September 12, 2005

Mr. Damian Peko
Senior Policy Advisor
International Programs
US Department of Energy
1000 Independence Avenue, S.W.
NA-26, Room 6G092
Washington, DC 20585

SUBJECT: NRC FUEL LICENSING REQUIREMENTS FOR REACTOR FUEL AND VALIDATION OF NEW FUEL TYPES

At a bilateral meeting held on October 11-14, 2004, between representatives of the Nuclear Industrial and Environmental Regulatory Authority of Russia (Rostekhnadzor), the United States (U.S.) Department of Energy (DOE), and the U.S. Nuclear Regulatory Commission (NRC), Mr. Mikhail Miroshnichenko of Rostekhnadzor commented upon the differences between U.S. and Russian policies pertaining to the acceptability of nuclear power plant fuel design. The enclosure to this letter is a white paper requested by the U.S. DOE, which summarizes NRC requirements for licensing reactor fuel, and summarizes how new fuel types are licensed in the U.S. without research reactor data.

Sincerely,

/RA/

James W. Clifford, Acting Chief
Special Projects Branch
Division of Fuel Cycle Safety
and Safeguards
Office of Nuclear Material Safety
and Safeguards

Enclosure: White Paper

cc: JOlencz, DOE
GSmith, DOE

CONTACT: Jeremy Tapp, 415-2258

Mr. Damian Peko
Senior Policy Advisor
International Programs
US Department of Energy
1000 Independence Avenue, S.W.
NA-26, Room 6G092
Washington, DC 20585

SUBJECT: NRC FUEL LICENSING REQUIREMENTS FOR REACTOR FUEL AND VALIDATION OF NEW FUEL TYPES

At a bilateral meeting held on October 11-14, 2004, between representatives of the Nuclear Industrial and Environmental Regulatory Authority of Russia (Rostekhnadzor), the United States (U.S.) Department of Energy (DOE), and the U.S. Nuclear Regulatory Commission (NRC), Mr. Mikhail Miroshnichenko of Rostekhnadzor commented upon the differences between U.S. and Russian policies pertaining to the acceptability of nuclear power plant fuel design. The enclosure to this letter is a white paper requested by the U.S. DOE, which summarizes NRC requirements for licensing reactor fuel, and summarizes how new fuel types are licensed in the U.S. without research reactor data.

Sincerely,
/RA/
James W. Clifford, Acting Chief
Special Projects Branch
Division of Fuel Cycle Safety
and Safeguards
Office of Nuclear Material Safety
and Safeguards

Enclosure: White Paper

cc: JOlencz, DOE
GSmith, DOE

DISTRIBUTION:
FCSS, r/f RPierson, FCSS JGiitter, FCSS FAkstulewicz, NRR
PClifford, NRR UShoop, NRR HScott, RES RMeyer, RES
JOlivier, FMS TCox, SPB

ML051540149

OFFICE	MOFLS	GCFLS	MOFLS	NRR	SPB
NAME	JTapp	LMarshall	SMagruder	JWermiel	BSmith for JClifford
DATE	6/3/05	6/6/05	6/8/05	6/13/05	9/12/05

OFFICIAL RECORD COPY

White Paper - NRC Licensing Requirements for Reactor Fuel and Validation of New Fuel Types

At a bilateral meeting between representatives of the Nuclear Industrial and Environmental Regulatory Authority of Russia (Rostekhnadzor), the United States (U.S.) Department of Energy (DOE), and the U.S. Nuclear Regulatory Commission (NRC), Mr. Mikhail Miroshnichenko of Rostekhnadzor commented upon the differences between U.S. and Russian policies pertaining to the acceptability of nuclear power plant fuel design. Mr. Damian Peko of the U.S. DOE requested the preparation of this white paper, which summarizes NRC requirements for licensing reactor fuel, and summarizes how new fuel types are licensed in the U.S. without research reactor data.

