UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICE OF THE
GENERAL COUNSEL

July 1, 2004

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

Dear Administrative Judges:

In accordance with the Board's Order of May 25, 2004, the Staff submits the following documents for inclusion in the instant proceeding's evidentiary record:

1. "NRC Staff Testimony of Undine Shoop, Dr. Ralph Landry and Dr. Ralph O. Meyer concerning BREDL Contention I," with their attached statements of professional qualifications;

2. NRC Staff's Proposed Exhibit 1, "Safety Evaluation by the Office of Nuclear Reacotr Regulation Renewed Facility Operating License NPF-35 and NPF-52," April 5, 2004;

3. NRC Staff's Proposed Exhibit 2, Letter from M.S. Tuckman (Duke) to the NRC, "License Amendment to Request, Implementation of Best-Estimate Large Break Loss of Coolant Analysis Methodology," August 10, 2000;

4. NRC Staff's Proposed Exhibit 3, Y. Yan et al., "LOCA Test Results for High-Burnup BWR Fuel and Cladding," Organization for Economic Cooperation and Development (OECD) Topical Meeting on LOCA Issues, May 25-26, 2004;

5. NRC Staff's Proposed Exhibit 4, N. Waeckel et al., "Does M5 Balloon More that Zircaloy-4 Under LOCA Conditions?," OECD Topical Meeting on LOCA Issues, May 25-26, 2004;

6. NRC Staff's Proposed Exhibit 5, Memorandum from Farouk Eltawila, RES, to Suzanne C. Black, NRR, RE: Response to User Need for Development of Radiological Source Terms for Review of Mixed Oxide Fuel Lead Test Assemblies, February 23, 2004

Sincerely,

Antonio Fernández Rivera for
/Susan Uttal
Counsel for NRC Staff

Enclosures: As Stated

cc with encls: Service List

July 1, 2004

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF TESTIMONY OF UNDINE SHOOP, DR. RALPH LANDRY AND DR. RALPH O. MEYER CONCERNING BREDL CONTENTION I

Q1. Please state your name, occupation, and employer.

A1a. (US) My name is Undine Shoop . I am employed as a Reactor Systems Engineer in the office of Nuclear Reactor Regulation at the U.S. Nuclear Regulatory Commission (NRC). A statement of my professional qualifications is attached hereto.

A1b. (RL) My name is Ralph Landry, I am a Senior Reactor Engineer employed by the NRC in the Office of Nuclear Reactor Regulation. A statement of my professional qualifications is attached hereto.

A1c. (ROM) My name is Ralph O. Meyer. I am employed as a Senior Technical Advisor for Core Performance and Fuel Behavior in the Office of Nuclear Regulatory Research at the NRC. A statement of my professional qualifications is attached.

Q2. Please describe your current responsibilities.

A2a. (US) In my position as a Reactor Systems Engineer at the NRC, I currently serve as the lead fuels reviewer for several projects involving technical evaluation of fuel designs, in-reactor fuel use, and core components. This work includes reviewing new fuel designs, fuel transition methodologies, core component changes (such as control elements), fuel pellet modifications, fuel assembly component changes, and cladding material.

A2b. (RL) I am currently assigned responsibility for leadership in the reviews of the thermal hydraulic analysis computer codes. This includes review of the advanced computing methodologies, Appendix K methodologies, advanced nuclear reactor system design analyses, and specific Loss-of-Coolant Accident (LOCA) application analyses.

A2c. (ROM) I am responsible for the technical content of all of NRC's research on fuel behavior under conditions of design-basis accidents. This work is currently being performed at three national laboratories and six cooperative international programs.

Q3. Please explain what your duties have been in connection with the NRC staff's (Staff) review of the license amendment request (LAR) filed by Duke Energy Corp. (Duke) for a license amendment to insert four mixed oxide (MOX) fuel lead test assemblies (LTAs) into the reactor core at Duke's Catawba Nuclear Station, Units 1 or 2.

A3a. (US) My duties in connection with the Staff's review of the LAR filed by Duke relative to the insertion of MOX fuel LTAs into the core at Catawba have been focused on the fuel rod design features, the fuel assembly design, and the exemptions for using MOX fuel and the M5 cladding material.

A3b. (RL) My duties in connection with the Staff's review of the LAR filed by Duke relative to the insertion of MOX LTAs into the core at Catawba have been focused on the LOCA analysis performed pertaining to the MOX LTAs.

A3c. (ROM) I was not involved in the Staff's review, but I am familiar with the technical issues.

Q4. Are you familiar with Contention I?

A4. Yes. As admitted by the Licensing Board, Contention I reads as follows:

> The LAR is inadequate because Duke has failed to account for differences in MOX and LEU fuel behavior (both known differences and recent information on possible differences) and for the impact of such differences on LOCAs and on the DBA analysis for Catawba.

Q5. Do you agree with the assertion in the contention that the Licensing Amendment Request (LAR) is inadequate because of failure to account for differences in MOX and LEU behavior and for the impact that such differences might have on design-basis LOCAs?

A5. No.

Q6. Do you agree that the LAR is inadequate because of failure to account for uncertainties in MOX fuel assembly behavior during LOCAs?

A6. No.

Q7. BREDL has stated that the experimental database for MOX fuel performance during LOCAs is woefully inadequate. Do you agree?

A7. (ROM) No.

Q8. What is the purpose of your testimony?

A8a. (US) The purpose of this testimony is to provide the NRC Staff's views concerning the acceptability of Duke's LAR, which is the subject of Contention I.

A8b. (RL) The purpose of this testimony is to provide the NRC Staff's views concerning the adequacy of Duke's LOCA analysis and the acceptability of the LAR, which is the subject of Contention I.

A8c. (ROM) The purpose of this testimony is to provide the bases for my answers to Q5, 6 and 7.

Q9. How many MOX LTAs does Duke's LAR request to load into the Catawba core?

A9. (RL, US) Four.

Q10. In addition to the four LTAs, how many other assemblies will be in the core?

A10. (RL, US) There will be 189 other assemblies in the core.

Q11. The contention raises concerns about aspects of fuel behavior during a LOCA. Could you give us a brief description of fuel behavior during a LOCA?

A11. (RL, ROM) Pressurized water reactors like Catawba use circulating water to take heat from the fuel, and they generate steam with the hot water. This removal of heat from the fuel keeps the fuel relatively cool in relation to temperatures that would cause fuel damage. For a licensing analysis, it is assumed that a large pipe breaks and the water (i.e., the coolant) starts escaping when the reactor is at full power. This loss of coolant automatically shuts down the reactor because the nuclear chain reaction cannot be sustained without the water, and power being produced by the fuel decays rapidly to very low levels. After sufficient coolant has boiled away from the core region, the fuel cladding begins to heat up because heat is no longer being adequately removed from the cladding surface. During the heatup, the cladding will soften, balloon, and burst because the internal pressure is high. As the cladding continues to heat up beyond the temperature for bursting, the cladding begins to oxidize rapidly. Eventually, cold water is injected into the core by an emergency core cooling system (ECCS) and the fuel cladding is cooled back down. Heat removal systems keep the reactor cool from that time on.

Q12. Did the Staff conduct an evaluation of Duke's LOCA analysis?

A12. (RL) Yes. The details of the evaluation are found in the Staff's Safety Evaluation for Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies (SE), sections 2.1.2 and 2.4.1, issued April 5, 2004. (NRC Staff's Proposed Exhibit 1, "Safety Evaluation by the Office of Nuclear Reactor Regulation Renewed Facility Operating License NPF-35 and NPF-52," April 5, 2004).

Q13. Please summarize the Staff's evaluation of the LOCA analysis.

A13. (RL) The Staff's evaluation is contained in section 2.4.1 of the SE. Briefly, the ECCS performance of the Catawba nuclear plant is contained in the analysis done for the current operating core. That analysis was performed by Westinghouse using their NRC approved realistic large-break loss-of-coolant accident analysis (LBLOCA) program, WCOBRA/TRAC. The analysis of record demonstrates that the Catawba nuclear plant complies with the acceptance criteria delineated in 10 CFR 50.46(b). When Westinghouse performed the analysis-of-record LBLOCA analysis, a series of sensitivity studies were performed. One study concerned the effect of co-resident fuel from another vendor. At the time of the Westinghouse study, Catawba was in transition from Framatome Mark-BW fuel to Westinghouse Robust Fuel Assemblies (RFA) and the core contained fuel from both vendors.

The Staff's review included a review of the analysis-of-record (NRC Staff's Proposed Exhibit 2, Letter from M.S. Tuckman (Duke) to the NRC, "License Amendment to Request, Implementation of Best-Estimate Large Break Loss of Coolant Analysis Methodology," August 10, 2000), the sensitivity studies, and the MOX LTA LOCA analysis, as discussed in the answers to questions 15-20, below.

Q14. Against what regulatory requirements was the LAR LOCA analysis evaluated?

A14. (RL) The regulatory requirements are provided in 10 CFR § 50.46. That is, an evaluation model must be used which either realistically describes the behavior of the reactor system during a LOCA such that the uncertainty in the calculated results can be estimated, or conforms with the required and acceptable features of 10 CFR 50, Appendix K. Whichever approach to the evaluation model is followed, the results must meet the acceptance criteria stated in 10 CFR 50.46(b). Specifically, the peak cladding temperature must not exceed 2200F, the maximum local oxidation must not exceed 17%, the hydrogen generated must not exceed that which could be produced by oxidation of 1% of the total cladding, the core must remain in a coolable geometry, and the core temperature must remain at an acceptable level for an extended period of time.

The following questions and answers will explain in further detail how the LAR has shown compliance with the requirements of 10 CFR 50.46.

Q15. What is the Peak Cladding Temperature (PCT)?

A15. (RL) The PCT is the highest temperature calculated to occur in the reactor's core and is specified by 10 C.F.R. § 50.46 to not exceed 2200°F. Compliance with this criterion, along with the oxidation limit, assures that the cladding will not become embrittled and lose its rod-like geometry during and after a LOCA. The PCT predicted by Westinghouse, for Catawba, is below the acceptance criterion of 2200°F specified in 10 CFR 50.46(b)(1).

Q16. Please describe the LOCA analysis that was performed for the MOX LTAs.

A16. (RL) The LOCA behavior of the proposed Framatome ANP MOX LTAs was evaluated in two ways. First, the analysis of record was shown to still be valid for the Catawba nuclear plant with the MOX LTAs in core. This was done by comparison of the hydraulic behavior of the MOX assembly, noted as Mark-BW/MOX1, with the Mark-BW assembly used in the analysis of record study. Comparison of the hydraulic behavior of the Mark-BW/MOX1 fuel assembly as referenced in the Duke February 27, 2003 submittal with the analysis of record performed by Westinghouse shows that the Mark-BW/MOX1 fuel assembly is much closer in hydraulic behavior, such as pressure drop, to the Westinghouse RFA fuel than is the Mark-BW fuel design. Thus, the effect of the Mark-BW/MOX1 fuel on the performance of the RFA fuel under LBLOCA conditions would be less than the effect of the Mark-BW fuel that was resident at the time of the transition to the RFA fuel. (NRC Staff's Proposed Exhibit 2). Further discussion of the comparisons between the AOR and the MOX LTAs is found in the LAR, section 3.7.1.7., submitted by Duke February 27, 2003, and in Duke's November 3, 2003 response to Staff RAI 14. The November 3, 2003 response to RAI 14 states that "...the MOX fuel lead assemblies are more similar hydraulically to the RFA fuel than the Mark-BW design fuel..." In addition, the mixed core sensitivities performed for the Westinghouse RFA fuel showed that "...the presence of the Mark-BW fuel assemblies had an insignificant impact on the calculated results."

The second evaluation performed was a LBLOCA analysis of the Framatome ANP Mark-BW/MOX1 fuel itself. Framatome ANP performed that analysis using their NRC approved

10 C.F.R. Part 50, Appendix K computer code, RELAP5/MOD2-B&W. As a sensitivity study, Framatome ANP also analyzed the Mark-BW/MOX1 assembly loaded with low enriched Uranium fuel pellets rather than the MOX pellets, thus obtaining a comparison for MOX versus LEU when installed in the same non-limiting core location. The results of those studies are that the Appendix K code analyzed MOX peak cladding temperature is 2018°F, while that of the same fuel assembly design containing LEU fuel is 1981°F. LAR, Table 3-5.

Q17. How did the Appendix K analysis account for MOX fuel?

A17. (RL) The requirements specified in 10 CFR Part 50, Appendix K for an acceptable evaluation model are not dependent upon the content of the fuel pellet except in limited areas: initial stored energy, fission heat, and decay heat. First, the fuel stored energy, which is a measure of the initial temperature of the fuel, was calculated by Framatome ANP using their NRC approved COPERNIC fuel code which has been modified and approved to include MOX properties. Stored energy will be discussed by Ms Shoop. Second, the rate at which the fuel continues to produce heat after the nuclear reaction has been shut down is determined by the decay heat model. The applicability of the Framatome ANP decay heat model, which provides the amount of heat generated in the fuel after the nuclear chain reaction has been stopped, to the MOX fuel was reviewed by the staff. The requirement of Appendix K, 10 CFR 50, Appendix K 1.A.4, is that the ANSI/ANS-5.1-1971 decay heat curve multiplied by 1.2 be used to predict the heat generation by uranium dioxide fuel following cessation of the nuclear chain reaction. Duke stated in their February 27, 2003 submittal, Section 3.7.1.1.2, that Framatome ANP utilized a decay heat curve that is based on the ANSI/ANS-5.1-1994 standard. (American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1994, American Nuclear Society, 1994.) That standard accounts for the fact that for long exposure times, LEU fuel produces the majority of its energy from the fission of plutonium. Thus, the 1994 standard would be expected to be the more appropriate model to use for the decay heat production of MOX fuel than the 1971 model since the

MOX fuel is producing heat from the fission of plutonium throughout its lifetime. Framatome ANP added conservatism by increasing the curve by a factor of 1.2. The resulting decay heat curve bounds the 1994 standard by a factor of 1.2, and bounds the 1971 standard that has been multiplied by a factor of 1.2. The staff concluded that the decay heat model used is conservative with respect to the decay heat generated by the MOX fuel as well as being more conservative than the regulation specifies in bounding the Appendix K specified 1971 curve multiplied by a factor of 1.2.

Q18. Does the LOCA calculation include the effect of fission heat?

A18. (RL) Yes, as specified by Appendix K fission heat is calculated based on the known reactivity (how readily fission takes place, that is, if the rate of fission increases or decreases) of the fuel which is well understood for both LEU and MOX, and the effects on reactivity that occur during a LOCA. Such phenomena as voiding of the reactor cooling water and fuel temperature increase cause the fission reaction to slow down and stop. Each of these factors is assumed to be at its worst value, that is, the fission heat is required to be maximized.

Q19. Did the staff perform independent LOCA calculations for the MOX LTAs?

A19. (RL) The staff did not perform independent LOCA calculations for the MOX LTAs. The staff normally performs those analyses as part of the review of the vendors' generic evaluation models and methodologies. Our practice is to review the documentation supporting the evaluation models, sample calculations, and perform calculations using the vendors' evaluation models as well as the staff's independent computer codes. Once that has been done, we review the plant-specific analyses that have been performed and submitted by the licensee. In the case of the MOX LTAs, we reviewed the evaluation model input description, the assumptions made by the vendor, Framatome ANP, and the results. The input and assumptions were found to be consistent with the staff's review of the generic evaluation model in the NRC staff "Safety Evaluation of Framatome

Technologies Topical Report BAW-10164P, Revision 4, 'RELAP5/MOD2-B&W, An Advanced Computer Transient Analysis,'" April 9, 2002.

Q20. What is your conclusion regarding the LOCA analysis performed in support of the Duke LAR?

A20. (RL) The staff concludes on the basis of its review of the LOCA analyses which form the analysis-of-record, and its review of the analyses in support of the use of MOX LTAs at Catawba in the LAR and supplements, that the analyses have been performed in an acceptable manner, are conservative, and demonstrate compliance with the stated acceptance criteria contained in 10 CFR 50.46.

Q21. Do you have an opinion regarding the safety of permitting the use of four MOX LTAs at Catawba Nuclear Station?

A21. (RL) Yes.

Q22. What is that opinion?

A22. (RL) The information submitted by Duke, including their responses to requests for additional information (RAIs) asked by the Staff, demonstrate that there is reasonable assurance that the four MOX LTAs to be installed at the Catawba Nuclear Station would not adversely affect the health and safety of the public. That opinion is based on the LOCA analyses that demonstrate that there is reasonable assurance that the MOX LTAs would behave in compliance with the NRC's regulations.

Q23. Earlier you stated that you do not agree with the assertion in Contention I that the LAR is inadequate. Please provide the basis for that conclusion.

A23. (RL) Duke has accounted for the effect of MOX on the LOCA through use of an approved method of determining the fuel stored energy at the initiation of the LOCA and also use of a conservative MOX-applicable decay heat model to determine the heat production of the fuel during the LOCA analysis. Use of these sources of heat ensure a conservative prediction of the behavior of the MOX LTAs in the unlikely event of a LOCA. In addition, Duke has performed specific analyses which provide a direct comparison of the response of MOX versus LEU fueled assemblies at the same core location. This was done by taking the physical design of the MOX-specific fuel assembly and performing the LOCA analysis with the assembly fueled with MOX pellets, and then repeating the analysis with the assembly fueled with LEU pellets. Those analyses were performed using the appropriate properties of each fuel pellet material and demonstrate that there is a difference in predicted PCT of only 37°F for the MOX versus LEU fueled assemblies.

Q24. Could you please describe your review of initial stored energy to ensure that MOX fuel was treated properly?

A24. (US) During the review of the COPERNIC code for MOX fuel, the staff used prediction to measurement comparisons at Linear Heat Generation Rate (LHGR) levels for LOCA stored energy calculations to estimate uncertainty including standard deviation in fuel performance codes. The uncertainty is applied to code predictions to obtain a conservative stored energy prediction at a 95/95 tolerance level for LOCA analysis. The staff used the FRAPCON-3.2 code to compare the results from the COPERNIC code for stored energy calculations.

COPERNIC supplies the stored energy and thermal conductivity values to the LOCA code; therefore, the MOX specific parameters are used in the analysis and the results account for the differences between uranium and MOX fuel in the stored energy and thermal conductivity calculations.

Q25. Did Framatome consider the clad ballooning properties of M5?

A25. (US) Yes. Framatome developed a model to specifically describe the clad ballooning properties of M5 and submitted it with topical report BAW-10227. The topical report was approved in December of 1999 and revised in February 2000. "Revised Safety Evaluation by The Office of

Nuclear Reactor Regulation, Topical Report BAW-10227P, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, Framatome Cogema Fuels, Inc," Feb. 4, 2000.

Q26. Do you have an opinion regarding the safety of permitting the use of four MOX LTAs at Catawba Nuclear Station?

A26. (US) Yes.

Q27. What is that opinion?

A27. (US) On the basis of the review of the fuel design, as articulated in the LAR and supplements, the staff concludes that the fuel has been analyzed in an acceptable manner with sufficient conservatism to demonstrate compliance with General Design Criteria 10 and that there is reasonable assurance that public health and safety will be protected.

Q28. Earlier you stated that you do not agree with the assertion in Contention I that the LAR is inadequate. Please provide the basis for your conclusion.

A28. (US) The parameters provided by the fuel performance code to the LOCA analysis in the LAR are MOX specific parameters, so the differences introduced from the use of MOX on the LOCA calculation are accounted for in the analysis.

Q29. BREDL has alleged that the LAR is based on out-of-date information relating to (a) the M5 Cladding, (b) the behavior of MOX fuel and (c) the interaction between the fuel and the cladding. Do you agree?

A29. (US, ROM) No.

Q30. Please provide the bases for this conclusion.

A30. (US, ROM) (a) The review of M5 cladding was done several years ago. A lot of work on niobium-bearing cladding alloys like M5 has been done since then and the NRC has been involved in that work. Although the Staff and the scientific community know more than they did then, nothing that has been learned invalidates the Staff's overall conclusion in 2000 that M5 is an acceptable cladding material. (b) The Staff used current data on MOX fuel for the review of the LAR. (c) Dr. Meyer will discuss the interaction between the fuel and the cladding.

Q31. One of the issues raised by BREDL concerned ballooned rods during a LOCA. Can you describe the ballooning and rupturing process?

A31. (ROM) Fuel rods are pressurized with helium during fabrication and further with fission gas during operation, and at normal full-power operation the internal rod pressure is high. Near the end of the fuel lifetime, the pressure on the inside is roughly balanced by the pressure on the outside (i.e., the reactor system pressure). When the coolant is lost in a LOCA, the system pressure is also lost, so the resulting large fuel rod pressure differential tries to expand the cladding. When the cladding temperature reaches 600 to 700C (approximately 1100 to 1300F), cladding expansion becomes rapid, and around 800C (approximately 500F) the fuel rod bursts just like a rubber balloon would burst. The fuel rod's internal pressure is then lost, and the deformation or ballooning ceases.

Q32. How large a diameter do these balloons have?

A32. (ROM) The diameter increase can be as big as 100%, but the balloons are usually smaller. Their size will depend on the open volume within the fuel rod, the amount of gas within the fuel rod, and the temperature at which the rupture occurs.

Q33. Will the diameter of the balloon be different for MOX fuel and LEU fuel?

A33. (ROM) No.

Q34. Please provide the basis for your answer.

A34. (ROM) During the last 18 months, researchers at Argonne National Laboratory completed the first four ballooning tests ever performed with actual high-burnup fuel rods (NRC Staff's Proposed Exhibit 3, Y. Yan et al., "LOCA Test Results for High-Burnup BWR Fuel and Cladding," Organization for Economic Cooperation and Development (OECD) Topical Meeting on LOCA Issues, May 25-26, 2004, p. 17). In each case, the size of the balloon was the same as the size of the balloon in the control test with an unirradiated specimen. In my opinion, the high cladding temperature during the LOCA transient loosens the pellet-to-cladding bond that develops at high burnup just like hot water loosens the cap on a jelly jar, and there is no effect of fuel-to-cladding bonding on ballooning. Therefore, the size of the balloon is determined by gas quantity, volume and temperature as described earlier. There would be no difference in ballooning between MOX fuel and LEU fuel.

Q35. Will the diameter of the balloon be larger for M5 cladding than for Zircaloy cladding?

A35. (ROM) No, but it is important to use the appropriate data in reaching a conclusion. Creep tests, for example, are conducted with pressurized tubes that are held at a constant temperature for a long time (10-1000 sec) until the tubes burst. During a LOCA, on the other hand, fuel rods would be experience a temperature that is rapidly increasing at the rate of about 5 C/sec at the time of bursting. These different temperature conditions can have a significant effect on the deformation process. Therefore, rod burst tests for LOCA applications are always done with increasing temperatures that are typical of a LOCA. N. Waeckel presented such test data for M5 and Zircaloy at a recent conference (NRC Staff's Proposed Exhibit 4, N. Waeckel et al., "Does M5 Balloon More that Zircaloy-4 Under LOCA Conditions?," OECD Topical Meeting on LOCA Issues, May 25-26, 2004, p. 10), and those data show that M5 does not develop larger balloons than Zircaloy under LOCA conditions.

Q36. After ballooning occurs during a LOCA, can fuel material move around inside the fuel rod?

A36. (ROM) Yes. Fuel pellets, which are about the size of little marshmallows, develop cracks during normal operation such that they can easily break apart. Tests have shown that broken pieces of fuel pellets can move down into the ballooned region of the cladding.

Q37. What, if any, is the significance of this fuel movement or relocation?

A37. (ROM) If extra fuel particles or fragments move into the ballooned region, they will increase the mass of fuel in that region and thereby increase the heat generated in that region. The increased heat generation would increase the cladding temperature in the balloon and thus increase the amount of cladding oxidation, which causes embrittlement.

Q38. Would there be any difference in the amount of fuel relocation for MOX fuel and LEU fuel?

A38. (ROM) It is possible, but in my opinion it would not matter. If the MOX fuel fragments were smaller than LEU fuel fragments, then you might be able to get more MOX fuel into the balloon.

Q39. Why might the fuel fragments be smaller?

A39. (ROM) There is probably a little more rim material in MOX fuel than in LEU fuel with the same burnup. This rim material, which forms in high-burnup regions around the circumference of LEU fuel and also around the agglomerates in MOX fuel, is the result of fission gas migration within the uranium-plutonium oxide crystalline grains. Fission gas migrates, coalesces, and precipitates in small bubbles, which attach themselves to the grain boundaries. As the number of bubbles increases with burnup, the grain boundaries subdivide to form more surface area to accommodate the bubbles, thus producing the smaller grained rim material. Because fission gas release, which is also related to the migration process, is a little higher in MOX fuel than in LEU fuel (e.g., 5% in MOX and 4% in LEU), the volume of rim material might be roughly 25% greater in MOX fuel than in LEU fuel. On the other hand, MOX fuel has a little more plasticity than LEU fuel, so I would expect fewer of the larger fragments in MOX fuel.

Q40. You said that in your opinion, the difference in the amount of relocated fuel between MOX and LEU fuel would not make any difference. Why is that?

A40. (ROM) In recent high-burnup integral tests in our program at ANL, we have observed a black deposit on the quartz tube of the apparatus just opposite the burst opening.

Large fuel fragments are also visible through the burst opening, and these particles have no small particles or fines around them (NRC Staff's Proposed Exhibit 3, Y. Yan et al., OECD Topical Meeting on LOCA Issues, May 25-26, 2004, p. 17). It thus appears that the small particles or fines are blown out of the burst opening when the rod depressurizes. Thus, there would be few or no small particles in the ballooned region, and it is these small particles that have been postulated to make a difference between the mass of fuel in the balloon in MOX fuel and LEU fuel.

Q41. For a given amount of fuel relocation, would the heat source in the balloon be greater for MOX fuel than for LEU fuel?

A41. (ROM) No. For the Catawba plant, the peak cladding temperature occurs a couple of minutes after the loss of coolant has shut down the power. By that time, most of the stored heat in the fuel has been dissipated and the chemical heat from the metal-water reaction is small, so the heat source is dominated by decay heat. Decay heat for MOX fuel is lower than it is for LEU fuel; therefore, the heat source in the balloon for MOX fuel would be less than it would be for LEU fuel.

Q42. In his ACRS presentation of May 6, 2004, Dr. Lyman used an average figure of 105C (190F) increase in peak cladding temperature and added that to the reported peak cladding temperature of 2018F for the MOX fuel in Catawba. He then concluded that this would bring the MOX peak cladding temperature well over the regulatory limit of 2200F. Do you agree with that conclusion?

A42. (ROM) No. If fuel relocation has any effect, it would increase the temperature only in the ballooned region of the fuel rod. Because of the larger surface area of the ballooned region, its cooling is enhanced and the ballooned region is seldom the location of the calculated peak cladding temperature when relocation is ignored. For the MOX fuel in Catawba, Duke reported a maximum cladding temperature in the balloon of only 1841F, and the balloon was located almost a foot below the location of the peak cladding temperature of the fuel rod. If you add 190F -- the number used by Dr. Lyman -- to the maximum cladding temperature in the balloon, you get 2031F, which is just 13F over the peak cladding temperature of 2018F reported by Duke. This is a small increase in peak cladding temperature, which would still be well below the allowable temperature of 2200F.

Q43. Was fuel relocation taken into account in the LOCA analysis for the MOX lead test assemblies?

A43. (ROM) There was no specific accounting for fuel relocation in the LOCA analysis submitted by the applicant, since it is not required by Appendix K.

Q44. How long are the ballooned regions in relation to the fuel rod length?

A44. (ROM) Fuel rods are 12-feet long. Because of local temperature variations, a localized bulge or balloon develops in a fuel rod under LOCA conditions and then ruptures. These ruptured balloons are only a few inches long.

Q45. Are the ballooned regions of the fuel rod treated in a special way in a safety analysis?

A45. (ROM) Yes. Appendix K requires that the inside of the cladding must also be considered to oxidize over a 3-inch length on the assumption that steam will enter the ruptured opening of the balloon and react with the inside of the cladding. Of course, the outside of the cladding is considered to oxidize over its full length. In the balloon, the thickness of the cladding is also reduced for the oxidation calculation, so the maximum oxidation that must be compared with the 17% limit is almost always in the balloon, as it is for the MOX fuel in Catawba. The bottom line is that only about 3 inches of the 12-foot fuel rod are threatened by embrittlement. Protecting these most vulnerable three inches of the limiting fuel rod ensures coolable geometry following a LOCA.

Q46. Earlier, you said that you did not agree that the database for MOX fuel performance during LOCAs is woefully inadequate. What is the basis for that statement?

A46. (ROM) As can be seen from the above testimony, fuel performance during LOCAs is almost entirely controlled by cladding behavior, which is unaffected by fuel type. The database for cladding behavior is the same for MOX and LEU fuel, and it is substantial. The only important fuel property that is different for MOX fuel and thus affects LOCA performance is the fuel thermal conductivity. Many measurements of MOX fuel thermal conductivity have been made during the past 35 years because of extensive research for breeder reactors as well as for MOX utilization in LWRs. However, the most critical measurements in relation to fuel temperatures and LOCA behavior are direct measurements of centerline temperature in instrumented fuel rods taken from commercial reactors. To this end, there have been 7 MOX instrumented fuel assemblies in the Halden test reactor, some of which tested fuel that operated to 65 GWd/t burnup (and some are continuing to test fuel above that burnup level). Each test assembly has produced hundreds of data points on fuel temperature and rod pressure over a long period of time. These results have provided an adequate database for validation of fuel rod codes for application to MOX fuel (NRC Staff's Proposed Exhibit 5, Memorandum from Farouk Eltawila, RES, to Suzanne C. Black, NRR, RE: Response to User Need for Development of Radiological Source Terms for Review of Mixed Oxide Fuel Lead Test Assemblies, February 23, 2004, Attachment B, Figure 1). Although additional data are being taken, especially at higher burnups, it is incorrect to say that the database for MOX fuel performance during LOCAs is inadequate.

Q47. Would you summarize your testimony and give us your perspective on the possible effects of fuel relocation in MOX fuel versus LEU fuel?

A47. (ROM) NRC is treating fuel relocation during a LOCA as a significant issue and we are investigating it actively in our research programs. If relocation is found to occur during that relatively small window of time between the bursting of the balloon and the rapid cooling (quenching) of the core, and if the average densities in the balloons are found to be relatively high, then the cladding temperature in the balloon might be increased by several hundreds of degrees

F. However, the peak cladding temperature of a fuel rod does not usually occur in the balloon, so the actual increase in the peak cladding temperature would probably be much smaller. Further, we should not lose sight of the objective of the LOCA analysis: to preserve long-term cooling of the core following a LOCA. Our regulatory logic is that we would be able to continue cooling the core if the fuel pellets stay within the cladding, and that we could accomplish this even if the cladding is bent or has holes in it as long as the cladding does not embrittle and fragment. Even if the balloon were to exceed the embrittlement criteria, the remainder of the 12-foot fuel rod would have lots of ductility because the embrittlement in the balloon is twice that in other comparable locations due to the doubling of the oxidation (inside oxidation plus outside oxidation). Realistically, the embrittlement and fragmentation of 3 inches of cladding out of 12 feet would not result in a core melt, and without core melt there could be no large releases and no major consequences. So in my opinion, the regulatory approach is conservative. The assumption that the relocation effect would be more severe for MOX fuel than for LEU fuel is speculative. Hypotheses have been put forth by Dr. Lyman that the balloons would be bigger, that the density of relocated fuel would be higher, and that the heat source for a given amount of fuel would be greater during a LOCA for MOX fuel than it would be for LEU fuel. I believe, based on the data discussed above and on my experience, training and knowledge, that those hypotheses are incorrect. I do not see any reason to believe, that potential relocation effects would be worse for MOX fuel than for LEU fuel.

