RAS 7635

RELATED CORRESPONDENCE

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OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

DUKE ENERGY CORPORATION'S RESPONSE TO BLUE RIDGE ENVIRONMENTAL DEFENSE LEAGUE'S FIRST DISCOVERY REQUEST

In accordance with the schedule established by the Atomic Safety and Licensing Board ("Licensing Board") in its March 30, 2004 "Order (Confirming Matters Addressed at March 25 Telephone Conference)," Duke Energy Corporation ("Duke") hereby files its Response to the March 31, 2004 "Blue Ridge Environmental Defense League's First Set of Discovery Requests Directed to Duke Energy Corporation." Consistent with the direction of the Licensing Board in its "Order (Confirming Matters Addressed at April 6 Telephone Conference)," dated April 8, 2004, this response does not address Contention III. (Order, at ¶8.)

This response includes certain documents that are proprietary and are governed by the Protective Order issued by the Licensing Board on April 8, 2004.[1] Such documents have been identified as proprietary in the index of documents (Attachment 3) included with this

[1] "Memorandum and Order (Protective Order Governing Non-Disclosure of Proprietary Information)," dated April 8, 2004.

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response and the copies of the documents provided have been marked accordingly. These documents must be handled by BREDL in accordance with the Protective Order.[2]

I. RESPONSE TO BREDL GENERAL DISCOVERY REQUEST

A. Duke Response To General Interrogatories

GENERAL INTERROGATORY NO. 1: *State the name, business address, and job title of each person who was consulted and/or who supplied information for responding to each of the interrogatories, requests for admission, and requests for the production of documents posed by BREDL herein. Specifically note for which interrogatories, requests for admissions and requests for production each such person was consulted and/or supplied information.*

If the information or opinions of anyone who was consulted in connection with your response to an interrogatory or request for admission differs from your written answer to the discovery request, please describe in detail the differing information or opinions, and indicate why such differing information or opinions are not your official position as expressed in your written answer to the request.

Duke Response to General Interrogatory 1:

The name, business address, and job title of each person who "was consulted and/or who supplied information" for Duke in response to each BREDL interrogatory and document production request is set forth in Attachment 1 to this Response. Attachment 1 also lists for each person the interrogatories and document requests for which they were consulted. Among those persons who supplied information for Duke's responses, no information or opinions received concerning the interrogatories were different from the information and opinions provided herein.

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[2] Counsel for BREDL and BREDL's technical consultant executed a Confidentiality and Non-Disclosure Agreement in accordance with the Protective Order on April 13, 2004.

GENERAL INTERROGATORY NO. 2: For Contentions I, II, and III, give the name, address, profession, employer, area of professional expertise, and educational and scientific experience of each person whom Duke expects to call as a fact or expert witness at the hearing. For expert witnesses, provide a list of all publications authored by the witness within the preceding ten years and a listing of any other cases in which the witness has provided fact and/or expert testimony and/or submitted affidavit(s) or declaration(s) within the preceding four years. For purposes of answering this interrogatory, the educational and scientific experience of expected witnesses may be provided by a resume of the person attached to the response. Fact and expert witnesses should be distinguished.

Duke Response to General Interrogatory 2:

For Contentions I and II, the requested information for Duke's expected witnesses is set forth in Attachment 2 to this Response. All witnesses are considered to be expert witnesses. Resumes for each prospective witness are included in Attachment 2. Note that Duke has not yet prepared testimony on Contentions I and II. If and when additional witnesses are identified, Duke will supplement its response to General Interrogatory 2 as appropriate.

GENERAL INTERROGATORY NO. 3: For each witness identified in response to General Interrogatory No. 2 above, describe the facts and opinions to which each witness is expected to testify, including a summary of the grounds for each opinion, and identify the documents (including all pertinent pages or parts thereof), data or other information which each witness has reviewed and considered, or is expected to consider or to rely on for his or her testimony.

Duke Response to General Interrogatory 3:

The facts and opinions as to which each Duke witness is expected to testify are summarized below:

- Steven P. Nesbit

Mixed oxide ("MOX") fuel is a well-established technology with widespread use in European nuclear power reactors. Duke's safety and environmental analyses submitted to the NRC have demonstrated that the MOX fuel lead assembly program can be conducted at Catawba without undue risk to the health and safety of the public. The substitution of four MOX fuel

assemblies for four low enriched uranium ("LEU") fuel assemblies will have no significant impact on Emergency Core Cooling System ("ECCS") performance, design basis event radiological consequences, or the probability of a severe accident at Catawba. In addition, changes in severe accident consequence impacts — as estimated based on various sources — should be less than such changes associated with power uprates. Power uprates are commonly approved by the Nuclear Regulatory Commission without requiring probabilistic risk assessments from the applicant.

Grounds for opinion:

- Duke Power analyses, as documented in the February 27, 2003 Duke Energy Corporation MOX fuel lead assembly license amendment request ("LAR") and subsequent related submittals (*e.g.*, responses to NRC requests for additional information).
- Reviews of assessments and sensitivity studies related to severe accident consequences involving MOX fuel lead assemblies.
- NRC regulatory reviews associated with power uprates.
- Professional experience in nuclear safety analysis.

➢ Robert C. Harvey

The AREVA Loss of Coolant Accident ("LOCA") analyses for the MOX fuel lead assemblies provide a conservative estimation of ECCS performance at Catawba. The AREVA analyses are consistent with approved 10 CFR Part 50, Appendix K methodologies, adjusted appropriately to reflect the properties of MOX fuel.

Grounds for opinion:

- Duke Power analyses, as documented in the February 27, 2003 Duke MOX fuel lead assembly LAR and subsequent related submittals.
- Professional experience in nuclear safety analysis.

➢ Bert Dunn

The AREVA LOCA analyses for the MOX fuel lead assemblies provide a conservative estimation of ECCS performance at Catawba. The AREVA analyses are consistent with approved 10 C.F.R. Part 50 Appendix K methodologies, adjusted appropriately to reflect the properties of MOX fuel. The LOCA analyses are documented in the February 27, 2003 MOX fuel lead assembly LAR and subsequent related submittals.

M5 cladding has been previously approved by the NRC and is currently used in the nuclear industry. The M5 cladding response is modeled in a conservative manner, and cladding ballooning does not pose a threat to core coolability. Clad ductility is not a MOX fuel effect and is not impacted by MOX fuel characteristics. In any event, the MOX fuel LOCA analyses assume the worst case clad ductility in the blockage model.

MOX fuel characteristics should not exacerbate any adverse fuel relocation effects in a LOCA or severe accident in a significant manner. VERCORS test results do not alter the conclusion that the MOX fuel lead assemblies can be safely utilized. One VERCORS test indicated fuel relocation, but that test was at temperatures far in excess of design basis LOCA temperatures. Moreover, any relocation at design bases temperatures is bounded by conservatisms inherent in the Appendix K criteria and the ECCS analysis models.

Grounds for opinion:

- Duke Power analyses as documented in the February 27, 2003 MOX fuel lead assembly LAR and subsequent related submittals.
- M5 Topical Report (BAW-10277-A), "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel" (AREVA/Framatome ANP), and related NRC submittals and safety evaluations.
- 36 years experience in the evaluation of accidents in nuclear reactors, including experimental evaluation, calculations, and licensing.

- Consultations with the international nuclear safety expert community through membership in several forums.

➢ H. Duncan Brewer

The substitution of four MOX fuel assemblies for four low enriched uranium fuel assemblies will have no significant impact on the probability of a severe accident at Catawba. Four MOX fuel lead assemblies will have no significant impact on the progression of a hypothetical core melt at Catawba.

Duke also appropriately estimated severe accident consequence impacts in the Environmental Report that was included with the February 27, 2003 MOX fuel lead assembly LAR. The estimated increase in public health consequences — assuming the radiological characteristics of four MOX fuel lead assemblies — is well within the uncertainties inherent in the analysis and is not significant in the context of a probabilistic risk assessment. For example, in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (December 1990), the NRC studied risks from severe accidents at five commercial nuclear power plants, including Sequoyah, an ice condenser containment plant similar to Catawba. As part of the work, NRC generated quantitative risk values, and it also examined the uncertainty inherent in the results. It is apparent that the results include considerable uncertainty, with a range of two orders of magnitude between the 5th and 95th percentile results, and a difference of approximately a factor of 3 between the mean and the median. In this context, the small change in public health risk associated with MOX fuel lead assemblies is much smaller than the uncertainty inherent in the calculations and therefore is not significant. With four MOX fuel lead assemblies, the risk due to operation of Catawba is well below the NRC's Safety Goals.

Duke's conclusions regarding the small increase in public health risk associated with four lead assemblies are being confirmed by Catawba-specific analyses, based on Duke's

Level 3 Probabilistic Risk Assessment ("PRA"), of severe accidents involving the four MOX fuel lead assemblies.

Grounds for opinion:

- Duke Power analyses, as documented in the February 27, 2003 MOX fuel lead assembly LAR and subsequent related submittals.
- Risk analysis work prepared by the Department of Energy ("DOE") in connection with the plutonium disposition program.
- Level III Probabilistic Risk Assessment analysis for MOX fuel lead assemblies at Catawba.
- Review of published risk analysis uncertainty analyses such as NUREG-1150.
- NRC Safety Goal Policy and other NRC guidelines and guidance documents on risk assessment.
- Professional experience in nuclear severe accident and probabilistic risk analysis.

➢ Robert Henry

The substitution of four MOX fuel assemblies for four low enriched uranium fuel assemblies will have no significant impact on the probability of a severe accident at Catawba. Four MOX fuel lead assemblies will have no significant impact on the progression of a hypothetical core melt at Catawba. Postulated MOX fuel assembly-specific radionuclide release fractions and release timing should not significantly perturb the overall plant response to design basis events or severe accident scenarios, or the resulting offsite consequences.

Grounds for Opinion:

- Duke Power analyses, as documented in the February 27, 2003 MOX fuel lead assembly LAR and subsequent related submittals.
- Level III Probabilistic Risk Assessment analysis of MOX fuel lead assemblies at Catawba.

- Professional experience in nuclear safety and severe accident analyses.

Documents That May Be Relied Upon:

Further in response to General Interrogatory 3, the documents, data or other information that each prospective witness may to rely upon in his testimony are listed below:[3]

1. Duke Energy Corporation MOX Fuel Lead Assembly License Amendment Request ("LAR") dated February 27, 2003, and related submittals and responses to NRC requests for additional information ("RAIs"). The LAR is available through ADAMS (ML030760734). BREDL has previously received copies of Duke's RAI Responses submitted in connection with the LAR, except for those that include NRC Safeguards Information.

2. BAW-10238 (NP), Rev. 1, *MOX Fuel Design Report* (AREVA/Framatome ANP), and related submittals and responses to NRC requests for additional information. The non-proprietary Fuel Design Report is available through ADAMS (ML031550349).

3. Technical Report No. 415, "Status and Advances in MOX Fuel Technology," International Atomic Energy Agency (2003). This document is publicly available through IAEA.

4. Edwin Lyman, "Public Health Risks of Substituting Mixed-Oxide for Uranium Fuel in Pressurized-Water Reactors," *Science and Global Security*, 2000. BREDL is in possession of this document.

5. ERI/NRC 02-202, "Accident Source Terms for Light-Water Nuclear Power Plants: High Burnup and Mixed Oxide Fuels," October 2002. This document is available through ADAMS (ML023500093).

6. DOE/EIS-0229, *DOE Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement ("PEIS")*, December, 1996. BREDL is in possession of this document.

7. DOE/EIS-0283, *Surplus Plutonium Disposition Environmental Impact Statement ("SPDEIS")*, Department of Energy, 1999. BREDL is in possession of this document.

[3] If additional responsive documents or other information are identified, Duke will supplement its Response to this interrogatory as appropriate.

8. October 23, 2003 Presentation by IRSN (Institut de Radioprotection et de Sûreté Nucléaire) to the NRC. This document is available through ADAMS (ML032970624).

9. Malgouyres, Ferroud-Plattet, Ducros, Poletiko, Tourasse, and Buolard, "Influence of MOX Fuel in Fission Product Release Up to Meltdown Conditions," American Nuclear Society Nureth-9 Conference, San Francisco, CA, October 3-8, 1999. This document is available through the American Nuclear Society.

10. DCS-FQ-1999-001, Rev. 2, Fuel Qualification Plan (AREVA/Framatome ANP), April 2001. This document is available through ADAMS (ML013390597).

11. Travers, W. D., Memorandum to the Nuclear Regulatory Commission, "Agency Plan for Confirmatory Research Associated with the Use of Mixed-Oxide Fuel in Commercial Light Water Reactors," February 11, 2000. This document is available through ADAMS (ML003676642).

12. Travers, W. D., Memorandum to the Nuclear Regulatory Commission, "Mixed-Oxide Fuel Use in Commercial Light Water Reactors," April 14, 1999. This document is available through ADAMS (ML993620025).

13. 66 Federal Register 1,158 (January 5, 2001), "Nuclear Regulatory Commission - Commonwealth Edison Company, Byron Station, Units 1 and 2, Braidwood Station, Units 1 and 2; Environmental Assessment and Finding of No Significant Impact."

14. BAW-10277-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel" (AREVA/Framatome ANP), and related submittals and responses to NRC RAIs, February 2000. This document is available through ADAMS (ML003686367).

15. Catawba Updated Final Safety Analysis Report, Chapter 15.6 (Loss of Coolant Accident Analysis).

16. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (December 1990). This document is available through ADAMS (ML040140729).

17. 51 Federal Register 30,028 (Aug. 21, 1986), NRC "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement."

