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October 3, 2003

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

Subject: Catawba Nuclear Station Units 1 and 2; Docket Nos. 50-413, 50-414
McGuire Nuclear Station Units 1 and 2; Docket Nos. 50-369, 50-370
Partial Response to Request for Additional Information Regarding the Use of Mixed Oxide Lead Fuel Assemblies

Reference: NRC Letter dated July 25, 2003, Request for Additional Information Re: Mixed Oxide Lead Fuel Assemblies (TAC Nos. MB7863, 7864, 7865, 7866)

Please find attached a partial response to the Nuclear Regulatory Commission Staff's Request for Additional Information (RAI) transmitted by the referenced letter.

By letter dated February 27, 2003 Duke Energy submitted an application to amend the licenses of McGuire and Catawba to allow the use of four mixed oxide fuel lead assemblies. In a subsequent submittal dated September 23, 2003, Duke notified the Nuclear Regulatory Commission of the decision to focus the lead assembly program on Catawba only. As part of the review of this application the Nuclear Regulatory Commission staff in a letter dated July 25, 2003 requested that Duke provide additional information related to the application. To facilitate the NRC Staff's review, Duke is providing responses to 23 of the 47 Reactor Systems questions contained in the RAI at this time. The remaining responses will be submitted by the committed date of October 31, 2003.

Inquiries on this matter should be directed to G. A. Copp at (704) 373-5620.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachment

A001

U.S. Nuclear Regulatory Commission
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Oath and Affirmation

M. S. Tuckman affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

M. S. Tuckman

M.S. Tuckman

Subscribed and sworn to before me on this 3rd day of October, 2003.

Michael T. Cash

Notary Public

My Commission expires:

MICHAEL T. CASH
Notary Public
Lincoln County, North Carolina
Commission Expires January 22, 2008

Jan 22, 2008

Date

Seal

cc: with Attachment

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P. M. Abraham – EC08I
D. E. Bortz – EC08G
L. F. Vaughn – PB05E
J. L. Eller – EC09A
S. P. Nesbit – EC09A
K.S. Canady – EC08H
M.S. Scott – EC08H
R.M. Gribble – EC08F
C. J. Thomas - MG01RC
M.T. Cash – EC05O
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A. W. Cottingham – Winston & Strawn
G. A. Meyer – Framatome ANP
P. T. Rhoads – Department of Energy
D. J. Spellman – Oak Ridge National Laboratory
NRIA File/ELL – EC05O
McGuire Master File – MG01DM
Catawba Master File 801.01 – CN04DM
Catawba RGC Date File (J.M Ferguson – CN01SA)
MOX File – 1607.3202

Attachment
Partial Response to NRC Staff Request for
Additional Information dated July 25, 2003

Reactor Systems

1. With respect to section 3.1, provide a full description of the post-irradiation examination planned to verify the mechanical properties of the lead test assemblies (LTAs) following irradiation. Describe the test methods planned and the acceptance criteria for each test as well as the frequency of each test. Additionally, please provide the same information for the hot cell examinations that will be performed following irradiation.

Response
Post-irradiation examination (PIE) of the lead assemblies will include both poolside and hot cell examinations. A description of the examinations, including test methods, acceptance criteria, and frequency of testing, is found in Section 8.5 of Reference Q1-1. This topical report is currently under NRC review.

The purpose of the hot-cell examinations is to collect data to confirm the applicability of fuel performance models for fuel burnups greater than the limit (50,000 MWd/MThm maximum rod average) that is proposed for batch application. Therefore, specific acceptance criteria have not been set for the hot-cell examinations.

Reference
Q1-1. BAW-10238(NP) Revision 1, *MOX Fuel Design Report*, Framatome ANP, May 2003.

3. Provide the specific burnup limit that is being requested for the LTAs.

Response
The requested burnup limit for the lead assemblies is 60,000 MWd/MThm for the maximum rod (axially averaged). This is greater than the 50,000 MWd/MThm maximum rod average that is proposed for batch application. Additional information on these limits is contained in Section 8.4 of Reference Q3-1.

Reference
Q3-1. BAW-10238(NP) Revision 1, *MOX Fuel Design Report*, Framatome ANP, May 2003.

4. Section 3.5.1.2 of the submittal states that the reduction in total plutonium concentration ensures that the macroscopic plutonium effects on fuel performance are bounded. Please define the subject macroscopic plutonium effects.

Response
The macroscopic effects referred to are changes in the following parameters:

- Fuel thermal conductivity
- Pellet thermal expansion
- Pellet thermal creep

- Fission gas release
- Fuel densification and swelling
- Helium gas accumulation and release
- Pellet radial power profile
- Fuel melting point

Each of these parameters is affected by the amount of PuO_2 in the MOX fuel pellets. By using approximately 40% less PuO_2 in weapons grade MOX fuel relative to reactor grade MOX fuel, the effects of changes to these fuel performance parameters relative to low enriched uranium (LEU) fuel will be bounded by the European experience with reactor grade MOX fuel.

These effects are also listed in Section 2.1 of Reference Q4-1, the MOX Fuel Design Report, which has been submitted to the NRC for review. Further discussion of these macroscopic effects is provided in Sections 2 and 3 of that document. The pellet radial power profile will be addressed further in the response to Reactor Systems RAI Question 7.

Reference

Q4-1. BAW-10238(P), Revision 1, *MOX Fuel Design Report*, Framatome ANP, May 2003.

5. Provide the appropriate regulatory criteria used for the parameters discussed in section 3.5.1.

Response

Section 3.5.1 provides a description of MOX fuel and fuel rod design features, and is consistent with Reference Q5-1, which is currently under NRC review. Similarly, Section 3.5.2 describes the MOX fuel assembly mechanical design features. These sections provide the necessary background for the design evaluation, which is presented in Section 3.5.3 and discusses how the design satisfies the design criteria. The parameters described in Sections 3.5.1 and 3.5.2 support a fuel design that satisfies the acceptance criteria in Section 4.2 of Reference Q5-2 that pertain to pressurized water reactor fuel assembly design.

References

Q5-1. BAW-10238(P), Revision 1, *MOX Fuel Design Report*, Framatome ANP, May 2003.
Q5-2. NUREG-0800, U. S. Nuclear Regulatory Commission *Standard Review Plan*, Revision 2, July 1981.

6. Provide a statistical analysis showing that the distribution of fissile material for the weapons-grade (WG) mixed oxide (MOX) fuel is the same as the distribution of fissile material for reactor-grade (RG) MOX fuel.

Response

The requested statistical analysis is not practical until after the fuel is manufactured. However, it is not necessary to perform such an analysis in order to show that the distribution

of fissile material is approximately the same for both types of MOX fuels when using the same manufacturing process.

As discussed in Section 2.3 of Reference Q6-1, the MIMAS process ensures homogeneity of the pellet. The plutonium-rich agglomerates are finely dispersed in the UO_2 matrix, and this distribution is not a function of isotopic content, only the final desired plutonium content. With regard to plutonium-rich agglomerate size, the relevant portions of the European (Reactor Grade) and American (Weapons Grade) Fuel Pellet Specifications are identical. This is also discussed in Section 2.3. Therefore, the distribution of plutonium-rich agglomerates is the same in reactor-grade and weapons-grade fuels.

As discussed in Section 3.3 of Reference Q6-1, the fissile plutonium contents of the agglomerates in RG and WG MOX fuel are approximately 22.5% and 19.2%, respectively. The fissile uranium content of the agglomerates is very small and is approximately 0.2% for both RG and WG MOX fuel, assuming a 0.25 weight percent tails uranium is used for feed powder.

