FRAMATOME ANP

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FRAMATOME ANP, Inc.

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

Ref.: 1. Letter, Drew G. Holland (NRC) to James F. Mallay (Framatome ANP), "Request for Additional Information (RAI) – Topical BAW-10238(P), Revison 1, 'MOX Fuel Design Report'," (TAC NO. MB7550), October 8, 2003.

Ref.: 2. 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.

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:03:080, November 24, 2003.

This letter responds to several of the questions contained in the RAI of Reference 1. Specifically, responses to questions 1, 2, 5, 8, 9, 12, 16, 17, and 26 are provided in the attachments – one proprietary and one non-proprietary. The remaining responses to reference 1 are expected to be submitted to the NRC by December 19, 2003.

Also included in the attachments are responses to two supplemental questions posed by the NRC during its audit of the topical report on November 18 through 20, 2003.

References 2 and 3 provided responses to many of the other questions in the RAI.

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
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Enclosures

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Responses to RAI 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 1: *On page 1-2, it mentions that Framatome will perform plant-specific evaluations of the performance of the Mark-BW/MOX lead assemblies in the mission reactors but does not specify which evaluations are being referred to. Please specify which plant-specific evaluations are being referred to. And provide any results that have been obtained from these evaluations.*

Response 1: The plant-specific evaluations are included in Chapter 6 of BAW-10238. As stated in Section 6.0, "the mechanical and thermal analyses presented in the following paragraphs will be redone, if necessary, when final fuel cycle design information is available."

From Chapter 6, the following evaluations are plant or cycle specific:

Fuel rod cladding strain (6.1.2)
Fuel rod cladding oxidation and hydriding (6.1.5)
Fuel rod end-of-life pressure (6.1.8)
Fuel assembly lift-off (6.1.9)
Fuel rod creep collapse (6.2.2)
Overheating of fuel rod cladding (6.2.3)
Overheating of fuel pellets (6.2.4)
Violent expulsion of fuel (6.3.2)
Fuel assembly structural damage from external forces (6.3.4)

The following evaluations are typically generic or bounding with cycle-specific confirmation of inputs, as needed:

Fuel rod cladding stress (6.1.1.2)
Fuel rod fatigue (6.1.3)
Fuel rod fretting (6.1.4)
Cladding rupture (6.2.6)
Cladding embrittlement (6.3.1)
Fuel rod ballooning (6.3.3)

The evaluations provided below are generic and typically are not re-evaluated without specific fuel design, manufacturing process, major plant design or operational changes:

Fuel assembly stress (normal operation) (6.1.1.1)
Fuel rod bow (6.1.6)
Fuel assembly/fuel rod axial growth (6.1.7)
Internal fuel rod hydriding (6.2.1)
Pellet/cladding interaction (6.2.5)

The numbers in parentheses above indicate sections in BAW-10238.

The results of the MOX evaluations are provided in Chapter 6 of BAW-10238 and are supplemented by a more detailed summary provided in the response to question 10. The fuel rod evaluations are for the MOX lead assemblies and are applicable to all of the mission reactors. These results are not expected to change but will be reassessed for batch application considering batch cycle design inputs. The fuel assembly lift-off evaluation was performed for the McGuire 1 MOX lead assembly application since this was the original target plant. A similar analysis will be performed for the planned Catawba 2 lead assembly application and for the future batch application for all of the mission reactors using approved methods. The fuel assembly structural damage assessment, which considers the faulted conditions, was performed for both the MOX lead assemblies and batch applications for all of the mission reactors.

Question 2: *On page 1-2, it mentions that the evaluation process described in this report may be used to justify small changes without specific NRC approval. Please list every small change that this statement is referring to.*

Response 2: Because of the large number of parts in a fuel assembly, it is not practical to list every possible design change. The following definition of small changes is provided:

> "Small changes" are those that (1) may be made in accordance with 10 CFR 50.59(c) without obtaining a license amendment and (2) meet all of the following criteria:

- The change does not result in an unreviewed safety question.
- No changes in plant technical specifications are required.
- The applicability of NRC-approved design/analysis or evaluation methods is not affected.
- Materials not previously qualified for in-reactor operation in a similar application are not introduced.
- Burnup limits are not extended beyond those previously approved.