18. Anderson, S.L., Laubham, T.J., Nesbit, S.P., and Gilreath, J.D., "Mixed Oxide Fuel Effects on the Integrity of the McGuire and Catawba Reactor Vessels." (This paper, presented at the Sept. 2002 American Nuclear Society (ANS) 5th Topical Meeting on U.S. Department of Energy Spent Nuclear Fuel and Fissile Materials Management, Charleston, S.C., is

included in the proceedings for that meeting, which are publicly available from ANS.)

19. Eller, J.L., "Reactor Core Model Benchmark for Partial MOX Fuel Cycles." (This paper, presented at the Oct. 2003 ANS meeting on Advances in Nuclear Fuel Management III, Hilton Head, S.C., is included in the proceedings for that meeting, which are publicly available from ANS.)

20. Nesbit, S.P., and Eller, J.L., "Basis for the Design of Reactor Cores Containing Weapons Grade MOX Fuel." (This paper, presented at the Oct. 2003 ANS meeting on Advances in Nuclear Fuel Management III, Hilton Head, S.C., is included in the proceedings for that meeting, which are publicly available from ANS.)

21. NUREG/CR-5249, "Quantifying Reactor Safety Margins - Application of Code Scale, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant-Accident," U.S. Nuclear Regulatory Commission, December 1989. This document is available through ADAMS (ML030380473 and ML030380495).

In addition, as described above, a Level 3 Probabilistic Risk Assessment analysis of MOX fuel lead assembly use at Catawba is being performed by Duke. Duke's responses to document requests described below include a final calculation of radionuclide inventories for a core that is all LEU and for a core with MOX fuel lead assemblies. This is a final document used as input to the Level 3 analysis. Duke will supplement its response with the final Level III analysis report when completed. Preliminary drafts and work papers are being treated as privileged, because the work is being prepared at the direction of counsel for preparation of testimony in this proceeding.

B. Duke Response To General Document Production Requests

REQUEST NO. 1: *All documents in your possession, custody or control that are identified, referred to or used in any way in responding to all of the above general interrogatories and the following interrogatories and requests for admissions relating to specific contentions.*

Duke Response to General Document Production Request 1:

See Response to General Interrogatory 3 above. Duke believes that all of the cited documents are publicly available or already in the possession of BREDL.

REQUEST NO. 2: All documents in your possession, custody or control relevant to each BREDL admitted contention, and to the extent possible, segregated by contention and separated from already produced documents.

Duke Response to General Document Production Request 2:

Duke's document production necessarily focused on documents that form the basis for Duke's position and that may be relied upon, and on issues raised in Contention I and II that are also addressed in other specific document requests. The documents identified in response to General Interrogatory 3 above, and in response to Specific Document Requests II-1 through II-6 below, appear to comprehensively respond to this very general request for "relevant" documents. A list of all documents being provided in response to BREDL's discovery requests is included as Attachment 3. That list notes specific discovery requests to which each document appears to be responsive. A Privilege Log is provided as Attachment 4.

Consistent with the Licensing Board Order of April 8, 2004, Duke has not construed documents related to dose analyses of non-LOCA or non-core melt design basis events to be within the scope of either Contention I or II. Duke also understands Contention I to cover only LOCA design basis accident analyses related to the criteria of 10 C.F.R. § 50.46, and not to cover analyses related to releases. Contention II is limited to core melt events, whether resulting from LOCA or another initiator, and resulting consequences.

Duke also has not included in its response analyses (*e.g.*, LOCA analyses) that are: (1) related only to potential batch-scale use of MOX fuel; or (2) related to LOCA analysis generally, but do not pertain to or involve differences between MOX fuel and LEU fuel.

Duke has not attempted to identify every pleading filed in this proceeding or in the completed McGuire/Catawba license renewal proceeding, or every draft of such pleading, whether including comments or not. While perhaps in a broad sense "relevant" to Contentions I and II, Duke believes that these documents are either publicly available (final pleadings) or are privileged under the attorney-work product and attorney-client communication privileges. These documents are not always shown on the Privilege Log, simply because of the number of documents involved and the time available to respond to the discovery requests.

Duke's responses to this and the other BREDL document requests also comport with the discussions of e-mail during the April 6, 2004 telephone conference and the Licensing Board's Order of April 8, 2004. (Order, at ¶ 4). Duke has limited its review of documents to relevant MOX Project files and files of other relevant Duke organizations and personnel responsible for input to the MOX Project on the lead assembly license amendment request.

> *REQUEST NO. 3: All documents (including experts' opinions, workpapers, affidavits, and other materials used to render such opinion) supporting or otherwise relating to testimony or evidence that you intend to use in the hearing on each BREDL admitted contention.*
>
> Duke Response to General Document Production Request 3:

See Response to General Interrogatory 3 and Request 2. Duke has not yet prepared testimony on Contentions I or II. If and when additional responsive documents are identified, Duke will supplement its response to General Document Production Request 3 as appropriate. In this regard, see the discussion above regarding Duke's ongoing Level 3 plant-specific risk assessment of MOX fuel lead assemblies.

II. RESPONSE TO BREDL SPECIFIC DISCOVERY REQUESTS

A. Response To Specific Interrogatories

SPECIFIC INTERROGATORY NO. I-1: *Identify and describe in detail all experimental data and analysis relevant to your claim that the differences between MOX and LEU fuel performance during a design-basis LOCA are fully accounted for in your application and RAI responses, including your claim in the 11/4/04 RAI response that the methods employed by Duke for performing LOCA calculations, including calculations of fuel clad ballooning, "are applicable to both MOX and LEU fuel because they are independent of the pellet type." RAI response at 8. Address this subject in the context of the NRC's statement that "chemical bonding between the pellets and the cladding, which may be different for MOX pellets and UO2, may affect the ballooning process and hence the fuel behavior." Memorandum from William. Travers, Executive Director for Operations, to NRC Commissioners re: Agency Plan for Confirmatory Research Associated With the Use of Mixed-Oxide Fuel in Commercial Light Water Reactors, Attachment at 2 (February 11, 2000).*

Duke Response to Specific Interrogatory I-1:

Provided there is reasonable similarity in the fuel pin design, the fuel pin (except for power and decay heat), is not dominant in determining the outcome of a LOCA. Reference 1 captures the importance of the various phenomena impacting the results of a LOCA. The identified controlling phenomena are listed in 13 categories, only one of which includes some phenomena that relate to the nuclear material. Within this latter category, fuel pellet enthalpy at operating conditions, fuel decay heat, gap conductance, and cladding oxidation are listed as significant contributors, and only the first three of those relate to nuclear material. All 12 of the remaining categories are related to the reactor system and independent of the fuel type. Thus, fuel pellet materials (*e.g.*, MOX ceramic pellets) with approximate equivalence to uranium dioxide ceramic pellets in operational enthalpy, fuel decay heat and gap conductance will perform equivalently during a LOCA.

To provide further assurance that MOX fuel would not differ significantly from LEU fuel during a LOCA, AREVA reviewed MOX fuel programs in the U.S., France, and

Britain for LOCA-significant phenomena. Areas of concentration during the review were reactor kinetics, decay heat, and thermal and mechanical properties. Reference 2, Section 3.7.1.1, documents (i) the phenomena that were given specific consideration for the MOX fuel lead assembly LOCA evaluation, and (ii) the disposition that was given to each phenomenon. It should be noted that Duke has chosen not to take credit for MOX-LEU differences such as lower decay heat that would be *beneficial* to the LOCA analysis for MOX fuel lead assemblies. This provides added assurance that the licensing basis analysis for the lead assemblies is conservative.

Concerning pellet-cladding bonding and the potential impact on cladding swelling, both MOX and LEU fuel can develop pellet-cladding bonding during a design basis LOCA. The differences between MOX and LEU fuel in this area, if any, lie in the strength of the bond and the integrity of the pellet surface-body connection. Duke is aware of no specific experimental data directly relevant to differences between MOX and LEU fuel with respect to pellet-cladding bonding. As described below, the MOX fuel lead assembly LOCA evaluation treats cladding swelling and rupture in a manner such that any postulated small difference in the degree of pellet-cladding bonding does not impact the conservative nature of the calculation.

The established requirements for deterministic LOCA calculational approaches, adopted in Appendix K to 10CFR50.46, require that the degree of cladding swelling and incidence of rupture not be underestimated. The general impact of pellet-cladding bonding is to provide some additional support or effective strength to the cladding, potentially reducing the occurrence of cladding rupture and limiting cladding strain. In order to ensure that cladding swelling and rupture are modeled conservatively, consistent with Appendix K requirements, deterministic LOCA evaluations are typically based on data taken from unirradiated cladding

with no provision for pellet-cladding bonding. This conservative approach was used for the MOX fuel lead assembly LOCA evaluation as described in Reference 2.

In addition, it should be noted that the phenomena of cladding swelling and rupture, although they occur during design basis evaluations, were not identified as being of high level importance in Reference 1. Therefore, differences in these phenomena, which could be postulated to occur because of alternative pellet or cladding materials, should not be expected to produce significant differences in LOCA performance.

Finally, this evaluation of the potential impact of MOX fuel on LOCA performance is consistent with the assessment performed by the NRC and documented in Reference 3 (see page 11 of the attachment, Fuel Behavior section). The NRC noted the importance of initial stored heat to the LOCA analysis; the AREVA MOX fuel lead assembly LOCA analyses factored this effect into account as documented in Reference 2, Attachment 3, Section 3.7.1.1.4. Concerning chemical bonding between the pellets and the cladding, the NRC noted that the effect ". . . may be different" and ". . . could be investigated." The NRC further observed that ". . . a major effect is not expected." As described above, any pellet-cladding bonding effect, if present, was accounted for conservatively in the AREVA calculations and should at any rate be minor relative to the overall evaluation of conformance with 10 C.F.R. 50.46.

References:

1. NUREG/CR-5249, "Quantifying Reactor Safety Margins - Application of Code Scale Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant-Accident," U.S. Nuclear Regulatory Commission, December 1989.

2. Duke MOX Fuel Lead Assembly LAR, February 27, 2003.

3. Travers, W.D., Memorandum to the Nuclear Regulatory Commission, "Mixed-Oxide Fuel Use in Commercial Light-Water Reactors," April 14, 1999.

SPECIFIC INTERROGATORY NO. 2: *Identify and describe in detail all experimental data and analysis justifying your omission of a design-basis LOCA consequence analysis in the license amendment application. In particular, please address the extent to which you took into account in your assessment the wide uncertainty bands for MOX early in-vessel release fractions for most fission product categories presented in the expert panel report "Accident Source Terms for Light-Water Nuclear Power Plants: High-Burnup and Mixed-Oxide Fuels," Energy Research, Inc., ERI/NRC 02/0202 at 38 (November 2002).*

Duke Response to Specific Interrogatory 2:

Duke objected to this interrogatory and any associated document production request, on grounds that dose impacts from a design basis LOCA is an issue beyond the scope of any admitted contentions. In its April 8, 2004 "Order (Confirming Matters Addressed at April 6 Telephone Conference)," the Licensing Board addressed this question at ¶ 6. In its Order, the Licensing Board directed that Duke respond to this interrogatory in the context of Contention II as it relates to any consequences resulting from LOCAs, design basis accidents, or severe accidents. Accordingly, Duke is providing the following response.

Contrary to BREDL's assertion in this interrogatory, Duke's February 27, 2003 MOX fuel lead assembly LAR did not "omit" evaluation of dose consequences following a design basis LOCA. The dose impacts are addressed in the LAR safety analysis, Attachment 3, Sections 3.7.2 and 3.7.3. Duke provided additional, revised assessments of dose consequences following a LOCA in responses to NRC Requests for Additional Information dated November 3, 2003, and March 16, 2004 (already provided to BREDL).

Concerning the impact of "wide uncertainty bands for MOX early in-vessel release fractions," the referenced report ERI/NRC 02-202 did not include quantitative uncertainty estimates. Therefore, there are, in that report, no "wide uncertainty bands" as

referred to in the interrogatory. BREDL is perhaps referring to the differences among values provided by different experts for some parameters. However, such differences are not unusual for expert elicitations, particularly in areas such as severe accidents that necessarily involve considerable inherent uncertainty. In addition, it should be noted that the ERI/NRC 02-202 report is relevant only to an Alternative Source Term ("AST") methodology for calculating doses following a design basis event. Catawba's current licensing basis for dose following a design basis LOCA is the TID-14844 source term, not AST. The TID-14844 source term is generally recognized to be more conservative than AST. Furthermore, any postulated "MOX fuel effect" on early in-vessel release fractions would be confined to 2% of the fuel in the core, and should therefore be minimal relative to offsite doses, as reflected in the Duke analyses. For these reasons Duke considers the Duke assessment of MOX fuel impact on dose following a LOCA to be suitably conservative to support the conclusion that public health and safety are adequately protected.

Duke Response To Specific Document Requests

REQUEST NO. I-1: *Any and all documents containing data or analyses described in response to Interrogatory I-1 above.*

Duke Response to Specific Document Request I-1:

The references listed in the response to Interrogatory I-1 are publicly available documents.

REQUEST NO. II-1: *Any and all documents containing results of severe accident consequence assessments that have been conducted by Duke for the Catawba nuclear power plant during the past five years. If documents responsive to this request include documents that have already been provided to BREDL in the license renewal case for Catawba and McGuire, it will be sufficient to identify the documents.*

Duke Response to Specific Document Request II-1:

The documents produced are listed in Attachment 3 to this response. Documents related to Generic Safety Issue ("GSI") 189 and severe accident mitigation alternatives ("SAMAs") for Catawba that were submitted to the NRC, or that are NRC documents, have not been provided unless they include specific substantive comments regarding severe accident consequences. Drafts of certain Duke documents related to GSI-189 and responsive to specific document requests below are provided, except to the extent they are privileged.

