LBP-04-32

RAS 9040

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED 12/22/04
SERVED 12/22/04

ATOMIC SAFETY AND LICENSING BOARD PANEL

Before Administrative Judges:

Ann Marshall Young, Chair
Anthony J. Baratta
Thomas S. Elleman

In the Matter of	Docket No's. 50-413-OLA, 50-414-OLA
DUKE ENERGY CORPORATION	ASLBP No. 03-815-03-OLA
(Catawba Nuclear Station, Units 1 and 2)	December 22, 2004

PARTIAL INITIAL DECISION
(Regarding BREDL Safety Contention I)

This proceeding involves Duke Energy Corporation’s (Duke’s) February 2003 application to amend the operating license for its Catawba Nuclear Station, located in York, South Carolina, to allow the use of four mixed oxide (MOX) fuel lead test assemblies at the station. After considering the parties’ evidence and argument on the one safety-related contention of Petitioner Blue Ridge Environmental Defense League (BREDL) admitted and remaining in this proceeding, we rule herein, based on the findings of fact and conclusions of law below, that

Duke has met its burden of persuasion with regard to all contested issues raised in this contention, showing by a preponderance of the evidence with regard to these issues that there is reasonable assurance that operation of Catawba with the four MOX assemblies will not endanger the health and safety of the public.

I. BACKGROUND and PROCEDURAL HISTORY

Duke's application is made as one part of a U.S.-Russian Federation nuclear nonproliferation program, in which it is proposed to dispose of surplus plutonium from nuclear weapons by converting it into MOX fuel (containing a mixture of plutonium and uranium oxides, with plutonium providing the primary fissile isotopes) to be used in nuclear reactors.[1] In its license amendment request (LAR) Duke seeks to modify certain technical specifications (TSs) to enable the use of four MOX fuel lead test assemblies in the Catawba plant, and also requests exemption from certain NRC regulations.[2] Duke is part of a consortium, Duke Cogema Stone and Webster (DCS), that has contracted with the Department of Energy (DOE) to perform various functions associated with this program.[3] Assuming approval of Duke's currently pending license application, DCS will, according to Duke, "provide for the design, construction, operation, and deactivation of a [MOX] Fuel Fabrication Facility (MFFF)," including agreements pursuant to which DCS "will process PuO_2 powder supplied by [DOE], blend it with depleted UO_2 powder, and fabricate it into MOX fuel pellets," which would then be loaded into

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[1] *See* 68 Fed. Reg. 44,107 (July 25, 2003); Letter from M.S. Tuckman, Executive Vice President, Duke Power, to NRC (Feb. 27, 2003), License Amendment Request [hereinafter LAR], Attachment 3 at 3-2 n.1.

[2] Duke's original license amendment request involved both the McGuire Nuclear Station, Units 1 and 2, and the Catawba Nuclear Station, Units 1 and 2. In September 2003 Duke revised the LAR to restrict the request to the Catawba facility. Letter from M.S. Tuckman to NRC (Sept. 23, 2003). *See* LBP-04-4, 59 NRC 129 (2004); LBP-04-10, 59 NRC 296 (2004), for more detailed information about Duke's application and the technical specification modifications and exemptions involved.

[3] LAR, Attachment 3 at 3-2.

MOX fuel assemblies.[4] Duke's ultimate plan, assuming all necessary approvals, is to use MOX fuel assemblies containing fuel manufactured by DCS in the McGuire and Catawba Nuclear Stations in "batch" quantities, with core fractions up to 40 percent MOX fuel.[5]

Petitioners Blue Ridge Environmental Defense League (BREDL) and Nuclear Information and Resource Service (NIRS) submitted petitions to intervene and requests for hearing regarding the current LAR in August 2003, in response to a July 2003 Federal Register publication of notice of opportunity for hearing; these were supplemented in October 2003 by contentions raising specific areas of dispute regarding the LAR.[6] After hearing oral argument on BREDL's safety-related and security-related contentions in December 2003 and March 2004, respectively,[7] the Licensing Board granted BREDL's request for hearing and, in Memoranda and Orders dated March 5 and April 12, 2004, admitted two safety-related contentions and one security-related contention.[8] After various prehearing activities including discovery among the parties, an evidentiary hearing was held July 14 and 15, 2004,[9] on the only safety contention then remaining in the proceeding, which we had admitted in the following form:

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[4] *Id.*

[5] *Id.*

[6] *See* 68 Fed. Reg. 44,107; [BREDL]'s Hearing Request and Petition to Intervene (Aug. 25, 2003); Nuclear Information & Resource Service's [NIRS] Request for Hearing and Petition to Intervene (Aug. 21, 2003); [BREDL's] Supplemental Petition to Intervene (Oct. 21, 2003) [hereinafter BREDL Contentions]; Contentions of [NIRS] (Oct. 21, 2003). *See also* [BREDL]'s Second Supplemental Petition to Intervene (Dec. 2, 2003); [BREDL]'s Contentions on Duke's Security Plan Submittal (Mar. 3, 2004).

[7] Tr. 71-576 (Dec. 3-4, 2004); Tr. 1263-1513 (Mar. 18, 2004) (Safeguards Information [SI]).

[8] LBP-04-4, 59 NRC 129 (2004); LBP-04-10, 59 NRC 296 (2004) (redacted public version of April 12, 2004, sealed Safeguards Memorandum and Order, issued May 28, 2004). None of NIRS' contentions were admitted. Of the three safety-related contentions admitted in LBP-04-4, the Board dismissed one in LBP-04-7, 59 NRC 259 (2004), and BREDL withdrew another, *see* Order (Regarding Proposed Redacted Memorandum & Order, and Proposed Schedule Changes) (May 25, 2004) (unpublished), leaving one for litigation, that at issue herein. *See also* CLI-04-19, 59 NRC **___???___** (2004).

[9] *See generally* Tr. 2072-2708.

> **Contention I**: The LAR is inadequate because Duke has failed to account for differences in MOX and LEU fuel behavior (both known differences and recent information on possible differences) and for the impact of such differences on LOCAs [loss-of-coolant accidents], and on the DBA [design-basis analysis] for Catawba.[10]

Subsequent to the hearing, the parties submitted proposed findings of fact and conclusions of law, and proposed reply findings.[11]

On the same date that the parties filed their reply proposed findings, Duke informed the NRC and the parties in this proceeding of certain information that has had the impact of delaying the issuance of this Memorandum and Order. This information concerned certain errors Duke had discovered, associated with certain calculated doses in a table in the Catawba Updated Final Safety Analysis Report (UFSAR), with regard to which Duke indicated that it was "working to provide . . . updated material by September 10, 2004."[12] On September 15, 2004, Duke counsel provided notification that the "new target date" for provision of this information was September 17, 2004.[13] On September 20, 2004, Duke submitted letters providing corrections to the LAR materials, and also committing to providing a summary of an independent review of the LAR that was done to give additional assurance that the LAR

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[10] LBP-04-4, 59 NRC at 167.

[11] [Duke]'s Proposed Findings of Fact and Conclusions of Law Regarding Contention I (Aug. 6, 2004) [hereinafter Duke Findings]; [BREDL]'s Proposed Findings of Fact and Conclusions of Law Regarding BREDL Contention I (Aug. 6, 2004) [hereinafter BREDL Findings]; NRC Staff's Proposed Findings of Fact and Conclusions of Law Concerning BREDL Contention I (Aug. 6, 2004) [hereinafter Staff Findings]; [Duke]'s Reply Findings of Fact and Conclusions of Law Regarding Contention I (Aug. 31, 2004) [hereinafter Duke Reply]; [BREDL]'s Proposed Reply Findings of Fact and Conclusions of Law Regarding BREDL Contention I (Aug. 31, 2004) [hereinafter BREDL Reply]; NRC Staff's Reply Findings of Fact and Conclusions of Law Concerning BREDL Contention I (Aug. 6, 2004) [hereinafter Staff Reply].

[12] Letter from David A. Repka to Administrative Judges (Aug. 31, 2004), and Attached Letter from W.R. McCollum to NRC Document Control Desk (Aug. 31, 2004) at 2.

[13] E-mail from Anne Cottingham to service list for proceeding (Sept. 15, 2004).

conclusions were accurate and adequately supported.[14] On October 6, 2004, Duke counsel notified the Licensing Board that Duke had on October 4 provided further information to the Staff regarding the independent review.[15]

During a closed session held September 1 to address certain matters related to the one security-related contention admitted by the Board, BREDL and Staff counsel were asked for their clients' responses to Duke's August 31 notification, with specific regard to any delay related to Contention 1 that might result from it. BREDL indicated that it would first need to see the information, and the Staff indicated that its review of the updated material would take "at least two weeks, and that is a minimum."[16] On October 4, 2004, the Staff indicated that it would respond to the materials provided by Duke within one month of that date.[17] On October 8 BREDL filed an e-mail statement that it would like an opportunity to review Duke's response to an expected Staff request for additional information (RAI) before commenting on the relevance of Duke's new information. On October 14, the Staff notified the Licensing Board by letter that it had concluded that the late-filed information provided by Duke had no impact on the Staff's testimony regarding Contention 1 and that the Staff's conclusions in its safety evaluation report (SER) on Duke's application and in Supplement 2 to the SER were unchanged with respect to fuel behavior and relevant LOCA analyses.[18] On October 25, during another closed session held to address various security-related matters, BREDL counsel indicated that it did not intend

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[14] Letter from David A. Repka to Administrative Judges (Sept. 20, 2004); Letter (with attachments) from Henry B. Barron to Document Control Desk (Sept. 20, 2004)

[15] Letter from David A. Repka to Administrative Judges (Oct. 6, 2004); Letter (with attachments) from Henry B. Barron to Document Control Desk (Oct. 4, 2004).

[16] Tr. 3080, 3082-83 (SI); *see id*. 3078-84 (SI).

[17] Letter from Susal L. Uttal to Administrative Judges (Oct. 4, 2004).

[18] Letter from Susal L. Uttal to Administrative Judges (Oct. 14, 2004).

to pursue anything further relating to Contention 1 in light of the Duke information, as BREDL did not consider the material relevant to Contention 1.[19]

Meanwhile, regarding the one security-related contention in the proceeding, admitted as Security Contention 5, both prior to and since issuance of LBP-04-10 the Licensing Board and parties have engaged, on a fairly intensive basis, in numerous activities involving various sensitive information, the relevance of particular pieces of such information, and access to such information. A number of closed sessions have been held to address issues related to such information, and the Licensing Board has issued a number of rulings on related discovery and other disputes, involving BREDL's access to and "need-to-know" regarding various sensitive information.[20] Some of these rulings have followed initial need-to-know determinations by the Staff and Duke, regarding documents held by each, and some Board rulings have been appealed to the Commission, leading to the issuance of several Commission Memoranda and

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[19] E-mail from Diane Curran to Administrative Judges (Oct. 8, 2004); Tr. 3575-77 (SI).

