FRAMATOME ANP

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FRAMATOME ANP, Inc.

November 24, 2003
NRC:03:080

Document Control Desk
ATTN: Chief, Planning, Program and Management Support Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Proj 693

Partial Response to RAI on BAW-10238(P), Revision 1, "MOX Fuel Design Report."

Ref.: 1. Letter, James F. Mallay (Framatome ANP), to Document Control Desk (NRC), "Request for Approval of BAW-10238(P), Revision 1, 'MOX Fuel Report'," NRC:03:037, May 30, 2003.

Ref.: 2. Letter, Drew G. Holland (NRC) to James F. Mallay (Framatome ANP), "Request for Additional Information (RAI) – Topical Report BAW-10238(P), Revision 1, 'MOX Fuel Design Report'," October 8, 2003.

Ref.: 3. Letter, James F. Mallay (Framatome ANP), to Document Control Desk (NRC), "Partial Response to RAI on BAW-10238(P), Revision 1, 'MOX Fuel Design Report'," NRC:03:072, October 27, 2003.

Framatome ANP requested NRC review and approval of the topical report BAW-10238(P), Revision 1, "MOX Fuel Design Report," in Reference 1. The NRC requested additional information regarding this topical report in Reference 2. The response to questions 4, 6, 7, 10, 11, 13, and 25 of this request are provided in two attachments - one proprietary and one non-proprietary. The remaining responses to reference 2 will be submitted to the NRC by December 19, 2003.

Reference 3 contains a previously submitted response to one of the questions in Reference 2.

Framatome ANP considers some of the information contained in Attachment 1 to be proprietary. The affidavit provided with the original submittal of the topical report satisfies the requirements of 10 CFR 2.790(b) to support the withholding of this information from public disclosure.

Very truly yours,

James F. Mallay, Director
Regulatory Affairs

Enclosures

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Project 693

Request for Additional Information on Topical Report BAW-10238(P), MOX Fuel Design Report

In all responses, "BAW-10238" means *MOX Fuel Design Report*, BAW-10238(P) Revision 1, May 2003.

Question 4: *Please provide references 6, 13, 25, 24, 27, 28, 29, 30, 31, 33, 35, 36, 37, and 38.*

Response 4: Copies of all references except 37 have been provided. Distribution of Reference 37 is restricted, but a copy was made available for inspection during the audit of Framatome ANP on November 18-20, 2003.

Question 6: *In the first paragraph of section 2.4 two types of mixed cores are defined and the topical states that both types of mixed cores have been considered and there are approved methods for handling them. Please provide the references to these approved methods.*

Response 6: The first paragraph of Section 2.4 was intended as an introduction, and the following paragraphs of the section discuss the methods. Two approved methods apply for thermal-hydraulic effects: For departure from nucleate boiling, the method is discussed in Reference Q6.1. Hydraulic forces for assembly liftoff were determined using the NRC-approved LYNXT code (Reference Q6.2). A topical report on the neutronic effects of mixing LEU and MOX fuel (Reference Q6.3) is under NRC review.
References Q6.1 and Q6.3 specifically address MOX fuel. Reference Q6.2 has been approved by the NRC for licensing applications for PWR core thermal-hydraulics. The methods for calculating hydraulic forces described in this report are applicable to MOX fuel because hydraulic forces are not dependent on fuel pellet material. Therefore, the methods for calculating hydraulic forces described in this report may be used to analyze design changes for MOX. Note that this report is not cited in BAW-10238, but it is used for supporting analysis.

1. Q6.1 DPC-NE-2005P-A Revision 3, *Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology*, September 2002.
2. Q6.2 BAW-10156-A Revision 1, *LYNXT: Core Transient Thermal-Hydraulic Program*, August 1993.
3. Q6.3 DPC-NE-1005P Revision 0, Duke Power Company Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX, August 2001.