REQUEST NO. II-2: *Any and all MACCS2 input files that Duke has used in consequence assessments for the Catawba nuclear power plant during the last five years, including the meteorological data file and the source term release fractions for all severe accidents considered.*

Duke Response to Specific Document Request II-2:

The documents produced are listed in Attachment 3 to this response.

REQUEST NO. II-3: *With respect to any consequence assessments identified in response to Request No. 1, any and all documents containing technical justifications for any assumptions made with respect to the quantities and characteristics of radionuclides released from core to containment.*

Duke Response to Specific Document Request II-3:

The documents produced are listed in Attachment 3 to this response. In this response, and the response to Request II-4, Duke is providing information on the Catawba Level 2 and Level 3 PRAs which contain the requested information and which references the MAAP code that is commonly used in the industry for these applications. The Level 2 and Level 3 documentation contains descriptions and assumptions about the core damage progression sequences and how they are used to select releases that are based on the MAAP analyses.

REQUEST NO. II-4: *With respect to any consequence assessments identified in response to Request No. 1, any and all documents containing technical justifications for any assumptions made with respect to the quantities and characteristics of radionuclides released from containment to the environment.*

Duke Response to Specific Document Request II-4:

The documents produced are listed in Attachment 3 to this response.

***REQUEST NO. II-5**: Any and all documents containing any analysis by Duke of the potential for or consequences of severe accidents when using MOX LTAs at the Catawba nuclear power plant.*

Duke Response to Specific Document Request II-5:

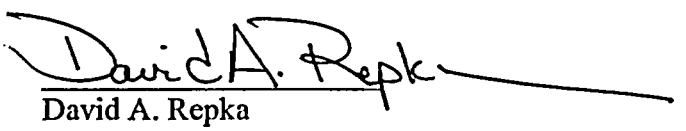
The documents produced are listed in Attachment 3 to this response.

***REQUEST NO. II-6**: Any and all documents discussing research regarding the potential for increased probabilities or consequences of severe accidents during use of MOX fuel at nuclear power plants, including but not limited to studies by the Institut de Radioprotection et de Sûreté Nucléaire ("IRSN") or its predecessor, IPSN.*

Duke Response to Specific Document Request II-6:

The documents produced are listed in Attachment 3 to this response.

Respectfully submitted,



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ATTORNEYS FOR DUKE ENERGY CORPORATION

Dated in Washington, District of Columbia
This 14th day of April 2004

April 14, 2004

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF BERT M. DUNN

Bert M. Dunn hereby declares under penalty of perjury that the following statements are true and correct of his own knowledge:

1. In the captioned proceeding, I have supplied information in response to Specific Interrogatory No. I-1, as set forth in Blue Ridge Environmental Defense League's First Set of Discovery Requests Directed to Duke Energy Corporation, dated March 31, 2004.

2. For 34 years, I have been employed by AREVA, working in areas related to nuclear energy. I currently hold the title of Advisory Engineer in the NSSS Engineering department.

3. The information in these responses is true and correct to the best of my knowledge and belief.

Bert M. Dunn

Subscribed to and Sworn before me personally, on this 14 day of April, 2004.

Notary Public

My Commission expires: 2/29/08

April 14, 2004

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT OF ROBERT C. HARVEY

Robert C. Harvey hereby declares under penalty of perjury that the following statements are true and correct of his own knowledge:

1. In the captioned proceeding, I have supplied information in response to Specific Interrogatory No. I-1, as set forth in Blue Ridge Environmental Defense League's First Set of Discovery Requests Directed to Duke Energy Corporation, dated March 31, 2004.

2. For 25 years, I have worked in areas related to nuclear energy. I am responsible for loss of coolant accident analyses for the Catawba Nuclear Station. I currently hold the title of Senior Engineer in the Duke Energy Nuclear Generation Department.

3. The information in these responses is true and correct to the best of my knowledge and belief.

Robt C Haney
[Name]

Subscribed to and Sworn before me personally, on this 14th day of April, 2004.

Mich T. Cash
Notary Public

My Commission expires: January 22, 2008

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

April 14, 2004

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

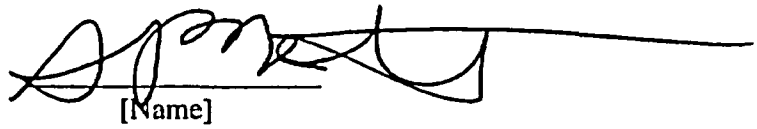
AFFIDAVIT OF STEVEN P. NESBIT

Steven P. Nesbit hereby declares under penalty of perjury that the following statements are true and correct of his own knowledge:

1. In the captioned proceeding, I have supplied information in response to Specific Interrogatory No. I-1 and No. 2, and General Interrogatories Nos. 2 and 3, as specified in Blue Ridge Environmental Defense League's First Set of Discovery Requests Directed to Duke Energy Corporation, dated March 31, 2004.

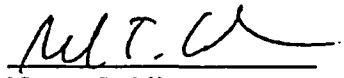
2. For 22 years, I have been employed by Duke Energy Corporation, working in areas related to nuclear energy. I currently hold the title of Engineering Supervisor II in the Nuclear Generation Department.

3. The information in these responses is true and correct to the best of my knowledge and belief.



[Name]

Subscribed to and Sworn before me personally, on this 14th day of April, 2004.



Notary Public

My Commission expires: ~~April~~ January 22, 2008

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

Attachment 1

General Interrogatory No. 1 — Respondents to BREDL Discovery Request

Name	Business Address	Job Title	Response Areas
Michael J. Barrett	Duke Energy 526 S. Church Street Charlotte, NC 28202	Senior Engineer	General Request 2 Specific Requests II-1 through II-6
H. Duncan Brewer	Duke Energy 526 S. Church Street Charlotte, NC 28202	Engineering Supervisor II	General Request 2 Specific Requests II-1 through II-6
Bryan C. Carroll	Duke Energy 526 S. Church Street Charlotte, NC 28202	Engineer	General Request 2 Specific Requests II-1 through II-6
Michael T. Cash	Duke Energy 526 S. Church Street Charlotte, NC 28202	Consulting Engineer	General Request 2 Specific Requests II-1 - II-6
Robert C. Harvey	Duke Energy 526 S. Church Street Charlotte, NC 28202	Senior Engineer	General Request 2 Specific Requests II-1 through II-6 Specific Interrogatory I-1
Melissa S. Moran	Duke Energy 526 S. Church Street Charlotte, NC 28202	Associate Engineer	General Request 2 Specific Requests II-1 through II-6

Steven P. Nesbit	Duke Energy 526 S. Church Street Charlotte, NC 28202	Engineering Supervisor II	General Interrogatories 2 and 3 General Request 2 Specific Interrogatories I-1 and 2 Specific Requests I-1, II-1, II-4, II-6
P. Doug Paul, Jr.	Duke Energy 526 S. Church Street Charlotte, NC 28202	Engineer	General Request 2 Specific Requests II-1 through II-6
Stephen P. Schultz	Duke Energy 526 S. Church Street Charlotte, NC 28202	Nuclear Engineering Manager	General Request 2 Specific Requests II-1 through II-6
Michael W. Scott	Duke Energy 526 S. Church Street Charlotte, NC 28202	Senior Engineer	General Request 2 Specific Requests II-1 through II-6
Gregg B. Swindlehurst	Duke Energy 526 S. Church Street Charlotte, NC 28202	Nuclear Engineering Manager	General Request 2 Specific Requests II-1 through II-6
F. Jay Verbos, Jr.	Duke Energy 526 S. Church Street Charlotte, NC 28202	Senior Engineer	General Request 2 Specific Requests II-1 through II-6
Robert Henry			
Bert M. Dunn	AREVA 3315 Old Forest Road Lynchburg, VA 24501	Advisory Engineer	Specific Interrogatory I-1

Attachment 2

General Interrogatory No. 2 - Prospective Witnesses (Contentions I and II)

Name	Address	Profession	Employer	Area of Expertise
Steven P. Nesbit	Duke Energy 526 South Church Street Charlotte, NC 28202	Nuclear Engineering	Duke Energy	MOX fuel Nuclear safety analysis
Robert C. Harvey	Duke Energy 526 South Church Street Charlotte, NC 28202	Nuclear Engineering	Duke Energy	Nuclear safety analysis Loss of coolant accident (LOCA) analysis
Bert M. Dunn	AREVA 3315 Old Forest Road Lynchburg, VA 24501	Nuclear Engineering	AREVA	Nuclear safety analysis Loss of coolant accident (LOCA) analysis
H. Duncan Brewer	Duke Energy 526 South Church Street Charlotte, NC 28202	Nuclear Engineering	Duke Energy	Nuclear safety analysis Severe accident analysis
Robert Henry	Fauske & Associates, Inc. 16W 070 West 83rd St. Burr Ridge, IL 60527	Mechanical Engineering	Fauske & Associates, Inc.	Nuclear safety analysis Severe accident analysis

STEVEN P. NESBIT
Duke Power
526 South Church Street
Charlotte, NC 28202

QUALIFICATIONS:

Mr. Nesbit has 24 years nuclear engineering and management experience in the commercial sector and on Department of Energy (DOE) projects. He is the Mixed Oxide ("MOX") Fuel Project Manager for Duke Power, which is playing a key role in the DOE program to dispose of surplus weapons plutonium. He has 22 years experience with Duke Power. In addition, Mr. Nesbit has managed activities for the Managing and Operating Contractor to DOE's Office of Civilian Radioactive Waste Management. He also has expertise in nuclear safety analysis technology. Mr. Nesbit has extensive experience interacting with the Nuclear Regulatory Commission and he has authored numerous topical reports and technical papers.

EDUCATION/TRAINING:

ME, Nuclear Engineering, University of Virginia, 1982
BS, Nuclear Engineering, University of Virginia, 1980
Graduate course work, Environmental Science

Supervisory Development Program, Duke Power

PROFESSIONAL AFFILIATIONS/CERTIFICATIONS:

Registered Professional Engineer, North Carolina
Registered Professional Engineer, South Carolina
American Nuclear Society

EXPERIENCE:

3/99-Present **Engineering Supervisor II – Duke Power**

Manages Duke Power's activities as part of the project to dispose of surplus United States weapons plutonium using mixed oxide (MOX) fuel. Directs technical, licensing, and business activities. Serves as a public spokesperson on the MOX fuel project.

09/96-3/99 **Consulting Engineer - Duke Power**

Led Duke Power's feasibility investigations regarding using MOX fuel at the company's three nuclear plants to support DOE's surplus weapons

plutonium disposition program. Served as a representative on the Nuclear Energy Institute's Working Group on Surplus Weapons Plutonium Disposition. Interacted with external groups (Congress, DOE, and the public) in support of the MOX fuel project.

11/95-09/96 **Engineering Supervisor II - Duke Engineering & Services (DE&S)**

Supervised the Design Basis and Project Integration Section of the DOE Office of Civilian Radioactive Waste Management (OCRWM) Management and Operating Contractor. Developed environmental design criteria and performed design basis accident evaluations for an interim storage facility for spent nuclear fuel.

05/94-11/95 **Manager, Regulatory Interactions Section - DE&S**

Manager of the Las Vegas Regulatory Interactions Section of the Regulatory and Licensing Department of the Management and Operating contractor for the DOE OCRWM. Responsibilities of the seven-person section included interactions with the Nuclear Regulatory Commission (NRC) staff and on-site representatives, the Advisory Committee on Nuclear Waste, and the Nuclear Waste Technical Review Board; development of regulatory positions; regulatory reviews; Site Characterization Analysis comment responses; regulatory commitments; and NRC issue resolution activities.

12/92-04/94 **Engineering Consultant - DE&S**

Licensing Engineer in the Las Vegas Regulatory and Licensing Department of the Management and Operating contractor for the DOE OCRWM. Provided nuclear power plant licensing experience and general support to the DOE Yucca Mountain Site Characterization Office. Assisted with interactions between the DOE, the National Academy of Sciences, the Environmental Protection Agency, and the NRC, related to the development of an environmental standard for the potential repository at Yucca Mountain.

1991-1992 **Utility Engineering Group (UEG) Site Engineer - DE&S**

Site Engineer in Washington, D.C., for the DE&S Utility Engineering Group. Provided utility perspective and experience to the DOE for the New Production Reactor Project. Served on the staff of the Chief Engineer of the project. Provided day-to-day liaison with the various project areas. Served as Project Engineer for the UEG. Managed the DE&S Washington, D.C., office.

1990-1991 **Senior Engineer - Duke Engineering & Services**

Worked in the safety review area of the UEG. Provided utility perspective and experience to the New Product Reactor Project in the area of nuclear reactor safety.

1988-1990 **Design Engineer - Duke Power**

Lead engineer in the area of nuclear safety analysis technology, a work group comprised of five engineers. Worked on developing mass and energy release analysis capability for high energy line breaks at Oconee, McGuire, and Catawba Nuclear Stations. Used the RELAP5/MOD002 transient analysis computer code and wrote in-house analytical codes. Worked to develop reactor building analysis capability for large dry and ice condenser containments, including applications of the FATHOMS (COBRA-NC) and CONTEMPT computer codes. Tested the upgraded Oconee training simulator and evaluated vendor performance. Represented the Babcock and Wilcox Owners Group (B&WOG) on the Project Management Group of the Multi-Loop Integral System Test Facility, a thermal-hydraulic research project sponsored by the B&WOG, the Electric Power Research Institute (EPRI) and the NRC. Served on the Duke Power Crisis Management Team.