[20] *See*, *e.g.*, Memorandum and Order (Protective Order Governing Duke Energy Corporation's September 15, 2003 Security Plan Submittal) (Dec. 15, 2003); Memorandum (Providing Notice of Granting BREDL Motion for Need to Know Determination and Extension of Deadline for Filing Security-Related Contentions) (Jan 29, 2004); Memorandum and Order (Ruling on BREDL Motion Regarding Staff February 6, 2004, Meeting with Duke Energy and Request for Need to Know Determination) (Feb. 4, 2004); Memorandum and Order (Ruling on BREDL Motion for Need to Know Determination Regarding Classified Documents) (Feb. 17, 2004); Memorandum and Order (Setting Schedule for Discovery and Hearing on Security-Related Matters) (April 28, 2004); Order (Ruling on [Duke] Objection to BREDL Document Production Request No. 2 Regarding BREDL Security Contention) (June 28, 2004); LBP-04-13, 60 NRC ___ (2004); Memorandum and Order Suspending Discovery Proceedings Pending Further Commission Guidance (July 28, 2004); Memorandum and Order (Confirming August 10, 2004, Bench Ruling Finding Need to Know and Order Provision of Documents Sought by Intervenor in Discovery) (Aug. 13, 2004); LBP-04-21, 60 NRC 357 (2004); Memorandum and Order (Ruling on Objections of Duke and Staff to BREDL Discovery Requests (Oct. 6, 2004); Memorandum and Order (Ruling on Redactions to Documents 67 and 68) (Oct 6, 2004); Memorandum and Order (Confirming Sept. 28, 2004, Bench Ruling Upholding Staff Need-to-Know Determination on Access to Security Plan Revision) (Oct. 15, 2004); Memorandum and Order (Confirming Matters Addressed and Ruled on at Oct. 25, 2004, Closed Session) (Nov. 5, 2004); Memorandum and Order (Ruling on BREDL Access to NRC Guidance Document) (Nov. 5, 2004); Memorandum and Order (Ruling on BREDL Need-to-Know Appeal Regarding Lessons Learned Report) (Nov. 22, 2004); Memorandum and Order (Granting in Part Motion for Interim Discovery Measures) (Nov. 23, 2004); Memorandum and Order (Confirming Actions Taken at November 23, 2004, Closed Session); Memorandum and Order (Ruling on BREDL Motion to Amend Protective Order) (Dec. 17, 2004); Memorandum and Order (Need-to-Know Ruling on SECY Document) (Dec. 17, 2004).

Orders.[21] Most recently, on December 17 and 20, 2004, the parties filed their prefiled direct testimony, along with various exhibits, for the evidentiary hearing on BREDL Security Contention 5. This hearing is scheduled to be held January 10-14, 2005.

II. RULINGS ON PENDING MATTERS

During the July 2004 evidentiary hearing Duke and NRC Staff counsel objected to BREDL's proposed Exhibit C, "Status of NSC Activities in the Field of Fuel Behaviour," Nuclear Energy Agency/Nuclear Science Committee, NEA/NSC/DOC(2003)12 (May 2003), on the grounds, respectively, that the document had not been timely disclosed and was beyond the scope of the proceeding.[22] The Board heard BREDL's testimony relating to this and one other document submitted by BREDL that had not previously been disclosed, taking under advisement objections to the documents as well as all testimony on them. The Staff subsequently objected in writing, based on the lateness of the submission, extraneous material in the exhibit, and the cumulative nature of the information in the document.[23] Duke submitted rebuttal testimony, per the Board's request, stating, *inter alia*, that Exhibit C "does not provide evidence of a difference in fuel pellet-cladding chemical interaction between MOX and LEU fuel," and that the exhibit "does not provide any evidence that, if such a difference actually existed, [ ] it would matter under LOCA conditions."[24]

BREDL's expert, Dr. Edwin Lyman, explained at the hearing that the document was submitted because of discussion therein regarding properties of pellet-clad bonding with MOX

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[21] *See* CLI-04-06, 59 NRC 62 (2004); CLI-04-19, 60 NRC 5 (2004); CLI-04-21, 60 NRC 21 (2004); CLI-04-29, 60 NRC 417 (2004); CLI-04-37, 60 NRC ___ (2004).

[22] Tr. 2541.

[23] Letter from Susan L. Uttal to the Atomic Safety and Licensing Board (July 20, 2004); Staff Findings at 4.

[24] Supplemental Rebuttal Testimony of Steven P. Nesbit and J. Kevin McCoy on Behalf of Duke Energy Corporation on Contention I, July 20, 2004.

fuel that may relate to fuel swelling and pellet expansion in transients.[25] Dr. Lyman stated that even though the information in the document related to a different kind of accident than a LOCA, he hypothesized that the differences with MOX fuel "could have an impact on the difference between MOX and LEU pellet clad interaction and the behavior in LOCAs," because MOX and LEU fuel "may have a pellet clad bond of a different nature."[26]

Given the hypothetical nature of Dr. Lyman's statements, we find they do not change our decision herein, and thus the impact of the evidence we heard and took under advisement is small. We have, however, considered the evidence and it is included in the record, in order to assure a more complete record on the matters of concern herein.

Duke has also filed some proposed corrections to the transcript of the July 14-15, 2004, hearing.[27] No objection having been posed to these proposed corrections, the transcript may be amended to reflect them.[28]

III. GOVERNING LEGAL STANDARDS

The legal standards that apply in this proceeding are found in various NRC regulations. First, under 10 C.F.R. § 50.90, whenever a holder of a license wishes to amend the license, including technical specifications in the license, an application for amendment must be filed, fully describing the changes desired. Under § 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate. Both the common standards for licenses and construction permits at § 50.40(a), and those specifically

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[25] Tr. 2553.

[26] Tr. 2554.

[27] [Duke]'s Proposed Corrections to the July 14-15, 2004 Hearing Transcript in the [MOX] Lead Assembly License Amendment Proceeding (Aug. 6, 2004).

[28] Duke counsel may wish to file a copy of the transcript with the corrections marked thereon, for the official record in the proceeding.

for issuance of operating licenses at § 50.57(a)(3), provide that there must be "reasonable assurance" that the activities at issue will not endanger the health and safety of the public.

In addition, 10 C.F.R. § 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," as its title indicates, defines the requirements that light-water reactors must meet with regard to their emergency core cooling systems (ECCSs); it relates specifically to boiling or pressurized light-water reactors that use low-enriched uranium (LEU) fuel consisting of "uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding." Duke seeks to utilize the same criteria of § 50.46 with regard to the proposed MOX fuel.[29]

Under § 50.46(a)(1)(i), an ECCS must be designed so that its "calculated cooling performance following postulated loss-of-coolant accidents" [LOCA] meets certain criteria. The criteria that are of particular relevance herein are found at §§ 50.46(b)(1), (2), and (4), and require that:

- The calculated maximum fuel element cladding temperature, or "peak cladding temperature" (PCT), shall not exceed 2200^{E}F;
- The calculated total oxidation of the cladding "shall nowhere exceed [17% of] the total cladding thickness before oxidation"; and
- Calculated changes in core geometry "shall be such that the core remains amenable to cooling."

These criteria essentially set limits on the extent of fuel damage that can occur during a design basis LOCA. The 2200EF limit on peak cladding temperature and the 17% limit on maximum local oxidation together assure that the cladding will not become embrittled and will not lose its rod-like geometry during and after a LOCA.

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[29] LAR, Attachment 6 at 6-8 – 6-12; *see also* 6-1 – 6-12, generally.

Section 50.46(a)(1) also provides as follows:

> (i) . . . . ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model. . . .
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> (ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models.[30]

BREDL has agreed that it is “generally appropriate to apply the requirements of [ ] § 50.46 to MOX fuel, as long as Appendix K [which does not include consideration of fuel relocation] is not strictly applied to exclude consideration of relocation of the fuel during LOCAs.”[31]

IV. FINDINGS OF FACT

A. General Information Relating to Matters at Issue

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[30] Required features under Appendix K include assuming the heat generation rates from radioactive decay of fission products to be equal to 1.2 times the values for infinite operating time in the ANS Standard, “Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors,” (approved by Subcommittee ANS-5, ANS Standards Committee, October 1971), *see* 10 C.F.R. Part 50, Appendix K, § I.A.4; calculation of the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction, using the Baker-Just equation (Baker, L., Just, L.C., “Studies of Metal Water Reactions at High Temperatures, III, Experimental and Theoretical Studies of the Zirconium-Water Reaction,” ANL-6548, p. 7, May 1962), *see* 10 C.F.R. Part 50, Appendix K, § I.A.5; and inclusion of a provision for predicting cladding swelling and rupture in each evaluation model — all of which, we note, provide for conservatism in an Appendix K analysis, *see* 10 C.F.R. Part 50, Appendix K, § I.B.

[31] Tr. 2257.

The MOX fuel assemblies at issue, which are currently being manufactured in France under the direction of AREVA,[32] will be based on the AREVA Advanced Mark-BW fuel assembly, a standard-lattice 17-by-17 fuel assembly specifically designed for use in Westinghouse reactors such as Catawba.[33] Duke's plans call for the assemblies, if ultimately approved, to be irradiated for a minimum of two cycles, in order to test the acceptability of the fuel assembly design, the ability of the Duke and AREVA models to predict fuel assembly performance, and the applicability of the European database on MOX fuel performance to Duke's use of MOX fuel.[34] As indicated above, the current proposal would, if successful, support the potential future use of larger, "batch" quantities of MOX fuel at either the Catawba or McGuire plant, which would require another license amendment application and associated licensing proceeding.[35]

In addition to the four proposed MOX fuel assemblies at issue herein, the reactor core for Catawba would, if Duke's LAR is approved, contain 189 other fuel assemblies, the majority of which would be Westinghouse Robust Fuel Assemblies (RFAs), and eight of which would be Westinghouse Next Generation Fuel (NGF) fuel assemblies. The NGF assemblies would be loaded into core locations that are not adjacent to the MOX fuel.[36]

The matters at issue herein concern the ability to keep the fuel in a reactor cooled and intact in a hypothetical "loss of coolant accident," or "LOCA." The Catawba reactor is a pressurized water reactor, or "PWR," in which water is circulated, under pressure, through the

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[32] AREVA is the trade name of the Société des Participations du Commissariat à l'Énergie Atomique, an organization consisting of several businesses including Framatome Advanced Nuclear Power (ANP), Siemens, Cogema, and AREVA T&D. AREVA Website at www.areva.com.