Question 7: In section 3.2.3, the first bullet states that SEM/microprobe examination of the fuel and cladding revealed no abnormal behavior. Please define abnormal behavior. Additionally, the third bullet refers to the gallium measurement uncertainty limit. Please state the uncertainty limit.

Response 7: "Abnormal behavior" with respect to the MOX test fuel is defined as any deviation from expectations based on the documented MOX fuel irradiation experience in Europe that cannot be explained solely by differences in fuel preparation or test conditions. For example, in the Average Power Test, the cladding crept outward rather than inward as in the commercial MOX experience, but this difference is readily explained by the absence of external coolant

pressure on the MOX test fuel pins. Therefore, the difference in creep direction is not abnormal behavior.

For the fuel pins withdrawn at 30,000 MWd/MTHM, the gallium measurements for both cladding and fuel are reported as parts per billion (ng/g) with an uncertainty limit of plus or minus 30 percent (see Tables 6.5 and 6.6 of Reference Q7.1).

4. Q7.1 R. N. Morris, et al., *MOX Average Power 30 GWd/MT PIE: Final Report*, ORNL/MD/ LTR-212, Volume 1, November 2001.

Question 10: *Fuel assembly stress is discussed in section 6.1.1.1 and the section includes a listing of the fuel assembly components evaluated. For each of the components listed, please provide the actual calculated stress value and the margin between the value and the limit.*

Response 10: Tables Q10.1 through Q10.10 provide the fuel component stresses and stress margins.

Table Q10.1. Fuel Rod Stress Margins for Normal Operation

Type	Condition	Maximum Applied Stress (psi)	Margin (%)
Compression	Primary membrane	[ ]	[ ]
Tension	Primary membrane	[ ]	[
Compression	Primary membrane + bending	[ ]	[ ]
Tension	Primary membrane + bending	[ ]	[ ]
Compression	Primary membrane + bending + local	[ ]	[ ]
Tension	Primary membrane + bending + local	[ ]	[ ]
Compression	Primary membrane + bending + local + secondary	[ ]	[ ]
Tension	Primary membrane + bending + local + secondary	[ ]	[
Compression	Buckling	[	[ ]
Tension	Fatigue	Usage factor = [ ]	[ ]

Note: psi = pounds per square inch

Table Q10.2. Fuel Rod Stress Margins for Faulted Conditions

Type	Condition	Maximum Applied Stress (psi) {Maximum Applied Load (lbs)}	Margin (%)
Axial compression	Primary membrane	[ ]	[ ]
Axial compression	Primary membrane + bending (due to axial load)	[ ]	[ ]
Fuel rod bending	SSE + LOCA, Primary membrane + bending	[ ]	[ ]
Axial compression	Buckling	[{ }]	[ ]

Note: SSE = safe shutdown earthquake; LOCA = loss-of-coolant accident

Table Q10.3. Guide Thimble Stress Margins for Normal Operation

Condition	Maximum Applied Stress (psi) {Maximum Applied Load (lbs)}	Margin (%)
GT buckling loads at 85 °F (lbs)	[{ }]	[ ]
GT buckling loads at 600 °F (lbs)	[{ }]	[
GT membrane (Pm) at 85 °F	[ ]	[ ]
GT membrane (Pm) at 600 °F	[ ]	[ ]
GT membrane (Pm) at 600 °F – Scram	[ ]	[ ]
GT membrane + secondary at 85 °F	[ ]	[ ]
GT membrane + secondary at 600 °F	[ ]	[ ]
Upper nozzle swage connection Compressive load, lbs (85 °F) Compressive load, lbs (600 °F)	 [{ }] [{ }]	 [ ] [ ]

Note: GT = guide thimble

Table Q10.4. Grid Restraint Sleeves and Holddown Spring Stress Margins for Normal Operation