Small changes will be developed in accordance with approved methods. A change in evaluation methods meeting any of the following criteria will be submitted for NRC review:

- An existing approved design code or method is replaced.
- A new core power distribution monitoring methodology is implemented.
- A method is extended beyond previously approved limits.

The definition of small changes above is consistent with that provided in Reference Q2.1.

Q2.1 BAW-10179P-A Revision 1, *Safety Criteria and Methodology for Acceptable Cycle Reload Analyses*, February 1996.

Question 5: *On page 2-3 in the second paragraph, it states that control of the agglomerate process sequence is verified through examination of a representative number of samples from each batch of pellets. Please define what a representative number is.*

Response 5: The European (MIMAS) process for fabrication of MOX pellets is sufficiently reproducible that the European specification requires measurement of agglomerate sizes in [ ]. The measurement frequency is based on experience and has been sufficient to ensure the satisfactory irradiation performance of European MOX fuel.

For the zoned design shown in Figure 5.2 of BAW-10238, the number of pellet lots for the lead assemblies is estimated to be [ ]. The specification can therefore be satisfied by examining [ ] pellets with low, medium, and high plutonium contents, respectively. Since the pellets for the lead assemblies will be fabricated by experienced personnel on an existing MOX fabrication line, the agglomerates in the lead assembly pellets will be the same as those in MOX pellets manufactured previously on the same line, and the sampling frequency given above should be sufficient. The European pellet manufacturing facility will go through a formal qualification process to ensure that the sampling frequency is sufficient to meet Framatome ANP (U.S.) requirements.

Pellets for batch assemblies will be produced in the U.S. at the MFFF. It is expected that the European requirements on measurement frequency will be applicable at the MFFF, but that will be determined as part of the MFFF startup and qualification process.

Question 8: *In section 6.0, the last sentence of the first paragraph states "Future analysis updates to incorporate design modifications or input changes will use the same methods and criteria." This appears to indicate that changes to the fuel assembly design are anticipated. Please indicate precisely what process is anticipated for use when making changes and where the request for approval for using the process with MOX fuel is made.*

Response 8: Changes made to the Mark-BW/MOX1 fuel assembly design will follow the design change process that is part of the NRC-approved Framatome ANP quality program. The program is described in Reference Q8.1, which fulfills the requirements of 10 CFR 50, Appendix B. The design change process, which is described in Reference Q8.2, requires that all design changes be approved by the cognizant engineering organizations within Framatome ANP and that any new evaluations needed to justify the changes be documented and use NRC-approved methods. As such, the approved methods and models in BAW-10238 will ensure that adequate margins are maintained for any proposed design change.

Submission of BAW-10238 implies a request to use the Framatome ANP quality program and design change process for use with MOX. The requirements of 10 CFR 50, Appendix B apply to production facilities for special nuclear materials, such as the MFFF, without a distinction between facilities for uranium and those for plutonium. Therefore, the Framatome ANP quality program and design change process are applicable equally to LEU and MOX fuel.

Q8.1 FQM Revision 1 U.S. Version, *Fuel Sector Quality Management Manual*, July 2003.

Q8.2 QAP-04, *Design Control*, July 2002.

Question 9: *In section 6.0, the third paragraph states that "COPERNIC is also used to provide pressures, oxide thicknesses, and strains for mechanical analyses that use approved methods from other sources." Please specify which approved methods are being referred to and the other sources.*

Response 9: COPERNIC is used in the following sections of BAW-10238:

- 6.1.2 Cladding Strain
- 6.1.3 Cladding Fatigue

- 6.1.5 Cladding Oxidation
- 6.1.8 Fuel Rod EOL Pressure
- 6.2.2 Creep Collapse
- 6.2.4 Overheating of Fuel Pellets
- 6.2.5 Pellet/Cladding Interaction

The analyses in Section 6.1.2, 6.1.5, 6.1.8, 6.2.4, and 6.2.5 are completed without recourse to any other approved methods.

