the transient. However, the overall cladding and fuel temperature trends were preserved and no significant peak cladding temperature differences were noted. The initial heatup of the cladding and cooldown of the fuel pellet occurred quicker with a high gaseous conductance multiplier. For reduced gaseous conductance, the opposite was true. After the initial heatup, however, the offset of the cladding and fuel temperatures is aligned to compensate for the differences in the gap conductance and the cladding temperature response are thereafter consistent in both timing and magnitude. Because the fuel energy decrease is delayed for the lower gap conductance, fuel temperatures tend to remain higher during the refill and reflood portions of the LBLOCA, resulting in a tendency for a slightly higher cladding temperature during this phase. Furthermore, because the cladding temperature response is, for the most part, consistent, it can be inferred that the core energy transmitted to the reactor system, which is initialized at steady-state conditions for the plant power, is consistent and that there is not a significant effect on the evolution of the remainder of the primary system during the LOCA transient. Therefore, because Figure Q25-2 shows that the gap opens quickly during a LBLOCA and Figure Q25-1 shows that there is little difference between the gaseous conductance of RELAP5/MOD2-B&W and COPERNIC for open gaps, the best solution is to apply no gaseous conductance multiplier (i.e. a factor of 1.0).

In conclusion, the system model in the MOX demonstration cases was initialized to steady state at the desired peaking conditions and the initial fuel temperatures were set to the COPERNIC-predicted temperatures with appropriate uncertainties added. The method ensures an appropriate specification of the initial fuel stored energy and a proper calculation of the gap conductance during a LBLOCA transient.