NRC FUEL LICENSING REQUIREMENTS

Fuel System Design

The Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG-0800, Chapter 4.2, “Fuel System Design”) provides guidance for conducting a review of the fuel system designs as described in the applicant's safety analysis report (SAR). Specifically, Chapter 4.2, Section II.C, “Design Evaluation,” identifies acceptable methods for demonstrating the fulfillment of design bases. This guidance is based on regulations, general design criteria, regulatory guides, industry standards, independent calculations, and staff judgements with respect to fuel system functions and component selections. The requirements relevant to the thermal, mechanical, and material design of the fuel system are as follows:

1. 10 CFR Part 50.46 and reference to 10 CFR Part 50 Appendix K, establishes fuel temperature limits that are used as a basis for the cooling performance requirements for the emergency core cooling system (ECCS) using an acceptable evaluation model, and establishing acceptance criteria for light water nuclear power reactor ECCS.

2. 10 CFR Part 100 provides acceptance criteria for determining the acceptability of a reactor site based on calculating the radiation exposure to an individual as a result of fission product releases to the environment following a major accident scenario. These criteria are used to establish fuel load limits based on source term assumptions for accident analyses.

3. 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 10, requires that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

4. 10 CFR Part 50, Appendix A, GDC 27, requires that the reactivity control system be designed with appropriate margin, and in conjunction with the ECCS, capable of controlling reactivity and cooling the core to assure fuel integrity under post accident conditions.

5. 10 CFR Part 50, Appendix A, GDC 35, requires an ECCS to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

The requirements in the Nuclear Design section that follows provide assurance through the requirements listed above that (a) the fuel system is not damaged as a result of normal operation and AOOs, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. A fuel system is "not damaged" when fuel rod cladding integrity is maintained, fuel system dimensions remain within operational tolerances, and functional capabilities of the fuel system are not reduced below those assumed in the safety analyses. The design limits that assure compliance with GDC 10 for normal operation and AOOs are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure," means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100, for postulated accidents. “Coolability,” which is sometimes termed coolable geometry, means that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat even after an accident.

Specifically, the following thermal, mechanical, and material fuel system damage mechanisms are analyzed to confirm that the design criteria are not exceeded during normal operation including AOOs: stress, cladding strain, cladding fatigue, fretting, oxidation, hydriding, crud buildup, fuel rod bow, axial growth, fuel rod internal pressure, and assembly liftoff. For example, GDC 10 requires “specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of (AOOs).” If the fuel rod was designed so that the internal fuel rod pressure became too high, the fuel rod could fail and the first barrier to fission products will have been breached. Therefore, this is an important damage mechanism to analyze in order to prevent fission product release. If the fuel is designed correctly it will remain below the design limits and satisfy GDC 10.

Nuclear Design

SRP, Chapter 4.3, “Nuclear Design,” is used in the reviews of the nuclear design of the fuel assemblies, control systems, and reactor core as an acceptable method to confirm that fuel design limits will not be exceeded during normal operation or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core. The requirements that assure the previous statement is true are as follows:

1. 10 CFR Part 50, Appendix A, GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of AOOs. The reactor core's nuclear design is one of several key design aspects that ensure fuel design limits will not be exceeded during normal operations. Compliance with GDC 10 significantly reduces the likelihood of fuel failures occurring during normal operations, including AOOs, thereby minimizing the possible release of fission products to the environment.

2. 10 CFR Part 50, Appendix A, GDC 11 requires that the reactor core and associated coolant systems be designed so that the net effect of prompt inherent nuclear feedback characteristics in the core tend to compensate for rapid increases in reactivity when operating in the power range. The nuclear design of the reactor core establishes the various reactivity coefficient

values that produce the desired feedback characteristics. Compliance with GDC 11 results in the reactor core being inherently safe during power range operations, thus eliminating the possibility of an uncontrolled nuclear excursion.

3. 10 CFR Part 50, Appendix A, GDC 12 requires that the reactor core and the associated coolant, control, and protection systems be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed. Power oscillations within the reactor core may result from conditions such as improper fuel design or loading, or improper reactivity control including control rod positioning, coolant flow instabilities, and moderator void formation or instabilities associated with nonhomogeneous reactor coolant density distributions.