Q48. What is your opinion regarding the safety of using four MOX LTAs at Catawba Nuclear Station?

A48. (ROM). I believe that the use of four MOX LTAs does not adversely affect the health and safety of the public.

Q49. Please provide a basis for your answer.

A49. (ROM) NRC has an aggressive research program that probes into potential weaknesses in reactor safety analyses and licensing procedures. We are investigating fuel-related issues associated with several accidents, including LOCA, and these issues are well known from documents that we have issued. The question is not whether we have issues, but, rather, the question is whether MOX fuel exacerbates these issues compared with LEU fuel. Based upon my experience, knowledge and training and the testimony I have given above, it is my opinion that these issues are not exacerbated by the use of MOX. Therefore, it is my opinion that the use of MOX LTAs will not have a deleterious effect on public health and safety.

Q50. What is your conclusion regarding Duke's LAR?

A50. Based upon the staff's evaluation and the testimony provided above, the Staff concludes that there is reasonable assurance that the public health and safety will be protected if the amendment is granted.

Q51. Does this conclude your testimony?

A51. Yes.

Professional Qualifications

Undine Shoop

Reactor Systems Engineer
US Nuclear Regulatory Commission

Education

B.S., The Pennsylvania State University, Nuclear Engineering, 1994
M.S., The Pennsylvania State University, Nuclear Engineering, 1996

Experience

Reviewed Lead Test Assembly (LTA) application for the use of down-blended weapons uranium fuel.

Completed multiple applications for insertion of LTAs into current LWR cores. Some of these reviews involved requests to use the LTAs above current approved burnup limits.

Reviewed multiple exemption requests for the use of M5 cladding in current LWRs.

Led the Reactor Systems Branch team for the MOX LTA application review.

Completed multiple requests to change fuel parameters in the Technical Specifications of current LWRs.

Reviewed new fuel designs, both for generic and plant specific applications.
Primary author of a commission paper on the agency technical and regulatory needs prior to the use of MOX fuel commercial LWRs for the office of research.

Member of the ANS 19.6 Standards Working Group which has evaluated the PWR Startup Physics testing program.

While working at the Pennsylvania Power and Light Co.

Generated SIMULATE-E generic beginning of life rod withdrawal sequence input decks for reactivity evaluation of core startup following outages.

Developed and guided a beginning of life core asymmetry check program through the quality assurance program requirements.

Evaluated local power range monitor (LPRM) trends to identify outage replacement candidates. Performed transversing in-core probe surveillances.

Verified acceptability of new fuel during fuel receipt procedures.

Publications

"TRAC-BF1 THREE DIMENSIONAL BWR VESSEL THERMAL-HYDRAULIC and ANSYS Stress ANALYSIS FOR BWR CORE SHROUD CRACKING" in Annuals of Nuclear Energy, Vol. 25, No. 1-3, pp. 65-81, 1998.

Professional Affiliations

Founding member of a new professional society, North American Young Generation in Nuclear, Incorporated December 1999. Currently serving as the Past President.
Member of the American Nuclear Society from 1992 to the present.
Chairman of the Professional Woman in ANS Committee, member from 1994 to the present. * Chairman of the Nuclear Installations Safety Division Program Committee in the ANS.

Professional Qualifications

RALPH R. LANDRY

PROFESSIONAL EXPERIENCE

October 1995 - Present: Senior Reactor Engineer (Nuclear) in the Reactor Systems Branch, DSSA, Office of Nuclear Reactor Regulation, NRC. Duties include:

> Group Leader for review of Siemens EXEM/PWR code, EPRI RETRAN-3D code, Framatome ANP S-RELAP5 code, GE update to SAFER/GESTR code, and the GE TRACG code. Lead reviewer and principle author of the Safety Evaluation Report of ESBWR TRACG application review. Review of the thermal hydraulic and analysis computer codes (LOFTRAN and NOTRUMP) for the Westinghouse AP600 advanced, passive reactor design. Work involves interaction with the applicant as well as management of contractor support and presentations before the ACRS. Assistance with the review of steam generator degradation problems experienced at the Combustion Engineering NSSS reactor sites. Inspection of the Yankee Atomic Electric Company LOCA analysis code pertaining to the Vermont Yankee Nuclear Plant. Inspection resulted in violations, enforcement action and civil penalties. Inspection of the Westinghouse LOCA Engineering Review Committee and Westinghouse analysis methods for the Westinghouse NSSS operating reactor plants. Inspection of the Siemens Power Corporation fuel and LOCA code methodologies. Member of Safety System Functional Inspection team for Crystal River Unit 3.

October 1994 - October 1995: Senior Reactor Engineer (Nuclear) in the Advanced Reactor Project Directorate, Office of Nuclear Reactor Regulation, NRC. Duties included:

> Thermal hydraulic and analysis computer code review of the CANDU standard design submittal. Work involves interaction with the applicant and the Atomic Energy Control Board of Canada. Development and management of technical assistance contracts in support of reactor physics and thermal hydraulic analysis code reviews. Assistance to the Office of Information Resource Management in design, specification, procurement and scheduling for the Technology Discovery Center. Mentor for NRC Fellow during his one year stay with PDAR.

January 1992 - October 1993: Senior Reactor Engineer (Nuclear) in the Reactor Systems Branch, Detailed to the Analytic Support Group, Division of Systems Safety & Analysis, Office of Nuclear Reactor Regulation, NRC. Duties included:

> Acquisition of engineering workstations for the Analytic Support Group's use in performing analysis computer code calculations. Workstations were acquired and networking and access to the Internet for the workstations was implemented. Administration of the engineering workstation network as backup to the IRM System Administrator. Installation and testing of computer codes to be used by the Analytic Support Group. Codes installed include: SCDAP/RELAP5/MOD3, the Nuclear Plant Analyzer, CONTAIN, MELCOR, CONTEMPT/LT28, COGAP, COMPARE, VIPRE, and MINET. Support of the Advanced Reactor Directorate in installation of the CATHENA code for analysis of the CANDU reactor. Analysis of the Shearon Harris Nuclear Power Plant degraded high pressure injection safety injection system condition. Analysis of the

steam generator tube rupture coincident with main steam line break generic issue. Evaluation of the AP600 advanced passive reactor design. Management of training contract for computer systems and code use training of Analytic Support Group staff. Development and management of contracts with national laboratories and universities to provide support to the Analytic Support Group.

April 1991 - January 1992: Senior Reactor Engineer (Nuclear) in the Reactor Systems Branch, Division of Systems Technology, Office of Nuclear Reactor Regulation, NRC. Duties included:

Application of advanced thermal-hydraulic analysis computer codes to evaluation of advanced reactor designs, including the AP600 and SBWR designs. Analyses are also performed in support of the design testing programs. Evaluate capability of analysis codes to adequately perform licensing safety reviews of advanced reactor designs. Develop capability within the branch to perform analyses of advanced designs. Includes determination of code needs, computer hardware configurations and training necessary to operate the required computer equipment and analysis codes. Develop the training program for the staff that will be performing the in-house reactor design analyses.

November 1989 - April 1991: Senior Research Engineer in the Accident Evaluation Branch, Office of Nuclear Regulatory Research, NRC. Duties included:

Program manager for the OECD/NEA TMI Vessel Integrity Project. This was a cooperative international program including the U.S. and ten partner countries under the auspices of the OECD/NEA. Preparation of revision to Severe Accident Research Program plan to include long term research plans. Program manager for lower head failure analysis research at INEL.

April 1987 - November 1989: Reactor Engineer in the Advanced Reactors Branch, Office of Nuclear Regulatory Research, NRC. Duties includedd:

Program manager for the PRISM and SAFR liquid metal reactor conceptual design reviews. Principal author of NUREG-1368, the PRISM Safety Evaluation Report. Principal author of NUREG-1369, the SAFR Safety Evaluation Report. Assist with development of Standardization Rule, 10 CFR 52. Preparation of Commission paper regarding proposed review of PIUS reactor design.

September 1986 - April 1987: Nuclear Engineer in the Regulatory Improvements Branch, Office of Nuclear Reactor Regulation, NRC. Duties includedd:

Preparation of Generic Letter for Individual Plant Examinations for implementation of the NRC's Severe Accident Policy Statement. Review of Guidelines and Criteria prepared for the NRC by a national lab for the Large Dry PWR containment design. Preparation of the implementation of the Severe Accident Policy Statement guidance for new and future plant designs. Preparation of the Severe Accident section for the NRC 1986 Annual Report. Preparation of letters and materials pertaining to international cooperative work for the Director and Deputy Director, Division of Safety Review and Oversight. Review and comment, including preparation of input, on an international report on approaches to nuclear safety for the Director, NRR.

September 1984-September 1986: Administrator, Nuclear Energy Agency, Organization for Economic Cooperation and Development, Paris, France. On leave from the NRC with responsibilities for:

Coordination of international cooperative research in nuclear reactor thermal hydraulics and fuel behavior. Definition of recommended procedures for thermal hydraulic analysis computer code assessment and validation, including criteria for successful completion of code assessment. Coordination of International Standard Problem exercises in thermal hydraulics, containment response, and fuel behavior. Coordination of the international cooperative research programs in the Loss-of-Fluid Test facility in the United States, and the Halden Reactor Project in Norway. Development of an international cooperative effort for examination of the debris material from the Three Mile Island Unit 2 facility, and standard problem analyses of the TMI-2 accident.

The work resulted in completion of an international code validation matrix for PWR and BWR analysis codes, completion of four standard problem exercises, definition of five future standard problem exercises, and publication of State-of-the-Art reports on BWR Pressure Suppression Containment Systems, and PWR Fuel Behavior Under LOCA Conditions. This work was based on coordination of the work in twenty OECD/NEA member countries.

June 1982-September 1984: Program Manager, Office of Nuclear Regulatory Research, NRC. Duties included:

Management of the Semiscale Project at the Idaho National Engineering Laboratory, Idaho Falls, Idaho. Management of the RELAP5 code development project at the INEL. Management of the PWR code assessment program at the INEL. Development of the MB-2 steam generator research program at the Westinghouse Tampa Facility, with Westinghouse and EPRI. Participation in the Test Advisory Group, comprised of NRC, B&W, B&W Owners Group, and EPRI studying the issues surrounding the B&W reactor design, leading to development of a research facility and program to resolve the identified needs for the B&W plant design. Preparation of the Integral Systems Tests sections of the NRC Annual Report and the Research Office's Long Range Research Plan.

May 1978-June 1982: LOFT Research Branch, Office of Nuclear Regulatory Research, NRC. Duties included:

Program management and technical direction for heat transfer and fluid dynamics analysis as related to integral systems experiments. Review of results of research and test programs to determine progress and to assure that the work is applicable to analysis methods. Assist the Assistant Director for Water Reactor Safety Research in developing research goals and objectives for the Division programs. Provide liaison and program management for the U.S. and international ECCS Standard Problem programs. Budget management of the LOFT Program; $54M per year.

January 1976-May 1978: Reactor Engineer, Reactor Systems Branch, Division of Operating Reactors, Office of Nuclear Reactor Regulation, NRC. Duties included:

Review of safety analyses associated with operating reactor fuel reloads. Safety evaluation of ECCS redesign work at San Onofre Unit 1. Evaluation of plant modifications for BWRs to meet ATWS requirements. Evaluation of isolation capability of PWR low pressure systems from the high pressure reactor coolant system.

February 1974-January 1976: Reactor Engineer, Reactor Systems Branch, Division of Technical Review, Office of Nuclear Reactor Regulation, NRC. Duties included:

Review of license applications, particularly the Westinghouse Standard Plant designs RESAR-41, RESAR-3S, and the Westinghouse 17x17 core design. Review of license application for the South Texas Project. Development of Branch Position on isolation capability and requirements for PWR Residual Heat Removal Systems. ECCS evaluation model review for acceptance under 10 CFR 50.46.

February 1972-February 1974: Nuclear Engineer, Bechtel Power Corporation. Duties included:

Analysis of containment design and subcompartment pressurization response for the Calvert Cliffs, Davis Besse, and Farley plants. Evaluation of control of post-LCOA hydrogen generation in BWRs. Development of hydrogen generation and transport code for BWRs. Development of containment and containment subcompartment thermal hydraulic codes. Evaluation of radiological consequences of all design basis accidents.

EDUCATION

BS in Mechanical Engineering, University of Missouri–Rolla.

PhD in Nuclear Engineering, University of Missouri–Rolla. Dissertation: A Study of the Effect of Rotation on the Nucleate Boiling from a Vertical Copper Cylinder.

Professional Qualifications

RALPH O. MEYER

AREA OF EXPERTISE

Reactor Fuel Behavior

EDUCATION

B.S. Physics, University of Kentucky, 1960
Ph.D. Physics, University of North Carolina, 1996

PROFESSIONAL EXPERIENCE

1965-1968	Research Associate, Department of Physics University of Arizona Tucson, AZ 85710
1968-1973	Assistant Metallurgist, Materials Science Division Argonne National Laboratory Argonne, IL 60439
1973-Present	Senior Technical Advisor, Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, DC 20555 (Formerly U.S. Atomic Energy Commission, various positions held)

PUBLICATIONS

D. E. Carlson and R. O. Meyer, "Assessment of Databases and Modeling Capabilities for the CANDU 3 Design," NUREG-1502, July 1994.

R. K. McCardell and R. O. Meyer, "Primary Factors Causing the Failure of High-Burnup LWR Fuel Rods During Simulated Reactivity Initiated Accidents," *Proceedings of the CSNI Specialist Meeting on Transient Behavior of High Burnup Fuel* (Cadarache 1995), OCDE/GD(96)197, 1996, pp. 167-184.

R. Meyer, "Summary of High Burnup Fuel Issues and NRC's Plan of Action," *Proceedings of the Twenty-Fourth Water Reactor Safety Information Meeting* (Washington 1996), NUREG/CP-0157, 1997, pp. 79-82.

R. O. Meyer, et al., "A Regulatory Assessment of Test Data for Reactivity-Initiated Accidents," *Proceedings of an International Topical Meeting on Light Water Reactor Fuel Performance* (Portland 1997), Amer. Nucl. Soc., 1997, pp. 729-744.

R. O. Meyer, "NRC Activities Related to High Burnup, New Cladding Types, and Mixed-Oxide Fuel," *Proceedings of an International Topical Meeting on Light Water Reactor Fuel Performance* (Park City 2000), Amer. Nucl. Soc., 2000, pp. 736-744.

R. Meyer, "NRC Program for Addressing Effects of High Burnup and Cladding Alloy on LOCA Safety Criteria," *Proceedings of the Topical Meeting on LOCA Fuel Safety Criteria* (Aix-en-Provence, 2001), NEA/CSNI/R(2001) 18, 2001, pp. 65-74.

R. O. Meyer, "Implications From the Phenomenon Identification and Ranking Tables (PIRTs) and Suggested Research Activities for High Burnup Fuel," NUREG-1749, September 2001.

G. M. O'Donnell, et al., "A New Comparative Analysis of LWR Fuel Designs," NUREG-1754, December 2001.

R. O. Meyer, "LOCA Ductility Tests," *Proceedings of the 2002 Nuclear Safety Research Conference* (Washington), NUREG/CP-0180, 2003, pp. 99-108.

J. R. Strosnider, Jr., and R. O. Meyer, "Safety Research in a Competitive World," *Proceedings of ENS TopFuel 2003* (Wurzburg 2003), European Nucl. Soc., 2003, Opening Session.

R. O. Meyer, "A Scaling Model for RIA Data," *Proceedings of the 2003 Nuclear Safety Research Conference* (Washington), NUREG/CP to be published 2004.

April 5, 2004

Mr. H. B. Barron
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28202

SUBJECT: SAFETY EVALUATION FOR PROPOSED AMENDMENTS TO THE FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATIONS TO ALLOW INSERTION OF MIXED OXIDE FUEL LEAD ASSEMBLIES (TAC NOS. MB7863, MB7864, MC0824, AND MC0825)

Dear Mr. Barron:

Enclosed is a copy of the Nuclear Regulatory Commission (NRC) staff's Safety Evaluation (SE) regarding your application submitted on February 27, 2003, as supplemented by letters dated September 15, September 23, October 1 (two letters), October 3 (two letters), November 3 and 4, December 10, 2003, and February 2 (two letters), March 1 (three letters), March 9 (two letters), March 16 (two letters), March 26 and March 31, 2004, to revise the Technical Specifications for the Catawba Nuclear Station to allow the use of four mixed oxide fuel lead test assemblies in one of the two Catawba units.

The issuance of this SE does not constitute NRC approval of your application to modify the licensing basis for the Catawba Nuclear Station. This SE documents the technical and regulatory disposition of the subject discussed within. NRC approval of your application, including its application for exemption from certain regulatory requirements, should it be appropriate, will be under separate correspondence. One or more supplements to this SE will be issued prior to or with the authorization of any change to the licensing basis for Catawba. These supplements will provide the publically available evaluation of security related issues and other matters as may be appropriate.

In the event of any comments or questions, please contact me at (301) 415-1493.

Sincerely,

/RA/

Robert E. Martin, Senior Project Manager
Project Directorate II-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosure: As stated

cc w/encl: See next page

NRC Staff's Proposed Exhibit 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RENEWED FACILITY OPERATING LICENSE NPF-35

AND

RENEWED FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated February 27, 2003, as supplemented by letters dated September 15, September 23, October 1 (two letters), October 3 (two letters), November 3 and 4, December 10, 2003, February 2, 2004, (two letters), March 1, 2004, (three letters), March 9, 2004, (two letters), March 16, 2004 (two letters), March 26 and March 31, 2004, Duke Energy Corporation, et al. (Duke, the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2 (Catawba), and to the McGuire Nuclear Station, Units 1 and 2 (McGuire), Technical Specifications (TS). The amendment request was revised by the licensee's letter dated September 23, 2003, to remove McGuire from the application. The licensee proposed to revise the TS to allow the use of up to four mixed oxide fuel (MOX) lead test assemblies (LTAs). Duke currently plans to load the four MOX LTAs into Catawba, Unit 1, in the spring of 2005. However, Duke has requested regulatory approval for both Catawba units to facilitate adjustments for changes in the LTA fabrication schedule, should any such changes occur. The supplemental letters provide additional clarifying information and did not expand the scope of the original Federal Register Notice (68 FR 44107, July 25, 2003).

This license amendment request is being made as part of the ongoing United States -- Russian Federation Fissile Material Disposition Program (FMDP). The goal of the FMDP is to dispose of surplus plutonium from nuclear weapons by converting the material into MOX fuel and using that fuel in nuclear reactors. In doing so, the plutonium will be rendered unsuitable for use in nuclear weapons and the increased radiation levels will reduce the threat of diversion of this material. Plutonium dioxide (PuO_2) powder supplied by the Department of Energy (DOE), will be blended with depleted uranium dioxide (UO_2) powder, and fabricated into MOX fuel pellets and MOX fuel assemblies. The four MOX LTAs will be loaded into Catawba instead of an equal number of low-enriched uranium (LEU) fuel assemblies for a minimum of two refueling cycles to be followed by post-irradiation examinations. These LTAs will be manufactured in France under the direction of Framatome Advanced Nuclear Power (ANP).

2.0 REACTOR SYSTEMS

This Safety Evaluation (SE) addresses the in-reactor performance and impact on the safety analyses for the MOX LTAs. The Nuclear Regulatory Commission (NRC) staff concludes that the MOX LTAs are capable of meeting the regulatory criteria addressed herein.

2.1 Regulatory Requirements

2.1.1 MOX Fuel LTAs Impact on Plant Operation

Fuel designs must ensure that the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 10, "Reactor Design," are met. Specifically, that appropriate margin be provided so that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Additionally, GDC 27, "Combined reactivity control system capability," and GDC 25, "Protection system requirements for reactivity control malfunctions," require that licensees maintain control rod insertability and core coolability. The NRC staff review process for new fuel designs is discussed in Standard Review Plan (SRP) 4.2, "Fuel System Design."

2.1.2 Loss-Of-Coolant Accident (LOCA) Safety Analysis

The requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," specify that each boiling or pressurized light-water cooled nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that the calculated cooling performance following a postulated LOCA conforms to the criteria contained within the rule.

The stated requirements can be met through an evaluation model for which an uncertainty analysis has been performed, as stated in 10 CFR 50.46:

> *(a)(1)(i)...the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. ...*
>
> *(ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models.*

Paragraph (b) of 10 CFR 50.46 specifies that: the calculated peak cladding temperature (PCT) shall not exceed 2200 degrees Farenheit (°F); the maximum cladding oxidation must not exceed 0.17 times the total cladding thickness before oxidation; the maximum hydrogen generation must not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel pellets were to react; the core must remain in a coolable geometry; and the core temperature shall be maintained at an acceptably low level and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

2.1.3 Non-LOCA Safety Analysis

According to 10 CFR Part 50, Section 34, "Contents of Applications; Technical Information," Safety Analysis Reports that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents must be submitted with an application. As part of the core reload process, licensees perform reload SEs to ensure that their safety analyses remain bounding for the design fuel cycle.

In addition, the licensee conducted all analyses using NRC approved codes and methods, resulting in conformance with GDC 11, "Reactor Inherent Protection," 10 CFR 50.46 (b) and other appropriate Updated Final Safety Analysis Report (UFSAR) Chapter 15 acceptance criteria. These acceptance criteria are addressed by the licensee in Tables Q12-1 through Q12-3 of the licensee's letter dated November 3, 2003.

2.1.4 Criticality Evaluation of MOX Fuel Storage in the Spent Fuel Pool (SFP)

Pursuant to 10 CFR Part 50, Appendix A, GDC 62, "Prevention of criticality in fuel storage and handling," the licensee must limit the potential for criticality in the fuel handling and storage system by physical systems or processes. The NRC staff reviewed the amendment request to ensure that the licensee will comply with GDC 62.

The regulatory requirement for maintaining subcritical conditions in SFPs is contained in 10 CFR Part 50, Section 50.68, "Criticality accident requirements." Since the licensee currently uses 10 CFR 50.68 as the licensing basis for its SFP, the NRC staff has reviewed the proposed changes against the appropriate parts of this section.

2.1.5 Technical Specification Changes

The regulations in 10 CFR 50.90 require a licensee to apply for an amendment to its license anytime a change to the TS is desired.

In an effort to reduce unnecessary changes to the TS not required by 10 CFR 50.36, "Technical Specifications," the NRC issued Generic Letter (GL) 88-16 (Reference 25), that provides guidance for relocating cycle-specific parameter limits from the TSs to a Core Operating Limits Report (COLR). This guidance allows a licensee to implement a COLR to include cycle-specific parameter limits that are established using an NRC approved methodology. The NRC approved analytical methods used to determine the COLR cycle-specific parameters are to be identified in the Administrative Controls section of the TSs.

2.2 Technical Evaluation

2.2.1 Description of MOX Fuel Lead Assembly Mechanical Design Features

The Mark-BW/MOX1 fuel assembly design is much the same as the Advanced Mark-BW fuel assembly design (Reference 26) with the exception that the Mark-BW/MOX1 design will use

MOX fuel rods instead of LEU fuel rods. The Advanced Mark-BW fuel assembly design is approved by the NRC's staff's SE in Reference 27, for use in Westinghouse three-and four-loop reactors that use a 17 x 17 fuel rod array. The Mark-BW/MOX1 fuel assembly design incorporates the same features as the Advanced Mark-BW fuel assembly design including: the TRAPPER bottom nozzle, Mid-Span Mixing Grids (MSMGs), a floating intermediate grid design, a low pressure drop quick disconnect top nozzle, and use of the approved M5 material for the cladding, structural tubing, and grids (Reference 28). The Mark-BW/MOX1 fuel assembly contains 264 fuel rods held in place by a structural cage of 11 spacer grids, 24 guide tubes, an instrument tube, and top and bottom nozzles. The MSMGs increase the flow turbulence along the hottest spans of the fuel rod. The intermediate spacer grids of the Mark-BW/MOX1 fuel assembly design are not mechanically attached to the guide thimble, instead they use ferrules around a third of the guide thimbles to limit the axial displacement of the intermediate grids. This allows the grids to float and reduces the axial forces on the guide thimbles and fuel rods.

2.2.2 MOX Fuel and Fuel Rod Design Features

The entire stack length of the Mark-BW/MOX1 fuel rod will be filled with MOX fuel pellets. The fuel rod uses a stainless steel spring in the upper plenum to prevent the formation of axial gaps during shipping and handling. The MOX fuel pellets are designed in a manner consistent with uranium oxide pellets. They are chamfered at the top and bottom to facilitate pellet loading into the rods and are dish shaped at the ends. This geometery configuration will reduce the tendency of the pellets to change into an hourglass shape under irradiation.

There are four differences between the Advanced Mark-BW and Mark-BW/MOX1 fuel designs. To accommodate the additional fission gas release from the MOX fuel, the fuel rod is slightly longer due to an increase in the upper plenum volume. This change has an impact on the required shoulder gap that the MOX fuel design topical report (Reference 29) addresses by stating that the axial gap between the top nozzle adapter plate and the fuel rods was analyzed to show that sufficient margin exists at the design rod average burnup to accommodate the fuel assembly growth and the fuel rod growth. The fuel pellet density is decreased from 96 percent theoretical density to 95 percent theoretical density. This change was made so that the theoretical density would be consistent with the MOX pellet density currently in use in Europe. Similarly, the dish and chamfer design uses the European design instead of the American design. The fuel rod burnup will also differ and be lower than the approved burnup of uranium oxide fuel rods. The lower burnup is consistent with current European burnup limits.

The isotopic mixture of weapons-grade plutonium differs slightly from the isotopic mixture of reactor-grade plutonium. Reactor-grade plutonium is derived from spent LEU fuel that is reprocessed after being discharged from a reactor core. Weapons-grade plutonium is irradiated for less time before being reprocessed. The difference in irradiation time affects the buildup of the plutonium 240, 241, and 242 isotopes. This difference in isotopes results in weapons-grade plutonium having a greater concentration of fissionable isotopes and lower concentration of absorber isotopes. This results in a decreased enrichment requirement for weapons-grade MOX fuel to achieve the equivalent burnup level and a different fuel reactivity change with burnup during the operating cycle. This difference in isotopes and their depletion with burnup has been modeled explicitly in the neutronics code.

For the MOX LTAs, Duke will use the approved CASMO-4/SIMULATE-3 MOX codes in Reference 30 that considers the effect of the weapons-grade MOX fuel isotopes in performing the core neutronic calculations. The NRC staff approval of these codes is contained in the related NRC staff SE (Reference 31) and will not be repeated here.

The use of weapons-grade plutonium instead of reactor-grade plutonium introduces slight differences into the fuel performance of the MOX fuel. These differences include the thermal conductivity, fission gas release, fuel pellet swelling, and pellet radial power distribution. These parameters have been investigated and models to predict the parameters have been developed and incorporated into the COPERNIC computer code (Reference 32). The NRC staff approval of this code is contained in the related NRC staff SE (Reference 33) and is not repeated here.

2.2.3 Design Evaluation

The fuel system design bases must reflect these four objectives as described in Section 4.2 of the SRP: 1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, 2) fuel system damage is never so severe as to prevent control rod insertion when it is required, 3) the number of fuel rod failures is not underestimated for postulated accidents, and 4) coolability is always maintained. A fuel system is "not damaged" when fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analyses. Fuel rod failure means that the fuel rod leaks and that the first fission product barrier (the cladding) has been breached. Coolability, which is sometimes termed coolable geometry, means that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat even after an accident. To satisfy these objectives, acceptance criteria are used for fuel system damage, fuel rod failure, and fuel coolability. The design bases and analyses demonstrating that the Mark-BW/MOX1 fuel design satisfies these objectives is contained in BAW-10238, MOX Fuel Design Report (Reference 29). The NRC staff approval of BAW-10238 is contained in the NRC staff's SE (Reference 34) and will not be repeated here.

2.3 Effects of MOX Fuel Lead Assemblies on Plant Operation

2.3.1 Nuclear Design

The primary active fuel material in MOX fuel is plutonium, which has different nuclear properties than conventional LEU fuel. However, even with these different nuclear properties, the impact of the four MOX LTAs will have an insignificant effect on core wide behavior. Core performance will be dominated by the nuclear properties of the remaining 189 fuel assemblies in the core. Duke performed a comparison of several key core wide physics parameters (critical boron concentration, control rod worths, moderator and fuel temperature coefficients) in a typical LEU core model that included the four MOX LTAs. The comparison showed that the physics parameters are very similar to those in a typical all-LEU core with no MOX LTAs (see Tables 3-7 through 3-10 of Reference 1).

The reload design process for a core with MOX fuel assemblies differs from the currently employed methods used with a LEU core, in the use of the NRC approved CASMO-4/SIMULATE-3 MOX code system (Reference 30), which is an update to the current CASMO-3/SIMULATE-3 code system. The licensee used the CASMO-4/SIMULATE-3 MOX codes to perform the required analyses of cycle-specific nuclear physics parameters and core transient behavior for both mixed LEU/MOX fuel cores and all-LEU fuel cores. Likewise, the licensee developed power distribution uncertainty factors that are used to evaluate predicted fuel performance with respect to established peaking limits by bench-marking the CASMO-4/SIMULATE-3 MOX code system against partial MOX fuel cores, all-LEU cores, and critical experiments. The licensee developed uncertainties for both LEU and MOX fuel assemblies. The detailed nuclear design methodology is described in Reference 30.

The presence of the Pu-239 in the MOX fuel impacts the fissionable isotopic contents of the MOX fuel. At the beginning of the cycle, the key difference between MOX fuel and LEU fuel is that Pu-239 is the predominant fissionable isotope in the MOX fuel. The substitution of a MOX fuel assembly for a LEU fuel assembly affects the assembly neutronic behavior, its neutronic interaction with the rest of the core, and the fission product concentrations. Neutronic interaction between MOX and LEU fuel assemblies arises through the energy spectrum of the neutron flux. The energy spectrum of the neutron flux for the MOX LTAs impacts the delayed neutron fraction (β_{eff}), the void reactivity effect, and the prompt neutron lifetime.

The fraction of delayed neutrons β_{eff} is lower in magnitude in MOX fuel than in LEU fuel. However, the use of four MOX LTAs will not result in any measurable decrease in the β_{eff} from a typical LEU core. Therefore, the NRC staff concludes that these slight differences in β_{eff} will not have any significant impact on plant operations.