[33] LAR at Attachment 3, § 3.5.

[34] Tr. 2112.

[35] Tr. 2111.

[36] Tr. 2293; 2112.

reactor core, taking heat from the fuel and keeping it cool enough to avoid fuel damage.[37] Licensees are required to have emergency core cooling systems (ECCSs) that can function in the event of water loss from the primary cooling system, and must perform certain analyses to show that their cooling systems can withstand designated hypothetical LOCAs.

Under 10 C.F.R. § 50.46, ECCSs must meet certain requirements, as indicated above. ECCS performance and its ability to cool the fuel must be calculated by analyzing a number of postulated LOCAs with various characteristics sufficient "to provide assurance that the most severe postulated [LOCAs] are calculated."[38] The results of any LOCA analysis must show that a reactor can meet particular criteria, including a maximum, or "peak," cladding temperature that does not exceed 2200E Fahrenheit; a total cladding oxidation in any location that does not exceed 17 percent of the total cladding thickness before oxidation; and changes in core geometry that are sufficiently limited to assure the core remains amenable to cooling.[39] The parties are in dispute over the impact of the use of the four proposed MOX fuel assemblies on Duke's LOCA analyses, and whether in light of such use Duke can satisfy the above-listed PCT, clad oxidation, and coolable core geometry requirements of 10 C.F.R. § 50.46, all of which are fundamental to providing reasonable assurance that such use will not endanger the health and safety of the public.

One postulated LOCA — a "large break" LOCA — assumes the 100 percent break, at full power, of the primary coolant line that carries the water from the reactor to the steam generator and back to the reactor. Such a break would result in escape of the water — i.e., a

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[37] This pressurized circulating water transfers the heat to a secondary water system (the water in which is contained a separate piping system from the primary system) by passing through a device called a steam generator. Steam produced in the secondary system operates steam turbines to produce power.

[38] 10 C.F.R. § 50.46(a)(1)(i).

[39] *Id.* §§ 50.46(b)(1), (2), (4).

loss of coolant. The reactor protective system would automatically shut down the reactor and lead to a rapid drop in the power being produced. At a certain point in time the forced flow through the core will be so reduced that water will begin to boil away in the core region, reducing cooling capability so that it is no longer capable of removing the decay heat energy released from radioactive byproducts of the fission process. This leads to a rise in the temperature of the fuel above its normal steady-state level, accompanied by a rise in the fuel cladding temperature. During this heat-up, internal pressure in the fuel rods will rise as gaseous fission products build up in the spaces between the fuel pellets and the cladding, and the cladding may weaken, could balloon in places, and may ultimately burst. As the temperature of the cladding rises it will also begin to oxidize rapidly. At some point in such a postulated accident, cold water would be injected into the core by the emergency core cooling system (ECCS) and the fuel cladding would cool down.[40] The results of the required LOCA analysis define the "LOCA limits," or the allowable core power peaking limits,[41] that are the constraints for meeting the requirements of § 50.46.[42]

Two related phenomena that can occur late in a LOCA scenario play central roles in analysis of the matters at issue herein — *clad ballooning* and *fuel relocation*. We summarize here some basic facts about these two phenomena. With regard to the first of these, during a LOCA, as the fuel and cladding heat up, fission product gases collect within fuel pellets and cause the pellets to expand. At first this has the effect of removing the normal gap between fuel and clad. Then, at higher temperatures, the clad begins to distort and expand more than the fuel, as gases move through the fuel matrix and along grain boundaries to produce a gas layer at the fuel-clad interface. As this occurs, gaps can form between pellet and the cladding,

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[40] Tr. 2294.

[41] The core power peaking limits define the maximum power at any one location in the core.

[42] Tr. 2119.

and the cladding expansion produces an increase in surface area as well as in pellet-to-clad gap. Stresses on the cladding can concentrate in a particular region as a result of temperature increase and fuel expansion, and a “balloon” can form in the cladding due to increasing gas pressure. If the LOCA progresses unimpeded (prior to ECCS injection beginning to become effective in reestablishing cooling), the cladding would continue to balloon and eventually rupture.

Meanwhile, the increase in the pellet-to-clad gap reduces the heat conducted from the pellets to the cladding, while the greater surface area of the cladding increases heat transfer from the cladding to the coolant. Also, flow turbulence develops around the area of cladding expansion and rupture, thereby generally increasing heat transfer to the coolant from the cladding and fuel. The net effect on the cladding temperature of (1) the reduction in heat transfer from pellet to cladding, and (2) the increase in heat transfer from the cladding to the coolant, can be short-term cooling of the cladding at the location of ballooning and rupture.[43] Because of this cooling effect, the ballooned or ruptured location is generally not the location of the peak clad temperature.[44]

With regard to the second phenomenon, however, BREDL asserts that *fuel relocation* may negate this cooling effect, and have an impact on PCT. Fuel relocation may occur during a LOCA when irradiation-induced cracks develop in fuel pellets, causing the pellets to lose their integrity and break into small fragments, which fall to lower portions of fuel rods where the cladding has swelled and ballooned. In order for this to occur, a pellet must have been irradiated to a sufficient level of burnup (BU)[45] for cracks and the potential for fragmentation to

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[43] Tr. 2123.

[44] Tr. 2305-08.

[45] “Burnup” is a measure of the amount of energy provided by the fuel at a given point in time, and is usually expressed as total energy produced per metric tonne of material, or GWD/t (gigawatt days (continued...)

develop, and the cladding must balloon sufficiently prior to rupture to provide room for the pellet fragments to collect.[46] Relevant with regard to Contention 1 is the concern that relocated fuel may generate too much power in a localized area and thereby increase the cladding temperature at that location.[47]

The effect of relocation will depend in part on the size of the cladding balloon and the "filling fraction" — i.e., the percentage of the expansion space filled by relocated fuel. Duke contends that the above-described cooling effect at the ballooning area of the cladding will mitigate any effects of fuel relocation so as to produce a "reasonable bound of uncertainty well within the conservatisms . . . under Appendix K."[48]

Duke has had LOCA analyses performed for Catawba using both of the alternative methods permitted under 10 C.F.R. § 50.46 — i.e., both a "best estimate" method under § 50.46(a)(1)(i), and an Appendix K method as permitted under § 50.46(a)(1)(ii). Westinghouse performed the § 50.46(a)(1)(I) best-estimate LOCA analysis for the Catawba core *without* the MOX assemblies, using its own WCOBRA/TRAC code.[49] Best-estimate methods are newer than those allowed under Appendix K, and are considered to be more advanced and state-of-the-art; they compute lower peak clad temperatures assuming the same input, which indicates a greater margin between the calculation results and the 2200E limit of § 50.46(b)(1).[50] Use of a best-estimate method requires a nominal calculation that considers an accounting of all

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[45](...continued)
per metric tonne).

[46] Tr. 2147.

[47] Tr. 2149.

[48] Tr. 2400; 2149.

[49] Tr. 2296. WCOBRA/TRAC refers to the Westinghouse "transient reactor analysis code" that is approved by the NRC for use with pressurized water reactors.

[50] *See* Tr. 2376.

variations and uncertainties. Under the Westinghouse approach, a large number of calculations are performed, and then a "Monte Carlo"[51] simulation is done to determine uncertainties.[52]

Framatome ANP also performed a LOCA analysis, assuming utilization of the MOX lead assemblies. In its analysis Framatome used its NRC-approved Appendix K deterministic evaluation model [EM][53] and its approved computer code, RELAP5/MOD2-B&W.[54] This method is more conservative than the best-estimate method in the sense of generally computing temperatures that are higher than should reasonably be expected to actually occur.[55] To address the fact that EMs permitted under Appendix K apply only to light-water reactors that use uranium oxide, or LEU, fuels, Duke in its LAR describes potential differences between LEU and MOX fuel that could affect the Appendix K analysis for Catawba, related changes made to the evaluation techniques and model, and its justification for the changes.[56]

According to Duke, the LEU-based EM did not require a great degree of adaptation in order to predict the LOCA behavior for MOX fuel, because, of the 13 categories of phenomena that have an impact on LOCA results listed in NUREG/CR-5249,[57] twelve are specifically related

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[51] The "Monte Carlo" simulation is so called because it is based on performing a large number of individual histories for a given technical process, and then using a random sampling of calculations to obtain both an average result and its uncertainty.

[52] Tr. 2377.

[53] According to Duke expert Bert M. Dunn, the NRC approved the EM for application to Westinghouse-designed four-loop PWRs (such as Catawba) that use LEU fuel. Tr. 2124.

[54] Tr. 2296-97.

[55] Tr. 2374, 2376. We note also in this regard BREDL's reference to several "known non-conservatisms" in Appendix K, including the effects of fuel relocation, particularly when applied to MOX fuel. Tr. 2258.

[56] LAR at Attachment 3, § 3.7.

[57] NUREG/CR-5249, EGG-2552, "Quantifying Reactor Safety Margins — Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident" (Dec. 1989).

to the reactor system and are independent of the fuel type. Only one category — Stored Energy and Fuel Response — includes phenomena that relate to the nuclear material. Four such phenomena are listed as significant contributors: fuel pellet enthalpy (heat content of the fuel) at operating conditions, fuel decay heat, gap conductance (potential for heat conduction across the gas-filled gap between the fuel and its cladding), and cladding oxidation. Only the first three of these relate to the nuclear material. Thus, Duke asserts, most of the approved EM was already appropriate to the modeling of MOX fuel with no adjustments.[58]

Framatome reviewed MOX fuel characteristics that have the potential to affect the results of an Appendix K LOCA calculation, concentrating on decay heat and reactor kinetics, and thermal and mechanical properties, including MOX pellet enthalpy.[59] In its LAR Duke documents these phenomena and how it addresses each one.[60] For the Catawba core with the MOX lead assemblies, Duke states, the reactor kinetics would be dominated by the LEU fuel because the MOX assemblies would comprise only 2 percent of the core. Those differences that would exist — lower beta effective value and a more negative moderator temperature coefficient in MOX fuel — would act to *reduce* the power generation in the MOX fuel relative to the surrounding LEU fuel.[61] Thus, in Duke's view, a MOX/LEU decay heat comparison could indicate a beneficial result for MOX fuel, in that for the first several thousand seconds (well

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[58] Tr. 2124.

[59] Tr. 2125.

[60] LAR at Attachment 3, § 3.7.1.1.