Figure Q25-1
Multipliers on RELAP5 to Match COPERNIC Gap Thermal Model

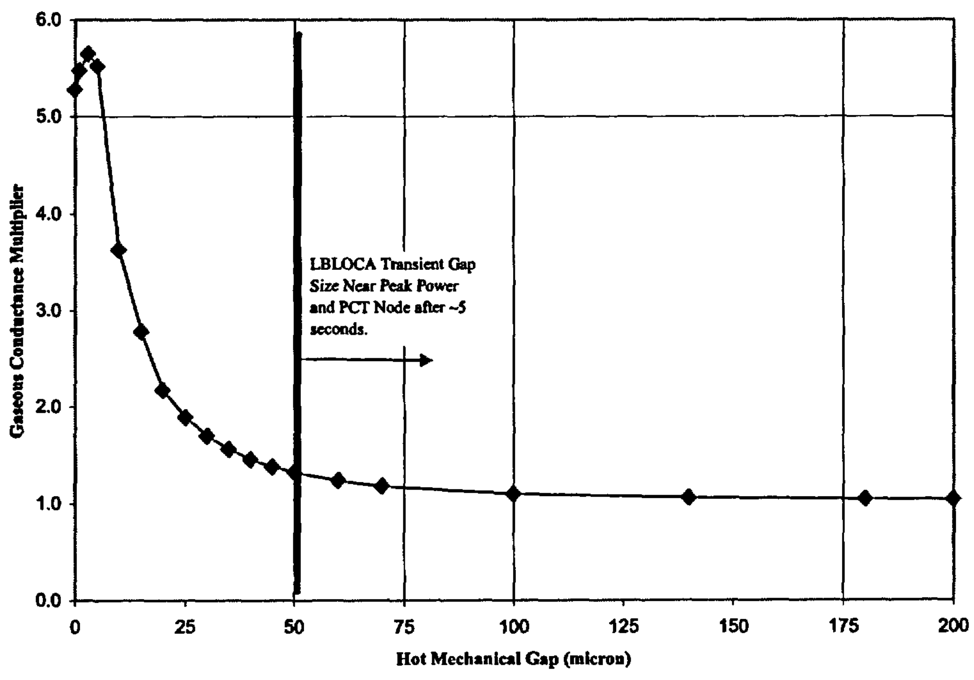


Figure Q25-2
LBLOCA Transient Hot Mechanical Gap Sizes

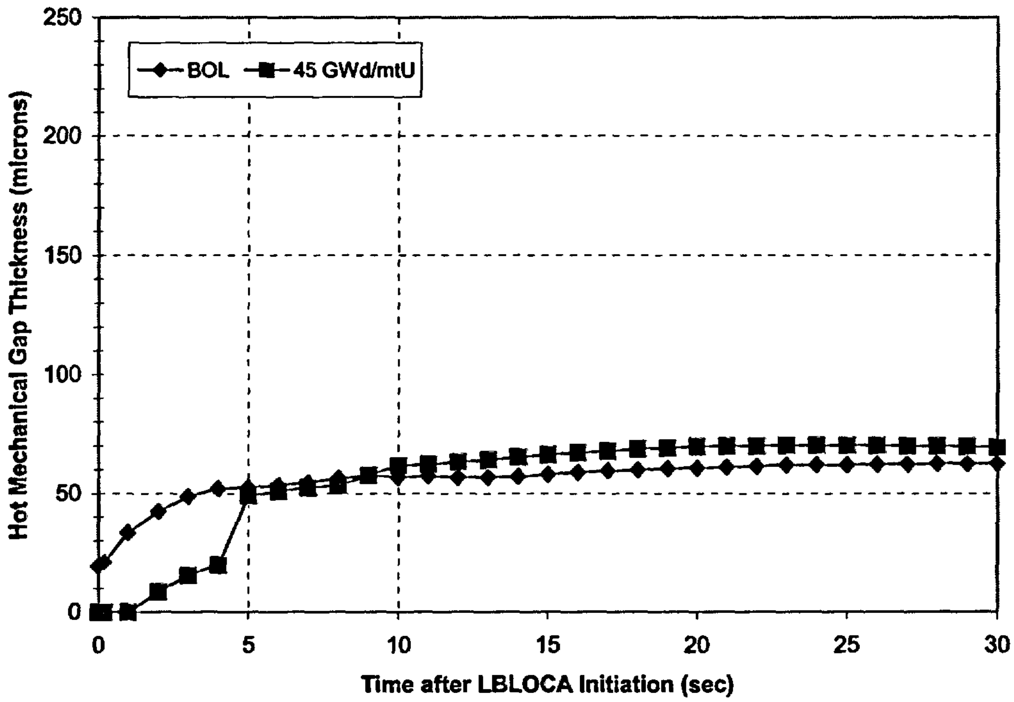
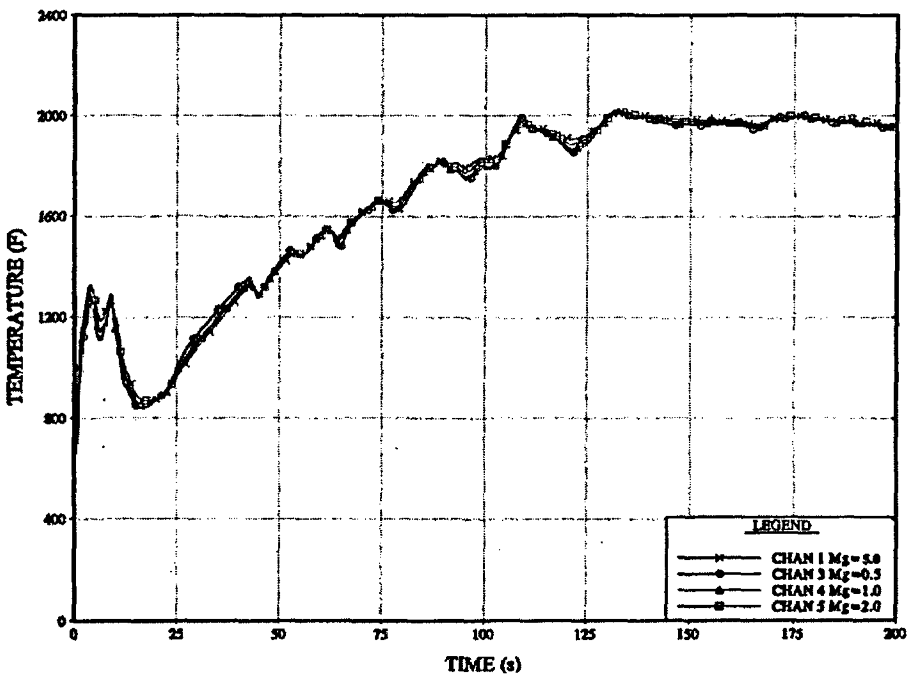