Calculations of cladding fatigue (Section 6.1.3) use the approved method described in BAW-10227P-A (Reference Q9.1). The design method requires that the cladding fatigue usage factor be less than 0.9. This criterion is independent of the fuel pellet material, so it is applicable to MOX fuel as well as LEU fuel. The Condition I, II, and III events are likewise independent of the fuel pellet material. The methods for analyzing cladding fatigue correctly reflect the properties of MOX fuel. Therefore, the method is applicable to MOX fuel.

Calculations of creep collapse (Section 6.2.2) use the approved methods described in BAW-10084P-A (Reference Q9.2). Differences in fuel behavior between LEU and MOX are reflected in the inputs to the program, which include time-dependent fuel rod internal pressures, cladding temperatures (at the inside and outside surfaces), and neutron fluxes. Apart from these inputs, the creep performance of the cladding is not dependent on fuel pellet material. Therefore, the methods for calculating creep collapse described in this report may be used to analyze design changes for MOX.

Q9.1 BAW-10227P-A, *Evaluation of Advanced Cladding and Structural Materials (M5) in PWR Reactor Fuel*, February 2000.

Q9.2 BAW-10084P-A Revision 3, *Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse*, July 1995.

Question 12: *In section 6.1.2, it discusses transients that induce a 1% cladding strain. Please specify which transients induce a 1% cladding strain and provide a discussion to quantify the difference between the 1% inducing transient and the maximum transient that the fuel rod is expected to experience.*

Response 12: As Section 6.1.2 indicates, the linear heat generation rate (LHGR) limit based upon the 1% transient cladding strain criterion is determined by simulation of design transient axial flux shapes imposed at [ ]. The transient axial flux shapes used in the analysis are generated by simulation of design xenon transients typical of severe load following reactor duty. For each burnup point analyzed, the magnitude of the axial flux shape determined by the xenon transient is scaled up to a level that causes the cladding strain to reach 1%, and the corresponding LHGR at that point defines the LHGR limit for the fuel.

The transients may also include regulating RCCA insertion in excess of the RCCA insertion limit specified in the Core Operating Limits Report (COLR) and an axial flux difference (AFD) in excess of the AFD limits specified in the COLR. The conservative nature of the transient simulation, coupled with scaling up the LHGR in order to reach a value that would produce a 1%

strain, indicates the LHGR at 1% strain would be similar to or greater than a level reached during a Condition II transient.
Table Q12.1 provides representative limiting LHGR values that induce a 1% cladding strain. The table includes results for burnups besides those discussed in BAW-10238.

Table Q12.1. Rod-Average Linear Heat Generation Rates Required to Induce 1% Cladding Strain

Burnup (MWd/MThm)	Rod-Average LHGR (kW/ft)
[ ]	[ ]*
[ ]	[ ]
[ ]	[ ]
[ ]	[ ]
[ ]	[ ]

* At this burnup, the maximum LHGR is limited by centerline fuel melting rather than cladding strain.

During power operation, reactor cores are constrained to operate within normal operating (Condition I) limits by the limits imposed upon core power distribution specified in the COLR. In addition, most commercial reactors operate in a mode typical of base load power generation. Load follow transients, if they occur, are performed within the bounds of normal operation and thus do not approach linear heat rate generation rates that challenge the 1% strain limit. Should a Condition II transient occur that causes local LHGR levels to increase, the reactor protection system would terminate the transient before any core safety limits, including transient cladding strain, are reached.

An indication of the difference that could be expected between the 1% strain inducing transient and normal operation can be estimated by comparing the LHGR limit values in the table above to the core average linear heat rate of the reactor increased by the maximum peaking factors that might be expected at steady-state operation. McGuire and Catawba reactors operating at a rated thermal power of 3411 MWt have a core average LHGR of about 5.58 kW/ft and a maximum local pin peaking factor (Fq) of about 1.70. This gives a peak nominal LHGR of approximately 9.5 kW/ft. These values provide an indication of the range of maximum LHGRs that might be expected for the fuel rod during normal operation. As indicated above, higher values could be expected for Condition I or II transients, but the reactor protection system would prevent core power peaking factors from exceeding fuel design limits.