[61] Tr. 2125. The term "Beta effective" (or "β_{eff}") refers to the fraction of the neutrons that are produced through the decay of fission products some time after the fission event has occurred. The word "effective" implies that the value of beta has been adjusted to reflect the fact that these neutrons are produced at lower energies than those in fission and are therefore more "effective" in producing subsequent fissions. "Moderator temperature coefficient" refers to the change in moderation caused by a change in the temperature of the coolant/moderator. The term "moderation" refers to the process by which the water in a reactor's cooling system slows down the neutrons that are emitted from the fissioning atoms in the fuel, which increases the likelihood of fission of the nuclei, thus sustaining the nuclear chain reaction in the fuel that produces a controlled level of heat, and ultimately power.

beyond the time of PCT following a LOCA), the decay heat for a MOX fuel assembly operated at the same power as an LEU assembly would be lower than the corresponding LEU fuel assembly. For both beta effective value and moderator temperature coefficient, however, the MOX assemblies were conservatively evaluated, according to Duke, using neutron power-related characteristics and decay heat characteristics appropriate for LEU assemblies.[62]

With regard to pellet thermal properties and the fuel-to-cladding heat transfer coefficient, or gap conductance, only thermal conductivity differs in any substantive way when comparing LEU and MOX pellets of the design planned for Catawba.[63] Because the initial enthalpy of a MOX pellet is slightly greater, Framatome used the MOX thermal conductivity from the COPERNIC computer code,[64] which supplies the stored energy and thermal conductivity values in the LOCA analysis code,[65] and has been approved for MOX applications.[66] Also, because MOX fuel has higher fission gas release rates and thus produces a slightly different gas composition and pressure, which can affect gap conductance, the Framatome LOCA analysis was based on MOX-specific gap gas compositions obtained from the COPERNIC computer code.[67]

Finally, also of note with regard to Duke's LOCA analysis, Duke points out that, because Appendix K requires that the degree of cladding swelling and incidence of rupture not be underestimated, and "[b]ecause all claddings tend to (i) embrittle with irradiation, and

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[62] Tr. 2125.

[63] Tr. 2126.

[64] The COPERNIC computer code is a fuel performance code used to analyze individual fuel rods. BAW-10231P, COPERNIC Fuel Rod Design Computer Code (September 1999). The computer code is used for mechanical analyses of MOX and LEU fuel. Tr. 2133.

[65] Tr. 2300.

[66] Tr. 2126.

[67] *Id.*

(ii) potentially accrue added strength [with irradiation] due to pellet-cladding bonding," it incorporated unirradiated cladding properties to maximize the predicted strain in its Appendix K LOCA analysis.[68] (This is a generally used approach in deterministic LOCA evaluation models, employed to maximize strain in the required calculations.)[69]

B. MOX Fuel Characteristics and Fuel Relocation

Contention I was originally derived from BREDL proposed Contention 10,[70] which was based on a presentation made to the NRC on October 23, 2003, by the Institute de Radioprotection et de Sûreté Nucleaire (IRSN).[71] The October 2003 IRSN presentation was offered as the basis for the assertion that fuel relocation may introduce a significant uncertainty with respect to the LOCA analysis for the MOX fuel lead assemblies.[72]

BREDL asserts through the testimony of its expert, Dr. Lyman, that Duke's LOCA analysis is inadequate because it does not address the uncertainties associated with fuel relocation effects that the MOX fuel, which will be manufactured using "M5" cladding rather than the Zircaloy-4 cladding more often used in the U.S., may experience under LOCA conditions. BREDL suggests these uncertainties counter Duke's assertion that the action proposed in the LAR will not result in a violation of the PCT, clad oxidation, and coolable core geometry criteria of 10 C.F.R. § 50.46.[73] Noting that fuel relocation has been observed in experiments with irradiated LEU fuel under LOCA conditions, Dr. Lyman points out that no such experiments

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[68] Tr. 2128.

[69] *Id.* The fuel-clad bonding issue is discussed in greater detail below in section IV.B(2), which includes a discussion of the testimony of BREDL expert Dr. Edwin Lyman.

[70] *See* [BREDL]'s Second Supplemental Petition to Intervene (Dec. 2, 2003), at 2-4 [hereinafter BREDL Second Supplemental Petition].

[71] Exhibit 28. IRSN is an agency in the French government that conducts nuclear research, formerly known as IPSN, the Institut de Protection et de Sûreté Nucléaire. *See* Tr. 2245-46.

[72] Exhibit 28; *see* BREDL Second Supplemental Petition at 3-4.

[73] Tr. 2256.

have been done with MOX fuel, but that “there are technical reasons to believe that the impact of fuel relocation effects during a LOCA may be more severe for MOX fuel rods than for LEU fuel rods of the same burnup, due to differences in characteristics such as fuel fragment sizes and fuel-clad interactions.”[74] Noting also calculations in Duke’s LAR indicating that MOX fuel is generally more limiting than LEU fuel with respect to design basis LOCAs, BREDL asserts that “the consequences of fuel relocation, and the non-conservatism associated with neglecting them, may be of greater concern for MOX fuel rods than for LEU fuel rods with respect to compliance with LOCA regulatory criteria.”[75]

Because Duke has failed to address these uncertainties in MOX fuel behavior, BREDL asserts, Duke’s LAR does not satisfy relevant § 50.46 requirements, and therefore Duke has failed to demonstrate compliance with the general reasonable assurance standard of § 50.40(a).[76] BREDL cites in support of its assertions information from various French sources, based on or related to use of MOX fuel in nuclear reactors in France.[77] BREDL points out that the authors of one of these sources state that “[a] lack of knowledge on theses [sic] parameters

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[74] Tr. 2243.

[75] *Id.*; BREDL Findings at 5.

[76] *Id.*

[77] Sources cited by BREDL include C. Grandjean, G. Hache and C. Rongier, “High Burnup UO_2 Fuel LOCA Calculations to Evaluate the Possible Impact of Fuel Relocation After Burst,” OECD/NEA Proceedings of the Topical Meeting on LOCA Fuel Safety Criteria, Aix-en-Provence at 7 (Mar. 22-23, 2001) [hereinafter Grandjean, Hache and Rogier] (Exhibit 29); Slides presented by A. Mailliat and J.C. Mélis, IRSN, at “PHEBUS STLOC Meeting” with NRC Staff (Oct. 23, 2003) (Exhibit 28); Grandjean and Hache “LOCA Issues Related to Ballooning, Fuel Relocation, Flow Blockage and Coolability” [hereinafter Grandjean, *LOCA Issues*], SEGFSM Topical Meeting on LOCA Issues, Argonne National Laboratory, slides at 8-9 (May 25-26, 2004) [hereinafter SEGFSM Topical Meeting, May 2004] (Exhibit 36); and V. Guillard, C. Grandjean, S. Bourdon and P. Chatelard, “Use of CATHARE2 Reactor Calculations to Anticipate Research Needs,” SEGFSM Topical Meeting, May 2004 (Exhibit 30). *See* BREDL Findings at 6.

[important for relocation] for irradiated UO2 and particularly MOX fuel may lead to reduce [sic] safety margins."[78]

Areas in which such fuel-relocation-related uncertainties are asserted to lie, on the significance of which the parties are in dispute, include (1) a collection of issues related to fuel composition, fragmentation, particle size, and "filling ratio"; (2) fuel-clad bonding; and (3) the effects of M5 clad characteristics on ballooning and fuel relocation. We discuss each of these areas next. In Sections IV.C, D, and E, we turn to the impact of fuel relocation on PCT, clad oxidation, and coolable core geometry. We next, in section IV.F, consider questions raised by BREDL on the adequacy of the current datebase on MOX fuel in LOCAs. We close, in section V, with our ultimate conclusions in the safety-related portion of this proceeding.

(1) Fuel Composition, Fragmentation, Particle Size, and "Filling Ratio"

The fuel relocation phenomenon, a generic issue that is not unique to MOX fuel,[79] has been observed in LEU fuel for rod burnups exceeding around 48 gigawatt days per metric tonne (GWD/t), which are considered to be "high" burnup levels.[80] The related phenomenom of *fuel fragmentation* has been observed by IRSN to be "clearly associated to [sic] burnup, with finer fragments at higher BU."[81] In LEU fuel, high-burnup "rim" regions are known to emerge with

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[78] *See* Exhibit 30 at Abstract.

[79] The possible impact of fuel relocation on LOCA analyses for LEU fuel was recognized by the NRC Staff in Generic Issue ("GI") 92, which was initially assigned a low priority and was subsequently dropped. More recently, the issue was acknowledged in a memorandum from Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, to Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, re: Information Letter 0202, Revision of 10 CFR 50.46 and Appendix K (June 20, 2002) (NRC ACN # ML 021720690) (Exhibit 27). *See* Tr. 2257.

[80] See Grandjean, Hache and Rondier at 2.

[81] *Id.*; Tr. 2247.

burnups exceeding about 40-45 GWD/t. Thus it has been suggested that vulnerability to fuel relocation in LEU fuel is associated with the development of such high-burnup "rim" regions.[82]

For MOX fuel, the situation is somewhat similar and possibly more severe. During manufacture of MOX fuel using the MIMAS[83] process to be used for the Duke LTAs, plutonium agglomerates — macroscopic clumps of plutonium-rich particles — develop in the fuel.[84] As pointed out by BREDL's expert, Dr. Lyman, because the fissile material is concentrated in these clumps, very high local burnups would result, due to the fact that the fission is occurring in a heterogeneous fashion. The ratio of local burnup within the agglomerates could be several times the rod-average burnup, depending on the irradiation time.[85] For instance, the agglomerate burn-up reaches about 60 GWD/t when the rod average is only around 18 GWD/t, and reaches 100 GWD/t when the rod average is only 28.4 GWD/t. As a result, BREDL asserts, high-burnup rim-like regions may emerge in the outer layers of the plutonium agglomerates for much lower rod-average burnups than 40-45 GWD/t, because the local burnups within the plutonium agglomerates increase much more rapidly than the rod-average burnups. Thus, BREDL argues, it is reasonable to expect that the onset of fuel relocation in MOX fuel may occur at lower rod-average burnups than in LEU fuel. This would imply that MOX fuel would be more vulnerable earlier in its irradiation history (and consequently for a longer time) than LEU fuel.

BREDL further asserts that the particle size distribution in fragmented MOX fuel will be smaller than in LEU fuel at the same rod average burnup, because of fine fragments that are

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[82] Tr. 2246.

[83] MIMAS stands for MIcronized MASter mix, an industrial process used by COGEMA to manufacture MOX fuel. COGEMA Home Page at www.cogema-inc.com/FAQs/MOX_fuel.htm.

[84] Tr. 2260.

[85] BREDL Findings at 13.

generated when micro-cracking occurs in the ultra-high burnup plutonium agglomerate regions as a result of fission gas accumulation and migration.[86] The smaller particle size in turn leads to greater "filling ratios," which refers to the ratio of the volume of relocated fuel material to the volume of a given ballooned region of a fuel rod in which the material has collected.[87] BREDL contends that greater filling ratios can lead to greater PCT, which we address in more detail below.