4. 10 CFR Part 50, Appendix A, GDC 13 requires that instrumentation and controls be provided to monitor variables and systems that can affect the fission process over normal operating ranges, AOOs, and accident conditions, and to maintain the variables and systems within the prescribed operating ranges. The nuclear design review includes verification that instrumentation and systems, along with the data processing systems and alarms, will reasonably assure maintenance of core power distributions within specified design limits. Compliance with GDC 13 provides assurance that instrumentation and controls systems can adequately monitor changes in core reactivity and maintain variables that affect core reactivity within designed operating ranges, thus minimizing the possibility of an adverse transient affecting the integrity of the fuel cladding.

5. 10 CFR Part 50, Appendix A, GDC 20 requires automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to assure automatic operation of systems and components important to safety under accident conditions. Review of the nuclear design verifies the adequacy of control systems and setpoints necessary to shutdown the reactor at any time during operation, which include AOOs or accident conditions. The automatic initiation of control systems during a reactor transient prevents damage to the nuclear fuel and in the early stages of a reactor accident will minimize the extent of damage to the fuel, thus reducing the release of fission products to the reactor coolant system and possibly the environment.

6. 10 CFR Part 50, Appendix A, GDC 25 requires that no single malfunction of the reactivity control system can cause violation of acceptable fuel design limits. The nuclear design review includes verification that no single malfunction of the reactivity control system results in the fuel design limits to be exceeded. Meeting the requirements of GDC 25 provides reasonable assurance that a malfunction in the reactivity control system would not result in exceeding fuel design limits.

7. 10 CFR Part 50, Appendix A, GDC 26 requires that two independent or redundant reactivity control systems of different or diverse design be provided. Review of the nuclear design verifies that two redundant and diverse reactivity control systems exist, and that one system can reliably control the rate of reactivity changes during normal power operation and AOOs. The review also verifies that one system uses control rods and one can hold the core subcritical under cold conditions. Compliance with GDC 26 provides assurance that core reactivity can be safely controlled and that sufficient negative reactivity exists to maintain the core subcritical under cold conditions, thus assuring that fuel design limits are not exceeded.

8. 10 CFR Part 50, Appendix A, GDC 27 requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods to assure that the capability to cool the core is maintained. The nuclear design review verifies that the reactivity control systems provide a movable control rod system and a poison addition system, and that the core has sufficient shutdown margin assuming a stuck rod. Meeting the requirements of GDC 27 provides assurance that the reactivity control system will be designed such that the capability to cool the core is maintained.

9. 10 CFR Part 50, Appendix A, GDC 28 requires that the effects of postulated reactivity insertion accidents not result in damage to the reactor coolant pressure boundary nor cause sufficient damage to the core, its support structures, or vessel internals to significantly impair the capability to cool the core. The nuclear design review assures that the proper reactivity coefficients and rod worths are used in the analysis of reactivity insertion events in Chapter 15 of the SRP. Compliance with GDC 28 provides assurance that the second barrier (i.e., the reactor coolant pressure boundary) that prevents the release of fission products to the environment will not be damaged in the event a reactivity insertion accident were to occur, and that core cooling will not be prevented by the structural collapse of fuel in the core.

The requirements above relate to the reactor physics, including the fuel assembly; reactivity control systems such as control rods or burnable poisons; and core loading requirements. In order to satisfy these criteria, the new fuel assembly and core design information must be analyzed to prove compliance with these requirements. For example, GDC 11 requires “that in the power operating range...the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.” To satisfy this requirement, the fuel design must assure the fuel assembly and core is designed so that in the power operating range, the nuclear feedback with a rapid increase in reactivity is negative, or the feedback decreased the reactivity. Meeting these requirements provides reasonable assurance that the public’s health and safety are not compromised.

<u>Thermal and Hydraulic Design</u>

In addition, SRP Chapter 4.4, “Thermal and Hydraulic Design,” provides an acceptable method to confirm that the thermal and hydraulic design of the core and the reactor coolant system (RCS) has been accomplished using acceptable analytical methods; is equivalent to or is a justified deviation from proven designs; provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs; and is not susceptible to thermal-hydraulic instability. The requirements relevant to SRP Chapter 4.4 are listed below:

1. 10 CFR Part 50, Appendix A, GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. Proper thermal-hydraulic design of the reactor core and associated systems is necessary to assure sufficient margin exists with regard to maintaining adequate heat transfer from the fuel to the RCS. Compliance with GDC 10 provides adequate assurance that the integrity of the fuel and cladding will be maintained, thus significantly reducing the potential for release of fission products during normal operation or AOOs.