Since the distribution of plutonium-rich agglomerates is the same in RG and WG MOX fuels, and the fissile plutonium content of the agglomerates is approximately equal, the distribution of fissile material will also be approximately the same for both types of MOX fuel of equivalent reactivity. However, note that it is common practice to adjust the master mix to UO_2 ratio in order to achieve the desired final plutonium content of a given rod. Each MOX fuel assembly is composed of three different plutonium content rods. For example, the three plutonium concentrations could be 2.40, 3.35, and 4.94 w/o plutonium for corner, periphery, and interior pins, respectively (see response to Reactor Systems RAI Question 18, Figure 18-2). The adjustment for these different pins, for which there is plenty of experience, dramatically changes the fissile material distribution within the pellet and far outweighs the small difference between RG and WG MOX fuel.

Reference

Q6-1. BAW-10238(P), Revision 1, *MOX Fuel Design Report*, Framatome ANP, May 2003.

8. In section 3.5.3, the statement is made that the Mark-BW/MOX1 fuel assembly design meets all applicable criteria to maintain safe plant operation. Specify all of the regulatory criteria being referred to in this statement.

Response:

A detailed analysis of the Mark-BW/MOX1 fuel assembly design is presented in Reference Q8-1. The design satisfies the acceptance criteria in Section 4.2 of Reference Q8-2 that pertain to pressurized water reactor fuel assembly design. The acceptance criteria in Reference Q8-2, in turn, align with the relevant requirements (regulatory criteria) in 10 CFR 50.46, General Design Criteria 10, 27, and 35, and 10 CFR 100.

The criteria in Reference Q8-2, Section 4.2.II.A, that are not addressed are as follows:

II.A.1(h): This criterion applies to control rods rather than fuel.
II.A.2(c): Fretting is treated in the response to criteria in paragraph II.A.1(c).

II.A.2(f): This criterion applies only to boiling water reactor fuel.
II.A.2(i): Mechanical fracturing is treated in the response to criteria in paragraph II.A.3(e).
II.A.3(c): More stringent criteria are already applied in paragraph II.A.3(a).

Reference
Q8-1. BAW-10238(NP) Revision 1, *MOX Fuel Design Report*, Framatome ANP, May 2003.
Q8-2. NUREG-0800, U. S. Nuclear Regulatory Commission *Standard Review Plan*, Revision 2, July 1981.

9. Section 3.5.3 refers to fuel rod analyses which follow previously approved methods. Please state the methods being referred to.

Response
Approved methods used for analysis of MOX fuel are listed below. A more detailed discussion of the methods used for analysis of MOX fuel is provided in Reference Q9-1.

The fuel rod cladding stress and cladding fatigue were analyzed using approved methodologies for M5™ cladding that are described in Reference Q9-2. This method is applicable to both LEU and MOX fuel because it is independent of the pellet type.

Cladding corrosion was analyzed using the COPERNIC code with models approved for predicting M5™ cladding oxide thickness as described in Reference Q9-2. This methodology is applicable to both LEU and MOX fuel because it is independent of the pellet type.

Axial growth calculations for fuel rods and fuel assemblies also used the methodology that was approved in Reference Q9-2.

The hydraulic forces for assembly liftoff calculations were determined using the NRC approved LYNXT code described in Reference Q9-3. This methodology is applicable to both LEU and MOX fuel because it is independent of the pellet type.

Fuel rod end-of-life internal pressure and centerline fuel melt temperatures were evaluated according to the methods and criteria presented in Reference Q9-4 for MOX fuel rods. Approval of Reference Q9-4 is pending. Revision 0 of this topical report has been approved for use with LEU fuel. A consistent methodology is used to describe the variations in properties between LEU and MOX fuel.

The fuel rod was analyzed for creep collapse using approved methods described in Reference Q9-5. This method is applicable to both LEU and MOX fuel because it is independent of the pellet type.

Overheating of cladding was analyzed with the approved BWU-N and BWU-Z critical heat flux correlations, which are described in References Q9-6 and Q9-7. This method is applicable to both LEU and MOX fuel because it is independent of the pellet type.

Fuel enthalpy during a rapid insertion of reactivity (control rod ejection) is calculated using the SIMULATE-3K MOX computer code as described in Section 3.7.2.4 of Attachment 3 to the Duke Power MOX fuel lead assembly license amendment request dated February 27, 2003. SIMULATE-3K MOX is not yet an approved methodology for MOX fuel, although the similar code SIMULATE-3K has been approved for application to rod ejection analyses in low enriched uranium fuel cores (Reference Q9-8).

LOCA calculations for cladding rupture, cladding embrittlement, and fuel rod ballooning used approved methods described Reference Q9-2. These methods are applicable to both LEU and MOX fuel because they are independent of the pellet type.

The axial and horizontal faulted analysis methodologies (for external forces) are consistent with the approved methodologies described in References Q9-9 and Q9-10, respectively. These methods are applicable to both LEU and MOX fuel because they are independent of the pellet type. Application of these methods to the Advanced Mark-BW fuel assembly design is discussed Reference Q9-11.

References
Q9-1. BAW-10238(NP) Revision 1, *MOX Fuel Design Report*, Framatome ANP, May 2003.
Q9-2. BAW-10227P-A Revision 0, *Evaluation of Advanced Cladding and Structural Material (M5™) in PWR Reactor Fuel*, February 2000.
Q9-3. BAW-10156-A Revision 1, *LYNXT: Core Transient Thermal-Hydraulic Program*, August 1993.
Q9-4. BAW-10231P Revision 2, *COPERNIC Fuel Rod Design Computer Code*, July 2000.
Q9-5. BAW-10084-A Revision 3, *Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse*, July 1995.
Q9-6. BAW-10199P-A Revision 0, *The BWU Critical Heat Flux Correlations*, December 1994.
Q9-7. BAW-10199P-A Addendum 2, *Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids*, June 2002.
Q9-8. DPC-NE-2009-P-A, *Duke Power Company Westinghouse Fuel Transition Topical Report*, September 1999.
Q9-9. BAW-10133P-A Revision 0, *Mark-C Fuel Assembly LOCA-Seismic Analysis*, June 1986.
Q9-10. BAW-10133P-A Revision 1, Addendum 1 and Addendum 2, *Mark-C Fuel Assembly LOCA-Seismic Analyses*, October 2000.
Q9-11. BAW-10239(NP) Revision 0, *Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report*, March 2002.

10. Section 3.5.3 states that the Mark-BW/MOX1 design preserves the original plant licensing bases for all reactor internal components. Specify the components and the bases for them that are being referred to in this statement. Also explain how the MARK-BW/MOX1 design preserves them.

Response

The reactor internal components that are affected by the fuel assembly design are the upper and lower core plates. The licensing basis is preserved through ensuring that the criteria established in Reference Q10-1 are met. These criteria were developed from the acceptance criteria of Reference Q10-2. The axial growth and assembly liftoff criteria in Reference Q10-1 must be met during the assembly lifetime. In addition, the fuel assembly must meet the following criteria for externally applied forces:

a) Operating basis earthquake: Allow continued safe operation of the fuel assembly following an event by ensuring the fuel assembly components do not violate their dimensional requirements.
b) Safe shutdown earthquake: Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies, control rod insertability, and a coolable geometry within the deformation limits consistent with the emergency core cooling system and safety analysis.
c) LOCA or safe shutdown earthquake plus LOCA: Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies and a coolable geometry within deformation limits consistent with the emergency core cooling system and safety analysis.

The Mark-BW/MOX1 design preserves the design basis because the growth models, nominal dimensions, and tolerances for the holddown springs, nozzles, and guide thimble assemblies are the same as those given in Reference Q10-3.