During a LOCA, the effect of the coolant voiding as the system depressurizes is responsible for achieving reactor shutdown and maintaining low fission powers in the unquenched regions of the core. Figure 3-2 of the February 27, 2003, submittal provides a comparison of a void reactivity curve (effect on assembly k_∞) for a reference Framatome ANP designed LEU fuel assembly with a void reactivity curve calculated for a weapons grade MOX fuel assembly at the same conditions. A larger negative reactivity insertion occurs for the MOX fuel assembly than for the LEU assembly for all void fractions. This effectively suppresses the MOX fuel assembly power relative to the LEU assembly throughout a LOCA.

2.3.2 Thermal-Hydraulic and Mechanical Design

The MOX LTAs will reside within a core of Westinghouse LEU fuel assemblies. The LTAs will be surrounded by resident LEU fuel assemblies having the same physical dimensions and very similar hydraulic characteristics. The MOX LTA design employs MSMGs and the resident fuel design uses intermediate flow mixing grids. The design of these mixing grids is such that the pressure drop from the entrance to the MOX LTA to its exit is less than 4 percent lower than the pressure drop for a resident Westinghouse fuel assembly at design flow rates. Hence, flow diversion favoring one fuel assembly at the expense of the other design is expected to be inconsequential. Therefore, there will be no mixed core impact on the LOCA performance of the resident Westinghouse assemblies. As part of its normal core reload analysis process, prior to loading of the MOX LTAs into the core, the complete set of LTA LOCA calculations will be done with the average core modeled to simulate the hydraulic performance of the resident assemblies, providing a direct evaluation of the resident fuel effects on the MOX fuel lead assemblies.

2.4 Safety Analysis of MOX Fuel Lead Assemblies

2.4.1 Impact of MOX Fuel Lead Assemblies on LOCA Analysis

The fuel resident in the Catawba core prior to the insertion of the MOX LTAs is Westinghouse Robust Fuel Assembly (RFA) LEU fuel. The LOCA analysis of record for the LEU fuel is composed of large break and small break LOCA analyses. The large break LOCA was evaluated using the approved Westinghouse realistic methodology based on the WCOBRA/TRAC computer code (Reference 35). As part of the analysis of record a sensitivity study was performed to account for the mixed sources of the fuel assemblies. The licensee found the Mark-BW fuel to have an insignificant affect on the performance of the RFA fuel. The limiting case PCT was found to be 2056 °F at the 95th percentile, and the maximum local oxidation was found to be 10 percent (Reference 1, Table 3-6).

The small break LOCA was evaluated using the approved Westinghouse NOTRUMP computer code (Reference 36). A mixed core penalty for the small break LOCA was assessed to be 10 °F, and was applied to the RFA fuel. The small break LOCA PCT was significantly lower than the large break LOCA PCT.

Evaluation of the LTA large break LOCA response was performed using the Framatome ANP approved Babcock & Wilcox Nuclear Technologies LOCA methodology for recirculating steam generator plants (Reference 37). This methodology conforms to the requirements of 10 CFR Part 50, Appendix K, "ECCS Evaluation Models." Evaluation of the LTA performance under large break LOCA conditions found that the LTAs could experience a PCT of 2018 °F, and a maximum local oxidation of 4.5 percent. The lower results are due to placement of the assemblies in non-limiting core locations yielding a local peaking factor and linear heat generation rate below those of the resident fuel.

The impact of the MOX LTAs on the thermal-hydraulics for the LEU fuel currently residing in Catawba will be small since the MOX LTA fuel utilizes the Mark-BW/MOX1 design with intermediate mixing grids that has similar pressure drop characteristics to the existing LEU resident fuel.

The licensee reported the results of additional sensitivity studies on the effect of Plutonium loading and fuel-cladding gap factor in Reference 1, section 3.7. Ranging the plutonium loading from 2.3 percent to 4.4 percent affected the PCT by 1 °F, and doubling the gap size reduced the PCT by 13 °F, while increasing the maximum local oxidation by 0.1 percent. The reported analyses utilized the worst conditions from the sensitivity studies.

Based on the NRC staff review of the information provided, the NRC staff concludes that the effect of four MOX LTAs has been conservatively evaluated and has been demonstrated to be in compliance with the requirements of 10 CFR 50.46.

2.4.2 Impacts of MOX Fuel Lead Assemblies on Non-LOCA Analyses

The licensee considered the impact of the MOX LTAs on the non-LOCA UFSAR Chapter 15 events. The addition of four MOX LTAs to an otherwise all-LEU core will not have significant impact on the core average physics parameters shown in Tables 3-7 through 3-10 of the February 27, 2003, application for a typical Westinghouse pressurized-water reactor (PWR) core such as that at Catawba. The data presented in Tables 3-7 through 3-10 summarized the differences in various core physics parameters between two representative core models. One core model (designated MOX in the tables) had four MOX LTAs in locations typical of the planned LTA core. The second core model (designated LEU in the tables) had all LEU fuel assemblies. In the second core model the four MOX LTA locations were replaced with four LEU fuel assemblies that were chosen so that the boron letdown and assembly powers were as close as possible to the first core model with the four MOX LTAs. Depletion simulations were then run on both core models and the core physics parameters were calculated at various effective full power days during the simulation runs. The results of the simulated runs are presented in the referenced Tables 3-7 through 3-10, demonstrating that the presence of four MOX fuel assemblies in an otherwise all-LEU core does not produce a significant change in any of the core physics parameters.

Duke stated in the submittal that for the first cycle of operation, the four MOX LTAs will be placed in symmetric core locations that have no control rods in them. The planned core design is a checkerboard reload pattern similar to that used in previous cycles. The reload value for each physics parameter used in the safety analysis and maneuvering analysis will be confirmed to be within the reference values previously calculated as described in References 30 and 38 prior to core reload with the MOX LTAs consistent with normal licensee reload analysis processes. If any of the reload values fall outside the reference values, the core design or safety limits will be modified or changes made to the core operating limits as allowed in the COLR.

The licensee also addressed the transients and accidents that are sensitive to local physics parameters, such as: 1) control rod ejection, 2) rod cluster control assembly misoperation (withdrawal/drop), 3) steam system piping failure, and 4) fuel assembly misloading. A brief discussion of each scenario is presented below.

As stated above, during the first cycle of operation, the four MOX LTAs will be placed in symmetric locations in the core that through core loading design techniques do not require the LTAs to be controlled with a control rod, (referred to as unrodded locations). In addition, they will be located away from fuel assemblies having significant ejected control rod worth. This action is intended to reduce the impact of the power increase that would occur in a MOX LTA located in the vicinity of a rod ejection assembly. Also as alluded to above, maintaining key core parameters within present design limits insures that both core wide and localized responses to a rod ejection in a core with MOX LTAs are no more limiting than for a core containing only LEU fuel assemblies.

The licensee performed an analysis to determine energy generated in the assemblies adjacent to assemblies with an ejected rod. Specifically, the licensee performed a control rod ejection simulation with the four MOX LTAs placed in their most likely locations in a representative core. The licensee performed this analysis with the NRC approved SIMULATE-3K MOX code (Reference 30) and included appropriate conservatisms on ejected control rod worth, delayed neutron fraction, fuel temperature coefficient, moderator temperature coefficient, control rod trip worth, and trip delay time. The calculations showed that the peak enthalpy in the core at end of life, hot zero power conditions was 54 calories per gram and occurred in an LEU fuel assembly located face adjacent to the ejected control rod location. The peak enthalpy predicted in a MOX LTA was 30 calories per gram. Therefore, for the core design contemplated for the MOX

LTAs, the control rod ejection accident calculation results are lower than the current regulatory acceptance criteria.

The licensee also looked at a single control rod withdrawal and control rod drop events. These events are not expected to be impacted by the introduction of four MOX LTAs, because, as previously noted, the MOX LTAs will be in unrodded locations during in the first cycle of operation. For later cycles the assembly reactivity and rod worth for any control rod inserted in a MOX fuel assembly will be reduced to values that are below the limiting values, since these assemblies will be at least once or twice burned. Consequently, the MOX LTAs will not be placed in limiting core locations for a single rod withdrawal or drop. The reload values for the control rod worths will be maintained within the reference values contained in the safety analysis.

The licensee stated that, steam system piping failure with the most reactive rod stuck will not be impacted. The introduction of the four MOX LTAs in unrodded locations will not significantly alter the rod worth of the most reactive rod. The core reload design will control the worth of the most reactive rod and the target value for the reload will be less than the Westinghouse reload design values contained in the safety analysis for this accident.

The licensee considered operation with a misloaded fuel assembly. The NRC staff's conclusion is that administrative measures already in place for detecting misloaded assemblies, plus additional assurance provided by core power distribution measurements during plant startup, will provide ample assurance against misloaded fuel assemblies.

The administrative measures imposed by the licensee during core reloads are as equally effective for MOX fuel as they are for LEU fuel loads. By design, these MOX LTAs have a much lower thermal neutron flux than LEU fuel assemblies for the same power level. Therefore, a MOX fuel assembly misloaded into an LEU location (or vice versa) would be even more apparent from a core flux map than a misloaded LEU assembly in a LEU loaded core. In addition, the planned reactivity for the MOX fuel assemblies was chosen to be similar to the reactivity of the co-resident LEU assemblies. Accordingly, the equally reactive MOX LTAs would have no more of an impact if misloaded than a similar misloaded LEU fuel assembly. As a result, a misloaded MOX LTA would be readily detected, given that the incore detector signal for an LEU assembly loaded in a MOX LTA location would be much higher than the expected signal for the MOX LTA. As a result, core operation with a misloaded assembly will not be significantly impacted by the introduction of four MOX LTAs.

In summary, the NRC staff finds that, neutronically, all analyses were conducted using NRC approved codes and methods, resulting in conformance with GDC 11, 10 CFR 50.46 (b) and other appropriate UFSAR, Chapter 15 acceptance criteria, as provided in the response to requests for additional information (RAIs) 12-1 through 12-3, dated November 3, 2003 (Reference 12).

2.5 Criticality Evaluation of MOX Fuel Storage in the SFP

2.5.1 Background

Catawba, Units 1 and 2 SFPs each contain a single storage region with one storage rack design. All of the storage racks have the same cell center-to-center spacing (13.5 inches) and have no Boraflex neutron absorbing panels. Currently, LEU fuel assemblies are qualified as "Restricted," "Unrestricted," or "Filler," based on initial enrichment and burnup criteria. "Restricted" storage allows storage of higher reactivity fuel when limited to a specified storage configuration with lower reactivity fuel (filler assemblies). Using the same subcriticality requirements, which is a $K_{eff} \leq 0.95$, unborated, the criticality evaluation performed by the licensee for this submittal has determined an acceptable "Restricted" storage configuration for MOX LTAs in the Catawba SFPs. In this evaluation "Restricted" storage is allowed for MOX LTAs when limited to a specified storage configuration with lower reactivity LEU fuel.

The licensee evaluated the storage of MOX LTAs in the Catawba SFPs. Specifically, the analysis was performed to determine whether the current LEU fuel storage configurations and strategies employed at Catawba will be adequate to store MOX LTAs in accordance with regulatory subcriticality limits.

The typical layout of the two fuel buildings at Catawba is provided in Figure A3-1 of the February 27, 2003, submittal (Reference 1). Fresh fuel is first received in the new fuel receiving area and stored temporarily prior to being removed from its shipping container. Upon removal from the shipping container LEU fuel assemblies are placed in a new fuel storage vault (NFV) location for inspection and then are either kept in the NFV or transferred to the SFP for storage prior to reactor irradiation. MOX fuel assemblies, on the other hand, will be placed directly in the SFP once they have been received on-site. The NFVs will not be used to store MOX fuel assemblies. Fresh fuel and irradiated reload fuel assemblies (both LEU and MOX) are transported to the reactor via the water-filled Fuel Transfer Area. Discharged fuel assemblies from the reactor are also returned to the SFP through the Fuel Transfer Area.

The Catawba SFP contains just one storage region, that is, all rack cells are of the same design. The Catawba racks are arranged in a flux trap pattern, and the spacing between storage cells is sufficiently large enough (13.5 inches), and the cell walls are thick enough, as to not require Boraflex poison material to ensure sub-criticality.

The reference MOX LTA evaluated for SFP storage contains a total plutonium concentration of 4.37 weight percent up to a maximum fissile plutonium concentration of 4.15 weight percent and a maximum U-235 enrichment of 0.35 weight percent, as discussed in Section A 3.7 of the application. The Mark-BW/MOX1 fuel design parameters important to neutronic analysis (pellet diameter, fuel density, active stack length, rod pitch, etc.) are identical or nearly identical to those parameters of the current LEU fuel assemblies being used at Catawba. Table A3-2 of Reference 1 provides the plutonium and uranium nominal isotopic fractions for the unirradiated Mark-BW/MOX1 fuel. Expected manufacturing variations from the nominal values are also listed, and these variations are considered in the mechanical uncertainty analysis provided in the February 27, 2003, submittal.

2.5.2 Neutronic Behavior of MOX Fuel in the Catawba SFP

The MOX LTA's principal fissile material is Pu-239. Pu-239 is a more effective thermal and epithermal neutron absorber than U-235 (larger absorption cross-section). As a result, other thermal neutron absorbers in the MOX fuel lattice (such as boron) are worth less than in a LEU fuel lattice. The boron atoms, whether dissolved in the coolant or in lumped burnable poison rods, do not compete for thermal neutrons as effectively with the Pu-239 in MOX fuel as they do with U-235 in LEU fuel.

Another important effect is the reactivity characteristics of MOX fuel. Higher plutonium isotopes build up more quickly with burnup in MOX fuel than in LEU fuel, because the MOX fuel assemblies start with appreciable amounts of Pu-239. This difference in the buildup and burnup characteristics of plutonium isotopes results in a flatter MOX fuel reactivity curve (reactivity drops off less steeply with burnup) than an equivalent LEU fuel reactivity curve.

2.5.3 MOX Criticality Analyses

The NRC defined acceptable methodologies for performing SFP criticality analyses are provided in two documents:

(a) Proposed Revision 2 to Regulatory Guide (RG) 1.13, "Spent Fuel Storage Facility Design Basis," (Reference 39) and

(b) Memorandum from L. Kopp (NRC) to T. Collins (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" (Reference 40).

In determining the acceptability of the Catawba amendment request, the NRC staff reviewed three aspects of the licensee's analyses: 1) the computer codes employed, 2) the methodology used to calculate the reactivity, and 3) the storage configurations and limitations proposed. For each part of the review the NRC staff evaluated whether the licensee's analyses and methodologies provided reasonable assurance that adequate safety margins in accordance with NRC regulations were developed and could be maintained in the Catawba SFP.

2.5.4 Computer Codes

The SCALE 4.4 / KENO V.a computer code system (Reference 41) was employed by the licensee for analyzing the MOX and LEU fuel assembly criticality. This code system is the industry standard for conducting SFP criticality applications, and has been extensively benchmarked to both MOX fuel and LEU fuel critical experiments as well as reactor operational data (References 42-44).

As an added measure of conservatism, the licensee performed the criticality computations for this evaluation of the MOX LTAs considering only unirradiated MOX fuel. That is, no burnup credit was taken, and so no reactivity-equivalencing curves were necessary. KENO V.a does have the capability of modeling burned fuel. This requires first generating isotopic number densities, and then putting that isotopic data into KENO V.a. However, as noted above, this was not necessary for the MOX LTAs at Catawba.

The licensee reviewed several benchmark reports for using SCALE with MOX fuel that had been previously developed. References 42 through 44 describe the results from benchmarking SCALE against MOX fuel critical experiments and against isotopic measurements from reactor-irradiated (Beznau and San Onofre) MOX fuel. The benchmarking of SCALE 4.4 to MOX fuel critical experiments yielded good agreement in K_{eff} predictions, with similar biases and slightly higher uncertainties than those previously determined for LEU fuel. Additional critical experiments were reviewed and evaluated by the licensee in order to enhance the benchmarking effort. All of these MOX experiments contained a mixture of plutonium oxide and uranium oxide fuel with plutonium oxide concentrations ranging from 2.0 weight percent to 19.7 weight percent (References 45 through 48).

2.5.5 Methodology

The analyses conducted by the licensee for storing MOX and LEU fuel in the Catawba SFP storage racks were reviewed against the regulatory requirements of 10 CFR 50.68, "Criticality Accident Requirements." Given the above regulatory requirement, the MOX fuel criticality analysis for the Catawba SFP comprises the following general steps:

- The design information is obtained for the MOX LTAs and LEU fuel assemblies that are being or will be stored in the Catawba SFP. Design details for the SFP racks themselves are also necessary, in order to properly model fuel storage in these racks.
- SCALE 4.4 / KENO V.a computer models for the MOX LTA design and the highest-reactivity LEU fuel assembly design are constructed. These assemblies are modeled in the Catawba SFP storage racks.
- From these nominal models, mechanical uncertainties are determined.
- With the nominal models, K_{eff} results are determined for each MOX or MOX/LEU assembly configuration considered for that particular SFP storage rack. Various reactivity penalties are added to each K_{eff} result to account for mechanical uncertainties (from the previous step) and code methodology biases/uncertainties, which gives the no-boron 95/95 K_{eff} for that storage configuration combination.
- In the Catawba SFPs, the maximum calculated 95/95 K_{eff} results must be less than 0.95 for the no-boron cases.
- Several potential SFP accident scenarios are also evaluated, including an assembly misloading event, accidents that increase or decrease the fuel pool water temperature, and a heavy load drop (weir gate) event. The amount of soluble boron needed to keep the 95/95 K_{eff} at or below 0.95 is determined for each of these accidents, and the maximum amount required is verified to ensure it does not exceed the minimum SFP boron concentration for normal operations (2700 parts per million (ppm)) for Catawba.

In accordance with the guidance contained in References 39 and 40, the licensee performed criticality analyses of the Catawba, Units 1 and 2 SFPs. The licensee employed a methodology that combines a worst-case analysis based on the most reactive fuel type, and statistical 95/95 analysis techniques. The major components in this analysis were a calculated K_{eff} based on the limiting fuel assembly, SFP design and code biases, and a statistical sum of 95/95 uncertainties and worst-case delta-k manufacturing tolerances.

In performing the criticality analysis, the licensee first calculated a K_{eff} based on nominal core conditions using the SCALE 4.4 / KENO V.a code package. The licensee determined this K_{eff} from the limiting (highest reactivity) fuel assemblies stored in the SFP. The licensee performed its reactivity analyses for various enrichments, cooling times, plutonium concentration uncertainties, and the rack cell wall thickness. In performing these calculations, the licensee assumed appropriately conservative conditions such as assuming plutonium isotopic fractions of 94 percent Pu-239, 5 percent Pu-240, and 1 percent Pu 241. The exact plutonium isotopics of the MOX LTAs are expected to be similar to those presented in Table A3-2 of the February 27, 2003, submittal and, therefore, are less reactive than the assumed isotopics in the criticality calculations.

To calculate K_{eff}, the licensee added the methodology bias as well as a reactivity bias to account for the effect of the normal allowable range of SFP water temperatures. The licensee determined the methodology bias from the critical benchmark experiments. For each of the proposed storage configurations, the licensee analyzed the reactivity effects of the SFP water temperature. The licensee calculated the reactivity bias associated with a temperature decrease to the maximum density of water, 4 degrees Celsius (°C).

Finally, to determine the maximum K_{eff}, the licensee performed a statistical combination of the uncertainties and manufacturing tolerances. The uncertainties included the computer code system benchmarking biases and uncertainty, Plutonium concentration uncertainties, fuel density uncertainties, cell wall thickness uncertainties, center-to-center cell spacing uncertainties and mechanical uncertainties. The licensee determined these uncertainties to a 95/95 threshold that is consistent with the requirements of 10 CFR 50.68. By using the most limiting tolerance condition, (upper limit of 95/95), the licensee calculated the highest reactivity effect possible. This results in conservative margin since the tolerances will always bound the actual parameters. Once the reactivity effects for each of the tolerances were determined, the licensee statistically combined each of the manufacturing tolerances with the 95/95 uncertainties. The NRC staff reviewed the licensee's methodology for calculating the reactivity effects associated with uncertainties and manufacturing tolerances as well as the statistical methods used to combine these values.

For normal conditions in the Catawba SFPs, the maximum no-boron 95/95 K_{eff} in the MOX/LEU Restricted/Filler configuration remained below 0.95, specifically 0.9217.

For three of the accident conditions that needed to be evaluated for fuel storage (fuel assembly misload, dropped fuel assembly, and abnormal SFP temperature changes), no addition of boron was needed to maintain the 95/95 K_{eff} below 0.95 in accordance with 10 CFR 50.68.

The other accident condition considered by the licensee is the heavy load drop onto the SFP racks. The largest loads that can be carried over the Catawba SFPs are the weir gates (see their locations in Figure A3-1). These 3000 to 4000 pound steel gates, if dropped onto the SFP racks, are capable of crushing up to seven fuel assemblies. In accordance with NUREG-0612 (Reference 49), heavy load drop evaluations must assume the racks and the fuel assemblies within them are crushed uniformly to an optimum pin pitch. Figure A3-7 of Reference 1 depicts the model for this weir gate drop into the SFP. The affected assemblies are crushed into a tighter and tighter configuration until maximum reactivity is achieved. For the Catawba racks, this worst-case 95/95 K_{eff} (0.9429) still remained below the 0.95 limit, with 2700 ppm boron in the SFP. The NRC staff finds the licensee's methods conservative and acceptable.

2.5.6 Storage Rack Configurations

As mentioned in the appendix section A3.2 of the February 27, 2003, submittal, the Catawba SFPs contain one storage region. The licensee performed criticality calculations for various storage patterns in the Catawba SFP. Different types of rack storage patterns were presented in Figure A3-5 of the February 27, 2003 submittal. These racks will be used for storing MOX and MOX and LEU fuel in the Catawba SFP. The one region SFP is designated as "Restricted/Filler Storage." Fresh or irradiated MOX LTAs qualify as Restricted assemblies in these storage regions. In addition, LEU fuel assemblies that exceed their LEU Unrestricted enrichment limit or do not meet the minimum required burnup for LEU Unrestricted storage can be stored as Restricted fuel in these storage regions. Low-reactivity "Filler" fuel assemblies in this configuration must be LEU fuel assemblies that meet the Filler minimum burnup requirements provided in Table A3-5 of Reference 1. The NRC staff finds this designation to be acceptable based on its review of the licensee's submittal as described above.

2.5.7 SFP Storage Summary

In summary, the licensee has examined the feasibility of MOX fuel storage in the Catawba SFP. The reference MOX fuel design (the Mark-BW/MOX1) was identified and evaluated for storage in the Catawba SFP. The analytical methodology used included conservatisms such as neglecting axial leakage and taking no credit for burnup in MOX fuel. The results from all of these Catawba SFP criticality analyses demonstrate that a reference MOX fuel design, with a maximum fissile plutonium concentration of 4.15 weight percent, and a maximum U-235 enrichment of 0.35 weight percent, can be stored fresh or irradiated in the patterns shown in Figure A3-5, of Reference 1, without any modifications to the existing SFP storage racks. This evaluation is consistent with the planned lead assembly fuel design of 4.37 weight percent total plutonium and 0.25 weight percent U-235, demonstrating that it also can be safely stored in the SFP storage racks. The NRC staff has reviewed the licensee's evaluation and concludes that all regulatory criteria are met.

2.6 Technical Specification Changes

The use of MOX LTAs necessitates revising TS on spent fuel storage, design features, and administrative controls. The licensee submitted the proposed TS changes and technical justification for the changes in Reference 1.

TS 3.7.16 Spent Fuel Assembly Storage

Currently, the Catawba Limiting Condition for Operation (LCO) 3.7.16 specifies allowable LEU fuel storage configurations by reference to TS Table 3.7.16-1 and Figure 3.7.16-1. A revision to this LCO is proposed in this license amendment request to also allow storage of MOX LTAs as Restricted Fuel in the Catawba SFPs. The description of the Restricted Fuel classification is in Figure 3.7.16-1 that is revised to include MOX assemblies as qualifying fuel.

In addition, Surveillance Requirement (SR) 3.7.16.1 is revised since the current language refers to initial enrichment and burnup criteria, neither of which applies to MOX LTA storage. SR 3.7.16.1 currently reads: "Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with the specified configurations." The intent of SR 3.7.16.1 is to verify that a fuel assembly meets the necessary criteria for storage in the SFP. The proposed change is to delete the current wording and insert the same language as contained in McGuire SR 3.7.15.1, that reads: "Verify by administrative means the planned spent fuel pool location is acceptable for the fuel assembly being stored."

The proposed change applies equally to an LEU or MOX fuel assembly and still requires verification that any fuel assembly meets the appropriate storage requirements identified in the associated LCO prior to moving it into the SFP. The NRC staff finds the proposed administrative change to be acceptable.

TS 4.2.1 Fuel Assemblies

TS 4.2.1, Fuel Assemblies, currently specifies that each fuel assembly consist of a matrix of ZIRLO or Zircaloy fuel rods with an initial composition of uranium dioxide as feed material. A revision to add a sentence describing the MOX LTAs is proposed. The proposed sentence states: "A maximum of four lead assemblies containing mixed oxide fuel and M5™ cladding may be inserted into the Unit 1 or Unit 2 reactor core." The proposed change would incorporate the description of the LTAs into the TS. The NRC staff finds the proposed change to the TS 4.2.1 description of the fuel assemblies to be acceptable because the description is consistent with the licensing application design provided by the licensee that has also been reviewed and approved by the NRC staff.

TS 4.3.1 Criticality

The licensee proposed to revise the current language of TS 4.3.1.1 that provides a limit on the enrichment of LEU fuel that can be stored in the fuel racks, with language that provides enrichment limits on MOX fuel as well as LEU fuel. The NRC staff finds the proposed change to the TS 4.3.1.1 to be acceptable because the description is consistent with the licensing application design provided by the licensee that has also been reviewed and approved by the NRC staff.

TS 5.6.5 Core Operating Limits Report

In accordance with the guidance provided by the staff in GL 88-16 (Reference 25), Duke requested to add two approved methodologies to the list in the COLR section of the TS. The two methodologies include the "Duke Power Nuclear Design Methodology Using CASMO-4/ SIMULATE-3MOX" and "COPERNIC Fuel Rod Design Computer Code" methods of analysis.

The core neutronic parameters are evaluated using the approved "Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3MOX" (Reference 30). These codes are approved for use in analyzing reactor cores that contain both LEU and four MOX LTAs. The NRC staff approval of these codes is contained in the related NRC staff SE (Reference 31) and will not be repeated here.

Fuel behavior is analyzed using the COPERNIC code. MOX parameters have been investigated through experimental results and models to predict these parameters have been developed and incorporated into the COPERNIC computer code, Reference 32. The NRC staff approval of the COPERNIC code with the MOX parameters is contained in the related staff NRC SE (Reference 33) and will not be repeated here.

Duke has identified the approved methodologies that are used to generate the cycle-specific parameters in accordance with GL 88-16. These methods are required for the analyses of the MOX LTAs in the Catawba core and have been approved by the NRC staff for MOX fuel analyses. Therefore, the NRC staff finds that adding these two methodologies to TS 5.6.5 - Core Operating Limits Report, is acceptable.

2.7 Reactor Systems Summary

The NRC staff reviewed the analysis methodology and supporting documentation presented by Duke in the licensing application and determined that the analysis methods are acceptable. The NRC staff finds the analysis in this licensing application to be acceptable based on the determinations provided in the evaluation section of this SE and concludes that associated modifications to the TS to implement the use of four MOX LTAs into one of the Catawba units are acceptable. The NRC staff's conclusion for the subjects addressed in this SE is based on a limitation of maximum fuel rod burnup to 60,000 MWD/MThm.

An LTA is designed to gather data on fuel performance. The LTAs are typically based on current production designs and are irradiated to obtain fuel performance data. In the past, as fuel performance data was obtained, it indicated that slight design modifications would be necessary. As a result, minor design changes have been implemented into the current production designs to retain high fuel reliability. Data from LTAs will also provide the basis for improved fuel designs and analytical models.

An LTA is a fuel assembly based on a currently available design. An LTAs' fuel cladding material is an NRC-approved cladding material. The assembly will receive pre-characterization prior to undergoing exposure in the "test" cycle that would permit the assembly to exceed the burnup limits of the COPERNIC fuel behavior code. The fuel assembly has been analyzed using currently approved fuel performance design models in COPERNIC and methods in BAW-10238 and demonstrated that the currently approved design limits are met for the extended burnup. Because the purpose of an LTA is to gather data on fuel performance including above approved burnup limits, the models and methods used for evaluation of the LTAs are not required to be approved to the projected bumups. The available data on MOX fuel performance above 50,000 MWD/MThm, while not statistically significant, indicates that the approved models can predict the fuel behavior and therefore are appropriate for use to this burnup so modifications to the approved models are not necessary. Use of the models above the approved burnup limit will only be used for analysis of the LTAs. Model performance will be shared with the NRC along with the PIE data results.

Pre-characterization measurements shall be assessed with the fuel performance design models and methods to ensure that the assembly will not exceed design limits after its cycle of exposure. Pre-characterization is the measurement of particular fuel performance parameters before the start of the cycle. Upon completion of the cycle of exposure, the LTA shall under-go a Post Irradiation Examination (PIE). Post Irradiation Examination of the LTA shall be documented in a PIE report and results of the PIE assessment shall be factored into future analysis to ensure that appropriate conservatisms are being maintained. In addition, tracking of the data results will provide the basis for developmental model creation to more accurately model fuel performance and to capture fuel performance fundamentals. Reports containing data gathered by the vendor/utility from the LTA program shall be presented to the NRC. Model performance shall also be tracked against data and presented to the NRC.

Because the fuel performance models are being extrapolated to burnups that have not been approved, the pre-characterization provides a measure of how much margin exists for a given design criterion to its limit, based on model predictions compared to the pre-characterization measurement. Comparison of pre and post cycle values, obtained from the PIEs, will yield the incremental effects that the cycle of exposure has on the LTAs. This provides a measure of whether an unknown phenomenon exists and is occurring in the LTAs. It also provides a very accurate measure of how well the predictive fuel performance models are behaving for the cycle of exposure.

3.0 DOSE CONSEQUENCES

3.1 Regulatory Evaluation

This SE section addresses the impact of the proposed changes on previously analyzed design basis accident (DBA) radiological consequences and the acceptability of the revised analysis results. The applicable regulatory requirements are the accident dose guidelines in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," as supplemented by accident-specific criteria in Section 15 of the SRP, the accident dose criteria in 10 CFR 50.67, "Accident Source Term," as supplemented in Regulatory Position 4.4 of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and 10 CFR Part 50, Appendix A, GDC 19, "Control Room," as supplemented by Section 6.4 of the SRP. Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance provided in the following documents in performing this review.

- Safety Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors"
- Safety Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"
- RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors"
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"
- SRP Section 15.0-1, "Radiological Consequence Analyses Using Alternative Source Term"
- SRP Section 15.1.5, "Steam System Piping Failures Inside and Outside Containment (PWR)," Appendix A
- SRP Section 15.3.3, "Reactor Coolant Pump Rotor Seizure"
- SRP Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Appendix A
- SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment"
- SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Rupture (PWR)"
- SRP Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," Appendix A and Appendix B
- SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents"
- NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook"

Since the guidance identified above was written for LEU fuel, the NRC staff considered appropriate changes related to the MOX fuel. These adjustments are addressed in this report.