1982-1988 **Design Engineer/Assistant Engineer/Junior Engineer - Duke Power**

Lead safety analysis engineer for the Oconee Nuclear Station, a work group of up to five engineers. Served as Duke Power representative on the B&WOG Analysis Committee. Participated in the Technical Advisory Group, a committee comprised of B&WOG, EPRI and NRC representatives, which evaluated the need for thermal hydraulic testing related to once-through steam generators. Helped develop symptom-oriented emergency procedures for Oconee. Performed extensive RETRAN benchmarks of plant transients and helped prepare a safety analysis methods topical report for submission to the NRC. Served as one of 12 auditors for the inaugural Duke Power Self-initiated Technical Audit, patterned after the NRC Safety System Functional Inspections. Participated in fuel loading and start-up physics testing at McGuire Nuclear Station. Participated in zero power physics testing at Oconee. Performed system and containment analyses of the Oconee plant. Prepared technical justifications for emergency Technical Specification changes which prevented unnecessary unit shutdowns.

1979-1982 **Reactor Operator/Reactor Operator Trainee - University of Virginia Reactor Facility**

Reactor Operator Trainee and licensed Reactor Operator for the 2-MW research reactor in Charlottesville, Va. Duties included shift operation work, training and fuel handling.

AWARDS/HONORS:

"Doer of Deeds," Yucca Mountain Site Characterization Office, February 2, 1994.
Newcomb/Thornton Fellowship, University of Virginia, 1980-1981.
Bachelor of Science with Highest Distinction, University of Virginia, 1980.

PUBLICATIONS:

Nesbit, S. P., Scott, M. W., Eller, J. L., Verbos, F. J., and Costello, M. V., "Non-LOCA Safety Analysis for Operation with Weapons Grade MOX Fuel Lead Assemblies," American Nuclear Society Winter Meeting 2003, New Orleans, LA, November 2003.

Nesbit, S. P. and Eller, J. L., "Basis for the Design of Reactor Cores Containing Weapons Grade MOX Fuel," Advances in Nuclear Fuel Management III, Hilton Head, SC, October 2003.

Anderson, S. L., Gilreath, J. D., Nesbit, S. P., and Laubam, T. J, "Mixed Oxide Fuel Effects on the Integrity of the McGuire and Catawba Reactor Vessels," Fifth Topical Meeting on Spent Nuclear Fuel and Fissile Materials Management, Charleston, SC, September 18, 2002.

Buckner, M. R., Bengelsdorf, H. D., and Nesbit, S. P., "American Nuclear Society Nonproliferation Position Statement," Fifth Topical Meeting on Spent Nuclear Fuel and Fissile Materials Management, Charleston, SC, September 18, 2002.

Clark, R. H., Dziadosz, D., and Nesbit, S. P., "MOX Fuel Irradiation Program for Disposition of Surplus United States Plutonium," Fourth Topical Meeting on Department of Energy Spent Nuclear Fuel and Fissile Materials Management, San Diego, SC, June 7, 2000.

Nesbit, S. P. and Bengelsdorf, H. D., "A Comparison of Surplus Weapons Plutonium Disposition Technologies," Third Topical Meeting on Department of Energy Spent Nuclear Fuel and Fissile Materials Management, Charleston, SC, September 9, 1998.

S. P. Nesbit, "A Utility Perspective on Surplus Weapons Plutonium Disposition in Existing United States Light Water Reactors," Advances in Nuclear Fuel Management II, Myrtle Beach, S.C., March 1997.

S. P. Nesbit, S. J. Brocoum, M. A. Lugo, J. A. Duguid, P. M. Krishna, "Regulatory Perspective on NAS Recommendations for Yucca Mountain Standards," 7th Annual International High-Level Radioactive Waste Management Conference, Las Vegas, NV, May 1, 1996.

J. Carl Stepp, Silvio Pezzopane, Quazi Hossain, Michael Hardy, Steven P. Nesbit, "Criteria for Design of the Yucca Mountain Structures, Systems, and Components for Fault Displacement," FOCUS '95 - Methods of Seismic Hazards Evaluation, Las Vegas, NV, September 20, 1995.

J. Carl Stepp, Michael P. Hardy, Quazi A. Hossain, Steven P. Nesbit, J. Timothy Sullivan, "Seismic Design Methodology for a Geologic Repository at Yucca Mountain," 6th Annual International High-Level Radioactive Waste Management Conference, Las Vegas, NV, May 4, 1995.

D. Stahl, S. P. Nesbit, L. Berkowitz, "Approach to Compliance with the NRC Substantially Complete Containment Requirement at the Potential Repository at Yucca Mountain," 6th Annual International High-Level Radioactive Waste Management Conference, Las Vegas, NV, May 3, 1995.

S. P. Nesbit, S. J. Brocoum, "New Public Health and Safety Standards for Yucca Mountain and Their Impact on the Carbon-14 Issue," Waste Management '95 Conference, Tucson, AZ, February 26, 1995.

S. P. Nesbit, R. J. Gerling, and G. B. Swindlehurst, "Qualification of the Oconee RETRAN Model by Comparison with Plant Transient Data," Nuclear Technology, Volume 83, December 1988.

TOPICAL REPORTS:

DPC-NE-1005P, "Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," Duke Energy, August 2001.

YMP/TR-003-NP, "Seismic Design Methodology for a Geologic Repository at Yucca Mountain," U. S. Department of Energy, October 1995.

DPC-NE-3003-P, "Mass and Energy Release and Containment Response Methodology," Duke Power Company, August 1993.

BAW-2079, "Technical Advisory Group Investigation of Once-Through Steam Generator Thermal-Hydraulic Data Requirements," Babcock and Wilcox, March 1989.

DPC-NE-3000, "Thermal-Hydraulic Transient Analysis Methodology," Duke Power Company, July 1987.

SECURITY CLEARANCE:

DOE "L" Clearance (active)

REFERENCES:

DOE and commercial references available upon request

ROBERT C. HARVEY
Duke Power
526 South Church St
Charlotte, NC 28202

QUALIFICATIONS:

25 years of Thermal Hydraulic and Safety Analysis experience supporting the reload and licensing of pressurized water reactors. Mr. Harvey has performed numerous safety analysis calculations using the RELAP4, RELAP5, RETRAN, TOODEE-2, CONTEMPT-LT, and MAAP computer codes.

EDUCATION:

Nuclear Engineering Graduate Studies, University of Lowell (1980-1982)
BS, Nuclear Engineering, University of Lowell, 1979
Supervisory Training, Yankee Atomic Electric Company, 1994
MAAP Code Utilization and Phenomena seminar, Fauske & Associates
Two-Phase Gas-Liquid Flow Seminar, University of Houston
Nuclear Power Reactor Safety Seminar, Massachusetts Institute of Technology (MIT)
Simulator Training, Combustion Engineering (CE)
Two-Phase Flow and Heat Transfer, Rensselaer Polytechnic Institute

EXPERIENCE:

Senior Engineer - **Duke Power Company**
2/99 - present

Lead Engineer responsible for the LOCA analysis supporting the Oconee, McGuire, and Catawba nuclear plants. Responsibilities include providing interface and oversight of the vendor analyses. In addition, performs LOCA mass and energy release calculation used as input to the containment analysis and performs UFSAR Chapter 15 non-LOCA safety analysis. Specific accomplishments include supporting the Oconee reanalysis to support steam generator replacement and the transition to best-estimate LOCA analysis methods for the McGuire and Catawba units. Serves as a member of the Emergency Operations Facility (EOF) in the position of Accident Assessment Manager.

Provided an independent assessment of the Texas Utilities LOCA analysis supporting the transition to Westinghouse fuel.

Engineer - **Duke Engineering** & Services - (12/97 - 1/99); *Senior Nuclear Engineer* - **Yankee Atomic Electric Company** (5/91 - 11/97)

Lead Engineer for pressurized water reactor (PWR) LOCA analyses supporting licensing for the Yankee Rowe, Maine Yankee and Seabrook Nuclear Power Stations. Areas of involvement included LOCA and containment analyses and severe accident analyses related to Individual Plant Evaluations (IPEs). Specific accomplishments included supporting the Maine Yankee small break loss of coolant accident (SBLOCA) analysis to justify a return to 2440 MWth operation, and providing oversight of Siemens Power Corporation SBLOCA re-analysis. Served as a response team member to Maine Yankee 1996 Independent Safety Assessment.

In addition, supported General Electric (GE) in severe accident analysis for simplified boiling water reactor (SBWR) certification and provided consulting to the Siemens fuel user group in the area of LOCA analysis.

Provided support to Northeast Utilities on severe accident management guidelines (SAMGs) for Millstone Units 2 & 3 and the Seabrook Nuclear Power Station and performed a technical review of ABB/CE reload analysis of St. Lucie Unit 2 for FP&L.

Senior Engineer - **Yankee Atomic Electric Company**
5/88 - 5/91

Lead Engineer for Yankee Rowe and Seabrook LOCA analysis related activities and for all severe accident analysis activities. Duties involved reload licensing analysis for the Yankee Rowe plant and vendor oversight of Seabrook LOCA analysis activities. Supported the plant specific model development and certification of the Yankee Rowe plant simulator. Participated in the Yankee Rowe plant life extension (PLEX) effort providing support in severe accident evaluations and pressurized thermal shock (PTS) analysis. Also, provided training to Texas Utilities personnel in LOCA analysis method applications.

Nuclear Engineer - **Yankee Atomic Electric Company**
5/85 - 5/88

Lead Engineer for Yankee Rowe LOCA analysis related activities. Activities included large break loss of coolant accident (LBLOCA) model development and applications related to reload licensing, steam generator tube rupture (SGTR) analysis and plant request responses. Participated in the development of Yankee Rowe, plant specific emergency operating procedures (EOPs) based on the generic Westinghouse Owners Group (WOG) emergency response guidelines (ERGs). Performed a plant specific analysis to support deviations from the generic WOG guidelines. Also, provided training to Korean Power (KEPCO) engineers in LOCA analysis methods.

Engineer - **Yankee Atomic Electric Company**
6/79 - 5/85

Performed LBLOCA analyses in support of reloads for the Yankee Rowe and Maine Yankee plants. Contributed to model enhancements of the LBLOCA methods used for Yankee Rowe. Participated in developing and assessing the RELAP5YA computer code used for PWR SBLOCA analysis.

PROFESSIONAL AFFILIATIONS/CERTIFICATIONS:

American Nuclear Society (ANS), Member
The Research Society of Sigma Xi, Associate Member
Registered Professional Engineer
North Carolina (Registration # 027387)
South Carolina (Registration # 22237)

SELECTED PUBLICATIONS:

1. Maine Yankee Steam Generator Tube Sleeving Thermal-Hydraulics and Safety Analysis Impacts, co-authors K. R. Rousseau, S. Palmer, P. A. Bergeron, presented at the American Power Conference, Chicago, Ill., 1995.

2. Maine Yankee Cycle 15 Core Performance Analysis, YAEC-1907, co-authors, January 1995.

3. Yankee Rowe Pressurized Thermal Shock, Thermal-Hydraulic Analysis, International Heat Transfer Conference, co-authors P. A. Bergeron, N. Fujita, August 1993.

4. Maine Yankee Level II PRA Results, ASME/JSME International Conference on Nuclear Engineering, co-author K. E. St. John, March 1993.

5. Thermal Hydraulics Analysis of the Yankee Plant Due to a Stuck Open PORV Using RELAP5/MOD3 Computer Code, RELAP5/TRAC-B International Users Seminar, co-authors W. S. Yeung, R. K. Sundaram, November 1991.

6. Yankee Plant Small Break LOCA Analysis, YAEC-1732, co-authors S. Mihaiu-Westerlind, R. K. Sundaram, July 1990.

7. Yankee Nuclear Power Station Core 21 Performance Analysis, YAEC-1731, co-authors, July 1990.

8. Yankee Nuclear Power Station Severe Accident Closure Submittal, YAEC-1711, co-authors, December 1989.

9. Plant-Specific Analysis to Support the Yankee Emergency Operating Procedures, YAEC-1663, co-authors, April 1989.

10. Seabrook Station Risk and Plant Response for Low Power Operating Conditions, YAEC-1623, co-authors, March 1988.

11. RELAP5YA Simulation of LOFT Small Break Experiments L3-6 and L5-1, Transactions American Nuclear Society, Volume 55, co-authors, L. Schor, November 1987.

12. Estimate of Peak Clad Temperature and Its Uncertainty in a Large Break LOCA at Yankee Nuclear Power Station, YAEC-1431P, co-authors, R. K. Sundaram, K. E. St. John, May 1984.

13. RELAP5YA - A Computer Program for Light Water Reactor System Thermal-Hydraulic Analysis, YAEC-1300P, co-authors, R. T. Fernandez, R. K. Sundaram, J. Ghaus, A. Husain, J. N. Loomis, L. Schor, R. Habert, October 1982.

14. RELAP4 and RELAP5 Calculation of LOFT L3-5 and L3-6 Experiments: Comparison to Data, ANS Specialists Meeting on Small Break Loss-of-Coolant Accident Analyses in LWRs, co-authors, L. Schor, J. N. Loomis, A. Husain, August 1981.

15. RELAP4 Analysis of CREARE Flashing Transients with Reverse Core Steam Flow, Transactions American Nuclear Society, Volume 38, co-authors G. J. Brown, A. Husain, August 1981.

16. Applications of a Lower Plenum Phase Separation Model to Yankee Rowe Large Break LOCA Analysis, YAEC-1231, Revision 1, co-authors, March 1981.