Question 16: *Section 6.1.9 states that conservative values for fast fluence on the holddown spring arising from the specific MOX fuel neutron spectrum are used to establish the holddown spring relaxation characteristics.*

A) *What were the conservative values used?*

B) *It later states that the liftoff will be minimal, and the holddown spring deflection will be less than the worst-case normal operating condition. Please provide a numeric definition for minimal and define the worst-case normal operating cold-shutdown condition.*

C) *Please provide the margin between the calculated value and the limit for all operating conditions.*

Response 16:

A) The maximum fast fluence calculated in the holddown spring region and used to determine the holddown spring relaxation is [ ] n/cm^2. This corresponds to a fuel assembly with a rod burnup of [ ] MWd/MThm.

B) Liftoff of the fuel assembly during the 120% pump overspeed condition is allowed at operating conditions. However, the fuel assembly must not compress the holddown spring to solid height during the event, and the holddown spring function should be maintained following the event. The maximum spring compression during the 120% pump overspeed condition is [ ] inches. The available spring stroke to solid height is [ ] inches minimum, thus sufficient margin exists to ensure that the 120% overspeed condition does not cause the spring to reach its solid height. The corresponding maximum net load on the holddown spring at the 120% overspeed operating condition is [ ] lbs. The elastic range of the holddown spring at operating conditions is [ ] lbs. The holddown spring remains fully elastic and therefore functional during and following the pump overspeed event. For comparison, the maximum (worst-case) holddown spring load during the normal operating cold shutdown at beginning of life is [ ] lbs. [ ]

C) Table Q16.1 below provides the holddown margins for the Mark-BW/MOX lead assemblies for the McGuire mission reactors for the stated conditions. A similar analysis will be performed for the planned Catawba 2 lead assembly application and for the future batch application for all of the mission reactors.

Table Q16.1. Holddown Margins for Mark-BW/MOX Lead Assemblies (McGuire)

Condition	Minimum Holddown Load (lbs)	Net fuel assembly lift (lbs)	Margin (%)
85 °F EOL 4th Pump Startup	[ ]	[ ]	[ ]
85 °F EOL 4th Pump Startup	[ ]	[ ]	[ ]
Hot operating BOL	[ ]	[ ]	[ ]
Hot operating EOL	[ ]	[ ]	[ ]
Hot operating 120% Pump Overspeed	[ ] (elastic load limit)	[ ]	[ ]

Note: BOL = beginning of life; EOL = end of life

Note that the minimum holddown load is conservatively based on minimum spring deflection, including maximum spring relaxation and zero fuel assembly growth, and minimum spring stiffness. The hydraulic lift is conservatively based on the mechanical design flow rate, bounding (low) core bypass flow, and bounding mixed core configurations.

Question 17: *In section 6.2.1, what is the European fill gas specification for MOX fuel? How does it compare to the proposed 15 ppm limit on total hydrogen in the pressurization gas?*

Response 17: Table Q17.1 lists the limits on impurities in the fill gas for reactor-grade MOX fuel, as given in the European specification. [ ] the 15 ppm limit in the U.S. specification on total hydrogen, including H_2O, H_2, and hydrocarbons. The European specification is less restrictive [ ].

Table Q17.1. Limits on Impurities from European Specification for Helium Fill Gas

[

]

Question 26: *In section 8.3.1.1, the plutonium feed material is discussed. Please specify the important chemical and physical properties of the PuO_2 powder and define the range of acceptability for each property.*

Response 26:

Chemical Properties: The important chemical properties for PuO_2 can be divided into two categories: those that affect fabrication and those that influence in-core operation. The limits defined in the specification cover both aspects.