References

Q10-1. BAW-10238(NP) Revision 1, *MOX Fuel Design Report*, Framatome ANP, May 2003.

Q10-2. NUREG-0800, U. S. Nuclear Regulatory Commission *Standard Review Plan*, Revision 2, July 1981.

Q10-3. BAW-10239(NP) Revision 0, *Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report*, March 2002.

11. Please respond to the following items related to section 3.6.1, so that the Nuclear Regulatory Commission (NRC) staff may evaluate the conclusion that the vessel fluence increase is limited:

(A) Identify the location of the LTAs.

(B) Identify the existing peak fluence azimuthal location of the vessel.

Response (A)
The MOX fuel lead assembly license amendment request is not tied to a specific unit and cycle. Current plans are to insert the lead assemblies in Catawba Unit 1 Cycle 16 (C1C16), which will start up in late spring 2005. The C1C16 core design will not be finalized until the first quarter 2004. The location of the lead assemblies will be influenced by numerous design considerations, and it is not feasible to commit to the location at this time. Duke has made a commitment (Reference Q11-1)to the Nuclear Regulatory Commission to place at least two of the four lead assemblies in core locations that contain incore flux mapping instrumentation. Core location C-08 (and its symmetrical locations) is a potentially desirable location because it is the only core location that will (i) allow full instrumentation for all four assemblies and (ii) maintain an eighth core symmetric arrangement. Peripheral core locations would not be desirable because the lead assemblies would not be subjected to sufficient fuel duty to achieve burnups representative of MOX fuel batch implementation. Duke has also made a commitment (Reference Q11-1) not to place the lead assembly in a core location with a control rod during the first cycle of use.

Figure Q11-1 depicts the Catawba one-quarter core geometry. The locations shaded in gray and red show typical fresh fuel placement in recent Catawba fuel cycles. The desirable lead assembly location C-08 (and its symmetrical location H-13) is shown in red.

Figure Q11-1
Representative MOX Fuel
Lead Assembly Core Design

	H	G	F	E	D	C	B	A
8		LEU Feed				MOX LTA Feed		
9	LEU Feed		LEU Feed		LEU Feed		LEU Feed	
10		LEU Feed		LEU Feed		LEU Feed	LEU Feed	
11			LEU Feed		LEU Feed		LEU Feed	
12		LEU Feed		LEU Feed		LEU Feed		
13	MOX LTA Feed		LEU Feed		LEU Feed	LEU Feed		
14		LEU Feed	LEU Feed	LEU Feed				
15								

<u>Response (B)</u>
For Catawba Units 1 and 2, the peak fast neutron fluence at the pressure vessel cladding/base metal interface is at the 30° azimuth relative to the core cardinal axis. Catawba cores are designed with eighth-core symmetry. Accordingly, relative to the diagram shown in Figure Q11-1, the core cardinal axis would be through row eight.

Figures Q11-2 and Q11-3 show the relative fast neutron flux differences between a core with four MOX fuel lead assemblies and an all LEU core at beginning of cycle and end of cycle, respectively (This information was not specifically requested but it shows the negligible impact of four MOX fuel lead assemblies on core exterior fast flux, which control vessel fluence).

<u>Reference</u>
Q11-1. Tuckman, M.S., June 26, 2003, Letter to U.S. Nuclear Regulatory Commission, Physics Testing Program in Support of Topical Report DPC-NE-1005P *Nuclear Design Methodology Using CASMO-4/ SIMULATE-3 MOX*, August 2001.

C01

Figure Q11-2
Fast Neutron Flux Comparison
All LEU Core vs Core with Four MOX Fuel Assemblies
(Beginning of Cycle)

Flux above 0.625 Mev **HFP, 4 EFPD, ARO, Nominal Conditions**

	H	G	F	E	D	C	B	A	
	3.619	3.561	3.652	3.654	3.688	3.709	2.970	1.429	
8	3.607	3.551	3.649	3.670	3.761	3.948	3.050	1.452	
	-0.3	-0.3	-0.1	0.4	2.0	6.4	2.7	1.6	
		3.597	3.565	3.626	3.604	3.442	3.121	1.629	
9		3.587	3.560	3.635	3.644	3.515	3.170	1.646	
		-0.3	-0.1	0.2	1.1	2.1	1.6	1.0	
			3.587	3.555	3.588	3.582	3.260	1.652	
10			3.581	3.552	3.593	3.594	3.271	1.657	
			-0.2	-0.1	0.1	0.3	0.3	0.3	
				3.584	3.680	3.664	3.055	1.220	
11				3.571	3.664	3.649	3.044	1.217	
				-0.4	-0.4	-0.4	-0.4	-0.2	
					3.975	3.678	2.240		
12					3.947	3.650	2.223		
					-0.7	-0.8	-0.8		
						3.006	1.325		No MOX
13						2.979	1.312		4 MOX
						-0.9	-1.0		% relative diff

Figure Q11-3
Fast Neutron Flux Comparison
All LEU Core vs Core with Four MOX Fuel Assemblies
(End of Cycle)

Flux above 0.625 Mev **HFP, 495 EFPD, ARO, Nominal Conditions**

	H	G	F	E	D	C	B	A	
	3.857	3.983	3.679	3.434	3.534	3.667	3.072	1.651	
8	3.872	3.997	3.692	3.438	3.514	3.647	3.047	1.648	
	0.4	0.4	0.4	0.1	-0.6	-0.5	-0.8	-0.2	
		3.811	3.916	3.644	3.804	3.540	3.410	1.876	
9		3.825	3.929	3.652	3.798	3.520	3.400	1.877	
		0.4	0.3	0.2	-0.2	-0.6	-0.3	0.1	
			3.827	3.968	3.782	3.909	3.548	1.872	
10			3.841	3.981	3.790	3.916	3.555	1.878	
			0.4	0.3	0.2	0.2	0.2	0.3	
				3.830	3.958	3.605	3.052	1.345	
11				3.844	3.973	3.619	3.065	1.352	
				0.4	0.4	0.4	0.4	0.5	
					3.790	3.524	2.196		
12					3.807	3.541	2.208		
					0.4	0.5	0.5		
						2.872	1.356		No MOX
13						2.887	1.364		4 MOX
						0.5	0.6		% relative diff

Note: Figures indicate fast neutron flux scaled by $1x10^{-14}$, in units of neutrons/cm^2/sec

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

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 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 acceptance criterion for LEU fuel in Reference Q12-2. 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.

10CFR 50.46 (b) Criteria	Applicability to 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:
Framatome ANP is submitting the input decks to the NRC Staff in compact disc format by separate letter. The 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