The NRC staff also considered relevant information in the Catawba UFSAR, TSs, and several technical reports. The technical reports are listed as References 50 through 63.

3.2 Technical Evaluation

3.2.1 Background

The LEU fuel used in U. S. nuclear reactors consists of uranium oxides in which the concentration of U-235 is increased over that in the naturally occurring distribution of the uranium isotopes during manufacture, such that U-235 constitutes about 4 to 5 percent of the uranium by weight. In fresh LEU fuel, U-235 is the fissionable component. The concentration of U-235 is specified by the fuel designer and produced during the enrichment process. Prior to irradiation, LEU fuel has no significant plutonium concentration. During irradiation, however, U-238 absorbs neutrons and transmutes to the various isotopes of plutonium. Some of these plutonium isotopes are fissionable and add to the power output of the LEU fuel.

In the beginning of the U. S. nuclear reactor program, it was anticipated that the fuel cycle would be closed by reprocessing spent fuel to recover the usable plutonium and uranium for use as MOX fuel in reactors. In the case of MOX fuel, Pu-239 rather than U-235 provides most of the fissionable material. The plutonium obtained from reprocessing is blended with natural or depleted uranium during manufacture to obtain the plutonium concentration specified by the fuel designer. Demonstration projects conducted in the 1970's and 1980's resulted in the irradiation of MOX fuel assemblies at several U.S. power reactors including San Onofre, Ginna, Quad Cities, and Dresden. Similar efforts proceeded in foreign countries during this period. Domestic MOX research ended by 1980 as a result of a presidential executive order against reprocessing irradiated fuel. However, foreign programs continued and commercial MOX use is a reality in Japan, India, and a number of European countries today. As of the end of 2001, more than 30 thermal reactors worldwide use MOX fuel. Since the plutonium in this commercial MOX fuel was obtained from reprocessing spent reactor fuel, this fuel is known as reactor-grade MOX fuel. Since the plutonium in the proposed MOX LTAs is obtained from weapons material inventories, this fuel is known as weapons-grade MOX fuel.

Table 1: Nominal Unirradiated Fuel Isotopics, %

Isotope	U.S. LEU	European MOX	Proposed MOX LTA
wt% ^{234}U / U	0.03	----	----
wt% ^{235}U / U	3.2	0.24 - 0.72	≤0.35
wt% ^{236}U / U	0.02	----	----
wt% ^{238}U / U	96.75	92.77	95.28
wt% ^{238}Pu / Pu	----	0.88 - 2.40	≤0.05
wt% ^{239}Pu / Pu	----	53.8 - 68.2	90.0 - 95.0
wt% ^{240}Pu / Pu	----	22.3 - 27.3	5.0 - 9.0
wt% ^{241}Pu / Pu	----	5.38 - 9.66	≤1.0
wt% ^{242}Pu / u	----	2.85 - 7.59	≤0.1
wt% Pu / HM	----	4.0 - 9.0	4.37
wt%Fissile / HM	3.2	3.65 - 5.25	≤4.15

HM = Pu + U. May not sum to 100% due to rounding and ranges. Derived from data in licensee submittal, ORNL/TM-2003/2 [Ref.1], NUREG/CR-0200 V1 [Ref.2]

The two MOX fuel types differ in that the relative concentrations of plutonium and uranium and the distributions of their isotopes differ. Table 1 above compares the distribution of fissile and non-fissile isotopes in typical LEU fuel, typical commercial reactor-grade MOX fuel, and the proposed MOX LTAs. The differences in the initial fuel isotopics are potentially significant to accident radiological consequence analyses since the distribution of fission products created depends on the particular fissile material. If the fissile material is different, it follows that the distribution of fission products may be different. For example, one atom of I-131 is created in 2.86 percent of all U-235 fissions, whereas one atom of I-131 is created in 3.86 percent of all Pu-239 fissions. This is an illustrative example only in that the radionuclide inventory in the fuel at the end of core life depends on more than fission yield. Nonetheless, this shift in the fission product distribution needs to be evaluated for its impact on the previously calculated radiological consequences of DBAs.

The LEU fuel is enriched in the U-235 isotope, an operation that occurs on a molecular scale while the UO_2 fuel is in the gaseous phase. This processing results in fuel pellets with a high degree of homogeneity and uniform grain sizes. The proposed MOX LTA fuel will be manufactured in a process that involves blending of UO_2 and PuO_2 powders to achieve the desired Pu content. The MOX fuel pellets, therefore, are not as homogeneous as an LEU fuel pellet. This difference in pellet structure has the potential to affect the diffusion of fission gases through the fuel pellet and may impact the fraction of the pellet fission product inventory that is in the fuel rod gap between the pellet outer surface and fuel clad inner surface (i.e., gap fraction). It is generally understood that the fission gas release (FGR) rate for MOX fuel is greater than that for LEU fuel, given comparable enrichments and burnups. This behavior is primarily explained by the lower thermal conductivity of MOX fuel pellets that results in higher fuel temperatures than in LEU rods. Since the gap fractions are an input to the analyses of calculated doses from non-core melt DBAs, changes to the gap fractions associated with MOX fuel need to be considered.

In addition to the possible impact on gap fractions, the increased FGR has an impact on the fuel rod internal pressurization. The proposed MOX LTAs have fuel design features intended to compensate for the increased FGR. The decontamination of radioiodine released from fuel rods damaged during a design basis fuel handling accident (FHA) is a function of the rate of a bubble rising through the overlaying water in the SFP and the bubble size distribution that are functions of the fuel rod internal gas pressure. If it can be shown that the internal pressurization is unchanged or increased only slightly, then current analysis decontamination assumptions remain valid.

In summary, the NRC staff's review was focused on the potential impacts of the following three characteristics of weapons-grade MOX fuel:

(1) The fission product inventory in a MOX LTA is expected to be different from that of an LEU assembly due to the replacement of uranium by plutonium as the fissile material.

(2) The fraction of the fission product inventory in the gap region of a MOX LTA is greater due to the increased FGR associated with higher fuel pellet centerline temperatures of MOX fuel.

(3) The increased FGR can result in higher fuel rod pressurization.

The configuration of the MOX LTA is nearly identical to that of the LEU fuel assemblies currently in use at Catawba. There is no change in rated thermal power or any significant changes to other plant process parameters that are inputs to the radiological consequence analyses. As such, the only impacts on these analyses would be from changes in the fission product inventory and the gap fractions, and in the case of the FHA, changes in the SFP decontamination factor, if any.

In performing this review, the NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of DBAs, performed by Duke in support of its proposed license amendment. Information regarding these analyses are provided in Section 3.7.3 of Attachment 3 of the February 27, 2003, submittal and in supplemental letters dated November 3 and December 10, 2003, and February 2, March 1, and 16, 2004. The NRC staff reviewed the assumptions, inputs, and methods used by Duke to assess these impacts. The NRC staff performed independent calculations to confirm the conservatism of Duke's analyses. However, the findings of this SE are based on the descriptions of Duke's analyses and other supporting information submitted by Duke. Only docketed information, supplemented by technical information in reports identified in the references, was relied upon in making this safety finding.

3.2.2 Radiological Consequence Analyses

The radiological consequences of postulated accidents were discussed in Section 3.7.3 of the February 27, 2003, submittal. In its response dated November 3, 2003, to the NRC staff's RAIs, Duke provided supplemental information on the evaluation of the impact of MOX LTAs on DBAs. In this response, Duke stated that it had performed a combination of evaluations and analysis to assess the impact of the MOX LTAs. Duke described a process in which the various DBAs were categorized on the basis of how many fuel assemblies would be affected by that event. Duke identified two major categories:

(1) Accidents involving damage to a few fuel assemblies. These include FHAs and the weir gate drop (WGD) accident. A small number of assemblies are involved such that if the MOX LTAs were in the damaged population, as conservatively assumed, they would comprise all or a significant portion of the population.

(2) Accidents involving damage to a significant portion of the entire core. These accidents range from the locked rotor accident (LRA) with 11 percent core damage, the rod ejection accident (REA) with 50 percent core damage, to the large break LOCA with full core damage. In this case, the relative effect of damaging all four MOX LTAs is reduced as the fuel damage population increases. For example, in a DBA LOCA, all 193 fuel assemblies are postulated to be damaged. The four MOX LTAs constitute just 2 percent of all the fuel assemblies in the core.

To these categories, the NRC staff would add a third:

(3) Accidents whose source term assumptions are derived from reactor coolant system (RCS) radionuclide concentrations. These include, steam generator tube rupture, main steamline break, instrument line break, waste gas decay tank rupture, and liquid storage tank rupture.

The radionuclide releases resulting from these events are based on established administrative controls that are monitored by periodic surveillance requirements, for example: RCS and secondary plant specific activity LCOs, or offsite dose calculation manual effluent controls. Increases in specific activities due to MOX LTAs, if any, would be limited by these administrative controls. Since the analyses were based upon the numerical values of these controls, there can be no impact of MOX LTAs on the previously analyzed DBAs in this category.

3.2.3 MOX LTA Fission Product Inventory

Duke calculated the fission product inventory of the proposed MOX LTAs using the NRC-sponsored SCALE (Standardized Computer Analyses for Licensing Evaluation) system, version 4.4. SCALE is a multi-purpose computational system for analyses of nuclear facilities and spent fuel packaging. SCALE contains analytical modules that address topics such as radiation source terms and shielding, criticality safety, high-level waste classification, lattice physics, and heat transfer. SCALE was developed at the Oak Ridge National Laboratory (ORNL) for the NRC. It is currently maintained by ORNL under the co-sponsorship of NRC and DOE, under a software quality assurance (QA) program that includes configuration management, module and data revision control, documentation, verification and validation programmatic elements. SCALE module results have been benchmarked against actual measurements and against other domestic and international analytical capabilities.

Duke selected the SAS2H control module of SCALE for performing this work. SAS2H uses the point depletion code ORIGEN-S to compute time-dependent concentrations of a large number of nuclides. The nuclides are simultaneously generated or depleted through neutronic transmutation, fission, radioactive decay, input feed rates, and physical or chemical removal rates. ORIGEN-S is a variant of the ORIGEN (and ORIGEN2) code that was modified to replace the "pre-packaged" cross-section libraries with the ability to access a cross-section library created specifically for the problem defined by the user's input. The SAS2H control module processes the user's input, calls several modules to produce the ORIGEN-S data input and time-dependent cross-section libraries, and calls ORIGEN-S to perform the burnup and decay analysis. Because of this structure, SAS2H and ORIGEN-S calculations can be based on parameters that precisely match those of the specific problem being considered. This is a significant advantage for the present evaluation since it would address nuclides and reactions not included in pre-packaged LEU libraries.

Duke applied SAS2H to a series of cases structured to model combinations of accident sequence, MOX LTA plutonium concentrations, and LTA power histories. Duke states that the models were built including conservatisms. In particular, the NRC staff notes that Duke assumed that the plutonium concentration of the pins in the LTA was 5 percent. The nominal LTA fuel design calls for 176 fuel pins with a plutonium concentration of 4.94 percent; 76 pins at 3.35 percent, and 12 rods at 2.40 percent. The nominal average plutonium concentration is 4.37 percent. Conservatively basing the calculation on a 5 percent plutonium concentration provides margin to compensate for differences (e.g., manufacturing tolerances and power history differences) between the nominal design and the actual fuel as loaded in the core. Duke described the modeling of these variables in greater detail in its RAI response dated November 3, 2003. Duke also defined and analyzed an equivalent LEU assembly based on assembly burnup, LEU enrichment, and MOX fuel plutonium concentration.

The NRC staff reviewed Duke's use of the SCALE code, the SAS2H modules and the general approach taken. The NRC staff also reviewed the input values Duke used with SAS2H. The NRC staff finds SCALE, ORIGEN-2 and SAS2H to be appropriate analytical methodologies. The NRC staff also performed some confirmatory analyses and comparisons. First the NRC staff compared the Duke results to data derived from a report prepared by Sandia (Reference 52). The calculations described in that report were performed using ORIGEN2 with a PWR plutonium cross-section library. The NRC staff performed its own SAS2H analysis. Based on its review and confirmatory calculations, the NRC staff concluded that the Duke inventory analysis, as described in the docketed materials, used an appropriate analytical methodology and appropriate input parameters to assess the fission product inventory of a MOX LTA.

3.2.4 Impact on Gap Fractions

The Catawba licensing basis is in transition between the traditional TID-14844 (Reference 53) source term and the alternative source term (AST) from RG 1.183. Duke revised the Catawba licensing basis to selectively implement the AST for the fuel handling and WGD accidents by License Amendments Nos. 198 and 191 dated April 23, 2002, for Units 1 and 2 respectively. The licensing basis gap fractions for the FHA and WGD were those provided in Table 3 of RG 1.183. There are no licensing basis gap fractions for the DBA LOCA as TID-14844 assumes an immediate full core melt release. For the remaining accidents, the gap fractions are those specified in Safety Guide 25 and RG 1.77. Duke proposed a 50 percent increase in the current guidance on gap fractions to bound the expected increase due to the MOX LTAs. In support of the conclusion that this assumed increase would be bounding, Duke advanced an argument based on the work of an expert panel convened by the NRC to evaluate the applicability of the fission product release fractions specified in NUREG-1465 (Reference 54).

Duke stated that:

> Since [RG 1.183] Table 3 is based upon expert panel work which was published in [NUREG-1465] and the panel saw similarities in gap release rates between LEU and MOX fuel, it could be inferred that the gap release rates in [RG 1.183] Table 3 should also be valid for MOX fuel gap releases.

The NRC staff does not believe that this inference is adequate justification for assuming that the non-LOCA gap fractions in Table 3 of RG 1.183 would be applicable to the MOX LTAs as stated by Duke. The expert panel was not tasked to consider gap fractions for non-LOCA events. The panel's deliberations were limited to LOCAs and other severe accidents involving a significant portion of the core. Finding that a core wide average gap fraction might not change does necessarily support a conclusion that the gap fraction for the limiting fuel assembly has not been affected. Duke appears to challenge its own inference by noting in Response Q3(g) of the November 3, 2003, letter "... current data comparisons show fission gas release from MOX fuel pellets is generally greater than the fission product release from LEU fuel ..."

Duke supplied a graph of measurements of FGRs from European (reactor grade) MOX fuel in its November 3, 2003 letter. By letter dated February 2, 2004, Duke provided an explanation of this graph. The majority of the plotted LEU data were obtained from 17 x 17 matrix fuel rods irradiated in Electricité de France (EdF) facilities. The MOX data were obtained from fuel rods irradiated in EdF PWRs operated in base-loaded or in load-following conditions. The fuel pellets were fabricated from depleted uranium and reactor-grade plutonium using the MIMAS process (Reference 55). The MOX fuel assemblies are radially zoned with typical plutonium concentrations ranging from 2 to 6 percent. The maximum axially-averaged linear heat generation rate (LHGR) during irradiation ranged from 4.7 kW/ft to 7.4 kW/ft. Following irradiation, the fuel rods were punctured and the gas collected and analyzed for helium, xenon, and krypton. With the exception of the origin of the feed plutonium, the irradiated fuel configuration and fabrication method of the MOX LTAs is closely comparable to the fuel assemblies that underwent the post-irradiation examinations for the EdF PWRs. This database is essentially the same as the fission gas data that was used to develop and qualify the COPERNIC FGR model. Although the maximum MOX LTA exposure will be 7.9 kW/ft (Reference 20, Table Q6-1), exceeding the range of the experimental data, this occurs only for a short period of time at the beginning of the cycle. Duke asserts that the FGR is generally insensitive to power peaking of this magnitude that occurs early in fuel lifetime. Given the relatively short decay half-lives of the more significant radionuclides, the NRC staff agrees.

The NRC staff had an independent analysis of FGR performed using the FRAPCON-3.2 computer code (Reference 56). Duke provided detailed fuel configuration data and projected power histories for the MOX LTAs. For this analysis, the FRAPCON-3.2 code was modified so that two FGR models were used. The primary model in FRAPCON-3.2 is the Massih model. The added model is based on the ANS-5.4 model that can predict the release of both stable noble gas elements and the radioisotopes. Although the Massih model is considered to be the more reliable model, it is only capable of predicting the release of stable noble gases. As such, both the ANS-5.4 and Massih models were run. The ANS-5.4 model calculation was structured to calculate the release values for the radioisotopes based on the Massih predictions for the stable isotopes. Consistent with ANS-5.4 recommendations, the diffusion coefficient for I-131 was assumed to be seven times that used for the noble gases and the diffusion coefficient for cesium isotopes was assumed to be two times that for the noble gases. Also consistent with ANS-5.4 recommendations, the release fractions for the longer-lived radionuclides Kr-85, Cs-134, and Cs-137 were calculated using the stable gas routine within the ANS-5.4 model and the diffusion coefficients identified above.

The accuracy of the FRAPCON-3.2 release fraction predictions is dependent on the data input for the analysis. It is particularly sensitive to the power history. Duke characterized the power history as being conservative and bounding for the expected MOX LTA power histories. The power history was tabulated as the fuel burnup at each time step and the radial peaking factor $F_{\Delta H}$ (F delta-H). From the values of time and burnup, the LHGR values for each time step can be calculated. The LHGR can also be calculated from the core average LHGR and the radial peaking factors. The two derived power histories are slightly different. As such, the release fraction calculation was performed for both power histories. Additionally, the average core power was increased by 5 percent to compensate for possible differences between the expected power history and the actual irradiation of the MOX LTAs. Peak gas releases and end-of-life gas releases were considered.

For a given power history, the uncertainty in the release fractions can be estimated based on the standard deviation of the FRAPCON-3.2 predictions of stable noble gases compared to the measured data from LEU and MOX fuel. The standard deviation for LEU fuel stable gas predictions is 0.026 absolute release fraction and the standard deviation for MOX fuel stable gas predictions is 0.048 absolute release fraction. The NRC staff has opted to use an overall standard deviation of 0.031 absolute release fraction for noble gases. The standard deviation for the radioisotopes was obtained by scaling the stable gas standard deviation by the ratio of the predicted nominal release of the radioisotope divided by the stable noble gas release value. The staff based this decision on the following considerations: (1) the mechanisms for release for LEU and MOX fuel is the same with the primary differences being the diffusion coefficients for MOX versus LEU–the uncertainties should be similar, and (2) the calculated value for MOX fuel is higher because of the limited number of MOX experimental data points that were considered compared to those considered for the LEU uncertainty.

Table 2: Release Fractions (Gap Fractions), in percent

	Kr-85	I-131	Other Noble Gases	Other Halogens	Alkali Metals
RG 1.183 Table 3	10.0	8.0	5.0	5.0	12.0
Duke Power Assumption	15.0	12.0	7.5	7.5	n/a
Staff Analysis EOC 3	13.5 (14.9)	0.2	0.1	n/a	17.7 (19.1)
Staff Analysis Peak Value	14.4 (16.8)	9.5 (10.5)	3.2 (3.5)	n/a	19.1 (21.6)

Table 2 shows the release fraction values obtained by this analysis. For comparison, the RG 1.183, Table 3 values and the release fractions assumed by Duke are tabulated. The bases of the NRC staff's values include the 5 percent power uncertainty factor discussed above and 2-sigma uncertainty adjustments. The peak values occur at the end of cycle two, corresponding to a projected burnup of about 47 GWD/MThm. The values in the parentheses are based on the power history derived from the $F_{\Delta H}$ values, as discussed above.

The only halogen considered by the ANS-5.4 model is I-131. The categorization of radioisotopes as noble gases, halogens, and alkali metals is on the basis of similarity in chemical behavior. The NRC staff believes that the chemical behavior of the iodine isotopes (and those of bromine) is sufficiently similar that the observed increase in the I-131 release fraction can be applied to the RG 1.183 "other halogens" value of 5.0 percent to obtain a value appropriate for the MOX LTAs. It is significant to note that the observed differences between the radionuclides Kr-85, Kr-87, Kr-88, Xe-133, and Xe-135 are correlated to the difference in the half-lives of these radionuclides. The iodine radioisotope half lives for I-132, I-133, I-134, and I-135, are much shorter than that for I-131. As such, the gap fractions for these radioisotopes would be less than that for I-131. Since the former radioisotopes are not significant contributors to dose, the NRC staff finds that the licensee's assumption of 7.5 percent as the gap fraction for the "other halogens" category is reasonable.

The NRC staff's peak values in Table 2 are bounded by the release fractions assumed by Duke with the following exceptions:

- Duke did not consider the increase in the gap fraction assigned to the "alkali metals" group. Duke stated that cesium need not be considered for the FHA and WGD in that the RG 1.183 acceptable assumptions for a FHA analysis provide that particulates, such as cesium, are retained by the SFP. The NRC staff agrees with this assessment. However, the NRC staff considered whether or not the significant increase in the gap fraction for cesium needed to be considered for the remaining accidents. At the present time, Duke has not been approved for use of an AST for DBAs other than the FHA and WGD. The current licensing basis analyses for the LOCA, LRA, and REA events are based on the TID-14844 source term that includes only noble gases and halogens. The increase in Cs-137 is not relevant to the current licensing basis at Catawba and is, therefore, not an issue for the present amendment request. The NRC staff notes that, if Duke should implement an AST at Catawba in the future, the gap fraction associated with Cs-137 will need to be explicitly addressed in the DBA analyses.

- The NRC staff analysis estimated a gap fraction for Kr-85 of 16.8 percent, which is an increase of 68 percent over the Kr-85 gap fraction for LEU and is greater than the 50 percent increase assumed by Duke. Given the relatively low significance of Kr-85 as a dose contributor in comparison to other radionuclides and the relatively small Kr-85 inventory in the core, the impact of this difference in the Kr-85 gap fraction on FHA and WGD postulated doses will be negligible. There is no impact on the comparative analysis of the LOCA, LRA, or REA events since Duke based this analysis on the difference in the I-131 inventories.

Based on the above, the NRC staff has determined that Duke's assumption of a 50 percent increase in the gap fractions in Table 3 of RG 1.183 is acceptable for the purposes of the present amendment request only, and should not be construed as a precedent for another licensing action at Catawba or any other reactor site. The gap fraction analyses are strongly dependent on the projected power history. If the actual power history is to deviate significantly from the projected power history, the gap fraction evaluation should be re-visited.

The NRC staff did not model reactivity insertion accidents in the FRAPCON-3.2 assessment of gap fractions. The NRC has a generic program plan for high-burnup fuel to address recent insights from reactivity insertion accident experiments performed on high burnup fuels. The criteria and analyses for reactivity accidents were identified for resolution (References 57 and 58). The issues identified in this program plan are generic to light-water power plants and fuel types and are, therefore, being resolved on a generic basis. They are not unique to MOX LTAs and need not be considered for the present amendment. The need for further regulatory actions, if any, will be determined based on the outcome of the program plan.

3.2.5 At-Power Core Damage Accidents

Duke considered the impact of the four MOX LTAs on the LOCA, LRA, and REA events. These DBAs were not explicitly re-calculated. Since the dose can be shown to be proportional to the fuel assembly inventory and gap fractions, Duke's approach to evaluating the potential impact of the MOX LTAs was to compare the relative differences in radionuclide inventory and determine a correction factor that could be applied to the results of the current analyses of record for these events. Duke used the MOX LTA and the equivalent LEU assembly source terms developed for the FHA and WGD accident re-analyses to perform this assessment. Duke selected the thyroid dose due to I-131 as the evaluation benchmark since the thyroid dose is typically more limiting than the whole body dose given the lesser margin between calculated thyroid doses and its associated dose criterion. Also, I-131 is generally the most significant contributor to thyroid dose due to its abundance and relatively long decay half-life. Duke determined that the I-131 inventory in a MOX LTA was 9 percent greater than that of an equivalent LEU fuel assembly. Since the observed increases in the other iodine isotopes were less than 9 percent, this factor could be conservatively applied to all iodines. Duke applied this 9 percent increase as a multiplier to the dose results in the current analyses of record for the LOCA, LRA, and REA events as discussed in its letter dated March 16, 2004. Duke also applied a correction factor of 1.5 to reflect the increased gap fractions associated with the MOX LTAs.

The current analyses of record assume that all fuel assemblies (193) are affected by a LOCA. For the LRA, 11 percent of the core (21 assemblies) are assumed to be affected; for the REA, 50 percent of the core (97 assemblies) are assumed to be affected. Duke assumes that the four MOX LTAs are in the affected fuel population replacing four LEU assemblies for each of these events. Duke's results are discussed below and are shown in Table 4 of this SE.

- For the LOCA, the four MOX LTAs represent only 2.1 percent of the 193 assemblies in the core. Thus, the potential increase in the iodine release and the thyroid dose is 1.32 percent. The thyroid dose increased to 90.2 rem at the exclusion area boundary (EAB), 25.3 rem at the low population zone (LPZ), and 5.37 rem at the control room. (Duke also applied this increase to the TEDE results from a LOCA analysis that was submitted as part of a separate proposed AST license amendment request that is still under review. Since a LOCA AST analysis is not part of the current licensing basis, and since the scaling did not consider the impact of the other nuclides that contribute to the TEDE, the staff did not rely on it in approving the MOX LTA amendment request.)

- For the LRA, the four MOX LTAs represent only 19 percent of the 21 affected assemblies in the core. Thus, the potential increase in the iodine release and the thyroid dose is 12 percent. The thyroid dose increased to 4.1 rem at the EAB, and 1.3 rem at the LPZ.

- For the REA, the four MOX LTAs represent only 4.1 percent of the affected 97 assemblies in the core. Thus, the potential increase in the iodine release and the thyroid dose is 2.63 percent. The thyroid dose increased to 1.03 rem at the EAB, and remained at 0.1 rem (increase masked by numeric rounding) at the LPZ.

Duke assessed the control room dose only for the LOCA since the control room doses from a LRA or REA are bounded by those for the LOCA. This is acceptable to the NRC staff.

A scaling approach is acceptable to the staff if the scaling represents the difference from the current licensing basis (LEU) to the proposed licensing basis (LEU plus MOX) and that the projected doses meet applicable acceptance criteria. In this case, Duke compared the I-131 inventory for a MOX LTA with that for an equivalent LEU assembly. Duke stated in its letter dated March 1, 2004, that the equivalent LEU source term was used in the interest of isolating the observed difference due to the difference in fuel isotopics between LEU fuel and the proposed MOX LTAs. Although the staff agrees that Duke's approach would isolate the differences, the staff believes that the before and after dose comparison is inconclusive since the "before" doses were not based on the equivalent LEU assembly. Duke also stated that the equivalent LEU assembly source term bounded the current licensing basis source term, which Duke characterized as a conservative situation. However, since the equivalent LEU assembly inventory appears in the denominator when calculating the dose multiplier, the dose result may not be conservative.

To address this, the staff performed an independent analysis. UFSAR Table 15-12 provides core inventory that is the basis of the current analyses of record source term. This table provides an I-131 core inventory of 8.9E7 Curies. This core inventory equates to an average fuel assembly I-131 inventory of 4.61E5 Curies. The MOX LTA I-131 inventory for a FHA or WGD analysis is 8.81E5 Curies. Dividing out the radial peaking factor of 1.65, to obtain a level comparison basis, yields an average MOX LTA I-131 inventory of 5.34E5 Curies. As such, the I-131 inventory in a MOX LTA is 15.8 percent greater than that used in the current analyses of record for the LOCA, LRA, and REA events.

- For the LOCA, the potential increase in the iodine release and the resulting thyroid dose is 1.53 percent. The thyroid doses increased to 90.4 rem at the EAB, 25.4 rem at the LPZ, and 5.38 rem at the control room.

- For the LRA, the potential increase in the iodine release and the resulting thyroid dose is 14.0 percent. The thyroid doses increased to 4.22 rem at the EAB, and 1.37 rem at the LPZ.

- For the REA, the potential increase in the iodine release and the resulting thyroid dose is 3.04 percent. The thyroid doses increased to 1.03 rem at the EAB, and 0.103 rem at the LPZ.

The NRC staff finds that the results of Duke's analysis for LOCA, LRA and REA events are acceptable in that the postulated accident doses will continue to meet applicable dose criteria. However, Duke's use of an equivalent LEU assembly is inappropriate. The NRC staff bases its finding on the minimal differences between the doses determined by Duke and those determined by the NRC staff, and on both sets of dose results meeting applicable dose criteria. This finding should not be construed as a precedent that Duke's comparative analysis approach will be found acceptable in another licensing action at Catawba or at any other reactor site.

The NRC staff considered the possible impact of radionuclides other than I-131 on the results obtained by Duke. Since the current analyses of record are based on the traditional TID14844 source term, only the krypton and xenon radionuclides need to be considered. The inventory of krypton isotopes in a MOX LTA is less than that in a corresponding LEU assembly. The inventory of some xenon isotopes in a MOX LTA increased between 7 to 11 percent with the exception of Xe-135, which increased by 189 percent. Using the MOX / LEU ratios (including a 1.5x gap fraction increase) developed above and conservatively considering only those noble gases that increased in concentration, the maximum increase in the whole body dose would be about 2 percent for the LOCA, 16 percent for the LRA, and 3.5 percent for the REA. The NRC staff found that the resulting whole body doses would remain within regulatory criteria.

3.2.6 Fuel Handling Accident and Weir Gate Drop Accident Radiological Consequences

Duke assessed the MOX LTA impact on doses for the FHA and WGD accidents by re-calculating the analyses of record with updated input data. Duke stated that with the exception of the fuel assembly isotopics the analysis models were basically the same as the FHA and WGD models described in Duke's license amendment request dated December 20, 2001. The staff reviewed those descriptions and approved that amendment request by letter dated April 23, 2002. That amendment selectively implemented the AST for the FHA and WGD at Catawba. Duke did revise the control room X/Q value for the unit vent releases from that approved in the earlier amendment. In lieu of assuming that the dual control room intakes have balanced flow rates, Duke assumes that 60 percent of the air being drawn into the control room is from a contaminated stream. This approach is consistent with the guidance of RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," and is, therefore, acceptable.

The results of these two re-analyses were tabulated in Tables Q3(b)-3 and Q3(b)-4 of Duke's submittal dated November 3, 2003, and are shown in Table 4 of this SE. Duke projected radiological consequences for the FHA of 2.3 rem TEDE at the EAB, 0.34 rem TEDE at the outer boundary of the LPZ and 2.1 rem TEDE in the control room, increases of about 64 percent over the previous analysis for LEU fuel. Duke projected radiological consequences for the WGD of 3.5 rem TEDE at the EAB, 0.5 rem TEDE at the outer boundary of the LPZ, and 3.3 rem TEDE in the control room, increases of about 58 percent over the previous analysis for LEU fuel. These results remain within applicable regulatory limits.