Component	Condition	Maximum Applied Stress (psi)	Margin (%)
Spacer sleeve A	Membrane	[ ]	[ ]
Spacer sleeve B	Membrane	[ ]	[ ]
Holddown spring	Maximum stress (EOL Cold Shutdown)	[ ]	[ ]
Holddown spring	Fatigue	Usage factor = [ ]	[ ]
Clamp screw	Shear at 600 °F (top nozzle threads)	[ ]	[ ]
Clamp screw	Fatigue	Usage factor = [ ]	[ ]
Quick disconnect sleeve	Membrane @ 600 °F	[ ]	[ ]

Note: EOL = end of life

Table Q10.5. Top Nozzle and Bottom Nozzle Stress Margins for Normal Operation

Component	Condition	Maximum Applied Stress (psi)	Margin (%) Pm + Pb
Top nozzle	Normal operating EOL HD force (600 °F) + Scram load	[ ]	[ ]
Top nozzle	Normal operating EOL holddown force @ 70 °F	[ ]	[ ]
Bottom nozzle	Normal operating (EOL HD) @600 °F	[ ]	[ ]
Bottom nozzle	Normal operating (EOL HD) @600 °F + Scram load	[ ]	[ ]
Bottom nozzle filter plate	Normal operating	[ ]	[ ]

Note: EOL = end of life; Pm + Pb = primary membrane + bending; HD = holddown

Table Q10.6. Guide Thimble Attachment Component Stress Margins for Normal Operation

Component	Condition	Maximum Applied Load (lbs) {Maximum Applied Stress (psi)}	Margin (%)
Bottom end spacer grid to sleeve B Weld	Normal operating	[ ]	[ ]
Bottom sleeve B to GT end plug crimp	Normal operating	[ ]	[ ]
GT to bottom end plug weld	Normal operating	[ ]	[ ]
Top end spacer grid to sleeve A weld tab	Normal operating	[ ] [ ]	[ ] [ ]
Intermediate ferrule to GT dimple	Normal operating	[ ]	[ ]
Mid-span mixing grid ferrule to GT Dimple	Normal Oper. (Handling)	[ ]	[ ]
Mid-span mixing grid ferrule to grid interface weld	Normal Oper. (Handling)	[ ]	[ ]
Ferrule to instrument sheath dimple	Normal Oper. (Handling)	[ ]	[ ]
Guide thimble bottom bolt	Normal Oper. Axial (Preload) Bearing Stress	[ ] [ ]	[ ] [ ]
Guide thimble bottom end plug	Normal Oper. Axial (Preload) Bearing Stress	[ ] [ ]	[ ] [ ]

Note: GT = guide thimble

Table Q10.7. Guide Thimble Stress Margins for Faulted Conditions

Faulted Condition	Maximum Applied Stress (psi) {Maximum Applied Load (lbs)}	Margin (%)
GT buckling, Vertical LOCA	[ ]	[ ]
GT membrane, Vertical LOCA	[ ]	[ ]
GT membrane + bending, SSE + LOCA	[ ]	[ ]
Upper nozzle swage connection Membrane + bending, SSE + LOCA	[ ]	[ ]
Instrument sheath Membrane + bending, SSE + LOCA	[ ]	[ ]

Note: GT = guide thimble; SSE = safe shutdown earthquake; LOCA = loss-of-coolant accident

Table Q10.8. Top and Bottom Nozzle Stress Margins for Faulted Conditions

Component	Condition	Maximum Applied Stress (psi) Pm {Pm + Pb}	Margin (%) Pm {Pm + Pb}
Top nozzle	OBE + Normal operating	[ ] [ ]	[ ] [ ]
Top nozzle	LOCA + SSE	[ ] [ ]	[ ] [ ]
Bottom nozzle	OBE + Normal operating	[ ] [ ]	[ ] [ ]
Bottom nozzle	LOCA + SSE	[ ] [ ]	[ ] [ ]
Bottom nozzle filter plate	LOCA + SSE	[ ] [ ]	[ ] [ ]

Note: Pm = primary membrane; Pm + Pb = primary membrane + bending; OBE = operating basis earthquake; SSE = safe shutdown earthquake; LOCA = loss-of-coolant accident