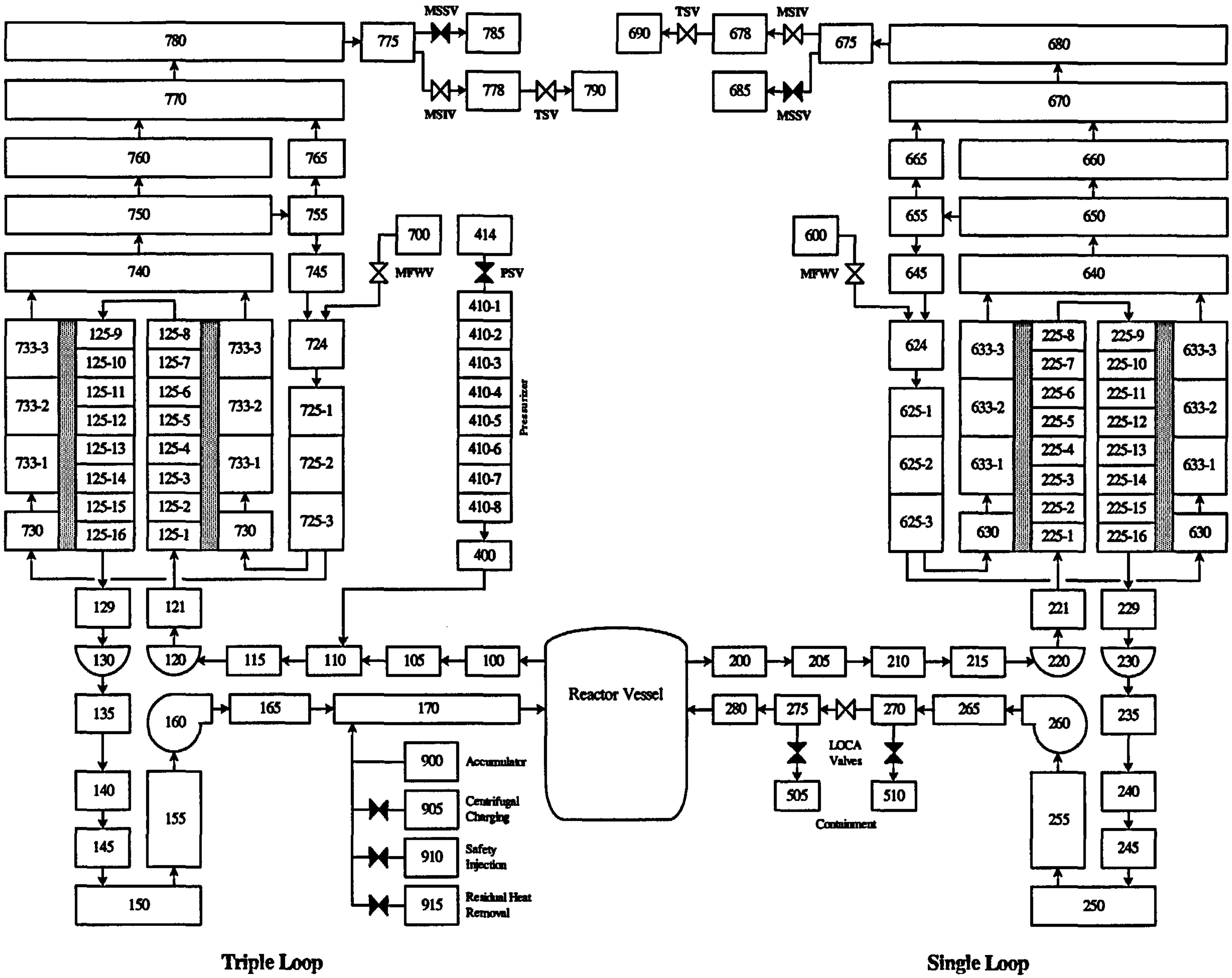
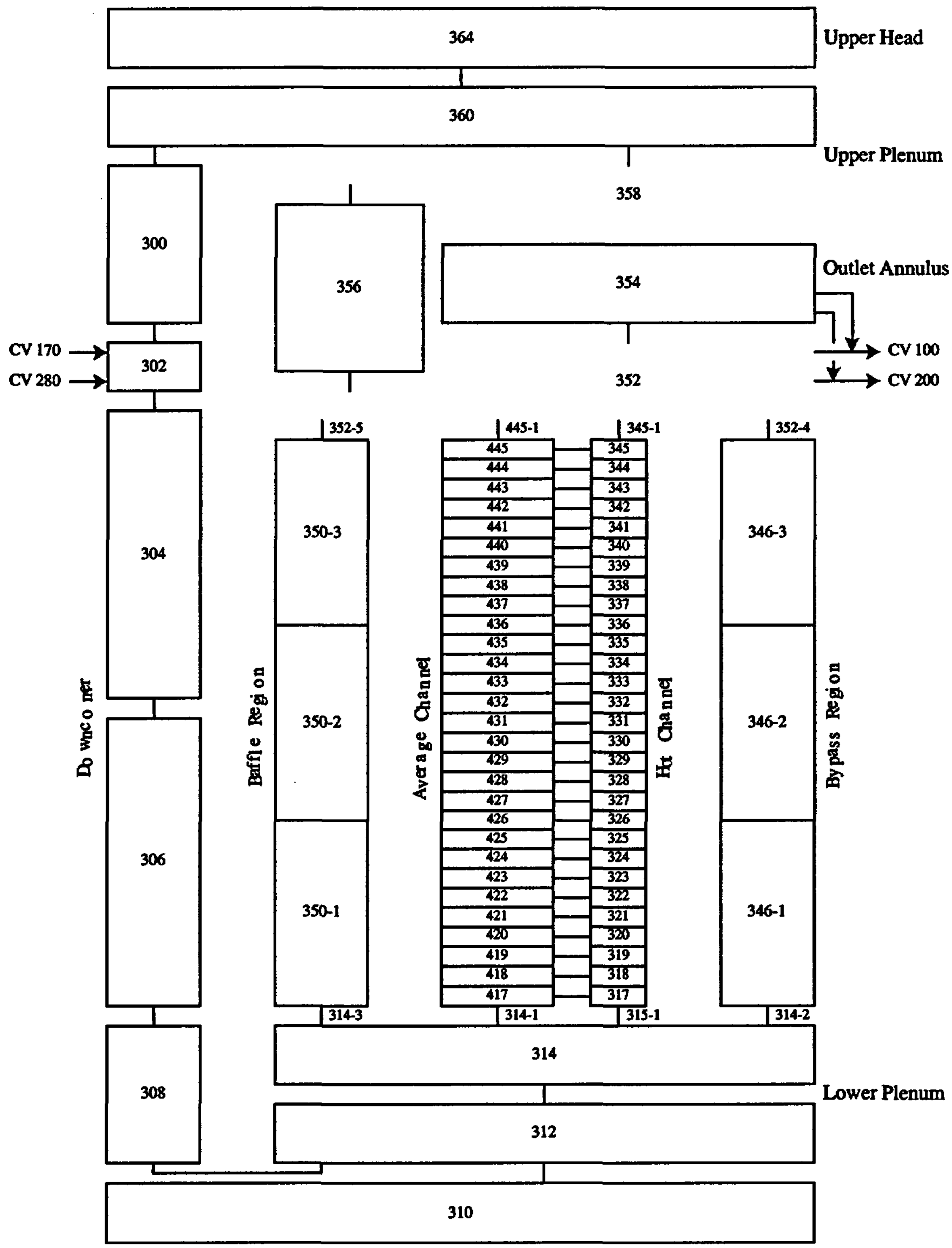


Figure Q13-2
Core Noding Diagram



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

Response
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.