As noted above, the FGR for MOX fuel is greater than that for LEU fuel. The increase in FGR will result in increased fuel rod pressure. The scavenging of released radioiodine by the SFP water is a function of the bubble transit time through the overlying pool water which, in turn, is a function of fuel rod pressurization. The acceptable effective pool decontamination factor of 200 is derived from data from tests involving fuel rods pressurized to no more than 1200 psig. Duke stated that its analyses of the internal rod pressure would remain below the 1300 psig that Duke states is their criteria. Duke uses a Westinghouse methodology to justify the acceptability of the 1300 psig pin pressure. The NRC has not endorsed the cited Westinghouse topical report as a generically acceptable methodology. As an adjunct to the gap fraction analyses, the NRC staff had an analysis performed of the fuel rod pressure. This analysis was based on a power history derived from the $F_{\Delta H}$ and core average LHGR data docketed by Duke, increased by 5 percent, with a two-sigma uncertainty added to the nominal FGR. This analysis showed that the rod pressure would be 1105 psia, which is less than the 1200 psig specified in Safety Guide 25. Based on the NRC staff's analysis, Duke's use of an effective pool decontamination factor of 200 continues to be acceptable.

The NRC staff performed confirmatory analyses of the FHA and the WGD. The NRC staff used a MOX LTA source term generated using the SCALE SAS2H computer code. For the FHA, this source term was decayed for 72 hours and multiplied by the radial peaking factor of 1.65. For the WGD, the NRC staff used the inventory from four MOX LTAs and three equivalent LEU fuel assemblies, decayed for 19.5 days, and multiplied by the radial peaking factor of 1.65. The results of the NRC staff's analyses confirmed the results obtained by Duke. Details on the assumptions found acceptable to the NRC staff are presented in Table 3 of this SE. The doses estimated by the licensee for the postulated FHA and WGD (See Table 4) were a small fraction, as defined in RG 1.183, of the 10 CFR 50.67 dose criteria and are, therefore, acceptable.

3.2.7 Fresh MOX LTA Drop

This accident analysis is not currently part of the Catawba licensing basis. Duke performed this analysis to assess the radiological consequences of a drop of a fresh MOX LTA prior to it being placed in the SFP. Duke correctly stated that plutonium isotopes have a much higher specific activity than uranium isotopes and could present a more severe radiological hazard if inhaled. Although the configuration of the MOX pellets and LTA fuel rods provides protection against inhalation hazards, it is conceivable that some plutonium might become airborne if the MOX LTA is severely damaged. Duke's analysis of this event was performed to be applicable to both McGuire Nuclear Station and Catawba Nuclear Station using values chosen to bound the parameters at either station.

Duke described the analysis as involving assumptions and methodologies that were used in the calculations supporting the MOX Fuel Fabrication Facility (FFF) construction authorization request. The NRC staff reviewing the FFF reviewed those calculations and found them to be consistent with NRC staff guidance in NUREG/CR-6410 and, therefore, acceptable. The review for the case of dropped fuel within the FFF was documented in the MOX FFF draft SE report dated April 30, 2003. Note that the specific case addressed in the present licensing action, dropped fuel in a reactor fuel building environment, was not considered during the MOX FFF review. The NRC staff has not previously used the guidance of NUREG/CR-6410 for DBA analyses for power reactors.

The following analysis description is from NUREG/CR-6410, revised to reflect the parameter values for the present application. The release of the radioactive material is found from the expression below:

Q = MAR x DR x ARF x RF x LPF

Where:

Q is the quantity of material that enters the environment, in kilograms

MAR is the quantity of the material at risk and is the kilograms of the uranium and plutonium isotopes in the fuel assembly that is postulated to be dropped. Duke documented the isotopic breakdown in Table Q3(a)-2 in the November 3, 2003, RAI response.

DR is the damage ratio of the material actually impacted by the event. Duke assumed that 1 percent of all the fuel pellets in the dropped fuel assembly are damaged from the fall. Duke states that the value is applicable to drops from heights up to 30 feet.

ARF is the atmospheric release fraction which is the fraction of the impacted material that can actually become airborne. Duke calculated a value of 1.96E-4 from curve fits to experimental data observed in a study performed by Sandia National Laboratory (Reference 59).

RF is the respirable fraction of the released material. Duke assumed that all of the material in the release was respirable.

LPF was defined as the fraction of airborne material that breaches the containment barrier. For the current application, this parameter is used for the fraction of airborne material not removed by the filters. Duke assumes credit for only one filter bank in the flow path from the SFP to the atmosphere and the control room and credits a filter efficiency of 95 percent.

The dose consequence of the release is found by the following expression:

$D = Q \times \chi/Q \times BR \times DCF \times Sp.A$

Where:

D is the dose. Although the FFF implementation called for the committed effective dose equivalent (CEDE), Duke opted to report the results in terms of TEDE. This is effectively equivalent in that TEDE is the sum of the CEDE and the deep dose equivalent, the latter being negligible in an accident involving a fresh fuel assembly.

Q is the release quantity solved above.

χ/Q is the atmospheric dispersion coefficient. Duke used a bounding value of 9.0E-4 sec/m^3. The Catawba value would have been 5.5E-4 sec/m^3. A bounding value of 1.74E-3 sec/m^3 was used for the control room.

BR is the breathing rate taken as 3.47E-4 m^3/sec. This value is consistent with regulatory guidance.

DCF is the dose conversion factor, rem/µCi

Sp.A is the specific activity of the plutonium or uranium isotope, µCi/kg. For its confirmatory calculations, the staff used the following relationship to determine the specific activity:

$$Sp.A = \lambda N = \frac{0.693}{T_{1/2}} \cdot \frac{m}{A} \cdot N_a$$

Where N_a = 6.025E23, m is 1 gram, A = atomic weight.

As noted, the overall methodology was previously accepted by the NRC staff reviewing the FFF and has been determined to be appropriate for the present application. Only two input values need to be considered further. The first, DR, was taken as 0.01. The FFF draft SE report found that the DR for pellets exposed to overpressurization gas flows and pressurized rods that are breached are 0.01 and 0.001 respectively. The NRC staff also considered the analysis of a dropped fuel assembly in Section 3 of Sandia Report SAND87-7082 (Reference 59). This evaluation postulated a 30-foot drop of a typical Westinghouse 17 x 17 irradiated fuel assembly. The evaluation concluded that the drop would result in fracturing the bottom 1.3 inches of the fuel pellets. The radial expansion of the fuel pellets causes the fuel rod clad to fail. Less than 1 percent (0.01) of the pellets are affected. The NRC staff notes that the experimental data were obtained with irradiated fuel. The physical properties of the fuel pellets and cladding (e.g., brittleness) are less limiting for fresh fuel. Based on these considerations, Duke's assumed value of 0.01 is acceptable for the present application.

The value of ARF is the fraction of particles released from the damaged pellets. The value of ARF derived by Duke was 1.96E-4. In developing this value, Duke used the methodology of Section 3.3.4.8 of NUREG/CR-6410. The method is based on the observation that when a hard, cohesive brittle solid material is impacted and crushed by some force (usually another solid) the first solid absorbs some or all of the impacting kinetic energy and can form fine particles. However, the impact might not make all the released material airborne. Following the guidance of NUREG/CR-6410, Duke used the Argonne National Laboratory data correlation shown below to arrive at the ARF value of 1.96E-4:

$$ARF = 3.27E\text{-}11 \times E^{1.131}$$

Where, E is the energy density in Joules per cubic meter. The energy density is the product of the drop height, the pellet density, and the gravitational constant.

The correlation assumes that the pellet is struck by a solid of equal or greater cross-sectional area in a free fall. In the actual case, the pellets in the lower portions of the fuel rods are compressed by the deceleration of the fuel assembly as the fuel rods impact the lower nozzle as the assembly strikes the floor. A portion of the energy exerted on the fuel pellets is dissipated in pellet-clad interaction and in the expansion of the fuel pellet causing bulging and rupture of the lower fuel rod cladding. Duke stated that by not considering the fuel assembly its analysis is conservative. Duke based this conclusion on the fact that including the fuel assembly structural materials would reduce the energy density projected for the drop, and that the analysis ignored the reduction in the energy density that would result from dissipation of momentum forces by pellet-clad interactions and the deformation of fuel assembly components. Based upon these considerations, the NRC staff has concluded that Duke's value for ARF is adequately conservative and consistent with the deterministic nature of this analysis.

Details on the assumptions found acceptable to the NRC staff are presented in Table 3. The EAB and control room TEDE estimated by the licensee for the postulated fresh fuel assembly drop were less than 0.3 rem. This is a small fraction of the 10 CFR 50.67 dose criteria and is, therefore, acceptable.

3.2.8 Summary

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by Duke to assess the radiological impacts of operation with four MOX LTAs at either Catawba unit. With the exception of deviations that are identified and dispositioned above, the NRC staff finds that Duke used analysis methods and assumptions that are consistent with the conservative regulatory requirements and guidance identified in Section 3.1 above. The NRC staff compared the doses estimated by Duke to the applicable criteria identified in Section 3.1. Based on its review as documented above, the NRC staff finds that the licensee's conclusion that the EAB, LPZ, and control room doses from postulated design basis accidents will continue to meet the acceptance criteria identified in Section 3.1 is acceptable. Therefore, the NRC staff concludes that use of four MOX LTAs at either Catawba unit is acceptable with regard to the radiological consequences of postulated design basis accidents.

3.3 Spent Fuel Pool Cooling

10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel storage and handling and radioactivity control," requires the SFP to be designed with provisions for decay heat removal. Using SRP Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," and the further updated guidance developed during extended power uprate reviews (Reference 84), the NRC staff reviewed the effect on the SFP cooling capability of adding four MOX fuel assemblies to the SFP. As shown in Figure 3-12 of Attachment 3 to the licensee's letter, dated February 27, 2003, the MOX fuel has a decay heat about 2 percent higher than that of the regularly used LEU fuel seven days (168 hours) after shutdown of the reactor, which is about the time of peak SFP temperature. Since the four MOX LTAs are only a small fraction of the fuel transferred to the SFP during refueling, the change in decay heat represents a negligible change in the total decay heat of all fuel stored in the pool. The effect of the relatively higher decay heat of the MOX fuel on the SFP cooling system will diminish with time because the decay of the fuel will lower the decay heat. Therefore, the NRC staff determined that placing four MOX fuel assemblies in the SFP will have a negligible effect on SFP cooling capability.

TABLE 3
ANALYSIS ASSUMPTIONS

Source Term

Core power (includes 2% uncertainty penalty), MWt	3479
Specific power level, MW/assembly *(Includes 1.65 peaking)*	29.745
Fuel assemblies	193
Fuel pins per assembly	264
Fuel pellet temperature, °K	1085.34
Cycle burnup, MWD/MThm	
MOX	16,950
LEU	60,000
Heavy metal per assembly, MThm	0.4626
Plutonium concentration wt% Pu / hm in MOX assembly	5.0
Uranium enrichment, wr% U / U in LEU assembly	4.0
Fuel isotopics	
^{238}Pu / Pu wt%	0.025
^{239}Pu / Pu wt%	92.5
^{240}Pu / Pu wt%	6.925
^{241}Pu / Pu wt%	0.5
^{242}Pu / Pu wt%	0.05
^{234}U / U wt%	0.0017
^{235}U / U wt%	0.25
^{236}U / U wt%	0.0012
^{238}U / U wt%	99.7471
Fuel clad	
Density, gm/cc	6.50
Temperature, °K	656.1
Zirconium, wt%	98.873
Nobium, wt%	1.0
Oxygen-16, wt%	0.127
Moderator	
Density, gm/cc	0.711
Boron, ppm (cycle average)	900
Temperature, °K	580.43
Model	
Lattice	squarepitch
Fuel pin pitch, cm	1.2598
Outside diameter of fuel in pin, cm	0.8191
Clad outside diameter, cm	0.95

Clad inside diameter, cm	0.8357

Fuel Handling Accident and Weir Gate Drop

Radial peaking factor	1.65
Number of damaged fuel assemblies	
FHA	1
Weir gate drop	7
Decay time, days	
FHA	3
Weir gate drop	19.5
Fuel rod gap fractions	
I-131	0.12
Kr-85	0.15
All other noble gases, iodines	0.10
Alkali metals	0.0
Iodine species fractions	
Elemental	0.9985
Organic	0.0015
Particulates	none
Water depth, ft	23
Pool scrubbing factor, effective	200
Release modeling	
Immediate release from fuel through pool to building / CNMT	
100% release from building / CNMT within 2 hours	
No credit for building holdup or filtration prior to release	
Control Room Volume, ft^3	117,920
CRAVS start delay time, minutes	30
Unfiltered inleakage, cfm	
Before CRAVS start	2100
After CRAVS start	100
CRAVS filter flow, cfm	
Recirculation	1500
Outside air makeup	2000
Total	3500
CRAVS filter efficiency, %	
Elemental iodine	99
Organic iodine	95
Control room occupancy factors	
0-24 hr	1.0
24-96 hr	0.6
96-720 hr	0.4
Control room breathing rate, m^3/s	3.47E-4

Offsite breathing rate, m^3/s	
0-8 hrs	3.47E-4
Atmospheric dispersion factors, s/m^3	
EAB 0-2 hours	4.78E-4
LPZ 0-8 hours	6.85E-5
Control Room 0-2 hours	1.04E-3

Drop of a Fresh MOX LTA

Number of dropped assemblies	1
Fraction of pellets in assembly that are affected	0.01
MOX LTA loading, kg U +Pu	462.6
Composition, %	
Pu	5
U	95
Weight percent	
wt% Pu-238 / Pu	0.025
wt% Pu-239 / Pu	92.50
wt% Pu-240 / Pu	6.925
wt% Pu-241 / Pu	0.50
wt% Pu-242 / Pu	0.05
wt% U-234 / U	0.0017
wt% U-235 / U	0.25
wt% U-236 / U	0.0012
wt% U-238 / U	99.747
Height of drop, ft	23
Airborne Respirable Fraction	1.0
Fraction of damaged pellet that becomes airborne	1.96E-4
Filter efficiency, %	95
Control room occupancy factors	
0-24 hr	1.0
24-96 hr	0.6
96-720 hr	0.4
Control room breathing rate, m^3/s	3.47E-4
Offsite breathing rate, m^3/s	
0-8 hrs	3.47E-4
Atmospheric dispersion factors, s/m^3	
EAB 0-2 hours	9.00E-4
Control Room 0-2 hours	1.74E-3

TABLE 4
DESIGN BASIS ACCIDENT DOSES BY LICENSEE

	All LEU Core	LEU Core Plus 4-MOX LTAs	Acceptance Criteria
LOCA, *rem Thyroid*			
EAB	89	90.2	300
LPZ	25	25.3	300
Control Room	5.3	5.37	30
Locked Rotor Accident, *rem Thyroid*			
EAB	3.7	4.14	30
LPZ	1.2	1.35	30
Rod Ejection Accident, *rem Thyroid*			
EAB	1.0	1.03	75
LPZ	0.1	0.103	75
Weir Gate Drop Accident, *rem TEDE*			
EAB	2.2	3.5	6.3
LPZ	0.31	0.5	6.3
Control Room	2.1	3.3	5.0
Fuel Handling Accident, *rem TEDE*			
EAB	1.4	2.3	6.3
LPZ	0.21	0.34	6.3
Control Room	1.3	2.1	5.0
Fresh LTA Drop, *rem TEDE*			
EAB	n/a	<0.3	2.5
Control Room	n/a	<0.3	5.0

3.4 Reactor Vessel Materials

Section 3.6.1 of the Technical Justification in the licensee's application dated February 27, 2003, contains an evaluation of the impact of using four MOX LTAs on the integrity of the reactor vessels in Catawba, Units 1 and 2. In a letter dated February 2, 2004, the licensee provided additional information that evaluated the impact of four MOX LTAs on the reactor vessel surveillance program.

3.4.1 Regulatory Evaluation

The NRC staff has established requirements in 10 CFR Part 50, Appendices G, "Fracture Toughness Requirements" and H, "Reactor Vessel Material Surveillance Program Requirements," (10 CFR Part 50, Appendices G and H) and 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," (PTS rule) to protect the integrity of the reactor vessel in nuclear power plants. 10 CFR Part 50, Appendix G requires the pressure-temperature (P-T) limits for an operating plant to be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Appendix G to the ASME Code) were applied. The impact of radiation embrittlement on P-T limits is determined using the methodology in RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." 10 CFR Part 50, Appendix G also requires the Charpy upper-shelf energy (C_vUSE) to be greater than 50 foot-pounds (ft-lbs) throughout the operating life of the reactor vessel unless lower values can be justified. 10 CFR Part 50, Appendix H requires nuclear power plants to establish a reactor vessel surveillance program to monitor changes in the fracture toughness in the reactor vessel beltline materials. 10 CFR 50.61 provides the fracture toughness requirements for protecting the reactor vessels of pressurized water reactors against the consequences of pressurized thermal shock (PTS). 10 CFR 50.61 requires licensees to perform an assessment of the reactor vessel materials' projected values of the PTS reference temperature, (RT_{PTS}), through the end of their operating license.

3.4.2 Technical Evaluation

3.4.2.1 Pressurized Thermal Shock

The PTS rule requires each licensee to calculate the RT_{PTS} value for each material located within the beltline of the reactor pressure vessel at the expiration of its license. The RT_{PTS} value for each beltline material is the sum of the unirradiated nil ductility reference temperature (RT_{NDT}) value, a shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation of the material (i.e., ΔRT_{NDT} value), and an additional margin value to account for uncertainties (i.e., M value). 10 CFR 50.61 also provides screening criteria against which the calculated RT_{PTS} values are to be evaluated. For reactor vessel beltline base-metal materials (forging or plate materials) and longitudinal (axial) weld materials, the materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 270 °F; for reactor vessel beltline circumferential weld materials, the materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 300 °F. RG 1.99, Revision 2, provides an expanded discussion regarding the calculations of the shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation and the margin value to account for uncertainties. In this RG, the shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation is the product of a chemistry factor and a fluence factor. The fluence factor is dependent upon the neutron fluence. The chemistry factor is dependent upon the amount of copper and nickel in the material. Since the amount of copper and nickel in the material does not change with MOX fuel, the only factor to be evaluated in determining the impact of radiation on reactor pressure vessel embrittlement is the neutron fluence.

In Reference 65 the NRC staff evaluated the protection from PTS for the Catawba, Units 1 and 2 reactor vessels. For Catawba, Unit 1, the NRC staff calculated an RT_{PTS} value for the limiting beltline material at the end of the extended operating term (60 years of operation) of 62 °F. For Catawba, Unit 2, the NRC staff calculated an RT_{PTS} value for the limiting beltline material at the end of the extended operating term of 133 °F. The neutron fluence at the end of the extended license term for Catawba, Units 1 and 2 are 3.12×10^{19} neutrons/square centimeter (n/cm^2) and 3.16×10^{19} n/cm^2, respectively. In order for these materials to reach the PTS screening criteria, the neutron fluence would have to increase more than ten times the value at the end of the extended operating term.

The only factor in the RT_{PTS} calculation affected by the MOX fuel is the neutron fluence. In a letter dated February 2, 2004, the licensee explained in Attachment 3 why the use of four MOX LTAs has a negligible impact on neutron fluence as follows:

> The use of four MOX fuel lead assemblies will have no significant impact on the end-of-life fluence experienced by a Catawba reactor vessel. While the neutron energy spectrum from plutonium fissions is slightly higher than the spectrum from uranium fissions, the four MOX fuel lead assemblies represent only about 2 percent of the 193 fuel assemblies in the core.
>
> Duke plans to use the MOX fuel lead assemblies for three operating cycles. For the first two cycles, the MOX assemblies will be loaded in the interior of the core (e.g., core location C8). For the third cycle, one or more MOX fuel lead assemblies will most likely be loaded in a core location at or near the core periphery (e.g., core location C14). A representative core loading map for the first cycle is shown in Figure Q11-1 of [the licensee's letter dated October 3, 2003]. It should be noted that the actual MOX fuel assembly core locations have not been finalized and will be determined as part of the cycle specific reload design. As discussed below, the incremental impact of the four MOX fuel lead assemblies on reactor vessel fluence will be insignificant.
>
> In [the licensee's letter dated October 3, 2003], response to Question 11, Duke showed that using four MOX fuel assemblies during the first cycle of operation will have a negligible impact on the fast flux in the core. At the beginning of the first cycle, Figure Q11-2 of [the licensee's letter dated October 3, 2003] shows that the maximum calculated impact is a fast flux increase of 6.4 percent in the MOX fuel location itself (C8). Peripheral core locations are the most important with respect to the leakage of neutrons out of the core, and the maximum increase in fast flux in a peripheral fuel assembly is only 1.6 percent at the beginning of the first cycle. The small incremental impact of using MOX fuel on fast flux decreases further with burnup, because conventional LEU fuels assemblies produce more and more of their power from plutonium fissions as their burnup increases. Figure Q11-3 shows that at the end of the first cycle the impact of using four MOX fuel lead assemblies on the fast flux is less than 1 percent in all core locations.

Burnup effects will make the incremental impact of using MOX fuel during the second cycle even smaller than during the first cycle. In the third cycle, with MOX fuel loaded in an exterior core location, any MOX fuel-related increase in the fast flux would have more potential to affect the fluence at the vessel. However, the difference between a twice-burned MOX fuel assembly and a twice-burned LEU fuel assembly is very small. As noted in Reference [66], at a burnup of 50 gigawatt-day per ton, "...only 36 percent of LEU fuel fissions are in uranium, so most of the power is coming from plutonium fissions. At this burnup the characteristics of LEU fuel have become very similar to those of MOX fuel." Accordingly, during the third cycle of irradiation there will be little difference between the neutron energy from a MOX fuel assembly and the neutron energy from a twice-burned LEU fuel assembly that would otherwise be loaded at the expected location on the core periphery. Therefore, the impact of four MOX fuel lead assemblies on vessel fluence should be negligible during all three cycles of operation.

The NRC staff concludes that using the lead MOX fuel assemblies as described by the licensee will have a negligible impact on the neutron fluence and the RT_{PTS} value. Since the neutron fluence would have to increase by more than ten times the value at the end of the extended period to reach the PTS screening criteria, the staff concludes that the Catawba, Units 1 and 2 reactor vessels will have adequate fracture toughness for protection against PTS while using MOX LTAs and the PTS analysis in Reference 65 will not be affected by the use of the MOX fuel lead assemblies.

3.4.2.2 P-T Limits and C_vUSE

P-T limits increase and C_vUSE decreases as neutron fluence increases. P-T limits were reviewed by the NRC staff in Reference 65. The NRC staff's evaluation of the P-T limits concluded that the limits satisfy the requirements in Appendix G to Section XI of the ASME Code, and Appendix G of 10 CFR Part 50 and that the licensee used the methodology in RG 1.99, Revision 2, for determining the impact of neutron radiation on the beltline materials. In Reference 65, the NRC staff concluded that the Catawba, Units 1 and 2 reactor vessels will have C_vUSE greater than 50 ft-lbs throughout the period of extended operation. Since the increase in neutron fluence using the MOX LTAs is negligible, as discussed in the previous section, this will have no impact on reactor vessel embrittlement, P-T limits and C_vUSE.

3.4.2.3 Reactor Vessel Surveillance Program

In References 64 and 65, the NRC staff reviewed the surveillance capsule withdrawal schedule for Catawba, Units 1 and 2. The NRC staff concluded that the surveillance program was being implemented in accordance with Appendix H of 10 CFR Part 50 and that the capsule withdrawal schedule for Catawba, Units 1 and 2 was acceptable. There will be no surveillance capsules in the Catawba, Units 1 and 2 reactor vessels during the use of MOX LTAs. However, an ex-vessel cavity dosimetry program is being implemented at both Catawba units. This program will supplement the surveillance capsule program and monitor the reactor vessel fluence. Ex-vessel dosimetry was installed in Catawba, Unit 1 in 2003, and will be installed in Catawba, Unit 2 in 2004. The ex-vessel cavity dosimetry program will confirm that the predictions of vessel fluence used to assess vessel embrittlement are conservative.

Since MOX LTAs will have a negligible impact on the neutron fluence received by the reactor vessel, as discussed above in Section 3.4.2.1, no change in the reactor vessel surveillance program is necessary.

3.4.2.4 Summary

Based on the NRC staff's review and evaluation of MOX LTAs, the NRC staff has determined that for Catawba, Units 1 and 2, the reactor vessel RT_{PTS} values will be less than the screening criteria in 10 CFR 50.61, the reactor vessel surveillance program, P-T limits, and C_vUSE will not be affected by the use of MOX LTAs. On the basis of the above regulatory and technical evaluations of the licensee's justifications for TS changes, the NRC staff concludes that the licensee's proposed TS changes are acceptable.

3.5 Occupational Dose, Routine Effluents

3.5.1 Regulatory Evaluation

The focus of the NRC staff's evaluation in Section 3.5 of this SE is with respect to whether the proposed changes to the TS are consistent with requirements of 10 CFR Part 20 and the criteria of Appendix I to 10 CFR Part 50 in the areas of occupational and public dose. The regulatory requirements and guidance on which the NRC staff based its acceptance are as follows:

Regulations

- 10 CFR 20.1101, "Radiation protection programs."

- 10 CFR 20.1201, "Occupational dose limits for adults."

- 10 CFR 20.1301, "Dose limits for individual members of the public."

- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents - nuclear power reactors."

Guidance

- 10 CFR Part 50, Appendix I, "Numerical guides for design objectives and limiting conditions for operation to meet the criterion "As Low As is Reasonably Achievable" for radioactive material in light-water-cooled nuclear power reactor effluents."

3.5.2 Technical Evaluation

This evaluation is on Sections 5.6.1 and 5.6.2 of the licensee's application, "Plant Effluents" and "Impacts to Human Health" respectively.

In Section 5.6.1, "Plant Effluents," the licensee has evaluated the overall impact that the proposed use of MOX LTAs would have on its radioactive gaseous and liquid effluent releases. The licensee concluded that there will be no anticipated changes in the type or amount of radiological effluents resulting from the use of MOX LTAs from that of its current LEU fuel. The licensee states that it will continue to maintain its radioactive gaseous and liquid effluents within license conditions and regulatory limits.

The licensee's conclusion is based on its evaluation of the similarity of MOX fuel to the current LEU fuel, both from a fuel design and fission product inventory perspective, and on the limit of having only four out of 193 fuel assemblies containing MOX fuel.

The licensee evaluated the types and amount of fission products available for release in effluents. As fuel is irradiated, both activation and fission products are created. The activation products are created in the reactor coolant and fission products are produced inside the fuel rods. Activation products that are created are a function of impurities and the chemistry of the reactor coolant and the thermal neutron flux that the materials encounter. Thermal flux is significantly lower in MOX fuel than in LEU fuel, which would tend to reduce the level of activation products. However, for four lead assemblies this is expected to be an insignificant effect.

Fission product inventories and fuel gap inventories in particular are of the same order or magnitude in both MOX fuel and LEU fuels. In particular, the amount of iodine and noble gas that would be released into the reactor coolant in the event of a leaking fuel rod would be similar. Additionally, any liquid or gaseous effluents would be processed by the plant liquid waste and waste gas systems prior to release into the environment. These waste treatment systems would limit radioactive discharges to the environment through the use of hold-for-decay, filtering, and demineralization. The licensee states that the plant treatment systems are capable of treating these radioactive effluents since the types of radioactive material in MOX and LEU fuel are the same and the curie content of MOX fuel is of the same order of magnitude as LEU fuel. Thus, the licensee is expected to maintain the same level of radioactive control and remain within regulatory limits with the MOX fuel as has been maintained with the LEU fuel.

In Section 5.6.2 of its application, "Impacts to Human Health," the licensee has evaluated the overall impact that the proposed modification would have on its workers (occupational exposure) and to members of the public.

For occupationally exposed workers, the licensee estimates that there will be slight increases in radiation exposure during the handling of MOX fuel during receipt and handling operations. The increase in dose is due to a higher dose rate from a fresh MOX LTA as compared to a fresh LEU fuel assembly. The total neutron and gamma dose rate at 10 centimeters from the face of a fresh MOX LTA averages about 6 mrem/hour, falling off to about 1.8 mrem/hour at 100 centimeters. This is a relatively low radiation field; however, it is larger than that associated with a LEU fuel assembly, which has virtually no radiation field at these distances. The initial receipt and handling activities for one MOX LTA could result in a conservatively estimated total occupational dose in the range of 0.020 to 0.042 person-rem. However, the licensee will use the application of the as low as reasonably achievable principle to try to effect lower doses than are estimated. Radiation doses of this magnitude are well within regulatory occupational exposure limits and do not represent an impact to worker health.

For members of the public, as discussed in Section 5.6.1 above, the licensee estimates that there will be no detectable increase in public dose during normal operations with the MOX LTAs. Use of the MOX LTAs in the reactor core will not change the characteristics of plant effluents or water use. During normal plant operation, the type of fuel material will have no effect on the chemistry parameters or radioactivity in the plant water systems. The fuel material is sealed inside fuel rods that are seal-welded and leaktight. Therefore, there would be no direct impact on plant radioactive effluents and the associated radiation exposure to members of the public.

3.5.3 Summary

Based on the NRC staff's review of the information provided in the licensee's application, the NRC staff concludes that there is reasonable assurance that the licensee will conduct its radiation protection and radioactive effluent release programs in a manner that maintains radiation exposures to plant workers and members of the public within the regulatory limits of 10 CFR 20.1301 and 10 CFR 20.1201.

3.6 Quality Assurance

3.6.1 Introduction

The licensee's application of February 27, 2003, included, in part, a description of the QA activities associated with the fabrication of the MOX LTAs by Framatome ANP, the supplier of the MOX LTAs to the licensee. This section of the SE addresses the programmatic aspects of the Framatome ANP QA program associated with the fabrication of the MOX LTA fuel pellets and fuel assemblies.

Section 3.5.4, "Quality Assurance," of the licensee's February 27, 2003, submittal contained a description of the QA process related to the fabrication and assembly of the MOX fuel pellets and fuel assemblies. As stated in the amendment request, Framatome ANP has the responsibility for the overall QA oversight of the entire fuel assembly fabrication process. As part of this effort, Framatome ANP will qualify every sub-vendor who operates under the technical requirements of the program and will verify that each sub-vendor and the sub-vendor's associated facilities meet the requirements in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants". The applicant further stated that the qualification of these vendors and facilities shall include a combination of system audits conducted by Framatome ANP, review of audits performed by other Framatome ANP facilities, and surveillance audits by other approved Framatome ANP quality auditors.

3.6.2 Regulatory Basis

The NRC staff review of Section 3.5.4, "Quality Assurance," of the submittal was conducted in accordance with the review requirements described in Chapter 17, "Quality Assurance," of the SRP (Reference 68) to assure that the requirements of 10 CFR Part 50, Appendix B, were adequately implemented. The NRC staff used additional guidance provided in RG 1.28, Revision 3, 1985, "Quality Assurance Program Requirements (Design and Construction)," (Reference 69), ANSI/ASME Standard N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities," (Reference 71), and ANSI/ASME Standard NQA–1 1983, "Quality Assurance Requirements for Nuclear Facilities," (Reference 70) respectively, in its review.