2. 10 CFR Part 50, Appendix A, GDC 12 requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed. Power oscillations within the reactor core may result from conditions such as improper fuel design or loading, improper reactivity control including control rod positioning, coolant flow instabilities, moderator void formation, and instabilities associated with nonhomogeneous reactor coolant density distributions. The occurrence of power oscillations can lead to excessive localized power peaking, cyclic thermal fatigue, potential for exceeding fuel design limits, and fuel failure. Compliance with GDC 12 provides reasonable assurance that the thermal-hydraulic design of the reactor core and associated systems protects the reactor from the consequences of power oscillations that could challenge the integrity of the fuel and result in the release of fission products.

The thermal and hydraulic design review complements the fuel and nuclear design reviews to assure the core, fuel, and the control and mitigative systems provide adequate assurance of fuel and coolant integrity. They are an integrated set of requirements mutually supportive of each other to assure the protection of the public's health and safety.

NEW FUEL LICENSING

Previous Use of MOX in the U.S.

The United States had experimented with prototypical MOX fuels during the 1960's and 1970's, before plutonium recycling programs were halted during the Carter administration. Prior to the halt of research, the U.S. tested plutonium fuels in several test reactors and developed basic information about irradiated plutonium fuel. The information included which types of MOX fuels perform the best in a reactor, both physically and economically. Defects were intentionally added to fuel rods in order to observe the fuel reactions. Post-irradiation studies included visual tests, gamma scans, burnup analysis, and fuel/cladding microscopy. Tests performed by the Electric Power Research Institute included loading MOX fuel assemblies into operating reactors (Quad Cities, San Onofre, and Big Rock Point) in order to examine the effects of incore irradiation on MOX fuel assemblies. If the NRC should receive an application for a batch loading (1/3 core or more) of MOX fuel, the nuclear utilities in the U.S., and the NRC, have not determined whether or not these previous U.S. prototype testing results will be used as part of the basis to license a batch load of MOX fuel.

Fuel Licensing

The process for all new fuel licensing in the United States requires three basic steps; (1) approval of the analytic model, (2) approval of the analytic method, and (3) verifying that the licensee has shown that the analytic model and method are applicable to its new fuel design and all conditions and limitations are met. It is possible to license a new fuel design without research reactor data. The NRC believes that irradiated fuel data is required, however, to show how the analytic model and method are applicable to the new fuel design and how all conditions and limitations are met. The process used in the U.S. to obtain this data without a research reactor is described in the following section using the new MOX fuel design as an example.

Fuel Licensing with Lead Test Assemblies (LTAs)

The requirements for licensing fuel in power reactors are described in the first section of this paper. The methods used to assure compliance with these requirements for new fuel types include operating experience, prototype testing, and analytical methods. The United States recognizes the need to get irradiated fuel data for new fuel types to assure fuel design limits are not exceeded, but has no batch load MOX operating experience of its own. This data will be obtained in two ways, both of which are important to licensing new fuel. First, the U.S. will acquire data from the reactor-grade (RG) MOX batch load operating experience of the European countries. Second, since conclusive operating experience is not available for weapons-grade (WG) MOX fuel, prototype testing will be conducted and reviewed. Belgium has processed and utilized RG MOX fuel for over forty years, and has invented the micronized master blend (MIMAS) manufacturing process that will be utilized in the United States MOX program. The MIMAS process has been in use since 1987 and has been used to produce pellets for 435,000 fuel rods. To date, European data has shown that RG MOX fuel rod performance has been similar to that of low enriched uranium (LEU) fuel rod performance characteristics. NRC representatives visited the COGEMA Cadarache facility in France in November 2004, to survey the processes used to fabricate mixed oxide fuel. Laboratory assessments of the fully irradiated MOX pellets showed that the pellets' physical limits would meet the NRC performance requirements outlined earlier in this paper, and that the rods and pellets are built to approved specifications, as related in the NRC approved Framatome-ANP (FANP) MOX Fuel Design Report.