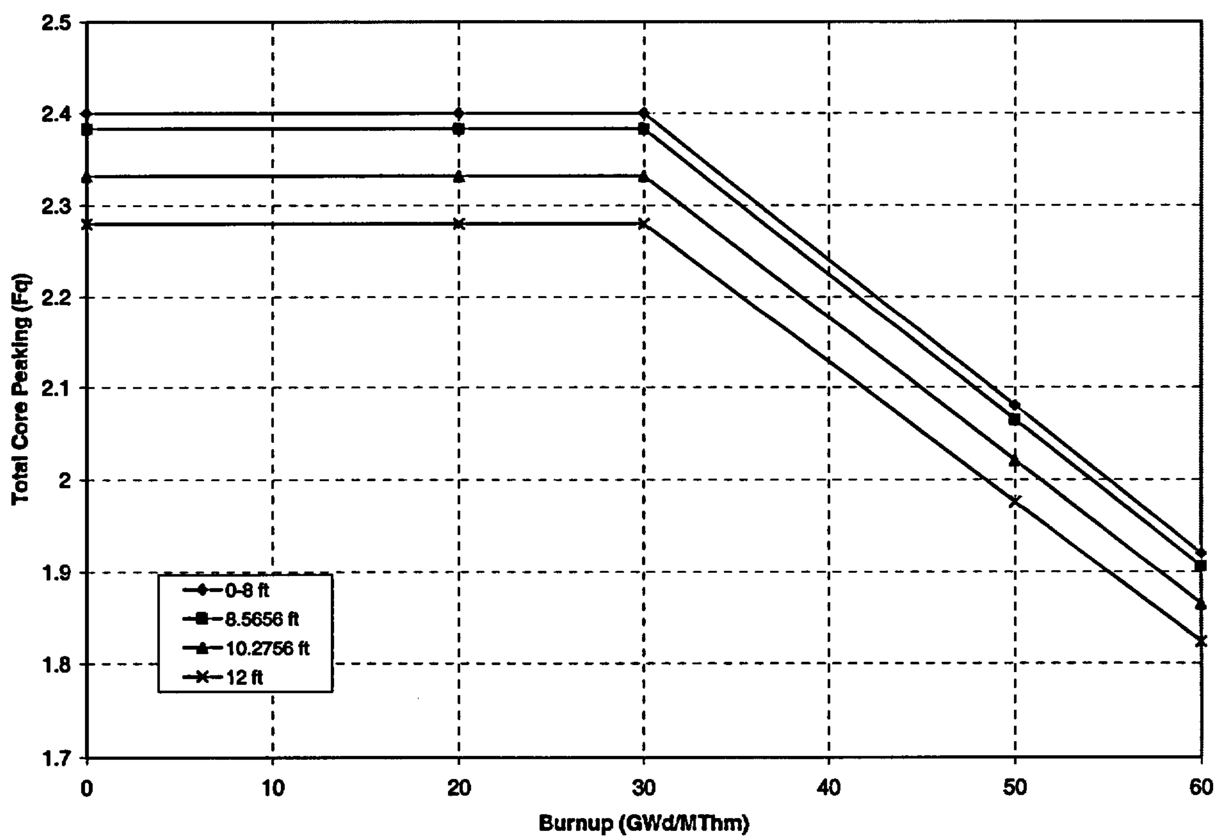
Q14-3. BAW-10168P-A, Revision 3, *RSG LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants*, December 1996.

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
   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:
   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
   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
   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 reactivity were placed in a checker board arrangement to simulate face adjacent MOX and LEU fuel assemblies (see Figure Q18-1). Two cases were considered – one using MOX fuel with weapons grade (WG) plutonium isotopics, and another using MOX fuel with reactor grade (RG) plutonium isotopics. The lead assemblies will be WG MOX fuel; the RG MOX fuel

cases are included for illustration purposes. Each case is examined at two burnup conditions: one with all fresh fuel, and the other with 20 GWd/MThm burnup on each assembly (typical of the end of one cycle of operation).

The MOX fuel assemblies used plutonium concentration zoning as shown in Figure Q18-2. This is standard practice in European pressurized water reactors that use MOX fuel. Low plutonium concentrations in the corner and peripheral pins serve to flatten the power distribution in the MOX fuel assemblies. The LEU fuel assemblies were a uniform lattice of 4.27 weight percent (w/o) ^{235}U. The MOX and LEU fuel share the same dimensional characteristics (e.g., pellet diameter, cladding inner and outer diameter, and lattice pitch).

Figures Q18-3 through Q18-6 show how neutron flux and power change in a single row of pin cells traversing the MOX/LEU fuel assembly interface. Pin cell locations -9 (minus nine) thru -1 (minus one) are LEU fuel pins. Pin cell locations 1 thru 9 are MOX fuel pins. The zero location on the x-axis of each graph corresponds to the center of the inter-assembly gap. Fast flux (> .625 MeV), thermal flux (< 0.625 MeV), and power are each normalized to the colorset average value.

Figures Q18-3 and Q18-4 show the results for a WG MOX/LEU colorset at burnups of 0 GWd/MThm and 20,000 GWd/MThm, respectively. Figures Q18-5 and Q18-6 show the corresponding results for a RG MOX/LEU colorset. The results are qualitatively similar between MOX fuel types and between burnups, with key points discussed below.

a) The fast flux is nearly uniform across the assemblies. Consistent with the results of the core simulations in the response to Reactor Systems RAI Question 11, the fast flux in MOX fuel is only slightly higher than the fast flux in LEU fuel.
b) There is a steep thermal flux gradient between the assemblies. The flux gradient results from the fact that the thermal neutron absorption cross-section in plutonium is larger than in uranium.
c) The thermal flux gradient is more pronounced for the RG MOX fuel case than for the WG MOX fuel case. The higher overall plutonium concentration in the RG MOX fuel assembly (7.07 w/o) depresses the thermal flux more than the WG MOX fuel assembly with only 4.37 w/o plutonium. Therefore, the thermal flux gradient from the LEU to the MOX fuel assembly is steeper in the RG MOX fuel case. It is not unexpected to find that WG MOX fuel behavior falls between that of LEU fuel and RG MOX fuel; this is consistent with studies of other neutronic characteristics of MOX fuel (Reference Q18-1).
d) Most importantly, the plutonium concentration zoning in the MOX fuel assemblies is effective in producing a relatively flat power profile across the LEU and MOX fuel assemblies. This conclusion is valid for both WG MOX fuel and RG MOX fuel. The edge and corner pins MOX pins see a much higher thermal flux than the interior fuel pins, but the lower plutonium concentration in the edge and corner pins makes the fission rate and power about the same as the interior pins. The response to a LOCA is driven by the pin power profile, not by the pin flux profile. Therefore, the neutron flux behavior at the interface of these MOX and LEU fuel assemblies should have no significant impact on the cladding temperature response following a loss of coolant accident.

<u>Reference</u>

Q18-1. BAW-10238(P), Revision 1, *MOX Fuel Design Report*, Section 3.1, Framatome ANP, May 2003.

Figure Q18-1
MOX Fuel and LEU Fuel Colorsets

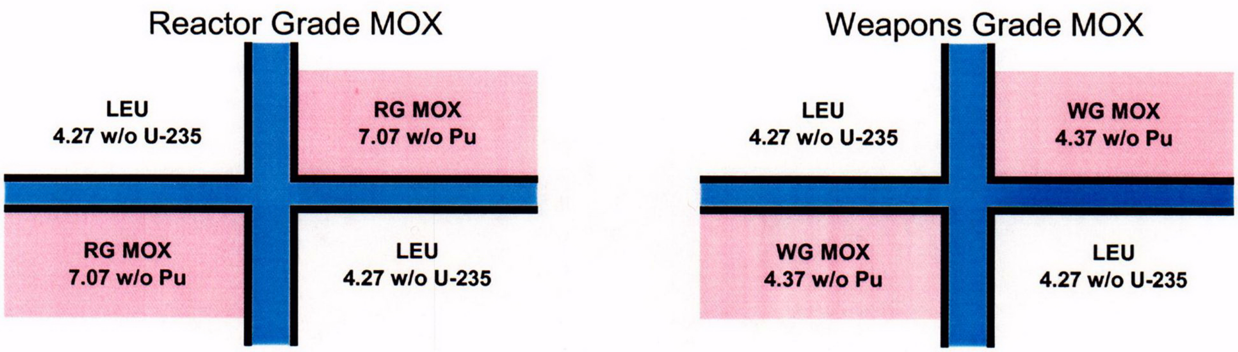


Figure Q18-2
MOX Fuel Assembly Zoning

3	3	2	2	2	2	2	2	2	2	2	2	2	2	2	3	3
3	2	2	1	1	2	1	1	2	1	1	2	1	1	2	2	3
2	2	1	1	1		1	1		1	1		1	1	1	2	2
2	1	1		1	1	1	1	1	1	1	1	1		1	1	2
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
2	2		1	1		1	1		1	1		1	1		2	2
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
2	2		1	1		1	1		1	1		1	1		2	2
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
2	2		1	1		1	1		1	1		1	1		2	2
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2
2	1	1		1	1	1	1	1	1	1	1	1		1	1	2
2	2	1	1	1		1	1		1	1		1	1	1	2	2
3	2	2	1	1	2	1	1	2	1	1	2	1	1	2	2	3
3	3	2	2	2	2	2	2	2	2	2	2	2	2	2	3	3