The NRC staff customarily reviews and evaluates an applicant's description of its QA program for the design and construction phases in each application for a construction permit, a manufacturing license, or a standardized design certification in accordance with applicable portions of SRP 17.1. The acceptance criteria in this section are based on the relevant requirements of 10 CFR Part 50, Appendix B; 10 CFR Part 50, Appendix A; 10 CFR Part 50.55a; 10 CFR Part 50.55(e); and 10 CFR Part 50.34(a)(7) with emphasis on activities associated with the design and construction phases. The acceptance criteria deal with the QA controls related to the 18 areas outlined in 10 CFR Part 50, Appendix B, and review guidance embodied in the regulatory guidance referenced by SRP 17.1. Appendix B of 10 CFR Part 50 identifies all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. The 18 elements described in Appendix B specifically describe those planned and systematic actions.

3.6.3 Technical Evaluation

Since the fabrication of the LTAs is but one of the activities of a consortium effort that also includes development of a MOX FFF, it is considered useful to provide background information on the QA program for the MOX FFF. The review of the QA Program for the construction of the MOX FFF has been performed and documented in an NRC Evaluation Report dated October 1, 2001, (Reference 74). The scope of this SE is limited to the QA aspects associated with the fabrication of the MOX LTA's for use in the Catawba, Units 1 and 2 as described in the subject amendment application. Additionally, prior evaluations and approval of the Duke QA program and the Framatome Topical Report, "Framatome Quality Assurance Program (for United States Applications)," have been completed by the NRC staff and have been documented in letters from NRC to the applicants (References 75 and 76).

The NRC staff requested additional information from the licensee in letters dated August 13, 2003, and December 24, 2003, to support the current review. The focus of those requests was on the QA aspects of the MOX manufacturing process. Specifically, the NRC staff requested the following information pertaining to the scope of the Framatome ANP QA program: (1) a description of the Framatome ANP QA plans governing the fabrication activities affecting quality, (2) identification of individual sub-suppliers of materials to Framatome ANP and information pertaining to their QA programs and qualifications, and (3) information related to the various verification activities of Framatome ANP to ensure adequate implementation of the QA program for all fuel fabrication activities affecting quality. A discussion of these three areas follows:

(1) Description of Framatome ANP QA Plans

The licensee responded to the request for a detailed QA program description for the fabrication of the MOX LTAs, in its letter dated October 1, 2003. As part of its response, the licensee included a copy of the Framatome ANP manual, "Fuel Sector Quality Management Manual," (FQM Revision 1, US Version - Applicable July 2003), that defines the quality program that applies to the fabrication of components within Framatome ANP and items purchased from suppliers. The licensee provided supplemental information regarding the specific QA plan for the assembly and certification of the fuel rods and assemblies by letter dated February 2, 2004.

The FQM contains a detailed description of each of the Framatome ANP QA program attributes including criteria and requirements established to ensure compliance with the 18 criteria of a QA program described in 10 CFR Part 50, Appendix B. The NRC staff finds that the document is of sufficient detail to adequately identify specific actions, roles, and responsibilities within the Framatome ANP organization to assure that the scope and breadth of activities affecting quality are adequate. Additionally, the manual contains an evaluation of the Framatome ANP QA program attributes with respect to the NRC's SRP Section 17.1 and pertinent regulatory and industry QA program guidance and the extent of applicability of those guidelines to the Framatome ANP program.

The NRC staff has reviewed the FQM QA program, as described in the Framatome ANP FQM, to ensure that the fuel supplier, Framatome ANP, has described a QA program consistent with requirements of 10 CFR Part 50, Appendix B. The NRC staff verified that the Framatome ANP FQM adequately describes the QA criteria consistent with the requirements of 10 CFR Part 50, Appendix B, and the regulatory guidance applicable to such programs. Furthermore the NRC staff verified that the FQM quality program applies to the fabrication of components within Framatome ANP and items purchased from their suppliers. In summary, the NRC staff has reviewed and verified that the Framatome ANP FQM QA program, adequately describes a series of actions necessary and sufficient to comply with 18 criteria for QA programs in accordance with regulatory requirements.

(2) Use of Qualified Suppliers

With respect to the second review topic regarding use of qualified suppliers, the submittals state that for the MOX LTAs, all hardware and materials will be purchased from suppliers that have been qualified and approved by Framatome ANP to meet the stringent requirements of the Framatome ANP quality program. The qualification process will be conducted by Framatome ANP in accordance with the Framatome ANP FQM through a series of activities required to qualify the facilities as approved suppliers. These activities will include: (1) review of the Caderache and Melox facilities' internal QA programs against the requirements of the Framatome ANP FQM; (2) successful resolution of any findings associated with those reviews; and (3) placement of the Cadarache and Melox facilities on the Framatome ANP-Fuels America approved suppliers list. Following this approval process, Framatome ANP will conduct implementation audits at each facility. These audits will verify that specific work instructions and procedures have been completed and put into effect to ensure compliance with the fuel supplier's fabrication contracts. These audits will also verify the product production will meet all design and quality requirements.

The NRC staff has reviewed the description of the planned activities by Framatome ANP to approve the Cadarache and Melox facilities as Framatome ANP approved suppliers, and the description of the implementation audits to be performed at each facility prior to fabrication activities. The NRC staff finds that these activities, if implemented as described, would provide an adequate process for qualifying these facilities consistent with the Framatome ANP FQM and applicable regulatory requirements and is, therefore, acceptable.

(3) Implementation of QA Processes

With respect to the third review area regarding verification of adequate implementation of the QA processes during the fabrication phase of the program, the licensee provided additional information describing these activities in its letter dated February 2, 2004. Specifically, the licensee described a multi-phased approach consisting of: (1) direct audit of the fabrication facilities at Cadarache and MELOX to verify that programmatic controls are in place; (2) on-site surveillance of the fabrication processes at both facilities by Framatome ANP-approved quality auditors, and (3) review of documentation generated in support of the fabrication and qualification processes including, but not limited to, process qualification reports, manufacturing and inspection process procedures, visual standards, certification and archive files, and non-conformance reports. This final review phase will incorporate both Framatome ANP and

Duke quality personal and will verify that the applicable documentation is available to support the fuel certification to be issued by Framatome ANP. The review will also include visual and dimensional inspections and surveillance of the completed fuel assemblies by qualified quality inspectors. As such, the fuel pellet and fuel rod characteristics will be documented as part of the on-site fabrication audits and final visual and dimensional inspections and surveillance of the completed fuel assemblies.

The NRC staff has reviewed the description of the planned activities including surveillance, inspection, and documentation review by Framatome ANP to verify adequate implementation of the QA processes during the fabrication phase of the program. The NRC staff has confirmed that the activities, if implemented as described, would provide adequate QA processes consistent with the regulatory requirements to support the fabrication of the MOX LTAs and is, therefore, acceptable.

As part of the NRC staff's continued oversight of the MOX LTA fabrication process, additional activities are envisioned. Specifically, the NRC staff plans on performing on-site verification activities associated with the implementation of the QA program of the fabrication facilities at Cadarache and MELOX during the fuel fabrication and assembly phases of the project. The purpose of these additional inspection activities is to confirm adequate implementation of the Framatome ANP QA processes as described and approved in this evaluation during the manufacturing phase of the MOX LTA program.

10 CFR Part 21, Reporting of Defects and Noncompliance

The Commission's regulations in 10 CFR Part 21, "Reporting of Defects and Noncompliance," requires suppliers of components to NRC licensed facilities within the United States to notify the Commission, immediately following discovery, of information reasonably indicating that the facility, activity, or basic component supplied to such facility or activity: 1) fails to comply with the regulatory requirements; or 2) contains defects, which could create a substantial safety hazard.

The NRC staff has verified that the identification and disposition of defects and noncompliance is included in the Framatome ANP FQM Section 5.3, "Control of Nonconforming Product." This section specifies that the requirements of 10 CFR Part 21 are fulfilled for products delivered to U. S. customers. These requirements include, but are not limited to, control of products which do not conform to preclude their inadvertent use, provisions for the identification, segregation, disposition, and notification to affected organizations, including establishment of a reporting process to ensure communication without delay of potential nonconformance. Additionally, the applicant has stated that the requirements of 10 CFR Part 21 are part of the Framatome ANP standard ordering requirements and shall be imposed on each supplier. As such, the supplier is required to notify Framatome ANP of any conditions that may be subject to a 10 CFR Part 21 review.

The NRC staff has verified that the Framatome ANP FQM describes a process for the control of nonconformance consistent with the requirements of 10 CFR Part 21 and that this process is to be implemented by Framatome ANP with respect to control of nonconformance associated with the MOX LTA fabrication. The NRC staff has confirmed that these activities, if implemented as described, would provide adequate control and reporting of nonconformance consistent with the regulatory requirements in support of the fabrication of the MOX LTAs and are, therefore, acceptable.

3.6.4 QA Summary

The NRC staff evaluated the scope of the QA activities involving the fabrication of the MOX LTA fuel pellets and fuel assemblies as described by the licensee, including the administrative controls governing those activities. The NRC staff finds that the proposed QA processes and activities described by the licensee in its amendment application as supplemented through letters dated October 1, 2003, and February 2, 2004, are consistent with the requirements of 10 CFR Part 50, Appendix B and the pertinent regulatory guidance described above and are, therefore, acceptable.

3.5 Security Plan

A non-safeguards information version of a safety evaluation will be provided in a supplement to this Safety Evaluation. The NRC staff's detailed conclusions will be provided in a document that, since it will contain safeguards information, will not be released to the public.

4.0 CONCLUSION

At the time of issuance of this SE, certain matters that are required to be completed to permit the issuance of any amendment to the operating licenses authorizing the use of MOX LTAs have not been completed. These include the completion of the NRC staff's review of the security plan as discussed in section 3.5 above, consultation with the State of South Carolina, and completion of the environmental consideration. The Commission has concluded, based on the considerations discussed in sections 1.0, 2.0 and 3.0 of this SE and subject to the completion of the matters discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter, M. S. Tuckman, Duke, to NRC, "Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50", dated February 27, 2003.

2. Letter, R. E. Martin, NRC, to Duke, requesting additional information on McGuire spent fuel pool storage racks, dated July 14, 2003.

3. Letter, R. E. Martin, NRC, to Duke, requesting additional information on reactor systems, radiological consequences, and environmental impacts, dated July 25, 2003.

4. Letter, R. E. Martin, NRC, to Duke, requesting additional information on quality assurance, dated August 13, 2003.

5. Letter, M. S. Tuckman, Duke, to NRC, "Revision 16 to Duke Energy Corporation Physical Security Plan and Request for Exemption from Certain Regulatory Requirements in 10 CFR 11 and 73 to Support MOX Fuel Use," dated September 15, 2003.

6. Letter, M. S. Tuckman, Duke, to NRC, amending the February 27, 2003 application to apply only to Catawba Nuclear Station, dated September 23, 2003.

7. Letter, M. S. Tuckman, Duke, to NRC, Responding the NRC staff's RAI of July 14, 2003, on Boraflex in McGuire, dated October 1, 2003.

8. Letter, M. S. Tuckman, Duke, to NRC, Responding to the NRC staff's RAI of August 13, 2003 and providing the Framatome ANP Fuel Sector Quality Management Manual, dated October 1, 2003.

9. Letter, M. S. Tuckman, Duke, to NRC, responding to 23 of the NRC staff's RAIs of July 25, 2003, dated October 3, 2003.

10. Letter, J. F. Mallay, Framatome ANP, to NRC, responding to RAI 13 on LOCA analysis from the July 25, 2003 RAI, dated October 3, 2003.

11. Letter, R. E. Martin, NRC, to Duke on the withdrawal of McGuire from the LTA program and review of security plan only for the LTA program, dated October 31, 2003.

12. Letter, M. S. Tuckman, Duke, to NRC, Responding to all of the NRC staff's RAIs of July 25, 2003, (Proprietary), dated November 3, 2003.

13. Letter, M. S. Tuckman, Duke, to NRC, Responding to all of the NRC staff's RAIs of July 25, 2003, (Non-Proprietary), dated November 4, 2003.

14. Letter, R. E. Martin, NRC, to Duke, requesting additional information on environmental and radiological consequences review, dated November 21, 2003.

15. Letter, R. E. Martin, NRC, to Duke, transmitting Notice of Withdrawal of McGuire from the LTA program, dated December 2, 2003.

16. Letter, K. S. Canady, Duke, to NRC, responding to the NRC staff's RAI of November 21, 2003 on radiological consequences, dated December 10, 2003.

17. Letter, R. E. Martin, NRC, to Duke, requesting additional information on materials, and radiological consequences, dated December 16, 2003.

18. Letter, R. E. Martin, NRC, to Duke, requesting additional information on quality assurance, dated December 24, 2003.

19. Letter, R. E. Martin, NRC, to Duke, requesting additional information on the physical security plan, dated January 30, 2004.

20. Letter, W. R. McCollum, Duke, to NRC, responding to NRC staff's RAIs dated November 21, 2003 on environmental impacts, December 16, 2003, on radiological consequences and materials, and December 24, 2003 on quality assurance, dated February 2, 2004.

21. Letter,W. R. McCollum, Duke, to NRC, "Additional Information Regarding Mixed oxide Fuel Lead Assemblies MOX Fuel Rod Free Volume," dated February 2, 2004.

22. Letter, R. E. Martin, NRC, to Duke, requesting additional information on radiological consequences, dated February 4, 2004.

23. Letter, H. B. Barron, Duke, responding to NRC staff's RAI dated February 4, 2004 on radiological consequences, dated March 1, 2004.

24. Letter, H. B. Barron, Duke, to NRC, responding to two NRC staff RAIs dated January 30, 2004, on the security plan, dated March 1, 2004.

25. USNRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 3, 1988.

26. Letter, J. F. Mallay, Framatome ANP, to NRC, "Request for Review of BAW-10239(P), Revision 0, Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report," dated April 30, 2002.

27. Letter, S. Dembek, NRC, to J. F. Mallay, Framatome ANP, "Draft Safety Evaluation For Framatome ANP Topical Report BAW-10239(P), Revision 0, Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report," dated March 30, 2004.

28. BAW-10227-P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000.

29. Letter, J. F. Mallay, Framatome ANP, to NRC, "Request for Approval of BAW-10238(P), Revision 1, MOX Fuel Design Report," dated May 30, 2003.

30. DPC-NE-1005, Revision 0, Duke Power Company Nuclear Design Methodology Using CASMO-4/SIMULATE-3MOX, August 2001.

31. Letter, R. E. Martin, NRC, to Duke, "Draft Safety Evaluation for Duke Topical Report DPC-NE-1005P, Nuclear Design Methodology Using CASMO-4/Simulate-3 MOX", dated February 20, 2004.

32. BAW-10239-P-A, "COPERNIC Fuel Rod Design Computer Code," September 1999.

33. Letter, H. N. Berkow, NRC, to J. F. Mallay, Framatome ANP, "Final Safety Evaluation for Topical Report BAW-10231P, COPERNIC Fuel Rod Design Code, Chapter 13, MOX Applications," dated January 14, 2004.

34. Letter, S. Dembek, NRC, to J. F. Mallay, Framatome ANP, "Draft Safety Evaluation for Framatome ANP Topical Report BAW-10238(P), Revision 1, MOX Fuel Design Report," dated March 31, 2004.

35. WCAP-12945P-A, "Code Qualification Document for Best-Estimate Loss of Coolant Analysis", March 1998.

36. WCAP-100564P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code", August 1985.

37. BAW-10168P-A, Revision 3, "RSG LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants", December 1996.

38. Duke Power Company Thermal-Hydraulic Transient Analysis Methodology, DPCNE-3000-PA, Revision 2, December 2000.

39. Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," December 1981.

40. L. Kopp, Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants, NRC Memorandum from L. Kopp to T. Collins, August 19, 1998.

41. SCALE 4.4, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (Rev. 5), CCC-545, Oak Ridge National Laboratory, March 1997.

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43. O. Hermann, Benchmark of SCALE (SAS2H) Isotopic Predictions of Depletion Analyses for San Onofre PWR MOX Fuel, ORNL/TM-1999/326, February 2000.

44. D. Hollenbach and P. Fox, Neutronics Benchmarks of Mixed-Oxide Fuels Using the SCALE/CENTRM Sequence, ORNL/TM-1999/299, February 2000.

45. Water-Reflected Mixed Plutonium-Uranium Oxide (20 wt. % Pu) Pins, EA/NSC/DCO/(95)03/VI, MIX-COMP-THERM-001, August 1996.

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78. Letter, H. B. Barron, Duke, to NRC, transmitting additional information on security, dated March 9, 2004.

79. Letter, H. B. Barron, Duke, to NRC, transmitting responses to spent fuel pool criticality questions, dated March 9, 2004.

80. Letter, W. R. Mc Collum, Duke, to NRC, transmitting additional information on radiological dose consequences, dated March 16, 2004.

81. Letter, W. R. Mc Collum, Duke, to NRC, transmitting additional information on fuel assembly bow, dated March 16, 2004.

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Dated: April 5, 2004

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April 10, 2000

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-001

ATTENTION: Document Control Desk

Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
McGuire Nuclear Station, Units 1 and 2
Docket Number 50-369 and 50-370
Implementation of Best-Estimate Large Break LOCA Methodology

Reference: 1) WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," March 1998.

Duke Power Corporation plans to implement the Westinghouse best-estimate large break LOCA methodology for McGuire and Catawba Nuclear Stations (Reference 1). This change in LOCA methodology will gain margin in the calculated large break LOCA peak cladding temperature, thus allowing more operational and core design margin. The first cycle to utilize Westinghouse fuel is Catawba Unit 2 Cycle 11. This cycle is scheduled to begin operation in early April 2000. Two more reload cycles, McGuire Unit 2 Cycle 14 and Catawba Unit 1 Cycle 13, are scheduled to begin operation in 2000. The Appendix K large break LOCA analyses for these cycles resulted in limited LOCA margins. Therefore, an expedited NRC review of the application of the best-estimate large break LOCA method is desirable. To help facilitate the review process three separate submittals are planned that will allow for the review process to get started while the plant specific analysis is being completed. This three step submittal process was discussed by phone with the NRC staff on March 9, 2000.

This is the first submittal, which provides the process of how the best-estimate LOCA model will be applied. The details of the modeling approach to be used for McGuire/Catawba are provided in the attachment to this letter. The second submittal, scheduled for early May 2000, will contain the Technical Specification changes required to implement the Westinghouse best-estimate LOCA method. The third submittal, scheduled for July 2000, will provide a summary of the McGuire/Catawba plant specific analysis.

Given the reload schedule for McGuire Unit 2 Cycle 14 and Catawba Unit 1 Cycle 13, approval of the implementation of the Westinghouse best-estimate large break LOCA method is requested by September 1, 2000.

Please address any comments or questions regarding this matter to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachment

xc w/Attachment:

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Attachment

McGuire/Catawba Nuclear Stations
Best-Estimate Large Break LOCA Model Development

The best estimate large break LOCA (BE LBLOCA) analysis for the McGuire and Catawba Units, incorporating a 1% higher power level, will be performed using a bounding unit approach, similar to what has been done for Farley and Diablo Canyon. In this approach, a WCOBRA/TRAC model is developed choosing bounding inputs for the plant configuration. Where the bounding direction is not known, sensitivity studies are performed to determine the limiting direction.

The differences between the four units will be divided into vessel and loop, as detailed below.

Vessel

Two vessel models will be built to capture the differences in the upper internals.

- McGuire Unit 1, with (14) 15x15 guide tubes
- Other three units, with (6) 15x15 guide tubes

Other minor differences will be bounded in the two vessel models as follows:

1. Barrel/baffle: All units are upflow, but the baffle plates and bypass flow fraction are different between them. A conservative composite approach will be used to model this area, including use of:

   - Thickest of the baffle plates (increases boiling rate, which decreases core flooding rate)
   - Maximum barrel/baffle volume (corresponding to thinnest plates, which decreases water available for core reflood)
   - Higher bypass flow (decreases water available for core reflood)

2. Cold leg nozzle loss coefficient (forward flow): Maximum value among the four units will be used which reduces the safety injection flow rate (higher injection pressure).

3. Balance of vessel: A bounding approach is used for other minor differences, similar to discussion of barrel/baffle region above. This approach will:

   - Maximize vessel volume where liquid is not available for core cooling, such as the lower plenum

   - Minimize vessel volume where liquid is available for core cooling, such as the upper head

   - Maximize metal mass

Based on these two vessel models, a limiting vessel will be chosen based on analysis results and determination of the phenomenological differences, which led to those results. Studies will then continue with the determination of limiting loop configuration.

<u>Loops</u>

The major differences between the loops for the four units are the accumulators and steam generators. For the limiting vessel model determined above, the following studies will be performed to determine the limiting configuration.

1. Accumulator line friction (L/D): The highest and lowest values will be analyzed to determine the limiting direction.

2. Accumulator pressure: Base transient will use a nominal pressure. The range of pressures to encompass all units will be included in the initial condition uncertainty calculations, so it will not be considered here.

3. Accumulator water volume: Base transient will use the minimum nominal water volume. The high nominal value will be analyzed to determine the limiting direction.

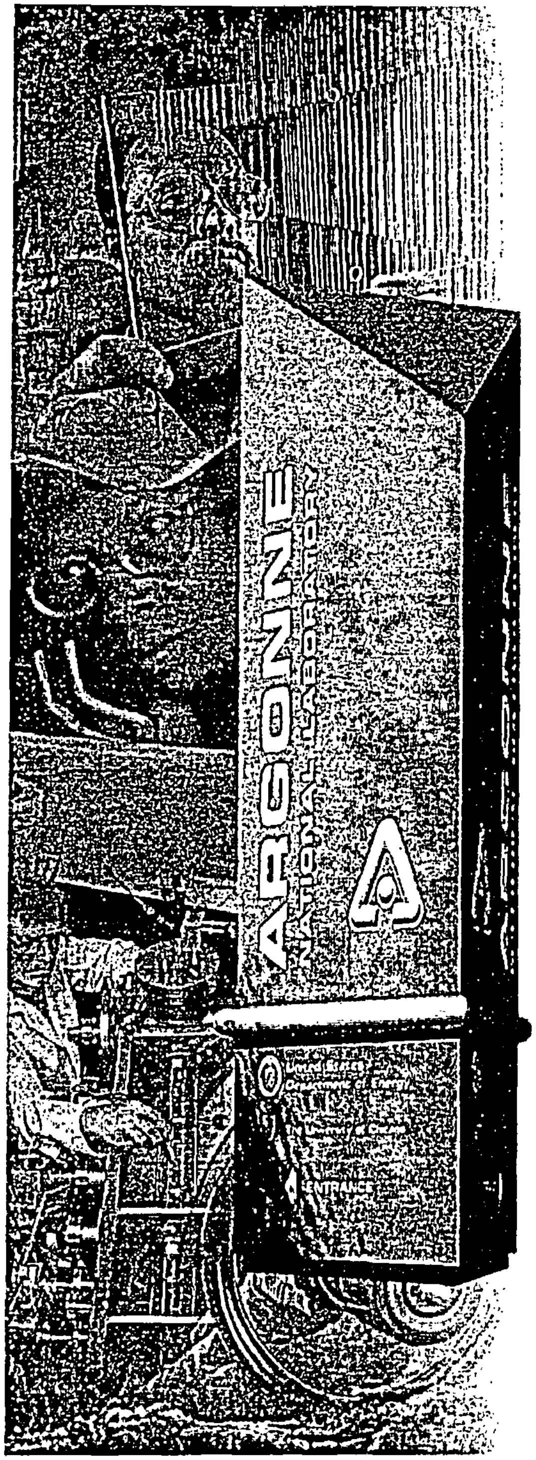
4. Steam Generators: The two types of steam generators (W and BWC) will be analyzed with the limiting vessel to determine the bounding type.

Limiting Composite Plant

At the completion of the loop sensitivity studies, a limiting composite plant configuration, which includes the limiting vessel model along with the limiting loop configuration, will be determined. The choice of limiting configuration will again be based on results, combined with an understanding of the phenomena that led to the results. This composite model will be used to perform a final composite initial transient. This model will form the basis for the remainder of the BE LBLOCA analysis. Other minor differences in plant initial conditions will be addressed in the initial conditions run matrix by ranging the parameters to bound all four units.

Transition Core Effects

The transition from Framatome to Westinghouse fuel will be addressed with a separate evaluation, similar to that performed for Point Beach. Two additional calculations will be performed to determine the effects of the transition core. One calculation will use a fresh Westinghouse assembly surrounded by once-burned (or more) Framatome assemblies. The second calculation will use a once burned Framatome assembly surrounded by Westinghouse assemblies.



LOCA Test Results for High-Burnup BWR Fuel and Cladding

Y. Yan, M. Billone, T. Burtseva
Energy Technology Division
Irradiation Performance Section

Argonne National Laboratory

Office of Science
U.S. Department of Energy

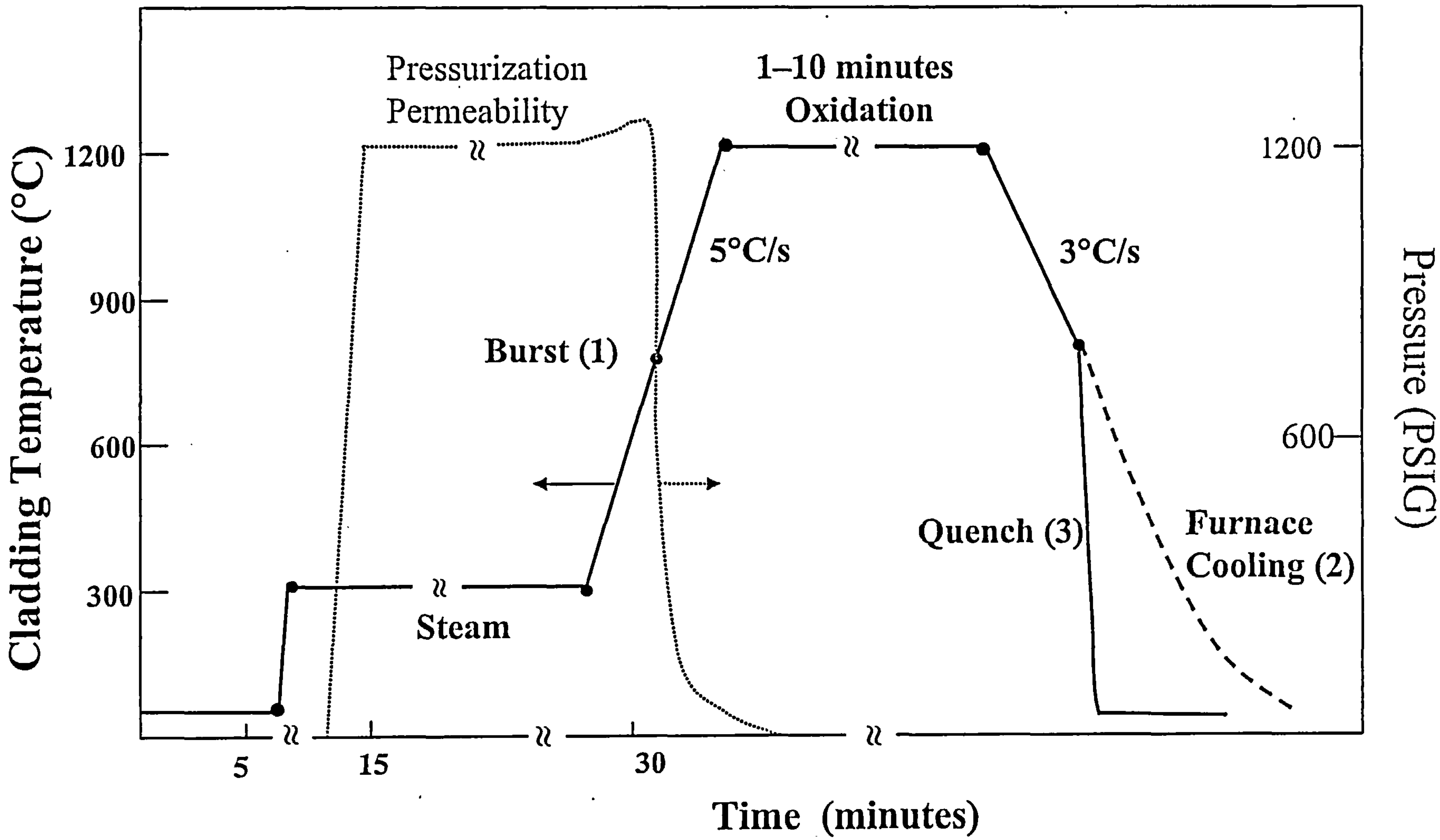
A U.S. Department of Energy
Office of Science Laboratory
Operated by The University of Chicago

NRC Staff's Proposed Exhibit 3

Background

- **Objective**
  - To evaluate the influence of burnup extension on fuel behavior under LOCA conditions
- **Licensing Issues Addressed**
  - 10 CFR 50.46 embrittlement criteria for maintaining residual ductility in Zircaloy (Zry) cladding:
    - ➢ Temperature limit: peak cladding temperature (PCT) ≤ 1204°C;
    - ➢ oxidation limit: effective cladding reacted (ECR) ≤ 17%
  - Confirming embrittlement criteria for high-burnup Zry-2 and Zry-4
- **High-Burnup Fuel Rod Segments**
  - H.B. Robinson 15×15 PWR rods at 67 GWd/MTU
    Corrosion layer ≤ 110 μm; H-content ≤ 800 wppm
  - Limerick 9×9 BWR rods at 56 GWd/MTU
    Corrosion layer ≈ 10 μm; H-content ≈ 70 wppm