17. Maine Yankee Cycle 5 Core Performance Analysis, YAEC-1202, co-authors, December 1979.

BERT M. DUNN
AREVA
3315 Old Forest Road
Lynchburg, VA 24501

EDUCATION: BS in Physics, Washington State University, 1968
MS in Physics, Lynchburg College, 1973

EXPERIENCE: AREVA Framatome ANP Inc. (formerly Framatome Nuclear Technologies Inc., Nuclear Power Division of the Babcock & Wilcox Company), Lynchburg, Virginia

3/84-Present Advisory Engineer

AREVA representative to high burnup industry forums responsible for providing recommendations on licensing criteria to NRC. Lead on AREVA consultation with NRC on irradiated fuel LOCA testing at Argonne National Laboratory.

Member of U.S. NRC High Burnup Fuel Design Basis Accident Technical Evaluation PIRT (Phenomena Identification and Ranking Table) Panel.

Technical lead for the development of LOCA and Safety Analysis techniques for the licensing of new fuel cladding materials.

Technical lead for the development of evaluation techniques to determine the outcome of inherent boron dilution events.

Technical co-lead for the evaluation of best estimate LOCA licensing techniques.

Project lead for development of Loss-of-Coolant-Accident analysis capability for Westinghouse and Combustion Engineering plants.

2/83-3/84 Senior Product Manager

Project Manager for Pressurized Thermal Shock (PTS) B&W Owners Group Task Force. Lead development of a probabilistic risk assessment for PTS in B&W plants.

1975-1980 Unit Manager, Emergency Core Cooling System Analysis

Responsible for all ECCS evaluations of the performance of B&W-designed nuclear power plants.

1970-1975 Engineer/Supervisory Engineer

Licensing of B&W's ECCS evaluation models and development of techniques for evaluating reactor building subcompartment pressure forces.

1968-1970 Engineer - Douglas United Nuclear Corporation, Richland, Washington

Reactor physics, fuel engineering, operations.

PUBLICATIONS: Major contributor or the principal author of:

a. BAW-10034, "Multinode Analysis of B&W's 2568 MWt Nuclear Plants During a Loss-of-Coolant Accident," October 1971.

b. BAW-10045, "Multinode Analysis of B&W's 205-Fuel Assembly Nuclear Plants During Loss-of-Coolant Accident," May 1972.

c. BAW-10052, "Multinode Analysis of Small Breaks for B&W's 2568 MWt Nuclear Plants," September 1972.

d. BAW-10064, "Multinode Analysis of Core Flooding Line Break for B&W's 2568 MWt Internals Vent Valve Plants," April 1973.

e. BAW-10091, "B&W's ECCS Evaluation Model Report With Specific Application to 177-FA Class Plants With Lowered-Loop Arrangement," August 1974.

f. BAW-10091, Supplement 1, "Supplementary and Supporting Documentation for B&W's ECCS Evaluation Model Report With Specific Application to 177-FA Class Plants With Lowered-Loop Arrangement," December 1974.

g. BAW-10102, "ECCS Evaluation of B&W's 205-FA NSS," June 1975.

h. BAW-10104, "B&W's ECCS Evaluation Model," May 1975.

i. BAW-10106, "- QUENCH - Digital Program for Analysis of Core Thermal Transients During Loss-of-Coolant Accident," May 1975.

j. Presentation for ANS Annual Meeting, Las Vegas, Nevada, June 8-12, 1980, "RC Pump Trip and Small Break LOCA," invited paper.

k. BAW-10227, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," September 1997.

HARRY DUNCAN BREWER
Duke Power
526 South Church Street
Charlotte, NC 28202

QUALIFICATIONS:

Mr. Brewer's nuclear power experience includes all aspects of probabilistic risk assessment, including base model development, project management and applications of Probabilistic Safety Analysis(PRA) to resolve regulatory and operational problems. He has 25 years experience with Duke Power. Mr. Brewer is currently the Section Manager of the Severe Accident Analysis Section, the group which provides PSA analyses and applications support for Duke's three nuclear stations.

EDUCATION/TRAINING:

BS, Nuclear Engineering, North Carolina State University, 1979
ME, Mechanical Engineering, University of South Carolina, 1983
Attended many continuing education courses, including leadership and professional development classes, decision analysis courses, and Advanced PRA Training.

PROFESSIONAL AFFILIATIONS/CERTIFICATIONS:

Registered Professional Engineer, NC #11574, SC #10040
Member of American Society of Mechanical Engineers

EXPERIENCE:

Title of Position: Section Manager
Department/Company: Nuclear Generation Department / Duke Power Company
From: 6/99 To: Present
Section Manager (first line supervisor) for nuclear plant Probabilistic Risk Assessment group which provides PRA models and PSA applications support for Duke Power's three nuclear power stations.

Title of Position: Senior Engineer
Department/Company: Nuclear Generation Department / Duke Power Company
From: 6/90 To: 6/99
Lead engineer for nuclear plant Probabilistic Risk Assessment group which provides PSA applications support and develops system and accident models. Team Leader for Catawba Probabilistic Risk Analysis project and IPE Submittal. Member of working group that developed the NUMARC Guidelines for Implementing the Maintenance Rule. Prepared and taught sessions of the NUMARC Maintenance Rule Workshops. Lead PRA analyst supporting Maintenance Rule implementation at Duke nuclear stations.

Title of Position: Design Engineer
Department/Company: Design Engineering Department / Duke Power Company
From: 6/84 To: 6/90
Lead engineer for group responsible for severe accident consequence analysis for McGuire, Catawba and Oconee Nuclear Stations. Performed extensive thermal/hydraulic computer analyses using the computer code MAAP to determine primary system and containment response to degraded core accidents. Also, have used CRAC2 and CRACIT to perform severe accident consequence modeling. Provided technical support to Industry Degraded Core Rulemaking Effort (IDCOR).

Title of Position: Engineer Associate
Department/Company: Design Engineering Department / Duke Power Company
From: 6/81 To: 6/84
Responsible for integrated safety analysis for McGuire, Catawba and Oconee Nuclear Stations. Coordinated review of nuclear power plant safety systems with Electrical, Mechanical and Civil engineering personnel to ensure an integrated, complete and safe design.

Title of Position: Assistant Engineer
Department/Company: Design Engineering Department / Duke Power Company
From: 7/79 To: 6/81
Responsible for effluent and radiation shielding analysis for McGuire, Catawba and Oconee Nuclear Stations. Designed radiation shielding for components and systems for nuclear plant systems. Performed radioactive effluent release analyses to ensure compliance with 10CFR100 and 10CFR50, Appendix I requirements.

Title of Position: Engineering Co-Op Student
Department/Company: Design Engineering Department / Duke Power Company
From: 8/75 To: 8/78
Assisted a group of engineers in performing effluent release and radiation dose analysis for McGuire, Catawba and Oconee Nuclear Stations.

Industry Support Activities

Sponsor	Description	Responsibilities	Dates
AIF	Sub-committee on Emergency Preparedness and Siting	Member	1984
AIF	Source Term Task Force	Chairman	1987-1988
EPRI	MAAP Users Group	Duke Representative/ Chairman	1987-1989 1990
EPRI	Safety Technology Task Force	Duke Representative	1989-1992
EPRI	Severe Accident Sub-Committee	Chairman / Duke Representative	1989-1992
NUMARC	Accident Management Ad Hoc Advisory Committee	Member	1989
NUMARC	Maintenance Rule Working Group	PRA Representative	1992-1996
EPRI	Risk and Reliability Workstation Users Group	Chairman / Duke Representative	12/93 - 12/95
B&WOG	Risk Informed Applications Committee	Duke Representative	1995-Present
WOG	Risk Based Technology Working Group	Past Chairman / Duke Representative	12/95 – 3/00
NEI	Risk Applications Task Force	WOG Representative	3/96 - Present
WANO	International Technical Exchange Peer	Peer Reviewer	1/6/97-1/10/97
INPO	Events Database Working Group	Member	4/97-12/97
INPO/MIT	Risk Informed Operational Decision Management Course	Guest Instructor	1/97-7/2001
ASME	BNCS Task Group on Risk Management for Nuclear Facility Application (Project Team)	Project Team Member	1/98 – 4/2002
ANS	PSA '99 Technical Program Committee	Committee Member	1998-1999
ASME	Committee on Nuclear Risk Analysis	Committee Member	1/98-Present
ASME	Sub-committee on PRA Technology	Chairman	3/2002-Present
ASME	Task Group on PRA Generic Database	Chairman	9/2002-Present
ANS	PSA '02 Technical Program Committee	Committee Member	2001-2002
EPRI	Risk/Safety Management Committee	Member	2004 - Present
ANS	Utility Working Conference Organizing Committee	Track Organizer	2004

Papers and Presentations

Sponsor	Description	Format	Date
EPRI Conference of Power Plant Simulators and Modeling	"MAAP-GRAAPH: A Nuclear Power Plant Severe Accident Simulation and Training Tool"	Presentation and Published Paper	June14,1988
NRC Containment Integrity Workshop	"Design Consideration for Severe Accident Containment Performance"	Co-Author, Published Paper	June 15, 1988
IAEA/ANL Interregional Training Course	"Use of PSA Optimization of Safety in Operation of Nuclear Power Plants", Module - "Use of PSA in Operator Training"	Classroom Lecture	1/21/93
ANS Annual Meeting	"The Role of PRA in Implementing the Maintenance Rule"	Presentation and Published Paper	6/21/93
ANS Annual Meeting	"Influence of Refueling Water Storage Tank Draining on Performance of the Catawba Nuclear Station Ice Condenser Containment"	Co-Author, Presentation and Published Paper	6/22/93
NUMARC Industry Workshop on Implementing the NRC Maintenance Rule	"Establishing Risk Significant Criteria" and "Risk Significant Systems Breakout Session"	Presentation / Training	8/5/93 and 8/23/93
ANS Utility Working Conference	"Operating Cycle Risk Management" Panel Discussion	Panel Member	8/14/95
EPRI ORSERG	"Applications of the IPE to ISEG Objectives"	Presentation	9/28/95
ACRS PSA Sub-Committee	"PSA Support for Maintenance Rule Implementation"	Presentation	10/26/95
ACRS PSA Sub-Committee	"PSA Model Methods and Results Comparison Program"	Presentation	7/18/96
ANS Utility Working Conference	"Risk Informed Regulation" Panel Discussion	Panel Discussion Organizer and Leader	8/5/96
Risk Informed Operational Decision Management Course Sponsored by INPO and MIT	"Application of PSA in Maintenance Rule Implementation", and "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications"	Guest Instructor	1997-2001

Reliability Engineering and System Safety	"Probabilistic Safety Assessment Support for the Maintenance Rule at Duke Power Company"	Scientific Journal	1998
PSA '99	"On-line Maintenance Issues and Insights" Panel Discussion	Panel Discussion Organizer and Leader	8/1999
ANS Utility Working Conference	"Maintenance Rule" Panel Discussion	Panel Discussion Organizer and Leader	8/2000
ANS Utility Working Conference	"License Renewal Requirements to Perform Severe Accident Mitigation Alternatives Analysis" Panel Discussion	Panel Discussion Organizer and Leader	8/2001
ANS Utility Working Conference	"The Risk-Informed Environment: Do We Think the Language" Panel Discussion	Panel Discussion Organizer and Leader	8/2002
PSA '02	"Nuclear Plant Safety Risk Management: A Case Study"	Co-Author, Published Paper and Presentation	10/2002
PSA '02	"PSA Standards" Panel Discussion	Panel Member	10/2002
PSA '02	"Present and Future PSA Challenges" Panel Discussion	Panel Member	10/2002
ANS Utility Working Conference	"Progress Toward a Risk-informed Environment" Panel Discussion	Panel Discussion Organizer and Leader	8/3/2003

ROBERT E. HENRY

EDUCATION:

University of Notre Dame	B.S. Mechanical Engineering, 1962
University of Notre Dame	M.S. Mechanical Engineering, 1964
University of Notre Dame	Ph.D. Mechanical Engineering, 1967

EXPERIENCE:

Fauske & Associates, Inc., (1980 to present), Burr Ridge, Illinois.

Senior Vice President and co-founder of FAI in 1980, Dr. Henry was involved in the Zion and Indian Point Probabilistic Safety Studies as well as the Limerick Probabilistic Risk Assessment. More recently, Dr. Henry made numerous contributions to the Industry Degraded Core Rulemaking Program in which FAI was responsible for initiating the development of the Modular Accident Analysis Program (MAAP) codes to analyze the response of pressurized and boiling light water reactors during severe accidents. The MAAP code has gained widespread acceptance in the domestic and foreign nuclear industry. This computer code specifically treats such phenomena as concrete attack, hydrogen formation, distribution and combustion within the containment, debris distribution, debris bed coolability, and fission product behavior. Dr. Henry was also the Manager for the Issue Resolution and Individual Plant Evaluation Tasks in the IDCOR Phase II Program which culminated with the Individual Plant Evaluation Methodology (IPEM). During the IDCOR Program, he was one of the industry representatives in the U.S. delegation to IAEA/Vienna to evaluate the Russian interpretation of the Chernobyl Unit 4 accident. Following Chernobyl, Dr. Henry served on review panels for both the N-Reactor and the Savannah reactors with respect to potential severe accident conditions.

Dr. Henry developed the Technical Basis Report for the Severe Accident Management Guidelines (SAMGs) that are being developed by the four owner's groups in the United States. As part of this, Version 4 of the MAAP code (MAAP4) was developed to provide analytical support for SAMG evaluations. He has also served on several NRC ad hoc committees for resolution of issues such as scaling, steam explosions, core melt relocation and direct containment heating. He is the author of approximately 90 journal articles as well as numerous other reports and abstracts.