Pin	Number	RG w/o Pu	WG w/o Pu
1	176	8.00	4.94
2	76	5.42	3.35
3	12	3.89	2.40
Total / Avg	264	7.07	4.37

C02

Figure Q18-3
Weapons Grade MOX/LEU Fuel Colorset (0 GWd/MThm)

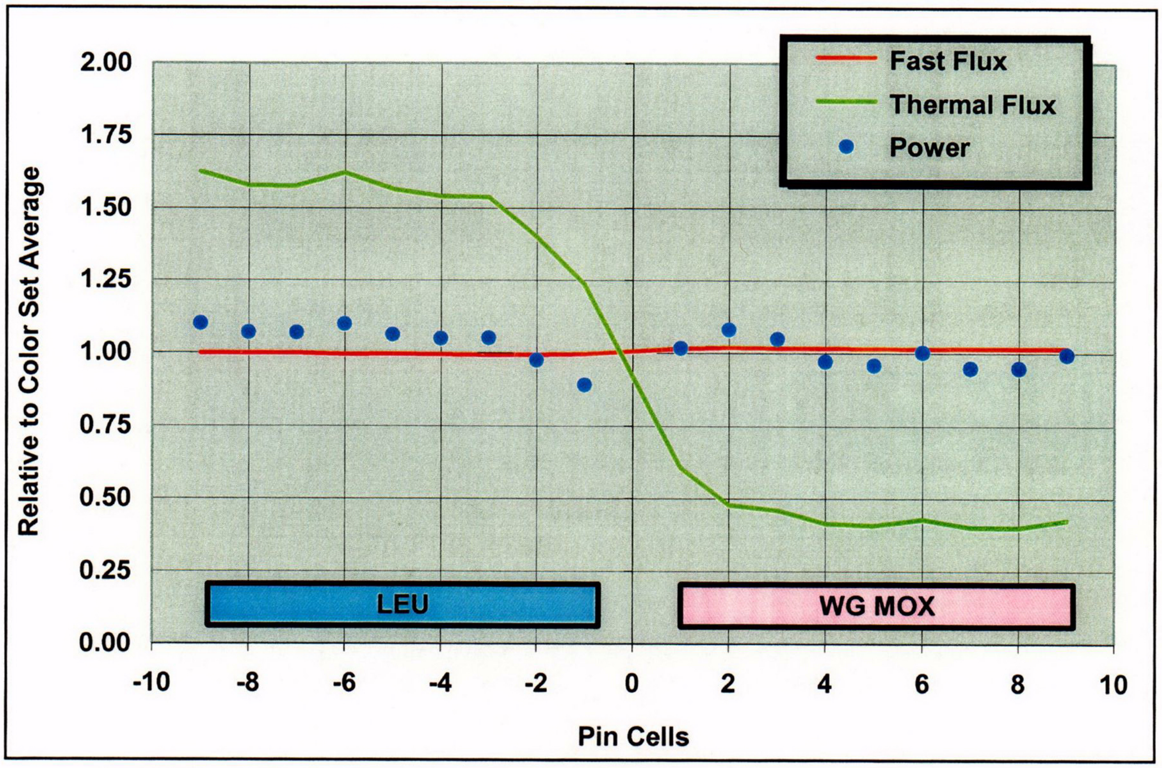
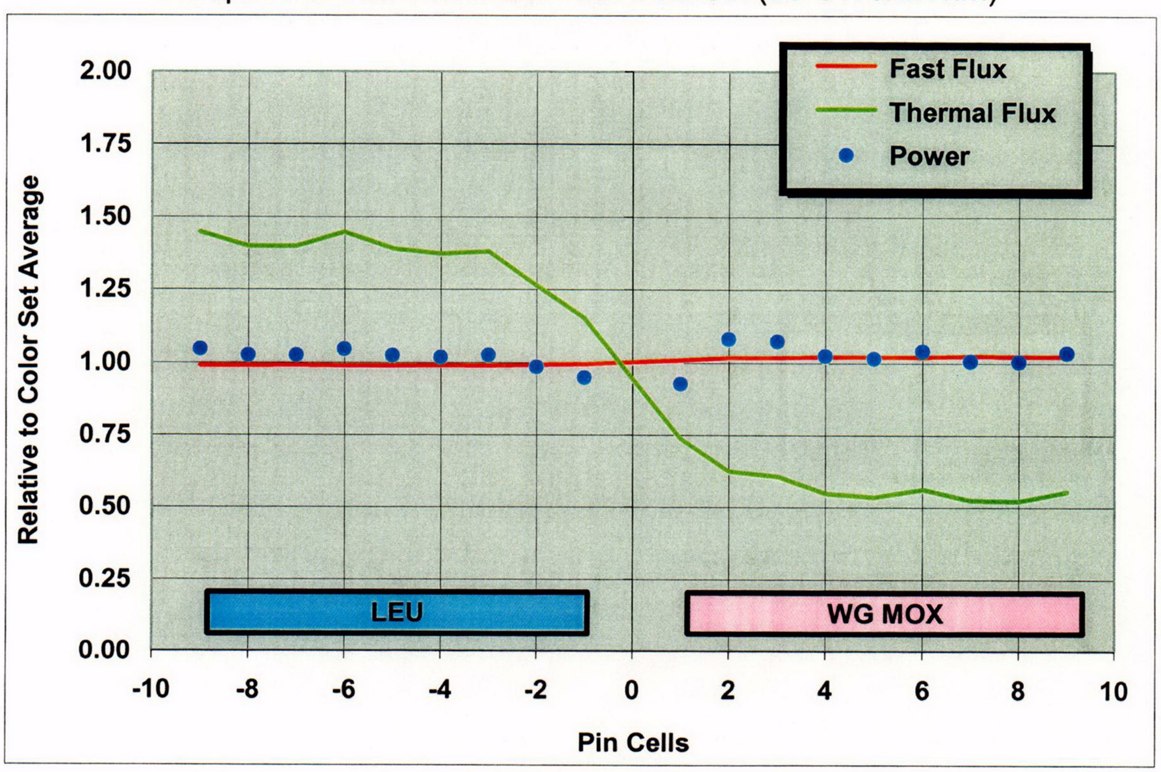


Figure Q18-4
Weapons Grade MOX/LEU Fuel Colorset (20 GWd/MThm)



C03

Figure Q18-5
Reactor Grade MOX/LEU Fuel Colorset (0 GWd/MThm)