Table Q10.9. Guide Thimble Attachment Component Stress Margins for Faulted Conditions

Component	Condition	Maximum Applied Load (lbs) {Maximum Applied Stress (psi)}	% Margin
Bottom end spacer grid to sleeve B weld	LOCA + SSE	[ ]	[ ]
Sleeve B to GT end plug crimp	LOCA + SSE	[ ]	[ ]
GT to end plug weld joint	LOCA + SSE	[ ]	[ ]
GT to collar weld joint	LOCA + SSE	[ ]	[ ]
Top end spacer grid to sleeve A weld	LOCA + SSE	[ ]	[ ]
Ferrule to GT dimple	LOCA + SSE	[ ]	[ ]
Quick disconnect sleeve	Bearing	[ ]	[ ]
	Shear	[ ]	[ ]
	Combined Stress Intensity	[ ]	[ ]
	Primary membrane	[ ]	[ ]
Guide thimble bottom bolt	Axial LOCA + SSE	[ ]	[ ]
	Shear LOCA + SSE	[ ]	[ ]
Guide thimble bottom end plug	Axial LOCA + SSE	[ ]	[ ]
	Shear LOCA + SSE	[ ]	[ ]

Note: SSE = safe shutdown earthquake; LOCA = loss-of-coolant accident

Table Q10.10. Grid Stress Margins for Faulted Conditions

Component	Condition	Maximum Applied Load (lbs) {Maximum Applied Stress (psi)}	Margin (%)
Intermediate grid	SSE	[ ]	[ ]
Intermediate grid	LOCA + SSE	[ ]	[ ]
Mid-span mixing grid	SSE	[ ]	[ ]
Mid-span mixing grid	LOCA + SSE	[ ]	[ ]
Grid restraint sleeves	Vertical LOCA Membrane	[ ]	[ ]
	LOCA + SSE Membrane + bending	[ ]	[ ]

Note: SSE = safe shutdown earthquake; LOCA = loss-of-coolant accident

Question 11: *In section 6.1.1.2, it states that "the cladding was shown to have an acceptable margin to the pressure that would cause buckling." Please provide for the worst cases, the pressure calculated and the margin between the value and the limit.*

Response 11: The calculated pressure difference (reactor coolant pressure minus fuel rod internal pressure) is [ ] psi. This value reflects the worst case reactor overpressure condition of [ ] psi minus the BOL rod hot internal pressure of approximately [ ] psi. This value of [ ] psi is a bounding lower value of the interior rod pressure under critical conditions for the initial design calculations. The pressure difference for buckling is calculated by Timoshenko's buckling criteria, and has a value of [2,460] psi. This results in a calculated margin of [ ].

Question 13: *In section 6.1.5, it states that the hydrogen pickup is controlled by the corrosion limit. Please explain how the hydrogen pickup is controlled by the corrosion limit.*

Response 13: During reactor operation, an oxide coating, ZrO_2, is formed on the outer surface of zirconium-based alloys by chemically reacting with the coolant. The reaction is:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

A fraction of the hydrogen that is generated during this reaction becomes dissolved in the zirconium alloy cladding. The quantity of hydrogen dissolved is generally accepted to be in the range of 0.12 to 0.18 of the total quantity of hydrogen generated by the corrosion process. The value is sometimes referred to as the "pickup fraction." Zircaloy-4 pickup fractions from different studies have been compiled by Garde and are shown as Figure 8 in Reference Q13.1.

Thus, the quantity of hydrogen absorbed by the cladding is controlled, or limited, by the combination of the pickup fraction, discussed above, and the limiting oxide thickness. For a given alloy and vendor, the pickup fraction is generally considered to have a fixed value, and probably depends on such variables as the cladding physical and chemical properties and the fabrication history. Hydrogen pickup depends only on the corrosion reaction between the cladding and the reactor coolant, so the pickup fraction is independent of fuel type. In the case of M5 cladding, the pickup fraction is much lower than that of Zircaloy-4 and is of the order of [ ]%. The limiting oxide thickness for the Mark-BW/MOX1 design at end-of-life (EOL) is 100 μm.