Recently Dr. Henry has promoted dynamic benchmarking of computer codes as a means of preserving insightful experimental data in a form that will remain abreast of the changing computer technology. Through such dynamic benchmarking with the large number of existing experiments, computer codes can be compared on a common basis and thoroughly qualified for their uses in supporting engineering analyses.

Dr. Henry has also initiated development of the MAAP5 code. This is focused on providing realistic, yet conservative, evaluations of the core, Reactor Coolant System and containment responses under postulated accident conditions.

Argonne National Laboratory, (1969 to 1980), Argonne, Illinois.

Dr. Henry held a number of responsible research and development positions at ANL. In March, 1979, he was appointed Associate Director of the Reactor Analysis and Safety Division, and was involved in the evaluation of the Three Mile Island-2 accident as part of the group formed by the Electric Power Research Institute's Nuclear Safety Analysis Center (NSAC).

OTHER EXPERIENCE:

Member: American Nuclear Society

Chairman: Mechanical Engineering Department
Midwest College of Engineering (1973-1978)

Dean: Graduate School
Midwest College of Engineering (1978-1979)

Non-Resident Research Associate: Massachusetts Institute of Technology

TECHNICAL AWARDS:

"Award for Outstanding Engineering Accomplishment", College of Engineering, University of Notre Dame (1990).

"Tommy" Thompson Award: The highest honor the American Nuclear Society gives in the field of reactor safety (1985).

OPEN LITERATURE PUBLICATIONS IN THE LAST 10 YEARS:

"Response of the Davis-Besse Backup Boron Mixing System," Paper presented at the 14th Pacific Basin Nuclear Conference, Honolulu, Hawaii, March 23-25, 2004.

"CVTR Revisited," Paper presented at the 14th Pacific Basin Nuclear Conference, Honolulu, Hawaii, March 23-25, 2004.

"Benchmarking of LOFT ATWS Experiments With MAAP4 Using a Point Kinetics Model," Paper presented at the 14th Pacific Basin Nuclear Conference, Honolulu, Hawaii, March 23-25, 2004.

"Waterhammer Experiments in Support of a Feedwater Leakage Control System," Paper presented at the 16th International Conference on Structural Mechanics in Reactor Technology (SMIRT-16), Washington, DC, August, 2001.

"The Role of Expanded Metal Network in Preventing BLEVES," Paper presented at the AIChE 2001 Spring National Meeting, Fifth Bi-Annual Process Plant Safety Symposium, April 22-26.

"Best Estimate Containment Assessment," Paper presented at the International Meeting on "Best-Estimate" Methods in Nuclear Installation Safety Analysis (BE-2000), Washington, DC, November, 2000.

"Dynamic Benchmarking of TREMOLO – A Program for Pipeline Two-Phase Flow Transient Analysis," Paper presented at the Ninth International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-9), San Francisco, California, October 3-8, 1999.

"Data Preservation and Dynamic Benchmarking," Paper presented at the ANS 1998 Winter Meeting and the Meeting of the Americas, Washington, DC, November 15-19, 1998.

"Waterhammer in Horizontal Lines During the Voiding Phase," Paper presented at the ANS 1998 Winter Meeting and the Meeting of the Americas, Washington, DC, November 15-19, 1998.

"Two-Phase Critical Flow Through Control Valves," Paper presented at the ANS 1998 Winter Meeting and the Meeting of the Americas, Washington, DC, November 15-19, 1998.

"An Assessment of a Fuel Channel Rupture Within a Calandria Tank in Candu Reactors," Paper presented at the 5th International Conference on Nuclear Engineering, ICONE5, May 26-30, 1997, Nice, France.

"Is Fuel Fragmentation Needed to Understand the KROTOS Experiments?," Paper presented at the 5th International Conference on Nuclear Engineering, ICONE5, May 26-30, 1997, Nice, France.

"Dynamic Benchmarking of Simulation Codes," Paper presented at the Canadian Nuclear Society Fifth International Conference on Simulation Methods in Nuclear Engineering, Montreal, Canada, September 8-11, 1996.

"Quenching of Metal Surfaces in a Narrow Annular Gap," Paper presented at the Canadian Nuclear Society Fifth International Conference on Simulation Methods in Nuclear Engineering, Montreal, Canada, September 8-11, 1996.

"Application of Uncertainty Analyses with the MAAP4 Code," Paper presented at the Canadian Nuclear Society Fifth International Conference on Simulation Methods in Nuclear Engineering, Montreal, Canada, September 8-11, 1996.

"Water Level Instrumentation Simulation," Paper presented at the Canadian Nuclear Society Fifth International Conference on Simulation Methods in Nuclear Engineering, Montreal, Canada, September 8-11, 1996.

"MAAP4 Uncertainty and Sensitivity Analyses," Paper presented at the 1996 ANS Annual Meeting, Reno, Nevada, June 16-20, 1996.

"MAAP4 Dynamic Benchmarking," Paper presented at the 1996 ANS Annual Meeting, Reno, Nevada, June 16-20, 1996.

"A Different Approach to Fragmentation in Steam Explosions," Paper presented at the Fourth International Conference on Nuclear Engineering (ICONE-4), New Orleans, Louisiana, March 10-14, 1996.

"How to Use Expert Judgement to Assess Uncertainties," Paper presented at PSA'95, Seoul, Korea, November, 1995.

"Formulation and Use of Uncertainty and Sensitivity Analyses," Paper presented at PSA'95, Seoul, Korea, November, 1995.

"MAAP4.0 Benchmarking with the TMI-2 Experience," Paper presented at PSA'95, Seoul, Korea, November, 1995.

"MARS Application and Benchmarking," Paper presented at PSA'95, Seoul, Korea, November, 1995.

"Overview: Uncertainties Remaining in Severe Accident Phenomenology," Paper presented at the OECD Specialist Meeting on Severe Accident Management Implementation, Niantic, Connecticut, June 12-14, 1995.

"Experiments on the Lower Plenum Response During a Severe Accident," Paper presented at ANS Topical Meeting on Thermal-Hydraulics, Taipei, Taiwan, 1994.

"Water in the RPV: A Mechanism for Cooling Debris in the RPV Lower Head," Paper presented at the OECD Specialist Meeting on Selected Severe Accident Management Strategies, Stockholm, 1994 and also submitted to Nuclear Engineering and Design for publication.

"Cooling of Core Debris Within a Reactor Vessel Lower Head with Integral Support Skirt," 9th Proc. of Nuclear Thermal-Hydraulics, 1993 ANS Winter Meeting, pp. 92-100.

"External Cooling of a Reactor Vessel Under Severe Accident Conditions," Nuclear Engineering and Design, 1993, Vol. 139, pp. 31-43.

"Externally Triggered Steam Explosion Experiments: Amplification or Propagation?," Paper presented at the CSNI-FCI Specialist Meeting, Santa Barbara, CA, 1993.

"Integrated Modeling of Debris Bed Heat and Mass Transfer in the PWR Lower Plenum for Severe Accident Analysis," 9th Proc. of Nuclear Thermal-Hydraulics, 1993 ANS Winter Meeting, pp. 38-49.

- 81 Publications Prior to the Above

IDCOR Reports

"MAAP User's Manual," IDCOR Technical Tasks 16.2 and 16.3, (January, 1985).

"MAAP Uncertainty Analyses," IDCOR Technical Task 23.4, (March, 1985).

"Evaluations of Containment Bypass and Failure to Isolate Sequences for the IDCOR Reference Plants," IDCOR Technical Task 23.5, (April, 1985).

"Sequoyah Nuclear Power Plant, Integrated Containment Analysis," IDCOR Technical Task 23.1, (1985).

"Zion Nuclear Generating Station, Integrated Containment Analysis," IDCOR Technical Task 23.1, (1985).

"Peach Bottom Atomic Power Station, Integrated Containment Analysis," IDCOR Technical Task 23.1, (1985).

"Grand Gulf Nuclear Station, Integrated Containment Analysis," IDCOR Technical Task 23.1, (1985).

"Technical Support for Issue Resolution," IDCOR Technical Task 85.2, (1985).

"The Attainment of a Safe Stable State for a Damaged Core," IDCOR Technical Task 22.1, (January, 1984).

"FAI Aerosol Correlation," IDCOR Technical Task 11.6, (August, 1984).

"Final Report on In-Vessel Core Melt Progression Phenomena," IDCOR Technical Task 15.1, (January, 1983).

"Final Report on Debris Coolability, Vessel Penetration and Debris Dispersal," IDCOR Technical Task 15.2, (January, 1983).

"Final Report on Core-Concrete Interactions," IDCOR Technical Task 15.3, (March, 1983).

"Final Report of Key Phenomenological Models for Assessing Explosive Steam Generation Rates," IDCOR Technical Task 14.1, (May, 1983).

"Final Report of Key Phenomenological Models for Assessing Non-Explosive Steam Generation Rates," IDCOR Technical Task 14.1, (June, 1983).

"Hydrogen Generation During Severe Core Damage Sequences," IDCOR Technical Task 12.1, (June, 1982).

Attachment 3

Index to Documents Produced in Response To BREDL Request

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
Isotopics Summary File for FHA BOC Fuel Assembly Activities	Excel		000001	000036	I-1	
Radionuclide Source Terms for an Environmental Critique of Reactor-Based Plutonium Disposition	Article		000037	000053	II-4	
draft NRC RAI Response	Word Documents	9/16/2003	000054	000058	I-1	
F-ANP input in LAR LOCA section	Word Documents	5/1/2002	000059	000068	I-1	
source team panel meeting	E-mail Hardcopy	3/20/2002	000069	000073	II-6	
	E-mail Hardcopy	2/20/2002	000074	000078	II-6	
MOX Fuel Project Overview	PowerPoint Handout	12/13/2001	000079	000119	I-1	
Overview of NRC Fission Product Research	Word Documents	9/23/2002	000120	000132	II-5	
MOX LEU Differences	Articles	noted on document	000133	000197	I-1	
Draft RAI Response	Word Documents		000198	000211	I-1	
Framatome Letter	Letter	5/30/2003	000212	000213	I-1	
Draft RAI Response	Word Documents		000214	000235	I-1	
Draft RAI Response	Word Documents		000236	000266	I-1	
Draft RAI Response	Word Documents	7/25/2003	000267	000296	I-1	
Draft NRC Submittal	Word Documents	3/16/2003	000297	000303	I-1	
Supplement to the Surplus Plutonium Disposition	US DOE document	Apr-99	000304	000343	II-6	X

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
draft NRC RAI Response	Word Documents	7/25/2003	000344	000349	I-1	
Engineering Calculation	Form	1/7/2004	000350	000399	II-4	
SBO Uncertainty Estimates	E-mail and document stapled to e-mal	10/10/2002	000400	000409	II-1	X
	E-mail Hardcopy	6/27/2002	000410	000411	II-6	
	E-mail Hardcopy and Hand written notes	2/22/2002	000412	000421	II-6	
disposition of comments	PowerPoint Handout	6/18/2002	000422	000451	II-6	
comments on draft alternate source term report	Word Documents	27-Mar-02	000452	000456	II-6	
comments on accident source terms	Excel		000457	000460	II-6	
draft panel report	E-mail Hardcopy	3/15/2002	000461	000546	II-6	
Phone Conversation w/ Schaperow	Hand written Notes	10/24/2002	000547	000547	II-6	
MOX Fuel and Cladding Temperatures During LOCA Compared to Severe Accident Fuel Melt Temperatures	Chart		000548	000548	I-1	
MOX Monthly Management Report	Word Documents	Jan-00	000549	000559	I-1	
MOX 4/9 Telecon	Hand written Notes		000560	000560	I-1	
MOX Quarterly Review Mtg	PowerPoint Handout	12/17/2002	000561	000563	I-1	
to John R. Biller	Letter	8/20/2004	000564	000568	I-1	
Questions 14, 15 & 16	Word Documents		000569	000572	I-1	
Questions 14, 15 & 16	Word Documents		000573	000576	I-1	
Questions 14, 15 & 16	Word Documents		000577	000580	I-1	

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Questions 14, 15 & 16	Word Documents		000581	000584	I-1	
Question 47	Word Documents		000585	000589	I-1	
Questions 14, 15 & 16	Word Documents		000590	000594	I-1	
Attachment 1 Partial Response to NRC Staff Request for Additional Information	Word Documents	7/25/2003	000595	000595	I-1	
to Jason Schaperow	Letter	3/13/2001	000639	000642	II-6	
Technical Paper: EdF Position on Radiological Consequences of MOX Fuel Accidents	Word Documents	1999	000643	000648	II-6	
NRC Confirmatory Research	PowerPoint Handout	10/4/2001	000649	000658	II-6	
Potential Research Needs	Word Documents	2/20/2002	000659	000661	II-6	
draft notes on PRA work	Word Documents	11/17/2003	000662	000665	II-6	
draft to Jason Schaperow	Letter	3/31/2001	000666	000669	II-6	
Comments on Draft of NRC Request for Additional Information	Word Documents	9/30/2003	000670	000682	I-1	
Isotopics Summary File for FHA BOC Fuel Assembly Activities	Excel		000683	000718	I-1	
Isotopics Summary File for FHA BOC Fuel Assembly Activities	Excel		000719	000754	I-1	
Isotopics Summary File for FHA BOC Fuel Assembly Activities	Excel		000755	000790	I-1	
Letter from WR McCollum to NRC	Letter	3/16/2004	000791	000796	I-1	
C1C16 Core Design	PowerPoint Handout	3/25/2004	000797	000807	I-1	X