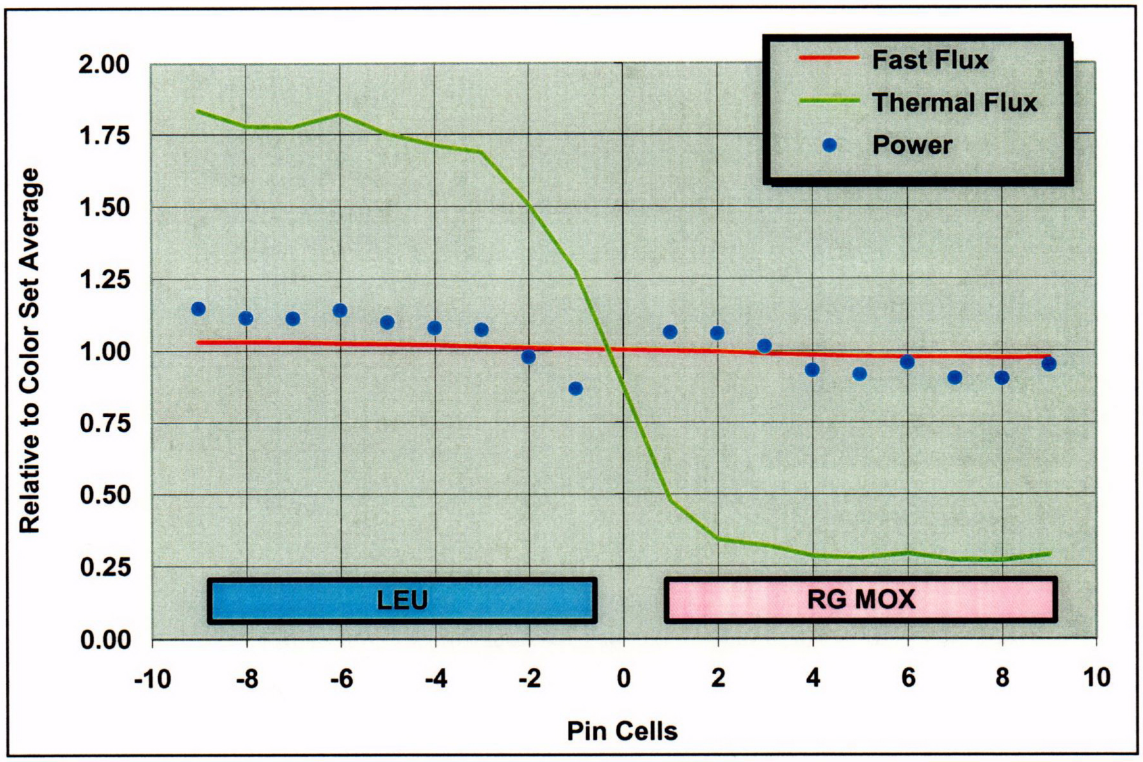
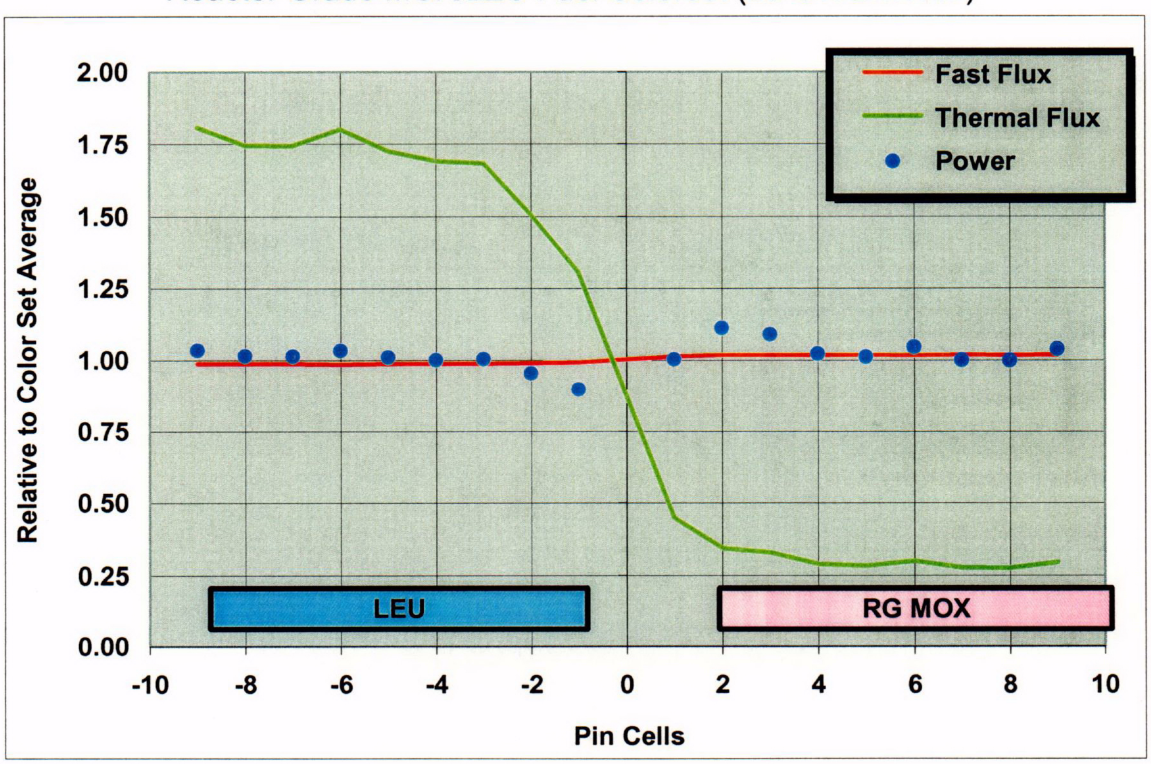


Figure Q18-6
Reactor Grade MOX/LEU Fuel Colorset (20 GWd/MThm)



C04

19. Section 3.7.1.1.1 discusses a variety of neutronic parameters. Provide additional detail about the differences between LEU and MOX parameters. Please use graphs, data, or any other visual representations to help clarify the impact of these parameter differences.

Response

Information on differences between LEU and MOX fuel neutronic parameters was provided in Tables 3-7 through 3-10 and Figure 3-2 in Attachment 3 to Reference Q19-1. Additional information is provided in the responses to Reactor Systems RAI Questions 11, 18, 20, and 23.

Reference

Q19-1. 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.

20. Also, in the second paragraph of sub-section 3.7.1.1.1, the change in the delayed neutron fraction is discussed at beginning of life. However, the behavior of the delayed neutron fraction at middle of life and end of life is not addressed. Provide a discussion on the delayed neutron factor change with respect to burnup.

Response

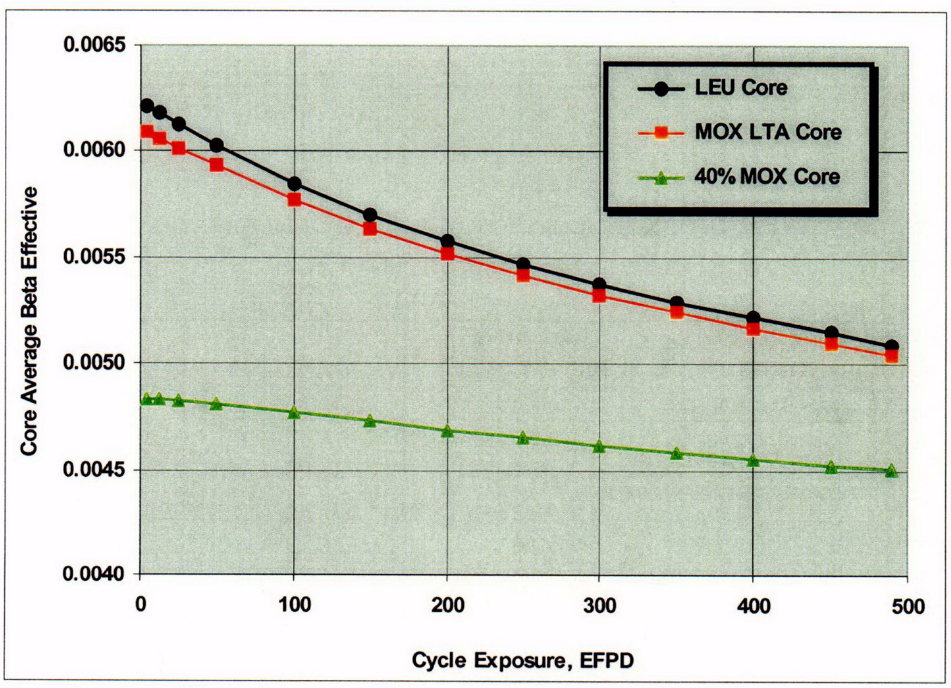
Table Q20-1 and Figure Q20-1 show the variation in core average effective delayed neutron fraction (β_{eff}) with cycle burnup and MOX fuel loading. The introduction of four MOX fuel lead assemblies results in a minor decrease in beginning of cycle (BOC) β_{eff}. The impact on end of cycle (EOC) β_{eff} is even smaller. The 40% MOX fuel core sees a more significant reduction in β_{eff} at BOC, but the change in β_{eff} with burnup is much smaller than the corresponding change in a core containing all low enriched uranium (LEU). The result is that the 40% MOX fuel core has more uniform kinetics over the cycle.