NRC Staff expert Dr. Ralph Meyer acknowledged that the particles produced by MOX fracture might be slightly smaller than particles from LEU as a result of the slightly higher fission gas release for MOX to the rim material. According to Dr. Meyer:

> This rim material, which forms in high-burnup regions around the circumference of LEU fuel and also around the agglomerates in MOX fuel, is the result of fission gas migration with the uranium-plutonium oxide crystalline grains. Fission gas migrates, coalesces, and precipitates in small bubbles, which attach themselves to the grain boundaries. As the number of bubbles increases with burnup, the grain boundaries subdivide to form more surface area to accommodate the bubbles, thus producing the smaller grained rim material. Because fission gas release, which is also related to the migration process, is a little higher in MOX fuel than in LEU fuel, the volume of rim material might be roughly 25% greater in MOX fuel than in LEU fuel. On the other hand, MOX fuel has a little more plasticity than LEU fuel, so I would expect fewer of the larger fragments in MOX Fuel.
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> . . . .
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> In recent high-burnup integral tests in our program at ANL, we have observed a black deposit on the quartz tube of the apparatus just opposite the burst opening. . . . It thus appears that the small particles or fines are blown out of the burst opening when the rod depressurizes. Thus, there would be few or no small particles in the ballooned region, and it is these small particles that have been postulated to make a difference between the mass of fuel in the balloon in MOX fuel and LEU fuel.[88]

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[86] Tr. 2234, 2260; *see* BREDL Findings at 14.

[87] Tr. 2258.

[88] Tr. 2304-05.

BREDL disagrees, pointing out that Dr. Meyer failed to take into account the fact that the tests in question were performed on BWR fuel rods and not PWR rods,[89] and, moreover, asserts that fuel fragmentation can also be caused by the stress induced by the stored-energy redistribution during the initial, “blowdown” phase of a LOCA.[90] Because MOX fuel has a lower thermal conductivity — i.e., lower ability to transfer heat from the center to the surface of a pellet — and therefore a higher temperature in the center of a pellet than LEU fuel, it could experience greater fuel fragmentation during the blowdown phase of a LOCA and more severe relocation effects as a result.[91]

NRC Staff experts who participated in a 2001 PIRT (phenomenon identification and ranking table) panel disagreed on the impact of fuel composition on fuel relocation, with one finding it to be of low importance, one finding it to be of medium importance (finding a greater impact with MOX fuel but also finding that the viscoelastic properties of MOX would mitigate the effect), and two finding the composition of MOX fuel to be of high importance for consideration of fuel relocation effects.[92] BREDL suggests this disagreement “highlights the inadequacies of

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[89] Tr. 2270; *see* BREDL Findings at 15.

[90] *Id.*; Exhibit 31, A. Mailliat and M. Schwarz, "Need for Experimental Programmes on LOCA Issues Using High Burn-Up and MOX Fuels," NUREG/CP-0176, Proceedings of the Nuclear Safety Research Conference at 436 (May 2002) (NRC ACN # ML021710793) [hereinafter "Mailliat and Schwarz"]. A large-break LOCA evolves through three phases, which overlap, although AREVA in its Appendix K analysis treated the phases as being distinct in order to make the analysis more conservative. The first phase, blowdown, initiates with the opening of a break in the coolant system and lasts until the system has depressurized to an approximate equilibrium in pressure with the reactor building. This phase ends, in an Appendix K analysis, with the coolant system nearly empty and the fuel and cladding beginning to increase in temperature. The second phase, refill, is defined as the time required to inject sufficient emergency coolant to fill the lower head and lower plenum of the reactor vessel and re-initiate flow of coolant into the reactor core. When the coolant in the reactor vessel rises to the bottom of the core, the third phase, reflood, initiates and continues for the remainder of an accident simulation. Tr. 2119-22.

[91] Tr. 2260.

[92] Tr. 2260; Exhibit 32, NUREG/CR-6744, “Phenomenon Identification and Ranking Tables for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High-Burnup Fuel,” Appendix D, Table D-1 at D-67 (Dec. 2001) (NRC ACN # 013540623) [hereinafter NUREG/CR-6477).

the experimental database with regard to integral tests of MOX fuel under design-basis LOCA conditions, and underscores the significant uncertainties in Duke's design-basis LOCA analysis."[93]

Duke expert Robert Harvey pointed out that members of the same PIRT panel "actually looked at a number of aspects of high burnup fuel with respect to a LOCA event," and that "under the category of 'plant transient analysis' none of the PIRT panel rated *fuel relocation* as important, two rated it as 'low medium importance,' and four rated it as low importance, stating that it has a small effect on the system analysis and it could make the burst (ruptured) node limiting."[94]

Based on the preceding, we find that fuel fragmentation and relocation can have an impact on PCT during a LOCA, and there is some uncertainty regarding the magnitude of such impact. We discuss *infra* in greater detail the extent of such impact, and whether in light of it Duke has performed adequate LOCA analysis relative to the applied-for LAR.

In addition, the particle size of fuel fragments entering the ballooned region during a LOCA is potentially important, as the energy generation rate in the ballooned volume increases as the mass of contained fuel increases, and this in turn potentially increases the surface temperature of the adjacent cladding. The calculation of surface temperature increase is a complex one, however, given that the heat dissipation rate also increases with increased clad surface area and thus can mitigate any temperature increase. We discuss these issues further in Section IV.C, below, "Impact of Fuel Relocation on PCT."

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[93] Tr. 2260.

[94] Tr. 2211 (emphasis added).

(2) Fuel-Clad Bonding

Dr. Lyman also expressed concern that differences between MOX and LEU fuel with regard to tightness of fuel-clad bonding prior to ballooning could have an impact on fuel relocation behavior during a design-basis LOCA. Tight bonding can delay the onset of ballooning, by constraining the movement of fission gases and the expansion of gas bubbles along the fuel-cladding interface.[95] Dr. Lyman noted that all four of the experts on the 2001 PIRT panel agreed that "chemical and mechanical bonding between the fuel pellet and the cladding . . . was of high importance to the fuel relocation phenomenon, because 'bonding could significantly affect the relocation characteristics by impeding pellet fragment movement.'"[96] Dr. Lyman further proposed that observed differences between MOX and LEU fuel behavior during irradiation — namely, MOX fuel being more resistant to clad failures due to pellet-clad mechanical interaction than LEU fuel[97] — "may imply that the pellet-clad bond is weaker for MOX fuel, in which case MOX fuel may have a greater propensity to earlier and more extensive fuel relocation than LEU."[98] Therefore, BREDL asserts, Duke's failure to account for this phenomenon "contributes another uncertainty to the safety margin associated with Duke's design basis LOCA calculation."[99]

Experts testifying for both Duke and NRC questioned whether this conclusion could be supported by the data and also questioned the concept that a MOX fuel pellet with only a small

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[95] Tr. 2261; BREDL Findings at 17; *see* Exhibit 31 at 433.

[96] Tr. 2261; Exhibit 32, Table D-1 at D-69.

[97] Tr. 2261; Nuclear Energy Agency, NEA/NSC/DOC(2004)8, *International Seminar on Pellet-Clad Interactions with Water Reactor Fuels*, at 20 (May 6, 2004).

[98] Tr. 2261. In BREDL's prefiled testimony Dr. Lyman also cites another report of the Nuclear Energy Agency, a specialized agency with the Organization for Economic Co-operation and Development (OECD), an intergovernmental organization of 28 industrialized countries including the United States. *See* www.nea.fr. *See* Exhibit 33, Nuclear Energy Agency, NEA/CSNI/R(2003)9, *Ongoing and Planned Fuel Safety Research in NEA Member States*, at 9.

[99] BREDL Findings at 18.

concentration of MOX could exhibit significantly different bonding behavior from a quite similar LEU pellet. Duke expert Dr. J. Kevin McCoy noted that the report cited by BREDL made "no suggestion that pellet-to-cladding chemical bonding is a possible explanation for differences that may exist between the PCMI [Pellet-Clad Mechanical Interaction] performance of MOX and LEU fuels."[100] He opined further that "the processes that lead to bonding are the same for LEU and MOX fuels, and the bond strengths are expected to be similar because of the similarities in fuel chemistry and in operating conditions."[101]

NRC Staff expert Meyer said that he and the PIRT experts originally thought that the very tight pellet-clad bonding in MOX fuel would have a significant effect on balloon size, but found that this was not the case when performing experiments at Argonne National Laboratory with 15-inch-long segments of fuel rods. "It must simply snap that layer when you start developing this uniform outward deformation as the temperature rises with the large pressure differential," according to Dr. Meyer.[102] He explained MOX fuel's greater resistance to cladding failures as being "the result of the greater plasticity of the MOX pellets."[103] Because MOX pellets are softer than LEU pellets, they are "able to deform somewhat and relax the stress they apply to the cladding," which "has nothing to do with bonding between the pellets and the cladding," according to Dr. Meyer.[104]

We find, with regard to any relationship between pellet-clad interactions and time occurrence and size of clad ballooning, that at this time the study of irradiated fuel materials is not sufficiently advanced to resolve the questions raised by BREDL. Moreover, it is unlikely that

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[100] Duke Findings at 23.

[101] Tr. 2165; *see* Duke Findings at 24.

[102] Tr. 2641-42.

[103] Tr. 2313.

[104] *Id.*

this can at this time be experimentally resolved since there is no obvious way to study fuel-clad bonding as an irradiation experiment is proceeding. And, finally, although uncertainty exists regarding fuel-clad bonding, the question before us is how much of an uncertainty is it, and how well does Duke's LOCA analysis allow for such uncertainty? We address these questions in our discussion below.

(3) Effect of M5 Clad Characteristics on Ballooning and Fuel Relocation

Dr. Lyman expressed concern that the M5 cladding used on the MOX fuel would produce larger balloons during a LOCA event than the Zircaloy-4 cladding on LEU fuel, and that this would contribute to a greater fuel relocation effect on PCT. Relying on IRSN, BREDL noted its prediction that "M5 will form larger balloons than Zircaloy-4 in a design-basis LOCA because it remains more ductile during irradiation."[105] According to Dr. Lyman, "[t]he greater retained ductility of M5 as a function of burnup compared to Zircaloy-4 can result in a greater M5 balloon size during a design-basis LOCA for fuel rods of the same burnup."[106] Further, "[l]arger balloons increase the space available for fuel fragments to fall and hence result in a greater propensity for fuel relocation during a LOCA, with an associated increase in PCT and local clad oxidation."[107]

Dr. Lyman subsequently acknowledged that Electricity de France (EDF) and Framatome ANP had used ramp creep tests to conclude that similar balloon sizes would occur for M5 and Zircaloy-4.[108] Duke notes that these tests showed "that M5 cladding actually does not develop larger balloons than Zircaloy-4 under LOCA conditions."[109] Dr. Lyman criticized the ramp tests,

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[105] Tr. 2262.