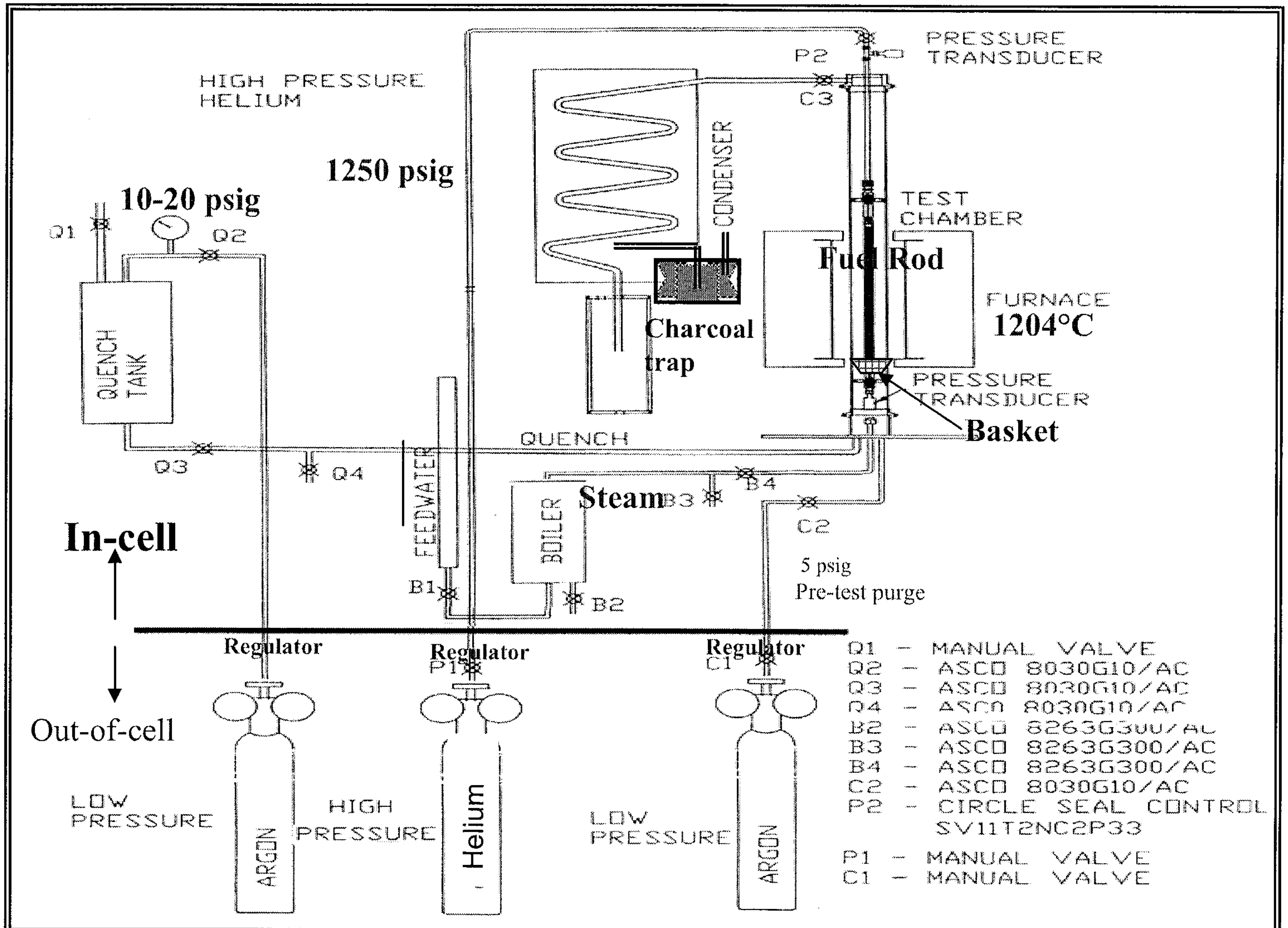
LOCA Integral Test Sequence



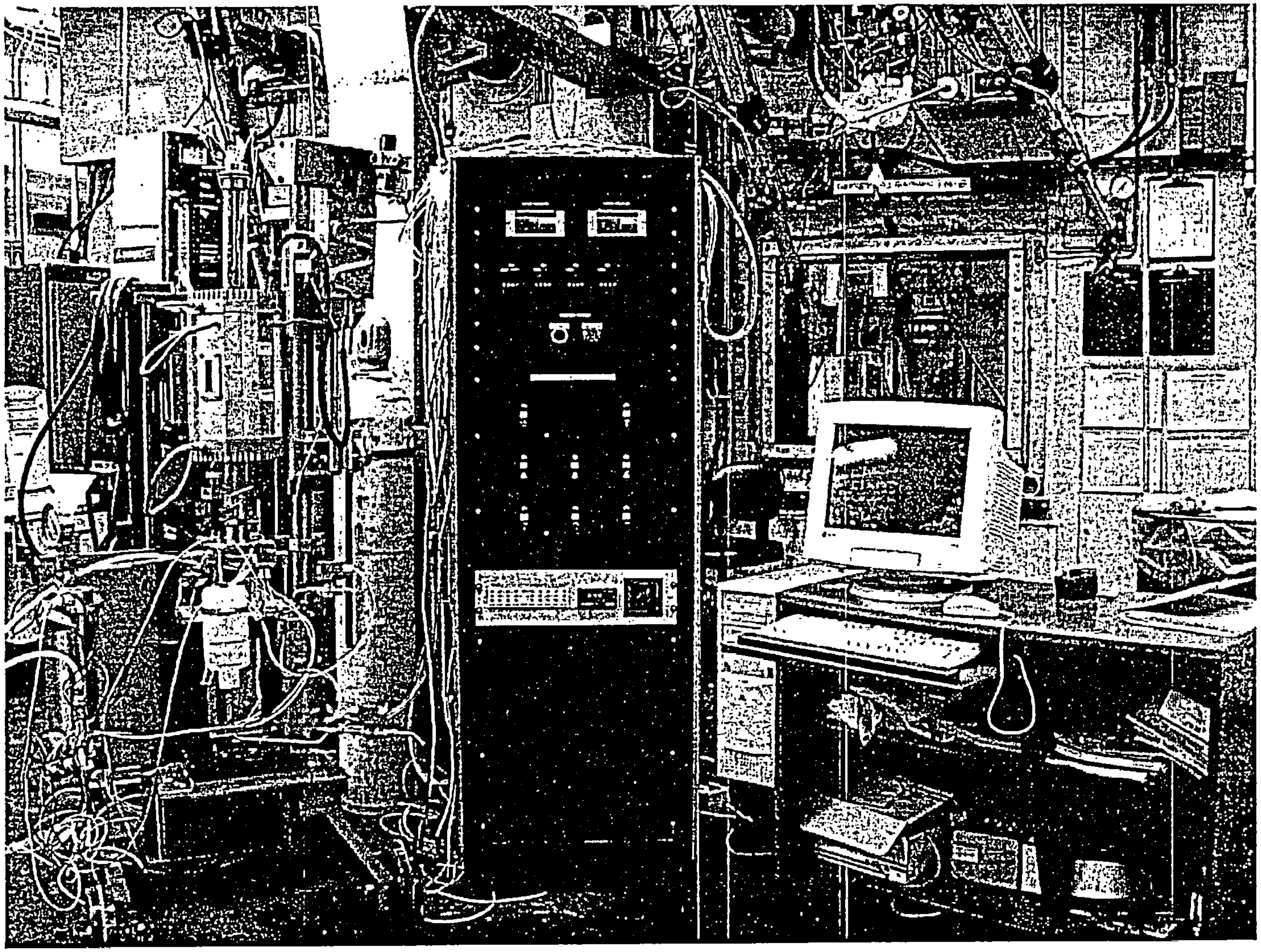
High-Burnup Phenomena Investigated

- **Fuel behavior and effects of fuel / cladding bond**
- **Effects of corrosion, hydriding, and irradiation hardening on cladding behavior:**
  - Ballooning and burst (profilometry, photography)
  - Secondary hydriding (LECO H determination)
  - Steam oxidation (Weight gain, LECO O determination, metallography)
  - Quench behavior and post-quench ductility
    - ✓ *Microhardness tests*
    - ✓ *Ring compression tests*
    - ✓ *Four-point-bend tests*
    - ✓ *Fractography*

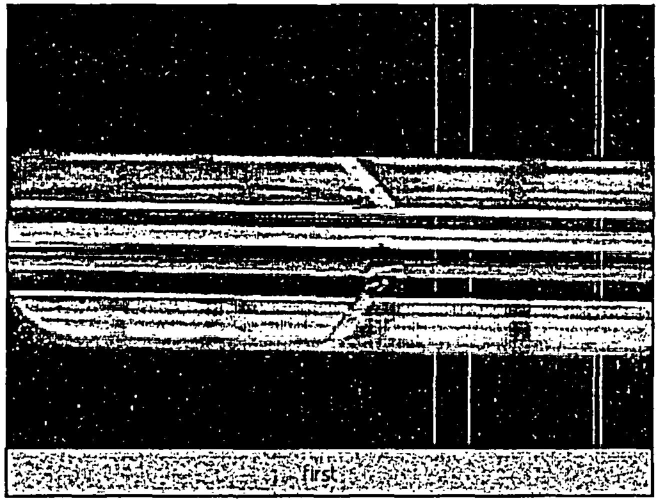
Schematic Illustration of LOCA System

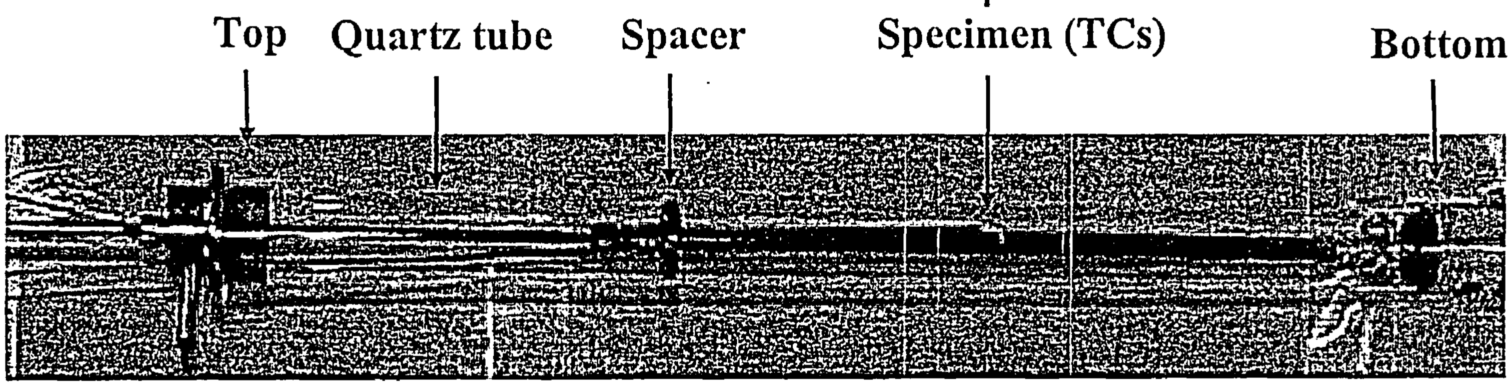


LOCA Apparatus



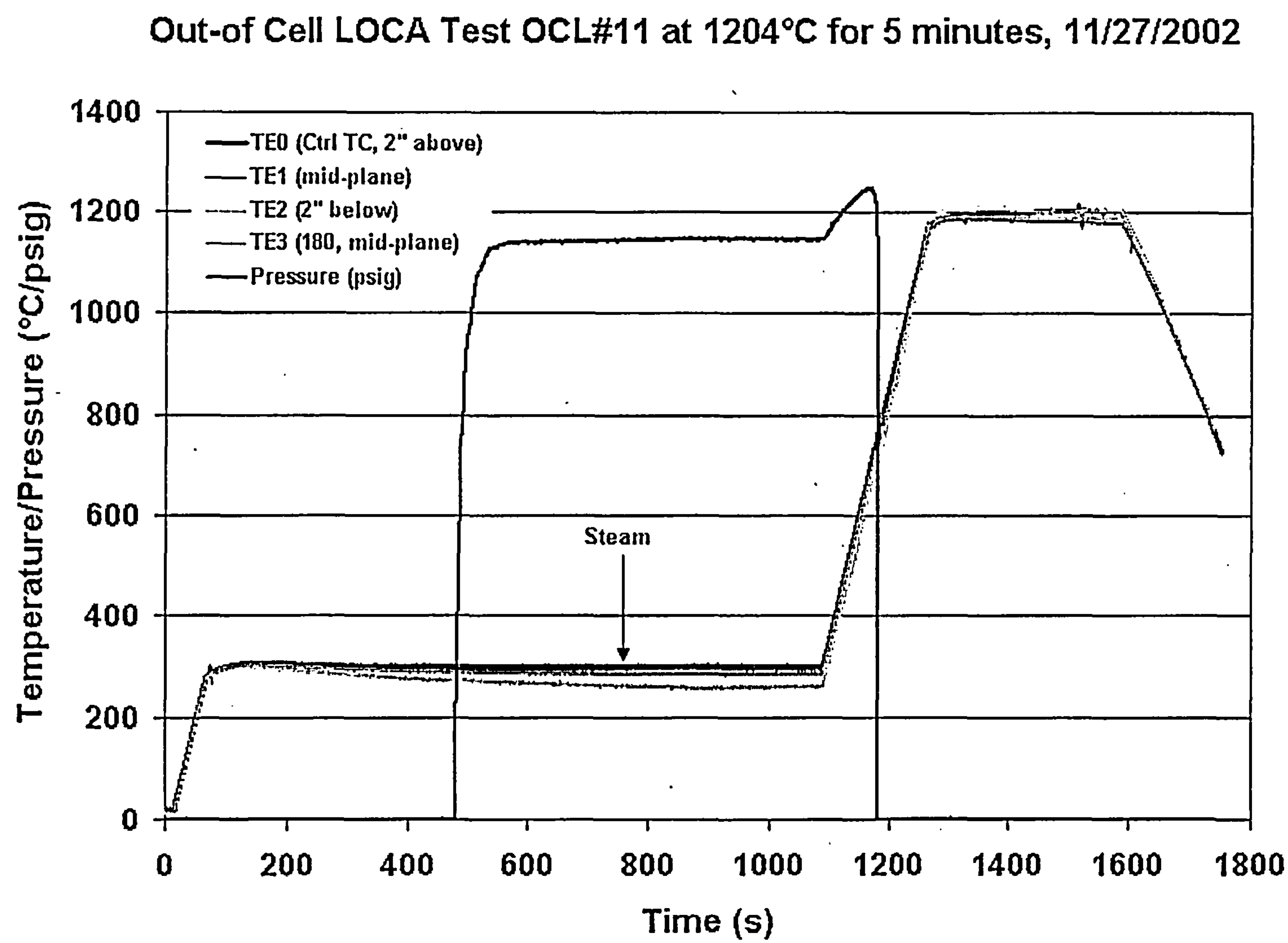
Real-time Observation of the LOCA Test



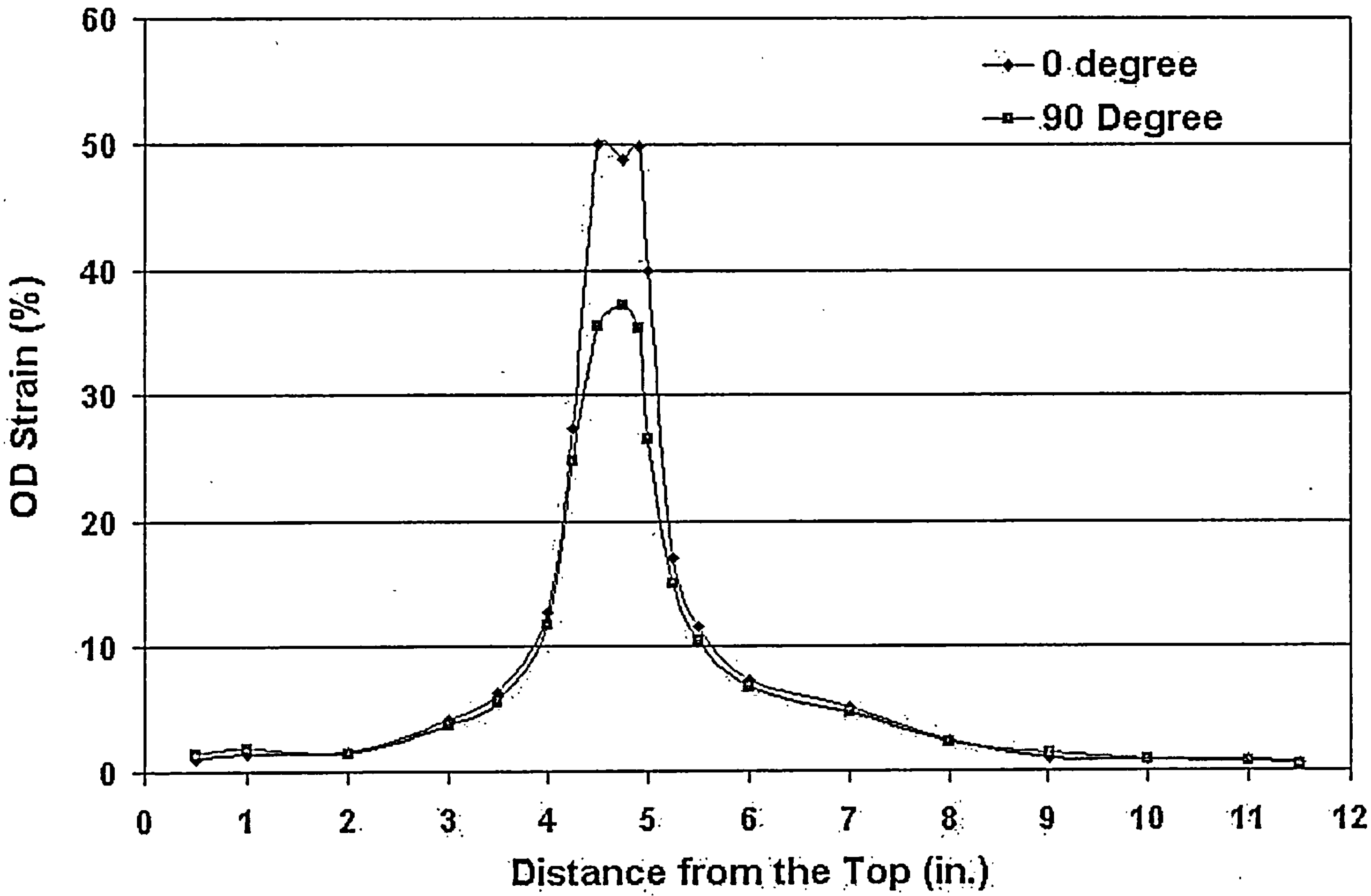


Hanging LOCA Test Train

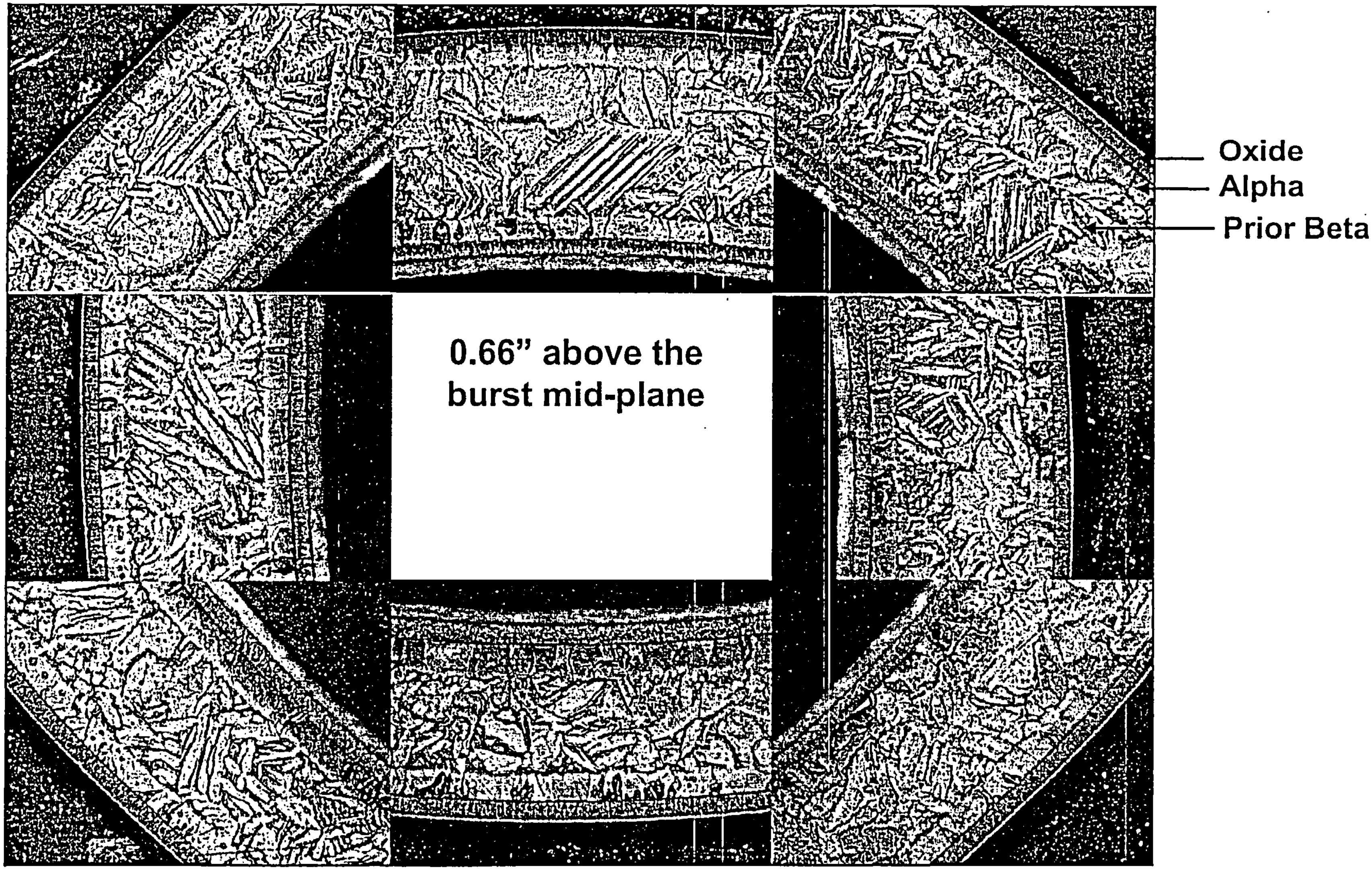
Temperature and Pressure Histories of Test OCL#11



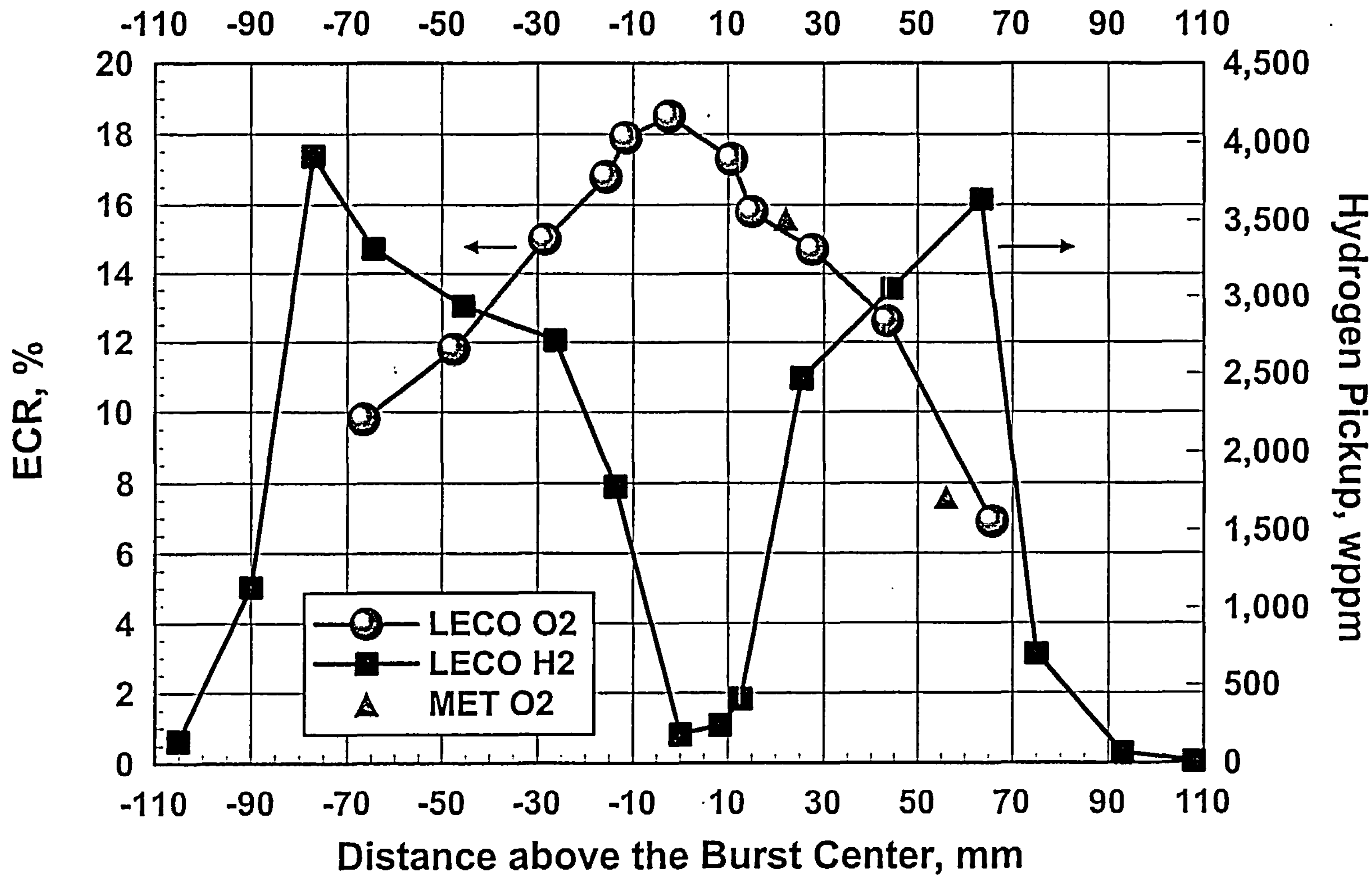
OD Strain of OCL#11 sample (1200°C for 5 min)



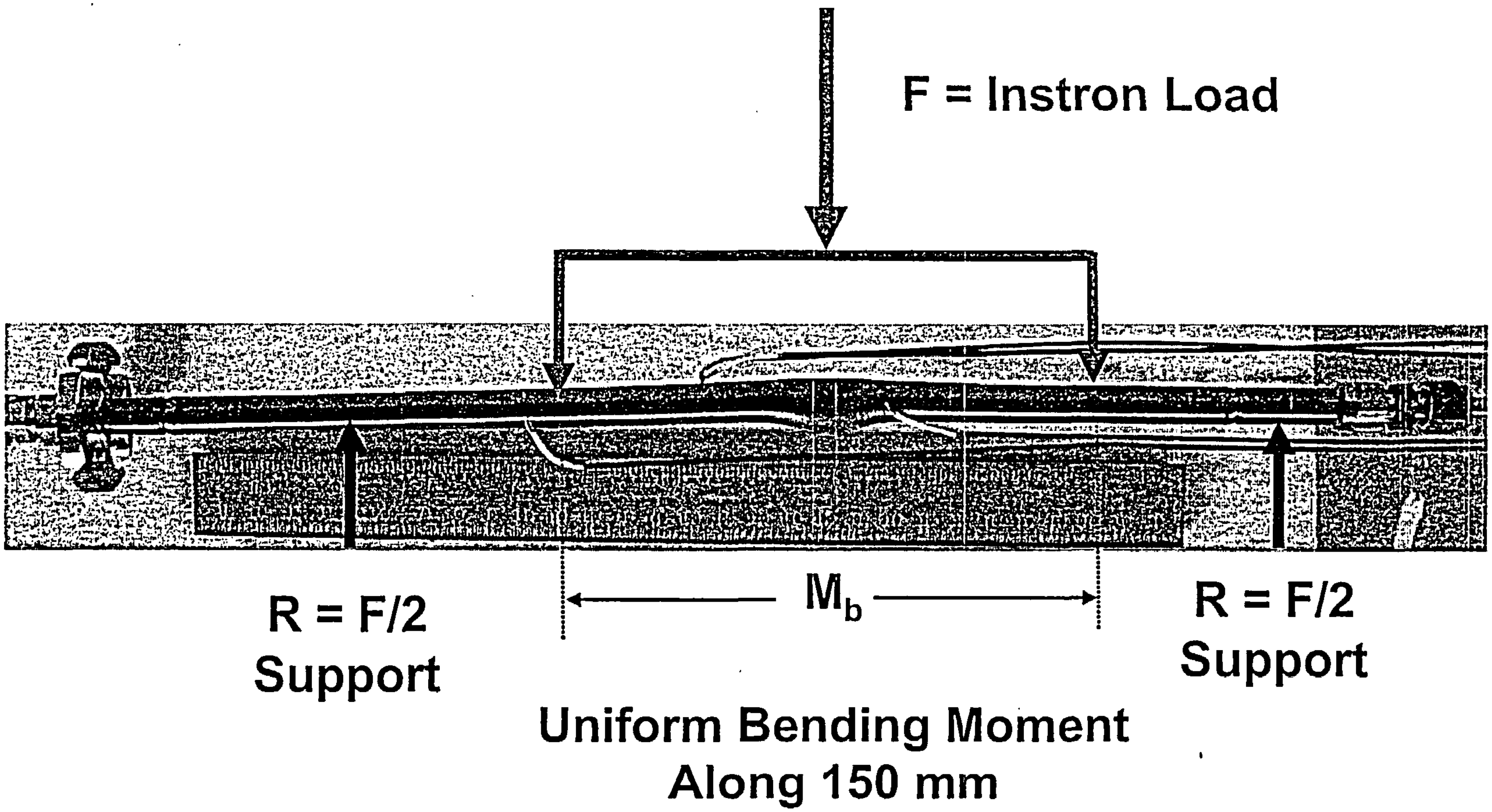
Cladding Metallographic Results for OCL#11 Specimen



LOCA Integral Test Results for Zry-2 (1200°C for 5 min)



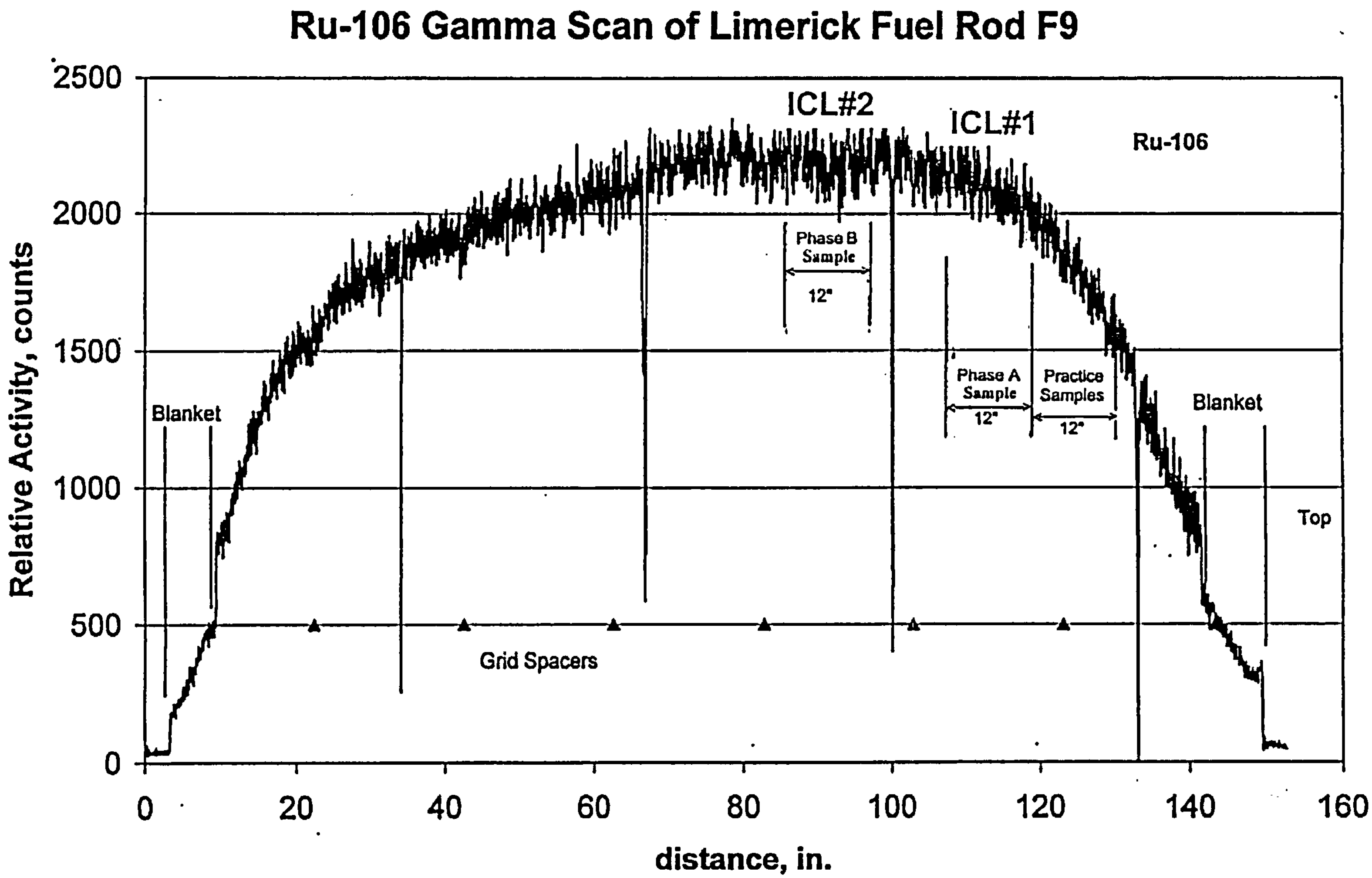
4-Point-Bend Test: Burst Area under Axial Tension