Figure Q25-3
LBLOCA Transient Cladding Temperatures at PCT Location



(Mg = Gaseous Conductance Multiplier)

26. Provide the basis for assuming that the uncertainty distribution for COPERNIC is a normal distribution.

<u>Response</u>
The actual assumption was that the COPERNIC uncertainty distribution was approximately normal. This assumption and the basis for this are explained in the response to Reactor Systems Question 24.

27. Please provide the basis for the COPERNIC temperature adjustments for core initialization in section 3.7.1.1.4. Additionally, please provide the basis for why the TACO3 temperature predictions are reasonable for application to COPERNIC predictions.

<u>Response</u>
TACO3 temperature predictions have no application to COPERNIC predictions. What was involved in the fuel temperature initialization of the LOCA core simulation was that the relative uncertainty for a specific region of the core, originally developed based on the TACO uncertainty distribution, was applied to the COPERNIC fuel temperature prediction. The application of the same relative uncertainty and the basis for it are

explained in the response to Reactor Systems Question 24.

28. In sub-section 3.7.1.6, the subject of mixed cores is discussed. In the middle of the paragraph it is stated that the MOX LTA pressure drop is less than four percent lower than the pressure drop for a resident Westinghouse fuel assembly at design flow rates. Please provide additional detail on the cause of this pressure drop difference, how it was calculated, and the impact including the consequences of this pressure drop. Also, please provide the design flow rate used for this analysis.

    Response (Previously submitted October 3, 2003)
    The pressure drop difference between the resident Westinghouse Robust Fuel Assembly (RFA) fuel and the MOX fuel lead assemblies is due to mechanical design differences in the grids and the top and bottom nozzles of the fuel assemblies. Even though the rod geometry, pitch, and axial grid locations are the same, unique design differences in the grids and nozzles themselves cause differences in hydraulic resistance. This overall difference was calculated by evaluating full core RFA and full core MOX models with the VIPRE-01 thermal-hydraulic code and comparing the overall calculated Δp. The code represents these hydraulic differences by means of vendor-provided form loss coefficients for each grid design, top, and bottom nozzles. The design flow rate for these evaluations was the current Technical Specification minimum flow rate of 390,000 gpm.

    The impact of this difference in pressure drop is flow redistribution between fuel types in a mixed core environment. This redistribution varies with axial elevation in the core as a direct effect of the difference in local grid form loss coefficients. The consequences of this pressure drop difference result in the need to account for this flow redistribution in the analyses of fuel assembly lift, departure from nucleate boiling ratio (DNBR) in steady state and transient analyses, and fuel assembly performance issues such as maximum allowable crossflow. Flow redistribution is accounted for in these analyses by modeling the hydraulic differences directly in a conservative representation of the mixed core fuel assembly geometry.

29. The staff presumes that a mixed core analysis will be performed to account for the use of four MOX LTAs in the core. Therefore, provide the mixed core penalty that was calculated. If a mixed core calculation was not performed, provide a technical justification for not performing the analysis.

    Response (Previously submitted October 3, 2003)
    The mixed core MOX fuel lead assembly DNBR penalty is explicitly calculated for the entire range of conditions analyzed in a reload cycle. With the currently licensed Duke Power analysis methodology, maximum allowable radial peaking limits are calculated for a range of axial peak locations and magnitudes as described in DPC-NE-2004P-A. This family of peaking limits is repeated for the various sets of reactor statepoints (power level, pressure, temperature, and flow) analyzed to support cycle reload analyses. This entire set of limits is used to represent the limiting fuel assembly in the core.

    To model the mixed core, a bounding model of a single high powered MOX fuel assembly at the center of the core surrounded by a remaining core of resident Westinghouse RFA fuel assemblies was used to calculate the explicit peaking limits. This model contained