[106] *Id.*

[107] *Id.*

[108] *Id.*

[109] Duke Findings at 22.

however, for not properly reflecting the true oxide film thickness that would occur on M5 during a LOCA, stating that he did not believe that the EDF presentation “fully addresses the differences that would be observed in actual irradiated fuel with regard to the ductility and the balloon size of M5 compared to that of Zircaloy-4." He also continued to emphasize “the absence of experimental data on the performance of high-burnup M5 clad fuel, under design-basis LOCA conditions.”[110]

NRC Staff expert Meyer acknowledged that the EDF presentation “does not entirely address the differences in the size of balloons between M5 and Zircaloy,” but insisted that it “clearly shows that the large difference claimed by IRSN is a consequence of using inappropriate data.”[111] Based on his knowledge and experience, he saw “no valid reason to expect that the size of the balloons will be affected by the type of fuel inside.”[112] Noting that, “[a]lthough confirmatory research on M5 cladding under LOCA conditions is continuing,” it was his opinion that Dr. Lyman’s concerns are not valid, and it is the Staff’s view that ballooning size has been adequately accounted for in Duke’s analysis.[113] Also, as pointed out by NRC Staff expert Undine Shoop, Framatome actually developed a model to specifically describe the clad ballooning properties of M5, which was approved in December 1999 and revised in February 2000.[114]

Duke responded further to the questions related to M5 cladding by noting that they are not specifically a MOX issue since M5 is used both on LEU and MOX fuel. Noting that M5

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[110] Tr. 2262-63.

[111] Tr. 2315.

[112] *Id.*

[113] *Id.*

[114] Tr. 2300; “Revised Safety Evaluation by The Office of Nuclear Reactor Regulation, Topical Report BAW-10227P, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, Framatome Cogema Fuels, Inc.” (Feb. 4, 2000).

experiences less strain at rupture than Zircaloy-4 in the unirradiated state, Duke expert Dunn pointed out that the two materials have "approximately equal strain potential near the end of irradiation," and asserted that "[o]verall, therefore, there is little expected difference in the consequences of fuel relocation due to cladding differences."[115]

We find that, as Dr. Lyman asserts, there are uncertainties in the current data base for both MOX and LEU fuel using M5 cladding. We do not, however, see a pattern in the data presented to us suggesting that the use of MOX fuel with M5 cladding poses a significant additional risk over LEU fuel in the event of a LOCA — particularly given that the application herein is for four lead test assemblies, and particularly in light of the fact that the use of M5 cladding is not strictly a MOX issue, as it is also used with LEU fuel. We find, to the contrary, that the preponderance of the evidence presented to us is that any risk relating to the M5 cladding alone is not significantly greater with MOX fuel.

We turn now to one of the more critical of the penultimate issues before us, the impact of fuel relocation on PCT, which involves consideration of combined effects of the preceding phenomena.

C. Impact of Fuel Relocation on Peak Clad Temperature

The NRC does not require modeling of fuel relocation under Appendix K.[116] BREDL contends that fuel relocation has a significant impact and should be modeled by Duke.[117] Duke insists, however, that any possible impact of MOX fuel relocation on compliance with the acceptance criteria must be considered in the context of all of the conservatisms already built into the Appendix K models and the criteria.[118] Relative to a best-estimate approach (the

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[115] Tr. 2160.

[116] *See* 10 C.F.R. Part 50, Appendix K; Tr. 2667.

[117] Tr. 2256-57.

[118] *See* Tr. 2132; 2376.

Westinghouse version of which does account for fuel relocation[119]), Duke suggests through its expert, Steven Nesbit, that an Appendix K methodology provides more than 600EF in margin for PCT.[120] Nevertheless, NRC Staff has more recently acknowledged that omission of fuel relocation effects is a non-conservatism in Appendix K with a very large potential impact on PCT, and that an early "resolution" of this issue (i.e., Generic Issue 92) may have been in error or is no longer applicable because of new information.[121]

As indicated above, BREDL suggests that the impact of fuel relocation effects during a LOCA may be more severe for MOX fuel rods than for LEU fuel rods of the same burnup, due to differences in characteristics such as fuel fragment sizes and fuel-clad interactions.[122] As also discussed above, the basis for Contention 1 as admitted was largely certain information from IRSN. One source of information was certain tests from the “VERCORS” series conducted by IRSN and its predecessor organization. These VERCORS tests involved irradiated MOX fuel, but Duke argues they are not directly relevant to a LOCA analysis because they dealt with severe accident consequences and were conducted at temperatures much higher than fuel temperatures experienced during a design basis LOCA.[123]

Separately, at a 2001 conference in Aix-en-Provence, France, IRSN presented a calculation of the possible effect of fuel relocation on cladding temperature at the ruptured location on an LEU fuel rod.[124] BREDL cites this 2001 IRSN report as illustrating the effect of fuel “filling ratio” on the PCT for cladding adjacent to the clad balloon. One parameter

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[119] Tr. 2374.

[120] *See* Tr. 2383-84.

[121] *See* Exhibits 26, 27, Attachment 4 at 4-5; *see also* Tr. 2520, 2532-33; Exhibit 53 (E-mail message from Ralph Meyer forwarded to various PIRT participants (October 13, 2000)).

[122] Tr. 2243; BREDL Findings at 5.

[123] *See* Tr. 2153-56.

[124] Exhibit 4.

evaluated was the ratio of the volume of relocated fuel material to the volume of the ballooned region, and the consequential effect of this “filling ratio” on the resulting surface temperature. Using the CATHARE2 computer code to calculate the impact of fuel relocation on a large-break LOCA PCT for a high-burnup UO_2 fuel rod as a function of the filling ratio, the authors among other things identified the importance of filling ratio to the resulting clad temperature.[125] As BREDL states, “For the scenario evaluated, the authors found that the PCT in the absence of relocation effects was 970EC. For a filling ratio of 70 percent, the maximum considered, the PCT was 1144EC. For a filling ratio of 40%, the PCT was about 20EC greater than for the no-relocation case.”[126] BREDL points out that the maximum impact on PCT of relocation was therefore, in this study, a ΔPCT of +174EC (or 313EF) for high-burnup UO_2 fuel with a filling ratio of 70 percent as compared to no relocation, noting further that it is “not clear from the study whether higher filling ratios, and hence larger impacts on PCT, are possible.”[127] Given that Duke’s calculated PCT is 2018EF, BREDL asserts, a relocation-associated increase in PCT of 313EF (associated with a 70 percent filling ratio for LEU fuel) would result in exceeding the 2200EF limit by 131EF.[128]

The 2001 IRSN calculation takes no credit for heat transfer and cladding cooling benefits associated with swelling and rupture of the cladding, which are demonstrated in the KfK tests discussed below. On the other hand, that 2001 calculation, as pointed out by BREDL,

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[125] Tr. 2459-60; *see also* Tr. 2223; Exhibit 4, "Discussion" following paper and presentation material, First “Answer by C. Grandjean."

[126] Tr. 2258.

[127] *Id.*; Exhibit 29.

[128] Tr. 2264. We note that in the 2001 study, the increase in local oxidation was 7% at the ruptured location under the calculation, which involved no actual test. Tr. 2174; *see* Tr. 2459, 2482.

was for LEU fuel, not MOX fuel, and included no estimate of an additional relocation effect on cladding temperature or oxidation due to LEU-MOX fuel differences.[129]

IRSN presented an updated calculation of relocation effects at Argonne National Laboratory in May 2004. This calculation suggests a difference in fuel relocation impact on PCT between LEU fuel and MOX fuel of only 18EF.[130] IRSN attributed the 18EF MOX fuel increment to the higher initial stored energy in MOX fuel.[131] The MOX fuel lead assemblies, however, as pointed out by Duke, will have lower initial stored energy than the LEU fuel assemblies in the Catawba core, due to operation at lower peaking. They will also have lower decay heat, due to the characteristics of MOX fuel.[132] Based on these factors, Duke suggests the MOX fuel assemblies should involve a benefit, not a penalty, relative to LEU fuel and the potential impact of fuel relocation.[133]

According to Duke, the most comprehensive experimental evidence available for evaluation of fuel relocation comes from certain “KfK” tests performed at the FR2 reactor in Germany.[134] These tests, which were done for LEU fuel that cracked and relocated as well as for fuel that did not relocate, show increased cooling at the ruptured location in the case where there is no relocation.[135] They further show, according to Duke expert Harvey, that with relocation the cooling benefits are a near match for the detrimental effect of relocation on

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[129] Tr. 2482.

[130] Tr. 2175.

[131] Exhibit 5 at 8.

[132] Tr. 2170-72; *see also* Tr. 2147-50.

[133] Tr. 2173.

[134] Tr. 2174; 2148; 2151-53; *see* Exhibits 15, 16.

[135] Tr. 2151.

cladding temperature.[136] In addition, the PCT still occurs at a non-ballooned location on the fuel pin; the relocation effect therefore remains bounded.

The NRC Staff utilized the Duke-calculated maximum temperature increase in the balloon of 1750EF, and adjusted that temperature for the effect of relocation by adding the temperature increment found in parametric studies of the relocation effect. This value of 270EF, when added to maximum clad temperature, yielded a maximum temperature in the balloon of 2020EF.[137] According to the Staff, "[t]his bounding increase in peak cladding temperature is almost identical to the peak cladding temperature of 2018EF reported by Duke and would still be well below the allowable temperature of 2200EF."[138]

According to NRC Staff expert Dr. Meyer, the lower decay heat of MOX fuel also mitigates PCT with MOX fuel. For the Catawba plant, Dr. Meyer stated, the peak cladding temperature occurs a couple of minutes after the loss of coolant has shut down the power, by which time most of the stored heat in the fuel has been dissipated and the chemical heat from the metal-water reaction is small, so that the heat source is dominated by decay heat.[139] Because decay heat for MOX is lower than it is for LEU fuel at the time the MOX peak cladding temperature occurs, the total heat source in the balloon would be lower for MOX fuel than for LEU fuel.[140] Finally, according to Dr. Meyer, "[i]f fuel relocation has any effect [for LEU or MOX fuel], it would increase the temperature only in the ballooned region of the fuel rod."[141]

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[136] Tr. 2174; *see* Tr. 2404.

[137] Tr. 2500.