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
Draft Attachment 3 Description and Technical Justification	Word Documents		000808	000862	I-1	
Draft Attachment 3 Description and Technical Justification	Word Documents		000863	000916	I-1	
Attachment 3 Description and Technical Justification	Word Documents		000917	000953	I-1	
Analysis Input Summary	Word Documents	Jan-03	000954	000973	I-1	
to U.S. NRC	Letter	2/27/2003	000974	000990	I-1	
Outline for Landry Telephone Call	Word Documents		000991	000995	I-1	
To Steve with Charts	Letter		000996	000999	I-1	
Comments on Section 6.1.1	Word Documents		001000	001002	I-1	
Question 24	Word Documents		001003	001015	I-1	
Draft ANP 02 Section 6.1.3	Word Documents		001016	001025	I-1	
Questions 14, 15 & 16	Word Documents		001026	001028	I-1	
	Word Documents		001029	001045	I-1	
Questions 14, 15 & 16	Word Documents		001046	001050	I-1	
Questions 14, 15 & 16	Word Documents		001051	001055	I-1	
draft NRC RAI Response	Word Documents	10/15/2003	001056	001058	I-1	
draft NRC RAI Response	Word Documents	10/15/2003	001059	001066	I-1	
draft NRC RAI Response	Word Documents	10/16/2003	001067	001072	I-1	
draft NRC RAI Response	Word Documents	10/30/2003	001073	001083	I-1	
draft NRC RAI Response	Word Documents	11/1/2003	001084	001094	I-1	
draft NRC RAI	Word	9/22/2003	001095	001099	I-1	

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
Response	Documents					
draft NRC RAI Response	Word Documents	9/22/2003	001100	001104	I-1	
draft NRC RAI Response	Word Documents	9/19/2003	001105	001109	I-1	
draft NRC RAI Response	Word Documents	9/19/2003	001110	001110	I-1	
draft NRC RAI Response	Word Documents	9/15/2003	001111	001115	I-1	
Mixed Oxide Fuel Project Probabilistic Risk Assessment Work	Word Documents	11/17/2003	001116	001119	II-1	X
5.0 Environmental Impacts of Postulated Accidents	NUREG-1437, Supplement 9	May-02	001120	001150	II-1	
Document of Numbers	Spreadsheet	9/27/2000	001151	001168	II-1	X
Severe Accident Release Fractions	Word Documents	1/15/2004	001169	001170	II-2	
Assessment of Differences Between MOX Fuel and Conventional Uranium Fuel	Word Documents	Mar-00	001171	001200	I-1 Contention II	
Rough Calculations by Verbos & Nesbit related to LOCA dose analysis	Hand written Notes		001201	001208	I-1 Contention II	
Ginna MOX SER pages	Typed document		001209	001211	I-1 Contention II	
Comments on Draft of NRC Request for Additional Information	Word Documents	9/1/2003	001212	001215	I-1	
From W.F. Jones, MOX Fuel Contractor -- LOCA Section for MOX LAR with Proprietary Info removed	Letter	9/5/2002	001216	001239	I-1	
Notes from initial Duke-F-ANP LOCA meeting	Hand written Notes	10/30/2000	001240	001256	I-1	

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
Skip Copp Notes from Duke-F-ANP Meeting	Hand written Notes on Word Document	10/31/2000	001257	001269	I-1	
draft NRC RAI Response	Word Documents	9/29/2003	001270	001276	I-1	
draft NRC RAI Response	Word Documents	10/10/2003	001277	001283	I-1	
draft NRC RAI Response	Word Documents	10/14/2003	001284	001289	I-1	
Figure Q11-3 Fast Neutron Fluence Comparison	Hand written Notes on Word Document		001290	001304	I-1	
Attachment 1 Partial Response to NRC Staff Request for Additional information	Word Documents		001305	001346	I-1	
Mox Quarterly Meeting	Word Documents	8/3/2000	001347	001349	I-1	
MOX Quarterly Review Mtg	PowerPoint Handout	12/17/2002	001350	001352	I-1	
Reactor Systems Draft	Word Documents		001353	001406	I-1	
Reactor Systems Draft	Word Documents		001407	001460	I-1	
Draft to US N.R.C. with attachments	Letter	Sep-00	001461	001594	I-1	
Attachment 6 Request for Exemptions from Certain Provisions of 10 CFR Part 50	Word Documents		001595	001605	I-1	
Draft license amendment submittal	Word Documents	5/6/2002	001606	001775	I-1	
Reactor Systems Draft	Word Documents		001776	001829	I-1	
Certification of Engineering Calculation -- MOX Isotopic Inventory Calculation	Form	2/22/2002	001830	002176	II-6	
Cr2b MAAP Runs for Quantification	Notebook		002177	002898	II-4	X

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
Calculation Summary Sheet	Form	1/7/2004	002313	006316	II-1	X
Attachment 1 Partial Response to NRC Staff Request for Additional information	Word Documents		002899	002943	I-1 Contention I	
Responses to Harvey Questions	Word Documents		002944	002945	I-1 Contention I	
Question 24	Word Documents		002946	002951	I-1 Contention I	
Draft	Word Documents		002952	002962	I-1 Contention 1	
	e-mail hard copy	6/3/2002	002963	002963	I-1 Contention I	
	e-mail hard copy	10/31/2002	002964	002964	I-1 Contention I	
	e-mail hard copy	10/31/2003	002965	002965	I-1 Contention I	
	Word Documents		002966	002976	I-1 Contention I	
	e-mail hard copy	5/3/2002	002977	002978	I-1 Contention I	
	e-mail hard copy	5/1/2002	002979	002979	I-1 Contention I	
	e-mail hard copy	5/5/2002	002980	002981	I-1 Contention I	
	e-mail hard copy	5/2/2002	002982	002983	I-1 Contention I	
	e-mail hard copy	2/23/2004	002984	002990	I-1	
from George Meyer	Letter	3/5/2004	002991	002992	I-1	
	e-mail hard copy	9/30/2003	002993	002994	I-1 Contention I	
	e-mail hard copy	9/23/2003	002995	002995	I-1 Contention II	
	e-mail hard copy	9/19/2003	002996	002996	I-1 Contention I	
	e-mail hard copy	10/23/2003	002997	002997	I-1 Contention	

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
					II	
	e-mail hard copy	10/24/2003	002998	002998	I-1 Contention II	
	e-mail hard copy	1/9/2002	002999	002999	I-1 Contention I	
	e-mail hard copy	1/9/2002	003000	003001	I-1 Contention I	
	e-mail hard copy	1/9/2002	003002	003003	I-1 Contention I	
	e-mail hard copy	1/9/2002	003004	003005	I-1 Contention I	
	e-mail hard copy	1/16/2002	003006	003006	I-1 Contention I	
	e-mail hard copy	2/25/2002	003007	003007	I-1 Contention I	
	e-mail hard copy	3/6/2002	003008	003009	I-1 Contention I	
	e-mail hard copy	3/8/2002	003010	003010	I-1 Contention I	
	e-mail hard copy	4/2/2002	003011	003011	I-1 Contention I	
	e-mail hard copy	4/3/2002	003012	003012	I-1 Contention I	
MOX Fuel and Cladding Temperatures During LOCA Compared to Severe Accident Fuel Melt Temperatures	Chart		003013	03013	I-1	
	e-mail hard copy	4/9/2002	003014	003014	I-1 Contention I	
	e-mail hard copy	10/16/2003	003015	003018	I-1	
License Amendment Request	Letter	2/27/2003	003019	003156	I-1 Contention I and II	
Westinghouse Fuel and LOCA Evaluation of R.E. Ginna	article	Dec-79	003157	003166	I-1 Contention I	

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
Draft RAI Response	Word Documents		003167	003207	I-1 Contention I	
Draft RAI Response	Word Documents		003208	003270	I-1 Contention I	
Draft RAI Response	Word Documents		003271	003314	I-1 Contention I	
MOX Lead Assembly Licensing Premise	Word Documents		003315	003315	I-1 Contention I	
	Word Documents		003316	003318	I-1 Contention I	
	Word Documents		003319	003349	I-1 Contention II	
Question 25	Word Documents		003350	003356	I-1 Contention I	
Question 35	Word Documents		003357	003358	I-1 Contention I	
Question 47	Word Documents		003359	003363	I-1 Contention I	
Response to Request for Additional Information	Word Documents		003364	003484	I-1 Contention I and II	
Response to NRC Staff Request for Additional Information	Word Documents		003485	003599	I-1 Contention I and II	
Attachment 3 Description and Technical Justification	Word Documents		003600	003609	I-1 Contention I	
Attachment 6 Request for Exemption	Word Documents		003610	003620	I-1 Contention I	
Attachment 3 Description and Technical Justification	Word Documents		003621	003707	I-1 Contention I and II, II-5	
Attachment 3 Description and Technical Justification	Word Documents		003708	003794	I-1 Contention I and II, II-5	
Draft Responses to Request for Additional Information	Word Documents		003795	003930	I-1 Contention I and II, II-5	
C1C16 Core Design with MOX LTA	PowerPoint Handout	3/25/2004	003931	003965	I-1 Contention I	

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
MOX Fuel Lead Assembly Safety and Environmental Evaluations	PowerPoint Handout	4/21/2004	003966	004007	I-1 Contention I and II	
Draft RAI Response	Word Documents	3/16/2003	004008	004014	I-1 Contention II	
Draft MOX LAR	Word Documents		004015	004103	I-1 Contention I and II, II-5	
Severe Accident Analysis Report	Notebook		004104	004396	II-2	X
SAAG File NO. 726	Report Binder		004397	004429	II-1	X
Documents of Numbers -- CNS MOX Analysis	Excel		004430	004580	II-1, II-5	X
Barrett Misc. Files and SNS PRA Rev. 2b Risk Spreadsheet	Disc		004581	004581	II-1	X
Rem Percentiles	Word Documents		004582	004586	II-1	X
To Steve Nesbit	Letter	1/31/2003	004587	004600	I-1	X
Calculation Summary Sheet	Form	7/30/2003	004601	004629	I-1	X
Draft Analysis Input Summary	Word Documents	Sep-02	004630	004654	I-1	X
Draft Duke Responses to NRC RAIs for Catawba SAMA	Word Documents	1/4/2002	004655	004682	II-1	X
From HB Barron	Letter with Attachments	11/3/2003	004683	004803	I-1 Contentions I and II	X
Summary of North Anna Licensing Needs	Bullets, notes		004804	004820	I-1	X
MOX Quarterly Review Mtg	PowerPoint Handout	12/17/2002	004821	004823	I-1	X
Partial Response to NRC Staff Request for Additional Information	Word Documents		004824	004867	I-1	X
Summary table for various SAMA	Excel		004868	004872	II-1	X

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
comparisons						
	e-mail hardcopy	9/7/2001	004873	004873	II-1	X
Effect of High Burn-up and MOX Fuel on Fission Product Release from VERCORS Tests	PowerPoint Handout	Jun-03	004874	004899	II-6	X
	Word Documents		004900	004907	I-1 Contention I	X
Summary of Modification Sensitivity Study Results	Word Documents		004908	004911	II-1	X
Summary of Modification Sensitivity Study Results	Word Documents		004912	004914	II-1	X
Fission Product Release and Fuel Behavior	Article		004915	004930	II-6	
Values for sob Contribution to Core Damage Frequency	Table		004931	004943	II-1	X
Engineering Information Record: LOCA Section for MOX LAR	Form		004944	004966	I-1	X
to NRC with attachments	Letter	12/10/2003	004967	004992	I-1 Contention II	X
Reactor System Questions	Word Documents		004993	005107	I-1 Contentions I and II	X
Duke Response to NRC RAI	Letter with Attachments	11/3/2003	005108	005344	I-1 Contentions I and II	X
Duke Response to NRC RAI	Letter with Attachments	11/3/2003	005345	005453	I-1 Contentions I and II, II-5	X
	E-mail Hardcopy	2/16/1999	005454	005454	II-5	X
LOCA Section for MOX LAR	Letter	3/22/2002	005455	005490	I-1	X

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
Calculation Summary Sheet	Form	7/30/2003	005491	005519	I-1	X
MOX Fuel Data for LOCA Calculations	Letter	3/7/2002	005520	005521	I-1	X
Physics Parameters Comparison	Memo	1/24/2002	005522	005525	I-1	X
Certification for Engineering Calculation	Form	3/20/2001	005526	005587	I-1	X
Proposed Procedure for Calculating Radionuclide Inventories for PRA Input	Word Documents	.11/12/2003	005588	005589	II-4	X
Best Estimate Fuel Batch Data	Internal Memo	12/3/2003	005590	005599	II-4	X
	e-mail hardcopy and notes	11/24/2003	005600	005618	II-4	X
Release fraction study	Hand written Notes		005619	005621	II-4	X
from W.F. Jones	Letter with Attachments	9/25/2002	005622	005650	I-1	X
Calculation Summary Sheet	Form	1/16/2004	005651	005686	I-1	X
Severe Accident Analysis Report	Word Documents	5/22/2001	005687	005705	II-1	X
Severe Accident Information	Spreadsheets		005706	005926	II_1	X
Accident Analysis	Word Documents		005927	006208	II-1	X
Duke Response to NRC RAI	Word Documents	1/4/2002	006209	006236	II-1	X
Conditional Person Rem	Word Documents		006237	006240	II-1	X
	Hand written note and article		006241	006312	II-1	X
from W.F. Jones	Letter with Attachments	3/22/2002	006317	006347	II-1	X
MAAP run CR2B20AS.Sum	Notebook		006348	006865	II-1	X
	Notebook		006866	007500	II-1	X