An LTA is designed to gather data on fuel performance. LTAs are typically based on current production designs, and in this case, the fuel cladding material is an NRC-approved cladding material. LTAs are irradiated to obtain fuel performance data. In the past, the data indicated when slight design modifications were necessary. As a result, minor design changes were implemented into the production designs to further improve the high fuel reliability. Data from LTAs will also provide the basis for improved fuel designs and validation and improvement of analytical models. Specifically, the use of four MOX LTAs for prototype testing in Duke Energy Corporation's (Duke's) Catawba reactor have the following purposes:

1. Demonstrating that the data and analytical models derived from the RG MOX fuel are consistent with WG MOX behavior.

2. Demonstrating the applicability of the MIMAS and impurity polishing processes to the use of WG MOX fuel.

3. Confirming that the lower trace levels of gallium do not impact the fuel cladding.

4. Demonstrating the acceptable performance of the Mark-BW/MOX1 fuel assembly design.

5. Demonstrating that the MOX fuel performance under U.S. pressurized water reactor (PWR) operating conditions is acceptable.

6. Confirming the validity of the neutronic models.

During the LTA program, the reactor licensee and the fuel vendor will ensure that each assembly undergoes a testing and inspection program that contains three major parts; (1) pre-characterization, (2) on-line fuel monitoring, and (3) poolside and hot cell post-irradiation examinations (PIEs).

First, pre-characterization is performed prior to undergoing exposure in the “test” cycle. Pre-characterization is the measurement of particular fuel performance parameters before the start of the cycle and provides a measure of how much margin exists for a given design criterion, based on model predictions compared to the pre-characterization measurement. Model performance will be shared with the NRC along with the PIE data results.

Second, when the MOX LTAs are inserted into Catawba, at least one assembly will be in an instrumented location to collect operational neutronic data and verify predicted operational neutronic performance.

Third, poolside PIEs will be performed between each refueling cycle and at discharge, while hot cell PIEs will be performed after two cycles and also after a third cycle if an LTA is irradiated for a third cycle. The poolside PIEs will be basic examinations that will include fuel assembly visual, fuel rod visual, fuel assembly growth, fuel rod growth, and fuel assembly bow and distortion examinations. In addition, following core discharge, a more extensive poolside PIE will be performed that includes grid width, fuel rod oxide thickness, grid oxide thickness, fuel assembly drag force, guide thimble plug gauge, and fuel rod bowing examinations. The hot cell PIEs will include rod puncture, metallography/ceramography, cladding mechanical tests, burnup analysis, and burnup distribution examinations. The PIE results will be documented in a PIE report and the results of the PIE assessment will be factored into future analysis to ensure that appropriate conservatisms are being maintained. In addition, tracking of the data results will provide the basis for developmental model creation to more accurately model fuel performance and to capture fuel performance fundamentals. Reports containing data gathered by the vendor/utility from the LTA program will be presented to the NRC. Model performance will also be tracked against data and presented to the NRC. These irradiated fuel assembly examinations will also provide MOX data that can be used to assess their behavior under U.S. pressurized water reactor (PWR) conditions.

Comparison of pre-cycle and post-cycle values, obtained from the PIEs, will yield the incremental effects that the cycle of exposure has on the LTAs. The comparison of the pre-cycle and post-cycle examinations of the LTAs provides a determination of whether an unknown phenomenon exists and is occurring in the LTAs. Also, comparison of the predictive fuel performance model values for the LTAs to the pre-cycle and post-cycle PIE values will provide a very accurate measure of how well the fuel performance models predict the physical fuel parameters for the cycle of exposure, which are explained below.

Analytical methods are another important aspect of licensing fuel in the United States. Before the license could be issued to approve the four LTAs going into Duke’s Catawba reactor, the NRC had to ensure these four LTAs would be safe. The U.S. experience base is limited to UO_2 fuel. With the differences between UO_2 and MOX fuel, which include for MOX fuel higher decay heat, less negative void coefficients, a harder neutron spectrum, higher fuel centerline temperatures, an increase in gas release, a smaller delayed neutron fraction, and differences in fission products, the NRC looked to the European MOX experience base. As mentioned earlier in this paper, European MOX experience is with RG MOX, not WG MOX, which the U.S. will be using. There is a small difference between the two, specifically, the isotopic distribution, but chemically they are the same. Therefore, using analytical methods based on RG MOX applications did not require substantial modifications before applying them to WG MOX.