[138] NRC Staff Findings at 18.

[139] Tr. 2305.

[140] *Id.*

[141] *Id.*

According to the MOX lead test assemblies LAR, the peak temperature at the hot pin rupture location is 1841°F.[142] If the 313°F increase in clad temperature associated with fuel relocation with a filling ratio of 0.7 is added to this value, the resulting clad temperature at the rupture location is 2154°F. According to Duke's calculations, the PCT in a rod where relocation occurs appears to be about 20°F greater than the maximum temperature at the rupture location.[143] Therefore, the peak clad temperature associated with an LEU rod with 0.7 filling ratio due to relocation could be as high as 2174°F — a value with substantially less margin to the 10 CFR § 50.46 limit of 2200E. The 313°F figure was, however, an upper bound number, not an average number.[144] This approach is an approximation, but according to NRC Staff expert Meyer, it is not "a bad way of getting some estimate on the outside effect" of fuel relocation on rupture node temperature during a LOCA.[145]

Duke's experts testified that the effect on Duke's LOCA analysis of using the LEU decay heat curve instead of the MOX decay heat curve is a conservatism of "up to 75°F on PCT."[146] BREDL expert Lyman agreed that this is a conservatism, but pointed out that the effect of using the LEU decay heat curve on PCT is "considerably smaller than the effect of relocation on PCT," which, as indicated above, BREDL contends could be on the order of several hundred Fahrenheit degrees.[147]

Duke and NRC Staff experts both expressed the view that the behavior of MOX fuel would be little different from LEU fuel in the effects of relocation,[148] with Duke challenging the

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[142] LAR, Attachment 3, Table 3-5.

[143] Tr. 2153.

[144] Tr. 2669.

[145] Tr. 2669.

[146] Tr. 2131.

[147] Tr. 2271.

[148] *See* Tr. 2164; 2304.

concept that a “rim” structure in MOX fuel would lead to smaller particle sizes following the fracture of the pellet surface. According to Duke expert J. Kevin McCoy, “the ‘rim’ regions are tougher than the balance of the fuel, so the reasoning [of a rim effect] fails,”[149] and micrographs of irradiated LEU and MOX pellets show similar cracking patterns. Also, the presence of Pu-rich agglomerates do not appear to alter the cracking pattern.[150]

With regard to filling ratio, Duke experts McCoy and Dunn cited the results of certain PBF and FR-2 experiments on LEU fuel, reported at a May 2004 meeting at Argonne, according to which “the [filling] ratio falls between 0.55 and 0.8. However, the upper portion of this range may be discounted. The most reliable relocation data is from the FR-2 tests, and those values lie in the 0.55 to 0.65 range.”[151] Dr. McCoy indicated that “available measures of relative fuel material density . . . indicate that a filling ratio less than 0.7 would be expected for LOCA conditions.”[152]

From the preceding, several observations follow. First, we note that the arguments for a significant effect of filling ratio on PCT come into play only at a relatively high filling ratio, on the order of 70 percent. Calculated PCT values drop significantly at lower filling fractions. Given the evidence that MOX fuel cracking patterns are shown to be quite similar to the patterns for LEU and the additional evidence that filling ratios will not likely be as high as 70 percent, it does not appear likely that this effect could produce a PCT increase that exceeds the allowed maximum value. This is particularly true when it is recognized that Duke implemented certain conservatisms in its Appendix K calculations that were not required — specifically, the

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[149] Tr. 2209.

[150] Tr. 2162.

[151] Tr. 2164.

[152] *Id.*

use of LEU decay-heat curves rather than MOX curves alone provide a conservatism of “up to 75EF” on PCT.[153]

Second, it may be found that, even if the cooling effects of the ballooning are ignored and it is assumed that the fuel relocates at a filling ratio of 0.7 to the ballooned region, the PCT in that area would not be expected to exceed 2174EF. While close to the 2200EF limit, this value is still below it. If the benefit of the lower decay heat is included, then this value would drop by 75EF to 2099EF. Such a calculation is conservative in light of the FR-2 tests suggesting the cooling effects of the ballooning are, if not equal, nearly equal to the heating effects associated with relocation, at least for LEU fuel. It would be expected that the cooling effects would be similar for LEU and MOX fuels, since the cooling is dominated by the thermal hydraulic phenomenon near the balloon and clad rupture.

In addition, as pointed out by Duke, despite its failure explicitly to account for fuel relocation, Appendix K is conservative in producing PCTs that are 600EF or more above best-estimate calculations.[154] Thus, a 300-325EF increase in temperature would be well within the conservatisms of an Appendix K calculation.

We make these observations in light of the fact that, for a best-estimate LOCA calculation for LEU fuel, the effects of fuel relocation are modeled explicitly. There would clearly be merit in gaining a better understanding of the effects of fuel relocation with MOX fuel by performing a best-estimate analysis, such as that used for the Westinghouse NGF, for a MOX fuel assembly. Development of improved MOX-specific models for fuel relocation would allow for a best-estimate analysis to be performed, and in turn would improve the understanding of MOX behavior during LOCA conditions. With that said, however, we find that Duke’s

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[153] Tr. 2131.

[154] *Id.*

analysis of PCT during a LOCA is sufficiently conservative to account for any uncertainties in fuel relocation characteristics of MOX.

D. Impact of Fuel Relocation on Cladding Oxidation

As indicated above, Appendix K limits the allowed cladding oxidation during a LOCA to 17 percent. The purpose of this limitation is to insure adequate cladding integrity in unruptured regions of the clad during a LOCA transient. BREDL contends that the use of M5 cladding on MOX fuel poses a risk of exceeding this limit during a LOCA.

We note that both surfaces of the clad in a LOCA would be exposed to an oxidizing environment at a clad rupture site, and this is proposed to be the region of maximum concern. In addition, oxidation rates are temperature-dependent. BREDL expert Lyman suggests that Duke may have underestimated the clad temperature at the rupture point in its safety analysis. Dr. Lyman asserts:

> According to the MOX LTA LAR at 3-43, the peak temperature at the hot pin rupture location is 1841EF. . . . If the 313EF increase in clad temperature associated with fuel relocation with a filling ratio of 0.7 is added to this value, the resulting clad temperature at the rupture location is 2154EF. From [Exhibit 16] in Duke’s testimony, the PCT in a rod where relocation occurs appears to be about 20EF greater than the maximum temperature at the rupture location. Therefore, the peak clad temperature associated with an LEU rod with [a] 0.7 filling ratio due to relocation could be as high as 2174EF — a value with substantially less margin to the 10 CRF §50.46 limit. Consideration of additional MOX effects, such as a greater filling ratio, could shrink this margin even further.[155]

Further, he notes, “[t]he oxidation rate for M5 is substantially greater at 2154EF than at 1841EF.”[156] BREDL in its proposed findings cites two studies substantiating an oxidation rate increase.[157]

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[155] Tr. 2269.

[156] *Id.*

[157] BREDL Findings at 12; Yan et al., Post Quench Ductility Results for Zry-4 and M5 Oxidized at 1000E C, and 1100E C (Jan. 15, 2004); and Post Quench Ductility Results for Zry-4 and M5 Oxidized at 1200E C, Slow Cooled to 800E C and Quenched (Mar. 23, 2004) (Exhibits 56, 57).

BREDL also cites an IPSN study in which the impact on maximum cladding oxidation for two-sided oxidation at a ruptured region is evaluated.[158] Using a rate law model to calculate the percentage of oxidation, the maximum cladding oxidation "was 12.6% for the no-[fuel-] relocation case, and 19.7% for the 70% filling ratio case. Thus the maximum impact on ECR (equivalent cladding reacted) resulting from relocation was calculated as ΔECR=7.1%."[159] BREDL also points out in its Proposed Findings that the IPSN calculation was terminated at less than 200 seconds, whereas the Catawba LOCA analysis was run for a longer period of time."[160]

BREDL expert Lyman suggests that clad oxidation may be even more severe for MOX fuel because of the fuel-clad interaction effects and pellet fragment size effects discussed above.[161] In response, Duke states that the calculated clad oxidation for M5-clad MOX fuel was 5.2 percent, based on the AREVA Appendix K LOCA evaluation model, which is consistent with all approved Appendix K models. Looking just at the rupture location, Duke asserts the following:

> The highest PCT at the ruptured location in the LOCA calculations for Catawba described in the MOX fuel lead assembly license amendment request was approximately 1750EF and the local oxidation on that fuel pin is 3%. Adding the IRSN (IPSN) predictions to the Catawba MOX fuel results gives an estimated PCT of 2070EF and a local oxidation of 10%. Thus, even if the pessimistic IRSN predictions are simply added to the current Catawba MOX fuel LOCA evaluations, the results remain well below the acceptance criteria of 10 C.F.R. 50.46.[162]

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[158] *See* Exhibit 29.

[159] Tr. 2259.

[160] *See* Tr. 2664; Exhibit 30; Exhibit 6. BREDL indicates that the Duke analysis was run for 400 seconds, but we note that Exhibit 6 indicates it was actually run for 600 seconds.

[161] Tr. 2259-62; 2549-50.

[162] Tr. 2175.

The NRC Staff only briefly addressed the oxidation issue, but noted that "[t]he increase in the amount of oxidation was estimated by IRSN in their parametric study to be about 10 percent."[163] This is a significant increase, Duke acknowledges, "but when added to the 3% [oxidation at the point of PCT] reported by Catawba, the total is still less than the 17% licensing limit."[164]

We note that Dr. Lyman did not predict a clad oxidation percentage that exceeds the allowed 17 percent (even when his proposals for increased PCT values were applied). Rather, he suggested several phenomena, which, if he is correct regarding their effects, would increase the percentage of clad oxidation above the values calculated by Duke in their LOCA analysis. We find, however, that this is not a sufficient basis for requiring revision to the LOCA model as presented, particularly given that the suggested corrections are themselves open to question.

E. Impact of Fuel Relocation on Ability to Maintain Coolable Core Geometry

BREDL has also raised questions about the ability of a MOX LTA core to preserve a coolable core geometry, noting that IRSN suggests that the impact of fuel relocation in fuel rod balloons on the coolability of blocked regions is "fully questionable and should be addressed by specific analytical tests with a simulation of fuel relocation."[165] Duke points out that the coolability of a blocked region given fuel relocation is not unique to MOX fuel, noting that in its LOCA analysis for a core with MOX LTAs, "[t]he maximum calculated cladding strain for the most limiting case [evaluated at a burnup of 30 GWD/t] is 51 percent and the flow blockage due to this ballooning is 52 percent of the coolant channel surrounding the hot pin."[166] This amount,

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[163] Tr. 2684.

[164] Tr. 2175.

[165] Tr. 2250.