Table Q20-1
Impact of MOX Fuel Loading and Burnup on
Effective Delayed Neutron Fraction (β_{eff})

EFPD	LEU Core[1]	MOX Lead Assembly Core[2]	Batch MOX Core[3]
4	6.216E-03	6.089E-03	4.827E-03
12	6.180E-03	6.060E-03	4.829E-03
25	6.125E-03	6.015E-03	4.825E-03
50	6.026E-03	5.929E-03	4.810E-03
100	5.852E-03	5.772E-03	4.768E-03
150	5.704E-03	5.634E-03	4.726E-03
200	5.579E-03	5.516E-03	4.686E-03
250	5.471E-03	5.413E-03	4.650E-03
300	5.377E-03	5.322E-03	4.615E-03
350	5.292E-03	5.240E-03	4.583E-03
400	5.216E-03	5.166E-03	4.551E-03
450	5.146E-03	5.098E-03	4.521E-03
490	5.086E-03	5.041E-03	4.502E-03

[1] No MOX fuel assemblies
[2] 4 MOX fuel assemblies
[3] 76 MOX fuel assemblies (~40% MOX fuel core)

Figure Q20-1
Impact of MOX Fuel Loading and Burnup on
Effective Delayed Neutron Fraction (β_{eff})



C05

23. All operating plants must have sufficient shutdown margin at the beginning and throughout a fuel cycle. Provide the predicted shutdown margin in graphical or tabulated form for the cores including the four MOX lead test assemblies. Show that the predicted shutdown margin will meet the Technical Specification limit for each of the plants that may load the LTAs.

Response:
Results of quantitative Mode 1 shutdown margin calculations are presented in Table Q23-1 for a representative core consisting of 189 LEU fuel assemblies and four unirradiated MOX fuel lead assemblies. The results are compared against the same core with 193 LEU fuel assemblies. Shutdown margins are shown at beginning, middle, and end of cycle.

The calculated shutdown margins include standard conservatisms. Operation at the control rod insertion limit is assumed. Power defect is increased and control rod worth is decreased to account for uncertainty in those values. It is assumed that the highest worth control rod remains stuck out of the core, and the worth of that rod is conservatively increased.

The resulting shutdown margins are substantially greater than the current Catawba Mode 1 minimum shutdown margin operating limit of 1300 pcm. Furthermore, there are no significant adverse impacts on calculated shutdown margin due to the presence of four MOX fuel lead assemblies. The differences observed are within normal fuel cycle design variations.

As part of the standard Catawba reload design process, Duke calculates control rod worths and verifies adequate shutdown margin. This same standard reload design process will be applied to the design of the actual MOX fuel lead assembly cores, thereby ensuring that shutdown margin limits are met.

Table Q23-1
Shutdown Margins Impact of
Four Fresh MOX Fuel Lead Assemblies at Catawba

Type of Core	Calculated Shutdown Margin (pcm)		
	4 EFPD	250 EFPD	490 EFPD
All LEU Fuel	3237	2498	2020
4 Fresh MOX Fuel Lead Assemblies	3255	2465	1988

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

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

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 the

correct geometry and local form loss coefficients to represent both fuel types (see response to Reactor Systems RAI Question 28). Therefore, the mixed core peaking penalty is calculated for each unique set of conditions and the appropriate conservative limits will be applied to the lead assembly core positions in the specific cycle reload analyses. This penalty magnitude varies as a function of the axial power distribution, with the overall average penalty equal to 3% in radial peaking or 10% in DNBR, relative to a full core of Westinghouse RFA fuel assemblies.

31. Section 3.7.2.2 discusses the thermal-hydraulic differences for the different co-resident fuel types. Please provide the limits analyzed for the different fuel types, including a discussion on how they were obtained.

    Response
    The co-resident fuel types, Westinghouse RFA and Mark Bw/MOX1, will be analyzed with their respective critical heat flux (CHF) correlations and limits. The RFA fuel will be analyzed with the Westinghouse WRB-2M CHF correlation with a design limit departure from nucleate boiling ratio (DNBR) value (95/95) of 1.14 for deterministic analyses and a design limit DNBR value of 1.30 for analyses where uncertainties are combined statistically.

    The Mark BW/MOX1 fuel will be analyzed with the Framatome BWU-Z CHF correlation with a design limit DNBR value (95/95) of 1.19 for deterministic analyses and a design limit DNBR value of 1.36 for analyses where uncertainties are combined statistically.

    Derivation of these limits is described in References Q31-1, Q31-2, and Q31-2.

    References
    Q31-1. DPC-NE-2005P-A, Revision 3 (Appendix E), *Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology*, September 2001.
    Q31-2. DPC-NE-2009P-A, Revision 2, *Duke Power Company Westinghouse Fuel Transition Report*, December 2002.
    Q31-3. WCAP-15025P-A, *Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids*, February 1998.

35. Provide information on how the peaking factor limits used for MOX fuel were developed.

    Response
    During the August 13, 2003 telephone conference among Duke, Framatome ANP, and the NRC staff, the staff clarified that the specific peaking factor limits of interest are LOCA peaking limits as provided in Table 3-6 of the February 27, 2003 License Amendment Request. This response was developed accordingly.

    The MOX fuel assemblies are limited to operation within the allowed peaking presented in Table 3-6 of the license amendment request. The allowed peaking was established using the same approach as for any fuel design. A target allowable peaking (Limiting Condition for Operation) was determined considering plant operability and the required fuel service. The acceptability of that target peaking was then confirmed through appropriate analysis

(design, safety, and LOCA). The considerations included in establishing the MOX fuel assembly target limit for normal operation were:

a) MOX fuel lead assemblies are demonstration lead assemblies and it is therefore appropriate to control them to somewhat reduced limits
b) Recognition that demonstration lead assemblies should be operated with fairly high duty in order to promote the discovery of issues that should be addressed prior to batch implementation,
c) The insertion of four demonstration lead assemblies should not impose an operational hardship on the irradiating plant.

As discussed in Section 3.7.1.4 of Reference Q35-1, additional LOCA studies were performed after the submittal of the license amendment request to further refine the LOCA peaking limits for the MOX fuel lead assemblies. These studies included:

a) Time-in-life (burnup) sensitivity to determine any burnup dependent limitation on the fuel, and
b) Axial power distribution sensitivity to confirm the K_Z curve.

The results of these studies are discussed in the response to Reactor Systems RAI Question 14.

<u>Reference</u>

Q35-1. 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.