As described in the MOX Fuel Design Report by FANP, the COPERNIC and CASMO-4/SIMULATE-3 MOX codes were used to license the four LTAs. The COPERNIC code, mentioned previously in the LTA section, predicts fuel performance for design purposes and was validated with the NRC audit code, FRAPCON-3.2, with WG MOX properties. FRAPCON-3.2 was verified against thermal, irradiated WG MOX fuel rod segment data. The properties specific to WG MOX fuel calculated by COPERNIC and verified by FRAPCON-3.2 are thermal conductivity, thermal expansion, thermal creep, fission gas release, in-reactor densification and swelling, helium gas accumulation and release, radial power profile, and melting point. To further verify COPERNIC, the thermal conductivity and fission gas release properties were compared to Halden test reactor data. European operational experience was also shown to agree with the COPERNIC code. Through FRAPCON-3.2, Halden test reactor data, and European operational data, the NRC approved COPERNIC to design the four LTAs.

The CASMO-4/SIMULATE-3 MOX code was also used to support core reload design, core follow, and calculation of key core parameters for reload safety analysis. The NRC conducted a detailed review of Duke's comparison of calculated key physics parameters to measurements obtained from several operating cycles of Catawba and McGuire, the St. Laurent reactor in France, and several MOX critical experiments that used a MOX fuel very close to that of WG MOX. These results were then used by Duke to determine the set of 95/95 (probability/confidence) tolerance limits for application to the calculation of the stated physics parameters. The benchmarking of CASMO-4/SIMULATE-3 MOX to the applicable U.S. PWR plant data, French MOX reactor data, and MOX critical experiment data validates this code to perform the design analyses required to license the four LTAs to operate in Catawba.

To support the licensing of a batch load of MOX fuel, if one should be requested, data must be obtained from the on-line monitoring of the MOX LTAs and the poolside and hot cell PIEs. The on-line monitoring of the MOX LTAs will confirm whether or not CASMO-4/SIMULATE-3 MOX can accurately predict key core physics parameters (within the uncertainties established during the code reviews) and produce accurate core follow results for WG MOX. The poolside PIE results will be compared to the new fuel inspections to assure the MOX LTAs performed as expected. Last, the hot cell PIEs will be performed and their results used to validate the analytical models, in COPERNIC, that were developed for WG MOX fuel.

Summary

Instead of using research reactors to test new fuel, the NRC uses a multifaceted approach to assure the adequacy of new fuel designs and associated models. The approach includes: (1) prototype testing such as LTAs and previous tests, (2) predictions from analytical methods that are benchmarked and validated through actual MOX fuel data and independent code calculations, and (3) new fuel inspections and PIE results of incore fuel behavior that will provide an accurate measure of how well the predictive fuel performance models and LTAs are behaving.

References

NRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800, Draft Revision 3, April 1996.

W. Travers, “Mixed-Oxide Fuel Use in Commercial Light Water Reactors,” white paper, U.S. Nuclear Regulatory Commission, April 1999. (ML993620025)

G. Galletti and U. Shoop, “Trip Report for Visit to the Cogema Cadarache Facility in Support of Mixed Oxide Fuel Lead Test Assembly Fabrication,” trip report, U.S. Nuclear Regulatory Commission, December 2004.

NRC, “Partial Initial Decision (Regarding BREDL Safety Contention I),” LBP-04-32, U.S. Nuclear Regulatory Commission, December 2004. (ML043650066)

Safety Evaluation for Topical Report BAW-10238(P), Revision 1, “MOX Fuel Design Report,” July 1, 2004. (ML041840220)

Framatome ANP to the USNRC, “BAW-10238(NP), Revision 1, MOX Fuel Design Report,” May 2003. (ML031550349)

Safety Evaluation for Topical Report BAW-10231P, “COPERNIC Fuel Rod Design Code” Chapter 13, MOX Applications, January 14, 2004. (ML040150701)

Safety Evaluation for Duke Topical Report DPC-NE-1005P, “Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,” August 20, 2004. (ML042370178)