Based on this evaluation, the maximum value of the hydrogen pickup (concentration of hydrogen dissolved) is controlled by the corrosion limit (assuming a fixed value for the pickup fraction).

5. Q13.1 Anand M. Garde, "Effects of Irradiation and Hydriding on the Mechanical Properties of Zircaloy-4 at High Fluence," *Zirconium in the Nuclear Industry: Eighth International Symposium*, ASTM STP 1023, L.F.P. Van Swam and C.M. Eucken, Eds., American Society for Testing and Materials, Philadelphia, 1989, pp. 548-569.

Question 25: *In section 8.0, it states that the core fraction will be increased with the maximum core fraction (approximately 40%) being achieved with the insertion of the third batch at each reactor. Please define precisely how many fuel assemblies will be in the core at the proposed equilibrium core. Also, please provide the data and discussion to support loading the core to approximately 40% given that the French have lower core fraction limits.*

Response 25: Representative partial MOX fuel core designs are described in Appendix A.2 of Reference Q25.1 and in Reference Q25.2. These equilibrium core designs have a MOX fuel core fraction of 39.4% (76 MOX fuel assemblies out of 193 total fuel assemblies).

The planned McGuire and Catawba core design approach, culminating in core fractions of approximately 40%, is consistent with European licensing and operational experience with MOX fuel. In France, 20 EDF reactors operate with MOX fuel core fractions of approximately 30% MOX fuel, or 48 MOX fuel assemblies out of 157 total fuel assemblies. German and Swiss reactors that use MOX fuel are licensed to a variety of MOX fuel core fractions up to 50%. Five units (Isar 2, Biblis A, Biblis B, Beznau 1, and Beznau 2) are licensed for core fractions of 40% or more (Reference Q25.3). The French, German, and Swiss experience is described in Reference Q25.4. To date, the maximum MOX fuel core fractions achieved in European nuclear power reactors are 36% at Goesgen, a Swiss pressurized water reactor, and 35% at Gundremmingen B, a German boiling water reactor. As described in Reference Q25.5, the two-unit Beznau plant in Switzerland is licensed to a 40% MOX fuel core fraction and has achieved core fractions of 33% MOX.

While MOX fuel is similar to LEU fuel, there are differences between the fuel types, as discussed in BAW-10238. Some of those differences (e.g., effective delayed neutron fraction) are a function of the MOX fuel core fraction. These differences will be addressed in any future application of batch quantities of MOX fuel as they may affect reactor safety. Duke Power identified 40% as the upper limit for MOX fuel core fraction to provide a reasonable boundary condition for safety analyses and environmental analyses that will be used to support regulatory approval of the use of batch MOX fuel.

6. Q25.1 DPC-NE-1005P, *Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX*, Revision 0, August 2001 (under NRC review).

7. Q25.2 Kenneth A. Naugle, "Cycle Design Work for Transition to Partial MOX Fuel Cores," American Nuclear Society, *5th Topical Meeting on Spent Nuclear Fuel and Fissile Materials Management*, Charleston, SC, September 17-20, 2002.

8. Q25.3 Dieter Porsch, Reinhard Lisdat, and Richard Stratton, "High Burnup MOX in Light Water Reactors," Presentation to the Plutonium 2000 Conference, Brussels, Belgium, October 9-11, 2000.

9. Q25.4 Dieter Porsch, Walter Stach, Pascal Charmensat, and Michel Pasquet, "Plutonium Recycling in LWRs at Framatome ANP - Status and Trends," American Nuclear Society, *Advances in Nuclear Fuel Management III*, Hilton Head, SC, October 5-8, 2003.

10. Q25.5 Tamer Bahadir and Raul Vielma, "Validation of Studsvik CMS for Beznau MOX Cores," American Nuclear Society, *Advances in Nuclear Fuel Management III*, Hilton Head, SC, October 5-8, 2003.