From the fabrication aspect, the powder needs to be free-flowing and have a minimal tendency to form large agglomerates, i.e., it must be capable of being blended with the diluent UO_2 powder to meet the MOX pellet specification limits on plutonium-rich agglomerate sizes. To ensure that the powder flows freely and does not form large agglomerates, the moisture content is controlled to less than [ ]

From the fuel performance aspect, the primary elements of concern are those with large neutron cross sections. Such elements influence the reactivity calculations and excessive quantities could require higher PuO_2 percentages to compensate for their poison effects. The rare earth elements (gadolinium, dysprosium, samarium, and europium), boron, cadmium, and silver are those of greatest significance and are controlled by the MOX pellet specification. Since the PuO_2 powder represents at most 5% of the total MOX pellet composition, some of the chemical requirements can be relaxed, provided the diluent UO_2 powder has sufficiently low impurity levels. The maximum concentration for boron has been relaxed from [ ].

All other impurity elements except gallium are specified at levels that are consistent with those permitted in the feed UO_2 powder. Gallium is rarely measured in UO_2 powder or pellets (and is not limited by specification for LEU fuel or for RG MOX fuel).

An early review of published data suggested that gallium concentrations in PuO_2 poweder could be reduced to levels no greater than 120 ppb, and that value was tentatively selected as the

specification limit. A subsequent analysis, which postdates BAW-10238, supported relaxing the limit to 300 ppb gallium in PuO_2, and the specification has been changed accordingly. The relaxation is supported by results from the Average Power Test. In that test, two batches of MOX pellets, with average gallium concentrations of 1.3 and 3.0 ppm, were successfully irradiated in the Advanced Test Reactor (ATR). The gallium had no impact on irradiation performance. Burnups in excess of 40 GWd/MThm have been achieved, with 50 GWd/MThm expected in early 2004.

Physical Properties: As stated above, the primary requirements for the PuO_2 powder are that the material flows freely and that it can be blended readily with the UO_2 powder to meet the specification criteria for the MOX pellets. Thus, the specification criteria have been established based on typical powder physical property measurement techniques, namely specific surface area (BET method), particle size analysis (Coulter Multisizer analyzer), and bulk and tap density measurements. Powder acceptance is on a lot-to-lot basis with one or more analyses being performed on each lot.

The specified limits for these measurements are given in Table Q26.1.

Table Q26.1. Requirements for Physical Properties of PuO_2 Powder

Criterion	Limits
Specific surface area	[ ]
Particle size	[ ]
Bulk density	[ ]
Tap density	[ ]

Supplemental Questions Posed by the NRC During its Audit of the Topical Report from November 18 through 20, 2003:

Supplemental question 1: *Figure 6.1 shows corrosion behavior only up to 50,000 MWd/MThm. Please discuss the corrosion of MOX fuel in the burnup range from 50,000 MWd/MThm to 60,000 MWd/MThm.*

Response 1: Reference SQ2.1 reports measurements of cladding oxide thickness for fuel rod burnups of up to 70,000 MWd/MThm, including numerous points in the range from 50,000 MWd/MThm to 60,000 MWd/MThm.

Cladding oxidation reflects chemical reactions at the interface between the cladding and the primary coolant. Since the cladding separates the fuel pellets from this interface, the oxidation behavior does not depend on the fuel pellet material. Therefore, LEU and MOX fuel will have the same oxide thicknesses for a given burnup.

SQ2.1 A. Seibold and J. P. Mardon, "M5™ Cladding Experience in European PWRs," *Jahrestagung Kerntechnik 2002, Annual Meeting on Nuclear Technology*, Stuttgart, May 14-16, 2002.

Supplemental question 2: *According to section 6.1.2, the maximum burnup for which transient cladding strain has been analyzed is less than the burnup limit for the lead assemblies, 60,000 MWd/MThm. Please discuss cladding strain for burnups between those discussed in section 6.1.2 and 60,000 MWd/MThm.*

Response 2: The transient cladding strain limit was analyzed at 60,000 MWd/MThm, but the results were not reported in BAW-10238. The results for 60,000 MWd/MThm are included in the response to question 12.