[166] Tr. 2128-29.

Duke states, "is well within the coolable geometry limit (specified by the AREVA LOCA evaluation model) of 90 percent."[167]

The maximum flow blockage that will preserve a coolable geometry depends, however, as BREDL points out, on the assumed heat source and the heat transfer properties of the fuel bundle. BREDL cites IRSN for the fact that bundle blockage ratios accepted and used by the nuclear industry have been derived based upon arrays of unirradiated fuel rods, and do not take into account fuel relocation and its associated impacts on the redistribution of the decay heat source within the fuel rods.[168] IRSN has also referred to the impact of fuel relocation on the coolability of a blocked region as "still fully questionable."[169] We find, however, that the 90 percent blockage figure used by Duke in its calculation is sufficient to bracket the calculated blockage plus uncertainty, and therefore is not a safety concern that would overcome Duke's showing on coolable core geometry.

F. Adequacy of Database on MOX Fuel in a LOCA

Irradiation testing of MOX fuel to burnups anticipated by the Catawba LTA has not been carried out, which raises potential questions over differences in performance under irradiation relative to LEU fuel. BREDL suggests that "currently, the data base is insufficient to permit a demonstration that the significant uncertainties associated with MOX fuel behavior in a LOCA do not undermine reasonable assurance of adequate protection of public health and safety."[170] In addition, Dr. Lyman has stated that because of the substantial differences in MOX fuel — including different microstructure and different physical material — the use of MOX should not be regarded in the same way as an LTA approval involving, for example, a change of mid-span

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[167] Tr. 2129.

[168] Tr. 2263 (citing Grandjean, *LOCA Issues*, at 23).

[169] *Id.*

[170] BREDL Findings of Fact at 24.

mixing grid position, or similar change.[171] Because there is a substantial change to the nature of the fuel in this case, Dr. Lyman contends, prior to approval of the proposed MOX lead test assemblies, there should be testing in actual test facilities, both in the United States and abroad, where experimental fuels are not only irradiated, they are also subject to severe accident, LOCA, and other accident testing.[172]

Dr. Lyman further suggests that irradiation and testing should be done in a test reactor prior to lead test assembly irradiation in a commercial reactor, because of questions and uncertainties involving the proposed MOX LTAs. According to Dr. Lyman:

> . . . the fact that there is relatively little on MOX fuel under accident conditions relative to the long experience with uranium fuel that's been acquired over the history of commercial nuclear power in this country, that leads to an effective discrepancy in the confidence we can have that we know enough about the LTAs to put them in a U.S. reactor. . . . [T]he goal is to make sure that U.S. commercial reactors are not test reactors. There are different safety criteria for test reactors. That's where testing of experimental fuels should be performed, not in a commercial reactor in a densely populated area. . . .[173]

According to NRC Staff expert Dr. Meyer, the problem with Dr. Lyman's suggestion is that, to do tests, one needs fuel specimens, and these would come from the LTA program. He noted that it is true that during normal operation in a commercial reactor, one can obtain only a limited amount of information from lead test assemblies because they are not fully instrumented, having only limited numbers of thermocouples, pressure transducers, and other instruments. Accident conditions must be created in a laboratory, as they are not created in power reactors to test fuel. Thus, for studies of LOCA behavior, one has to do all of the testing in the laboratory or in a test reactor. But to do so, one needs fuel specimens, which are

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[171] Tr. 2588.

[172] *See* Tr. 2592-2600.

[173] Tr. 2589.

obtained from a lead test assembly program. The process is not a step-by-step sequential process but rather a parallel one with each type of testing complementing the other.[174]

Moreover, according to the testimony of NRC Staff expert Shoop, there is actually a preference on the part of the NRC Staff for prototypical irradiations of the sort produced in operating reactors, assuming all relevant safety considerations are addressed. Such prototypical irradiations typically cannot be produced in a test reactor because the irradiation spectrum differs from that of a power reactor. The NRC Staff wants to see the results of fuel performance under thermo-hydraulic and irradiation conditions of a power reactor as part of its approval for any new fuel design before allowing batch loading, and thus, in NUREG 800, in-reactor testing is preferred, whether the change is to a component of a fuel assembly or in the fuel material itself.[175]

Taking all of the evidence presented in this matter into account, we recognize that there would be considerable value to having test reactor irradiation results on well-instrumented test specimens at burnups at or above the values anticipated for commercial application of any new reactor fuel.[176] We also recognize, on the other hand, that evidence presented by both Duke and NRC Staff experts provides strong support for their argument that the behavior of MOX fuel would be little different from LEU fuel in the effects of fuel relocation, the principal phenomenon under consideration in this case.[177] We note in this regard the quite low concentration of plutonium oxide fuel in a MOX fuel core — necessarily considerably less than the 2 percent figure for the percentage of MOX fuel lead assemblies in the core.[178] In addition, the NRC

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[174] *See* Tr. 2696-99.

[175] Tr. 2699-2700; *see* NUREG 800 at § 4.2.

[176] It appears that no such studies are now under way or planned for the near future.

[177] *See* Tr. 2164; 2304.

[178] *See* Tr. 2125.

Staff's preference for prototypical irradiations of the sort that are produced in operating reactors, and its reasons for such preference, we find to be persuasive argument against requiring testing in test reactors — absent safety concerns that would cause Duke's application to fail to meet relevant criteria for approval.

What we are left with is a situation in which there are, without question, some uncertainties about MOX fuel and its performance under irradiation. We find, however, that the preponderance of the evidence with regard to the matters at issue in Contention 1 is that all of the uncertainties raised by BREDL are within sufficient ranges, particularly given the conservatisms in Appendix K, to assure to a reasonable degree that the use of the proposed four MOX lead test assemblies will not endanger the health and safety of the public. Under these circumstances, and given that only four LTAs are to be irradiated, with post-irradiation examination of them planned, we do not find the absence of test reactor performance data to be sufficient reason not to permit use of the MOX LTAs at Catawba.

V. CONCLUSIONS OF LAW

We conclude, under the provisions of 10 C.F.R. §§ 50.90, 50.92, 50.40(a), and 50.57(a)(3), that Applicant Duke Energy Company has shown, by a preponderance of the evidence, that with regard to the matters at issue in Safety Contention 1 there is reasonable assurance that its proposed use of the four MOX lead test assemblies in the Catawba plant will not endanger the health and safety of the public. In reaching this conclusion we observe that Duke's LOCA analysis under Appendix K, although it does not take into account fuel relocation, is informed by sufficient compensating conservatisms that such reasonable assurance is provided.

Appendix K has been in effect since 1974, and over the years some extra conservatisms and some non-conservatisms have been identified in it.[179] The Appendix K model does not account for relocation, which has a non-conservative impact, but it also does not take credit for other known extra-conservative factors.[180] For example, among the required features in Appendix K is the use of October 1971 American Nuclear Society "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors."[181] Also included is the use of the Baker-Just equation for the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction of the cladding.[182] Appendix K also requires that each evaluation model shall include provisions for predicting cladding swelling and rupture during a LOCA.[183]

We find that these and other conservatisms provide adequate compensatory effect for the fuel relocation effects presented to us in this proceeding. Thus, we do not find the absence of an accounting for fuel relocation in Appendix K to invalidate Duke's Appendix K LOCA analysis for Catawba with use of the four MOX lead test assemblies.

Further, although BREDL has demonstrated various uncertainties related to the use of MOX fuel, fuel relocation, and related phenomena, to which we expect Duke will be alert in its irradiation of the applied-for MOX lead test assemblies, we find the preponderance of the evidence in this proceeding to be that use of the four MOX test assemblies as planned will not, as required under 10 C.F.R. §§ 50.46(b)(1), (2), and (4), cause peak cladding temperature to exceed 2200EF, nor cause total cladding oxidation to exceed 17 percent of total cladding

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[179] Tr. 2312.

[180] Tr. 2669-70.

[181] 10 C.F.R. Part 50, Appendix K, § I.A.4.

[182] 10 C.F.R. Part 50, Appendix K, § I.A.5.

[183] 10 C.F.R. Part 50, Appendix K, § I.B.

thickness before oxidation, and that calculated changes in core geometry will be such that the core will remain amenable to cooling with use of the MOX LTAs. This showing, we conclude, provides the requisite reasonable health and safety assurances with regard to the issues arising from Contention 1, by a preponderance of the evidence as a whole.

VI. ORDER

1. This Memorandum and Order is effective immediately and, in accordance with 10 C.F.R. § 2.760 of the Commission's Rules of Practice, shall become the final action of the Commission forty (40) days from the date of its issuance (on ***January 31, 2005***), unless any party petitions the Commission for review in accordance with 10 C.F.R. § 2.786 or the Commission takes review on its own motion.

2. Within fifteen (15) days after service of this Memorandum and Order, any party may seek review by filing a petition for review with the Commission on the grounds specified in 10 C.F.R. § 2.786(b)(4). The filing of a petition for review is mandatory for a party to exhaust its administrative remedies before seeking judicial review. 10 C.F.R. § 2.786(b)(1).

3. Any petition for review shall be no longer than ten (10) pages and shall contain the information set forth at 10 C.F.R. § 2.786(b)(2). Any other party may, within ten (10) days after service of a petition for review, file an answer supporting or opposing Commission review. Any such answer shall be no longer than ten (10) pages and, to the extent appropriate, should concisely address the matters in 10 C.F.R. § 2.786(b)(2). 10 C.F.R. 2.786(b)(3). A petitioning party shall have no right to reply, except as permitted by the Commission. *Id*.

It is so ORDERED.

THE ATOMIC SAFETY
AND LICENSING BOARD

/RA/
____

Ann Marshall Young, Chair
ADMINISTRATIVE JUDGE

/RA/
Anthony J. Baratta
ADMINISTRATIVE JUDGE

/RA/
Thomas S. Elleman
ADMINISTRATIVE JUDGE

Rockville, Maryland
December 22, 2004[184]

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[184] Copies of this Memorandum and Order were sent this date by internet e-mail to counsel for all parties.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)
)
DUKE ENERGY CORPORATION) Docket Nos. 50-413-OLA
) 50-414-OLA
(Catawba Nuclear Station, Units 1 and 2))

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing LB PARTIAL INITIAL DECISION (REGARDING BREDL SAFETY CONTENTION I) (LBP-04-32) have been served upon the following persons by deposit in the U.S. mail, first class, or through NRC internal distribution.

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Docket Nos. 50-413-OLA and 50-414-OLA
LB PARTIAL DECISION (REGARDING BREDL
SAFETY CONTENTION I) (LBP-04-32)

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Office of the Secretary of the Commission

Dated at Rockville, Maryland,
this 22nd day of December 2004