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
ORNL and SAIC Re-Work of Accident Consequences	Internal Memo		007501	007516	II-1	X
	e-mail hardcopy	2/16/1999	007517	007519	II-1	X
Risk Impact of MOX Fuel Lead Assemblies	Word Documents		007520	007520	II-1	X
Draft LAR	Word Documents	5/6/2002	007521	007690	II-1	X
	Word Documents		007691	007693	II-1	X
	Word Documents		007694	007696	II-1	X
	Word Documents		007697	007698	II-1	X
Duke Responses to NRC RAIs	Word Documents		007699	007726	II-1	X
CNS_State.xls;Land Fraction	Excel	4/7/2004	007727	007751	II-1	X
CNS Pop Data	Excel	4/7/2004	007752	007765	II-1	X
cns_calc.xls:State Information	Excel	4/7/2004	007766	007778	II-1	X
CNSSFRAC-- State sector fractions	Excel		007779	007875	II-1	X
CNS_ISO.xls	Excel		007876	007877	II-1	X
Release Category Frequency Matrix	Excel		007878	007916	II-1	X
Summary of Risk Results	Excel		007917	007949	II-1	X
Rem Chart	Excel		007950	007950	II-1	X
McGuire and Catawba Nuclear Station PRA System Documentation Level 2 & 3 Analysis	Notebook		007951	008504	II-1	X
Public Health risks of Substituting Mixed-Oxide for Uranium Fuel in Pressurized Water Reactors	Article		008505	008528	II-6	

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
Public Health risks of Substituting Mixed-Oxide for Uranium Fuel in Pressurized Water Reactors	Article	1/20/1999	008529	008582	II-6	
	Hand written note and article		008583	008596	II-6	
Cover Memo and NRC Document			008597	008618	II-6	
Impact of MOX Fuel Lead Assemblies on LOCA	Word Documents		008621	008663	II-1	
Impact of MOX Fuel Lead Assemblies on LOCA	Word Documents		008664	008674	I-1	
MOX Fuel Lead Assembly License Amendment Request	Word Documents	10/9/2002	008675	008682	I-1	
	Article	2/1/1999	008679	008620	II-6	
Contention 10	Word Documents		008683	008683	I-1	
Draft Document	Word Documents		008684	008698	I-1	
Comments on Section 6.1.1	Word Documents		008699	008701	I-1	
6.1 Impact of MOX Fuel Lead Assemblies on LOCA Analyses	Word Documents		008702	008701	I-1	
6.1 Impact of MOX Fuel Lead Assemblies on LOCA Analyses	Word Documents		008711	008719	I-1	
Attachment 3 Description and Technical Justification	Word Documents		008720	008722	I-1	
6.1 Impact of MOX Fuel Lead Assemblies on LOCA Analyses	Word Documents		008723	008730	I-1	
Attachment 3 Description and Technical Justification	Word Documents		008731	008739-	I-1	

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
6.1 Impact of MOX Fuel Lead Assemblies on LOCA Analyses	Word Documents		008740	008765	I-1	
Draft Document	Word Documents		008766	008818	I-1	
Attachment I Partial Request to NRC Staff Request for Additional Information	Word Documents		008819	008848	I-1 Contention II	
Cr2b MAAP Runs for Quantification	Diskette		Disc 002177-002898	Disc 002177-002898	II-4	X
SAAG No. 726 Supporting Files	Disc		Disc 004397-004429	Disc 004397-004429	II-1	X
CNS MOX Analysis 1999	Disc		Disc 004430-004580	Disc 004430-004580	II-1, II-5	X
GSI 189 Discs	Disc		Disc 004582-004586	Disc 004582-004586	II-1	X
DPC-1607.00-00-0008	Disc		Disc 005526-005587	Disc 005526-005587	I-1	X
CNS_FTB_Cost SAAG #669	Disc		Disc 005687-005705	Disc 005687-005705	II-1	X
SAAG #586 CNS MACCS2 Model	Diskette		Disc A 004104-004396	Disc A 004104-004396	II-2	X
SAMA RAIs Supporting files	Disc		Disc A 005927-006208	Disc A 005927-006208	II-1	X
SAAG #586 CNS MACCS2 Model	Diskette		Disc B 004104-004396	Disc A 004104-004396	II-2	X
Same Files	Disc		Disc B 005927-006208	Disc B 005927-006208	II-1	X

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
SAAG #586 CNS MACCS2 Model	Diskette		Disc C004104-004396	Disc A 004104-004396	II-2	X
Technical Reports Series No. 415	Public Document	2003	Not Produced -- Publicly Available		I-1 Contention I and II	
License Amendment Request	Public Document	2/27/2003	Not Produced -- Publicly Available		I-1 Contention I and II	
Letter from HB Barron to NRC	Public Document	3/26/2004	Not Produced -- Publicly Available		I-1 Contention II	
Letter from WR McCollum to NRC	Public Document	3/16/2003	Not Produced -- Publicly Available		I-1 Contention II	
Letter from WR McCollum to NRC	Public Document	2/2/2004	Not Produced -- Publicly Available		I-1 Contention II	
Accident Source Terms for Light-Water Nuclear Power Plants: High Burnup and Mixed Oxide Fuels	Public Document	Dec-02	Not Produced -- Publicly Available		II-6, I-1 Contention II	
Request for Additional Information on Application for MOX Lead Test Assemblies Duke Power Co.	Public Document		Not Produced -- Publicly Available		I-1 Contention II	
NRC Memo on Plan for Confirmatory Research Associated with the Use of MOX Fuel	Public Document	2/28/2000	Not Produced -- Publicly Available		I-1 Contention I and II	
NRC Memo on Plan for Confirmatory Research Associated with the Use of MOX Fuel	Public Document	2/11/2002	Not Produced -- Publicly Available		II-6, I-1 Contention II	
Summary December 12, 2000 Meeting Memo from NRC	Public Document	1/27/2001	Not Produced -- Publicly Available.		I-1 Contention I and II	

Document Description	Document Type (letter, calc, etc.)	Date of Document	Beginning Bates Number	Ending Bates Number	Responsive To	Proprietary
Draft NUREG-1437, Supplement 9	Public Document	May-00	Not Produced -- Publicly Available. This copy contains a few identifying marks and minimal underlining, none of which reflect substantive thoughts		II-1	
2/28/00 memo transmitting 2/11/00 NRC Memo on Agency Plan for Confirmatory Research Associated with the Use of Mixed-Oxide Fuel in Commercial Light Water Reactors	Public Document	2/28/00 and 2/11/00	Not Produced-- Available Publicly		II-6	
Technical Report Series NO. 415, IAEA Publication	Public Document		Not Produced-- Available Publicly		II-5	
NRC Panel on Source Term--IPSN presentation Part 1: Overview on French Studies on Source Term	Public Document	12/11/2001	Not Produced-- Available Publicly		II-6	
Note To File: Information Provided by Duke Energy corporation Related to Severe Accident Mitigation Alternatives in its License Renewal Application for the Catawba Nuclear Station, Units 1 and 2	Public Document	3/14/2002	Not Produced-- Available Publicly		II-1	

Attachment 4

PRIVILEGE LOG

Document Description	Document Type (letter, calc, etc.)	Date of Document	Owner	Privilege Claim
Risk Analysis Chart	Word Document		Steve Nesbit	Prepared at direction of counsel
Contention 2	Word Document		Michael Barrett	Attorney Work Product
Contention 2	Word Document	5/3/2003	Duncan Brewer	Attorney-Client and Attorney Work Product
Draft Brief	Word Document	2/25/2002	Duncan Brewer	Attorney Work Product
Draft Affidavit	Word Document	12/11/2002	Michael Barrett	Attorney Work Product
Comment Matrix On MOX Source Term Calc	Spreadsheet		Steve Schultz	Created at direction of Counsel
Best Estimate Fuel Batch Data	Word Document	12/3/2003	Franklin Verbos	Created in support of litigation per D. Repka
Mox LTA Level III Calc	Spreadsheet		Franklin Verbos	Created at direction of Counsel
E-mail and engineering calculation	E-mail and form	3/29/2004	Melissa Moran	Created at direction of Counsel
	Draft pleading	12/16/2003	Robert Harvey	Attorney Work Product
	E-mail	1/8/2004	Steve Nesbit	Legal Advice/Strategy for a pre-hearing conference
E-mail with attached document on answers to IR 03-022 Rev 1	E-mail	1/19/2004	Steve Nesbit	Legal Advice/Strategy related to preparing responses to contentions
Handwritten notes on conversations	Handwritten notes	12/8/2003	Steve Nesbit	Legal Advice/Strategy DAR

Document Description	Document Type (letter, calc, etc.)	Date of Document	Owner	Privilege Claim
with Ralph Meyer				
Screen print of index of Discovery 1 Privileged documents	Screen print	4/7/2004	Steve Nesbit	Claim of not discoverable. Does not reference question #
Evaluation of Consolidated Contentions	Word Document	3/8/2004	Steve Nesbit	Claim of not discoverable. Does not reference question #
BREDL's 1st Set of Discovery Requests to Duke	Word Document	3/31/2004	Steve Nesbit	Claim of not discoverable. Does not reference question #
Admissibility Insert	Word Document	3/11/2004	Steve Nesbit	Claim of not discoverable. Does not reference question #
NRC Support of Duke's Appeal (Draft)	Word Document	3/11/2004	Steve Nesbit	Claim of not discoverable. Does not reference question #
Duke Response to BREDL's Discovery	Word Document	4/2/2004	Steve Nesbit	Claim of not discoverable. Does not reference question #
Rev 1 of Technical / Factual Inputs to Respond to BREDL's Supplemental Petitions of 12/2/03	Word Document	12/11/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Technical / Factual Inputs to Respond to BREDL's Supplemental Petitions of	Word Document	12/8/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #

Document Description	Document Type (letter, calc, etc.)	Date of Document	Owner	Privilege Claim
12/2/03				
MOX Fuel Issues Matrix	Word Document	10/30/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Notes for Pre-Hearing Conference	Word Document	1/14/2004	Steve Nesbit	Claim of not discoverable. Does not reference question #
Rev 1 of Proposed BREDL Settlement	Word Document	3/29/2004	Steve Nesbit	Claim of not discoverable. Does not reference question #
Proposed BREDL Settlement	Word Document	12/4/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Response to BREDL's 2nd Supplemental Petition to Intervene	Word Document	12/16/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Response to BREDL's 2nd Supplemental Petition to Intervene red-lined by SPN	Word Document	12/17/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Response to BREDL's 2nd Supplemental Petition to Intervene red-lined by SPN & BMD	Word Document	12/18/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Rev 1 of Response to BREDL's 2nd Supplemental	Word Document	12/18/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #

Document Description	Document Type (letter, calc, etc.)	Date of Document	Owner	Privilege Claim
Petition to Intervene				
Rev 1 of Response to BREDL's 2nd Supplemental Petition to Intervene red-lined by SPN	Word Document	12/19/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Rev 2 of Response to BREDL's 2nd Supplemental Petition to Intervene	Word Document	12/20/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Rev 2 of Response to BREDL's 2nd Supplemental Petition to Intervene red-lined by SPN	Word Document	12/22/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Rev 3 of Response to BREDL's 2nd Supplemental Petition to Intervene	Word Document	12/22/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Rev 3 of Response to BREDL's 2nd Supplemental Petition to Intervene red-lined by SPN	Word Document	12/22/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #
Risk Perspective	Word Document	11/7/2003	Steve Nesbit	Claim of not discoverable. Does not reference question #

Document Description	Document Type (letter, calc, etc.)	Date of Document	Owner	Privilege Claim
HDB's Editorial Comments on NRC's Response of Duke to Proposed Late-filed Contentions	Hard Copy Document	6/10/2002	H.D.Brewer	B-4 (BREDL II-1)
Affidavit of H.Duncan Brewer	Hard Copy Document	12/11/2002	H.D.Brewer	Attorney Work-Product
MOX Intervenor Discovery Responses	Word Document	5/20/2002	Stephen Schultz	Input to Duke's response to intervenor's (NIRS) Request for discovery in the McGuire/Catawba License Renewal Case.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of "DUKE ENERGY CORPORATION'S RESPONSE TO BLUE RIDGE ENVIRONMENTAL DEFENSE LEAGUE'S FIRST DISCOVERY REQUEST," in the captioned proceeding, have been served on the following by deposit in the United States mail, first class, this 14th day of April, 2004. Additional e-mail service, designated by **, has been made this same day, as shown below.

Documents being provided in connection with this response have been delivered directly from Duke Energy Corporation by overnight courier for receipt this same day by counsel for Blue Ridge Environmental Defense League and the NRC Staff (identified by †).

Ann Marshall Young, Chairman**
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
(e-mail: AMY@nrc.gov)

Anthony J. Baratta**
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Atomic Safety and Licensing Board
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Thomas S. Elleman**
Administrative Judge
5207 Creedmoor Road, #101
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Office of the Secretary **
U.S. Nuclear Regulatory Commission
Washington, DC 20555
Attn: Rulemakings and Adjudications Staff
(original + two copies)
(e-mail: HEARINGDOCKET@nrc.gov)

Office of Commission Appellate Adjudication
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Adjudicatory File
Atomic Safety and Licensing Board Panel
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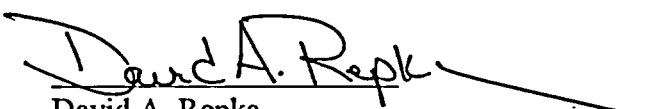

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