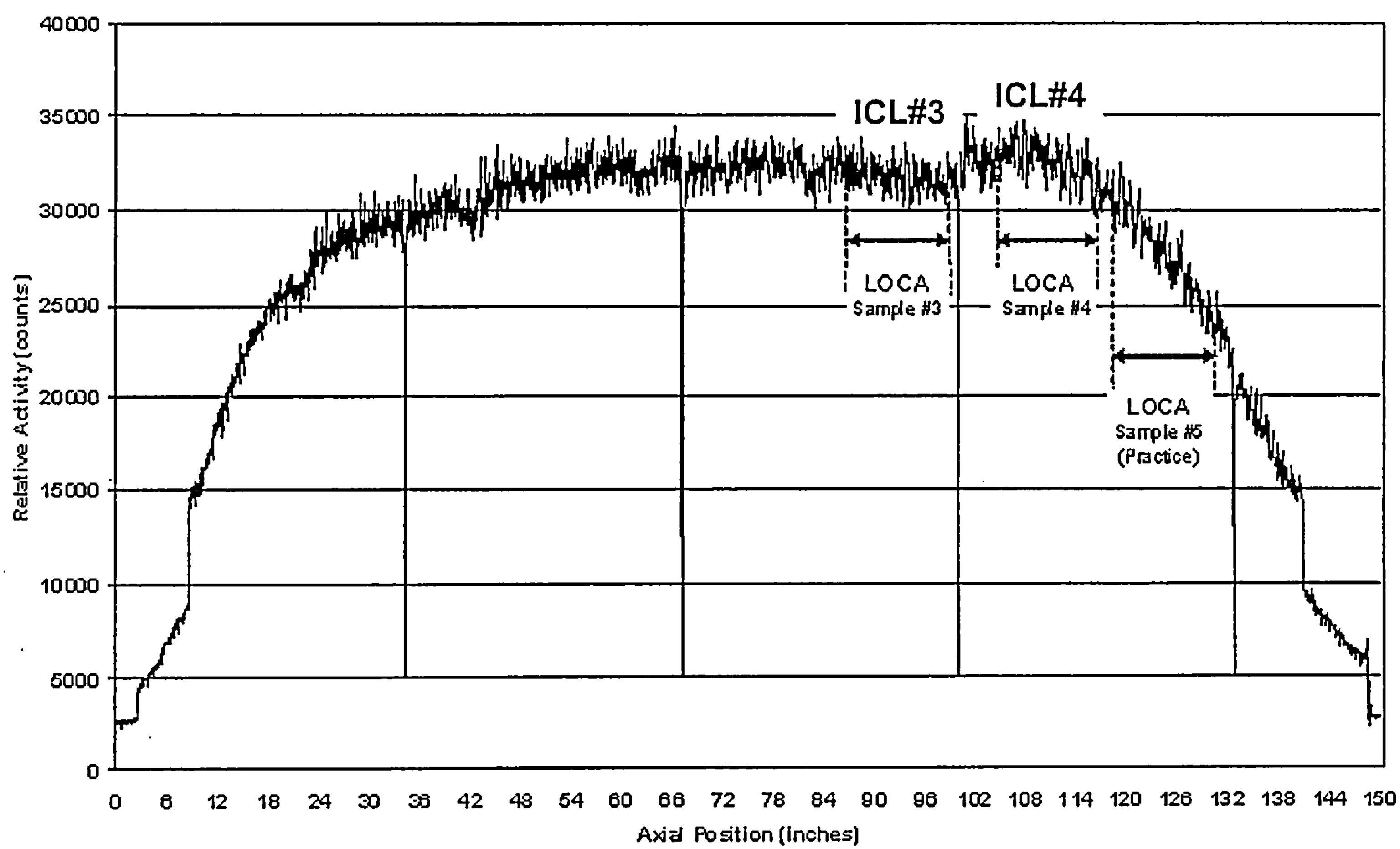
LOCA Tests with High-Burnup BWR Fuel

- **In-cell LOCA Test ICL#1**
  - Ramp-to-Burst Test Conducted in Argon
- **In-cell LOCA Test ICL#2**
  - LOCA Sequence with 5-minute Oxidation at 1204°C and Slow-Furnace Cooling
- **In-cell LOCA Test ICL#3**
  - 5-minute Oxidation at 1204°C Followed by Quench at 800°C (quarts tube failed at 480°C)
- **In-cell LOCA Test ICL#4**
  - Full LOCA Sequence (5-minute Oxidation at 1204°C) with Quench at 800°C

Gamma Scan of Limerick Fuel Rod F9

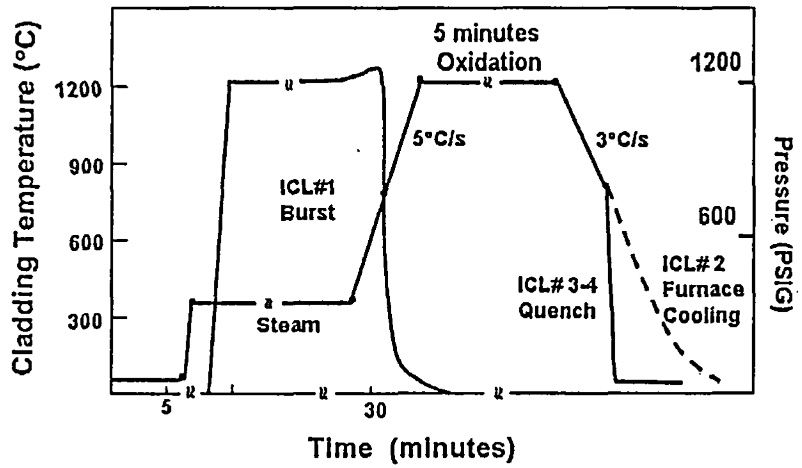


Gamma Scan of Limerick Fuel Rod J4

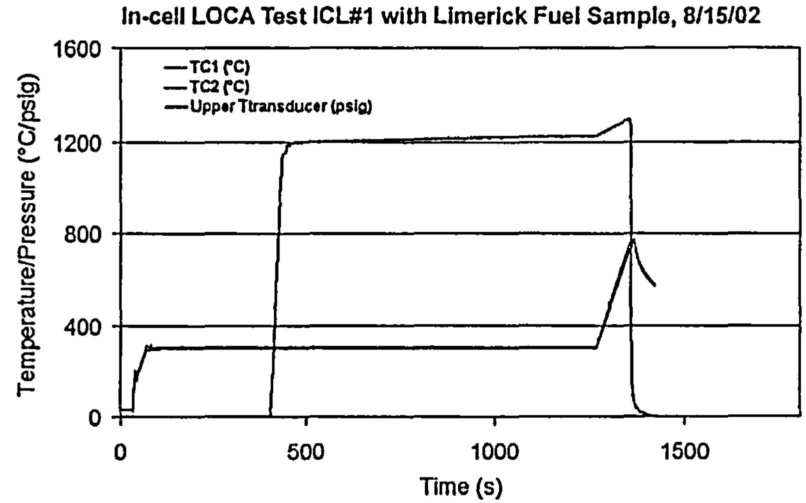


Temperature and Pressure of In-cell LOCA Tests

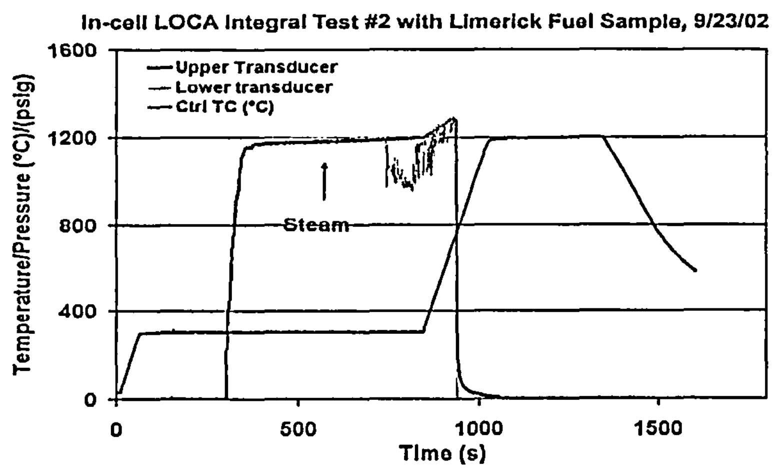
LOCA Integral Test Sequence



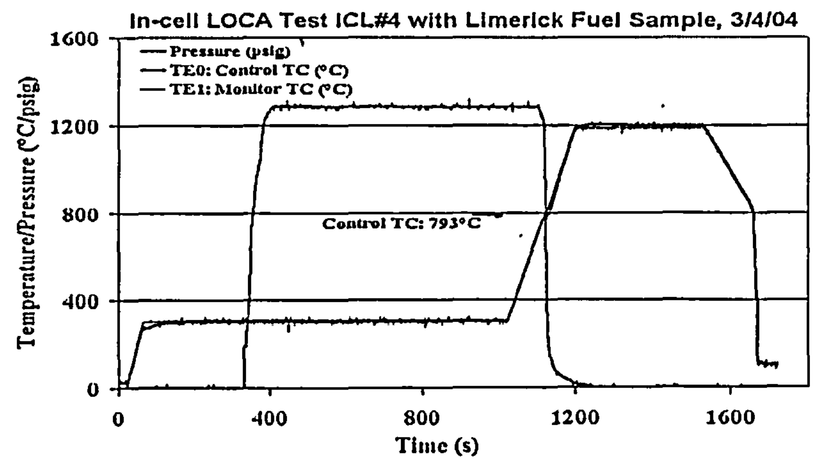
In-cell LOCA ICL#1



In-cell LOCA ICL#2

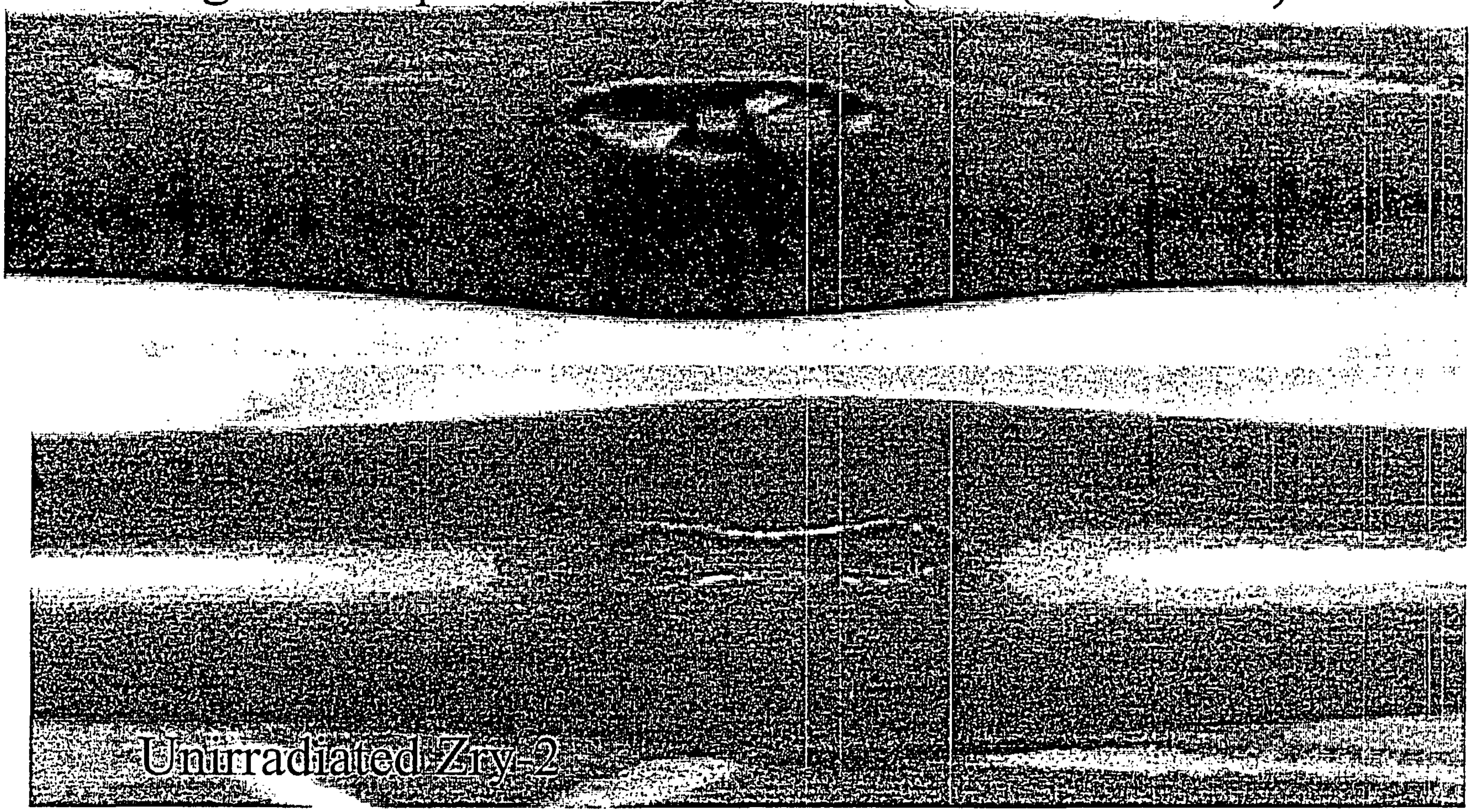


In-cell LOCA ICL#4

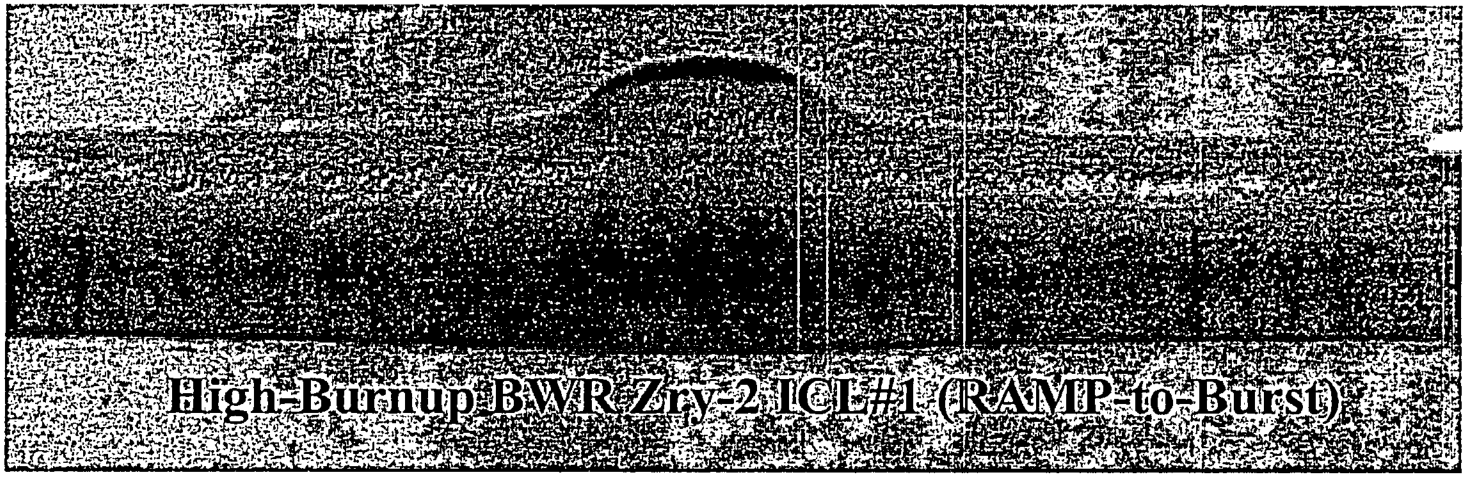


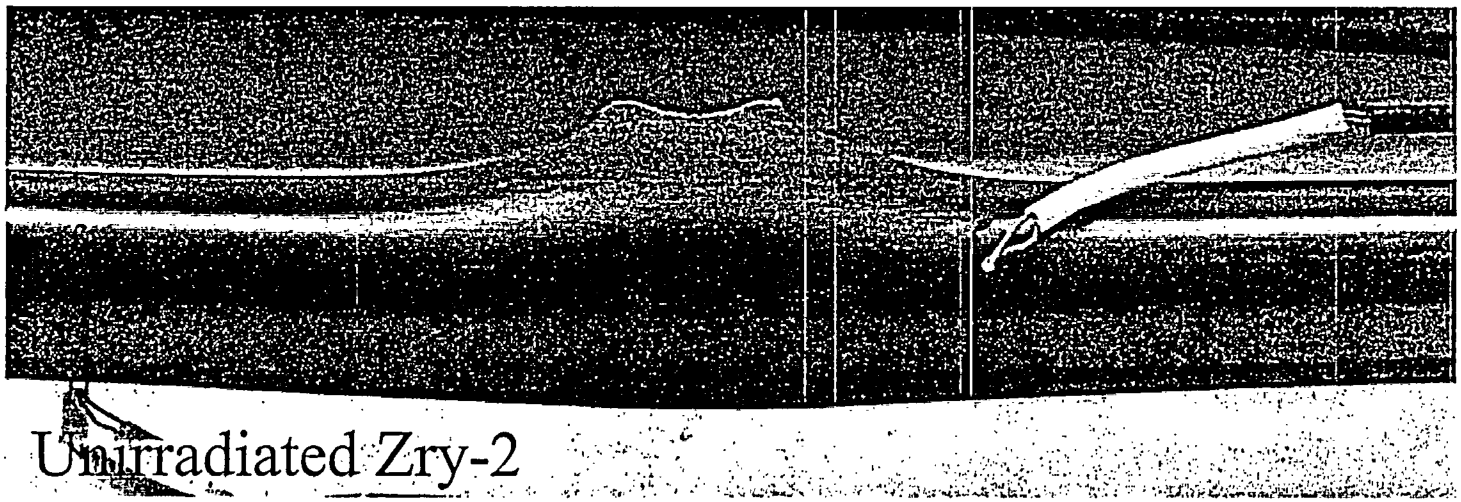
Burst Opening Comparison

High-Burnup BWR Zry-2 ICL#1 (RAMP-to-Burst)

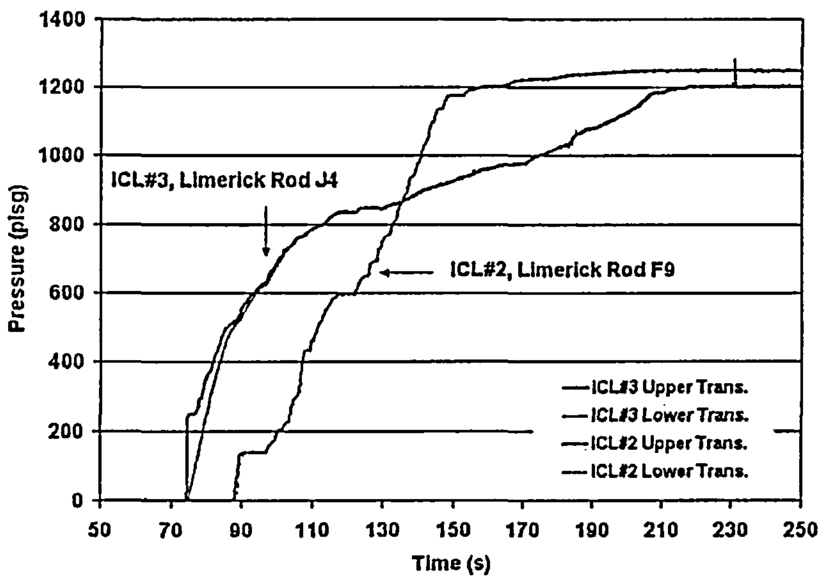


Side View of Burst Opening

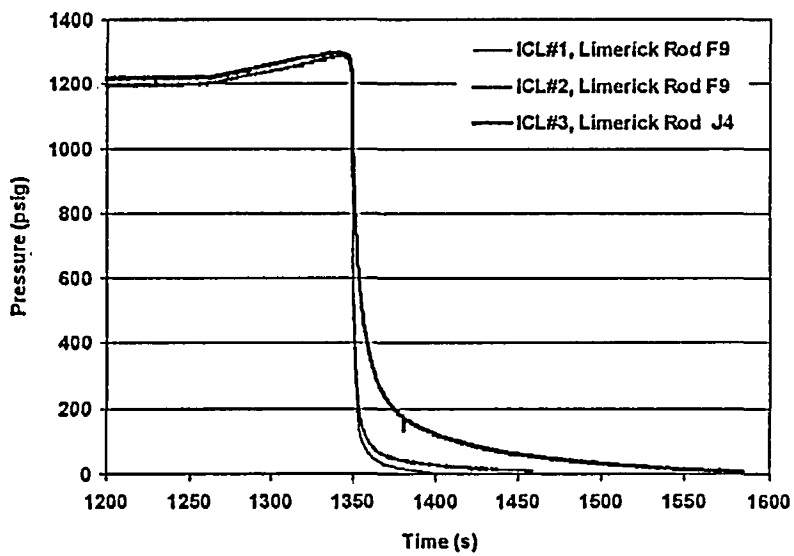




Gas Communication in High-burnup BWR Rods

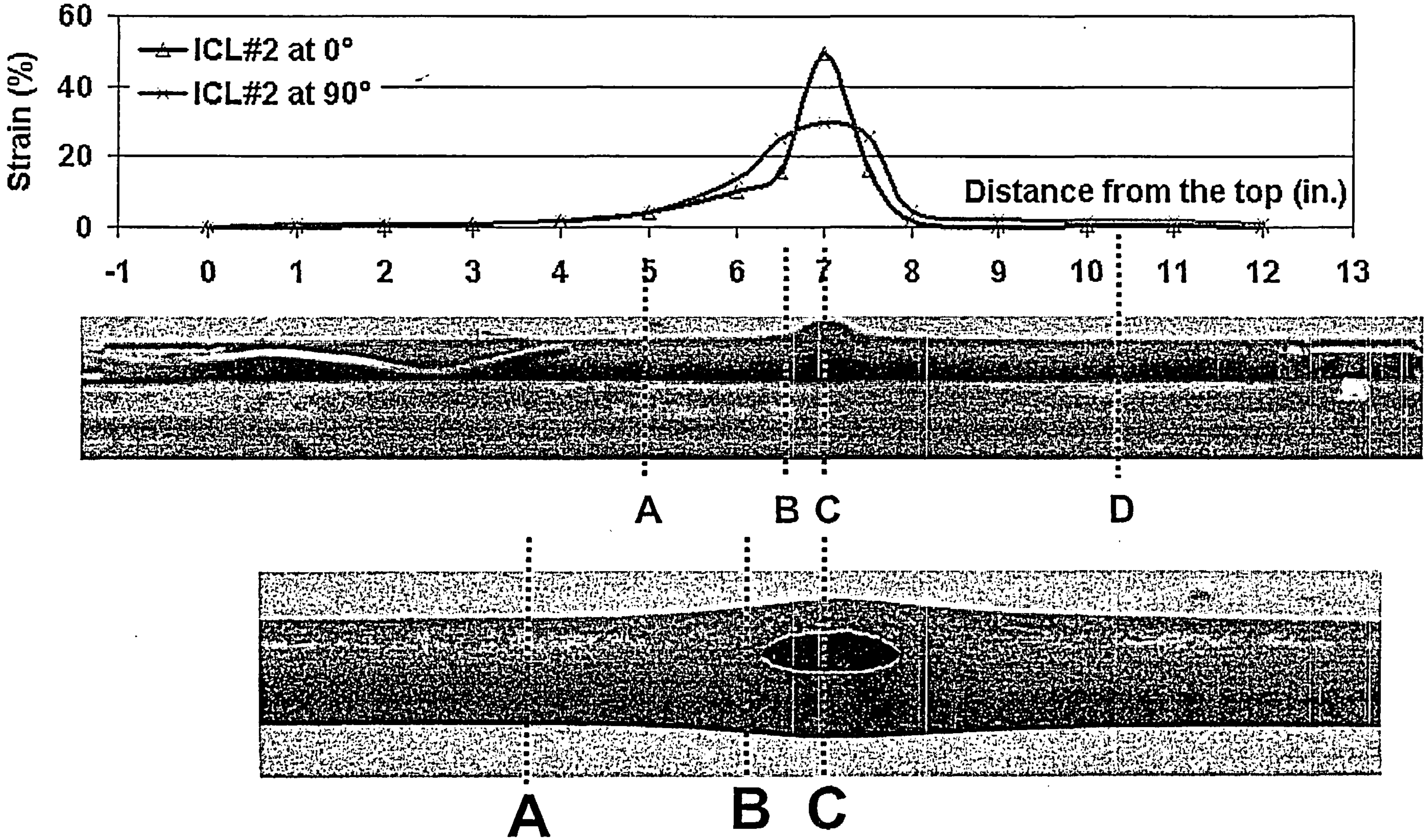


Upper and lower transducer reading at room temperature

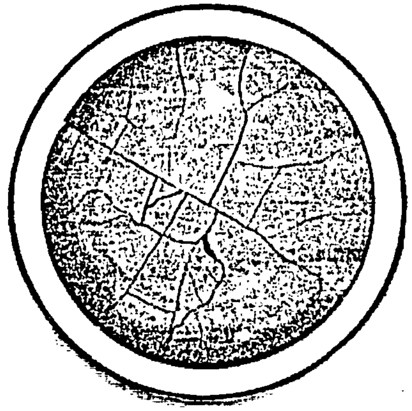


Upper transducer readings after burst

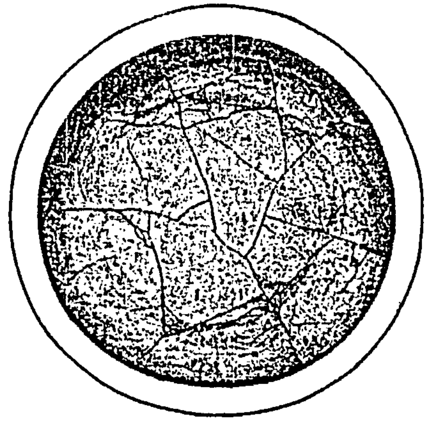
Post-test Characterization for ICL#2 Specimen



Fuel Metallographic Results for ICL#2 Specimen

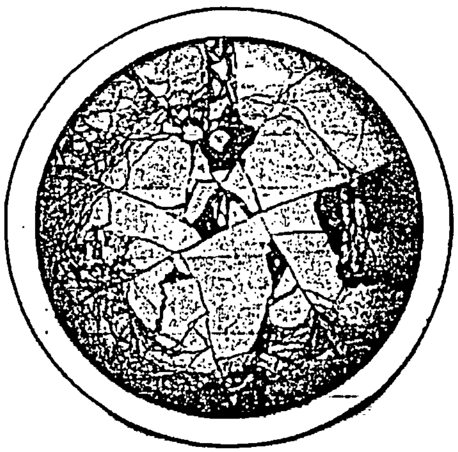


Pre-test Specimen



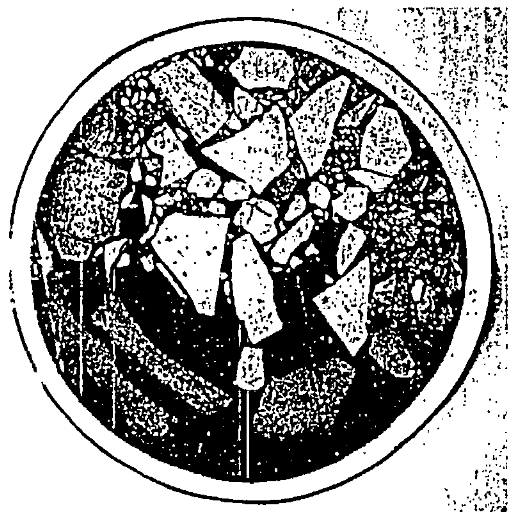
≈80 mm away from the burst

Post-test Specimen D, Strain: 0% - 2%



≈ 50 mm above burst mid-plane

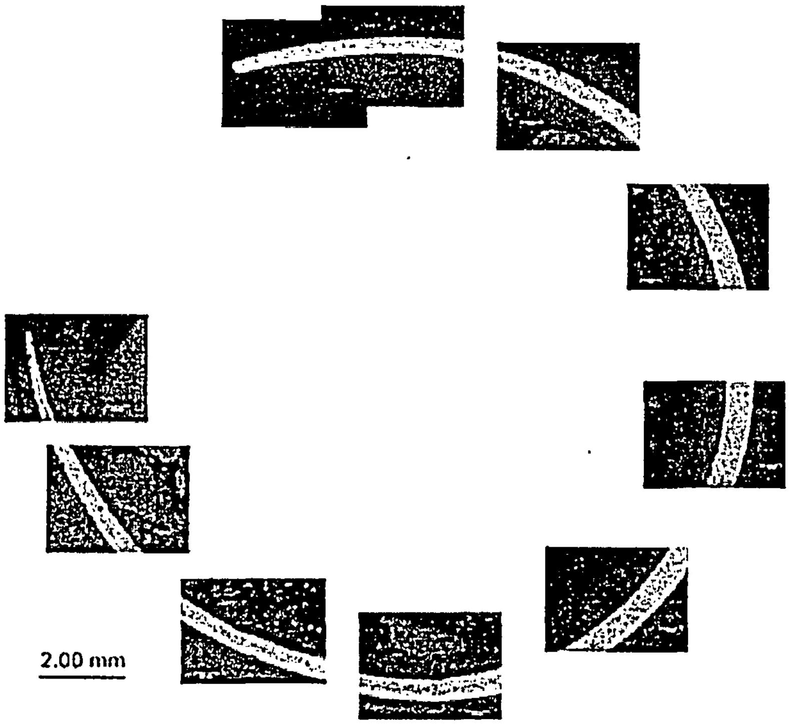
Post-test Specimen A, Strain: 2% - 4%



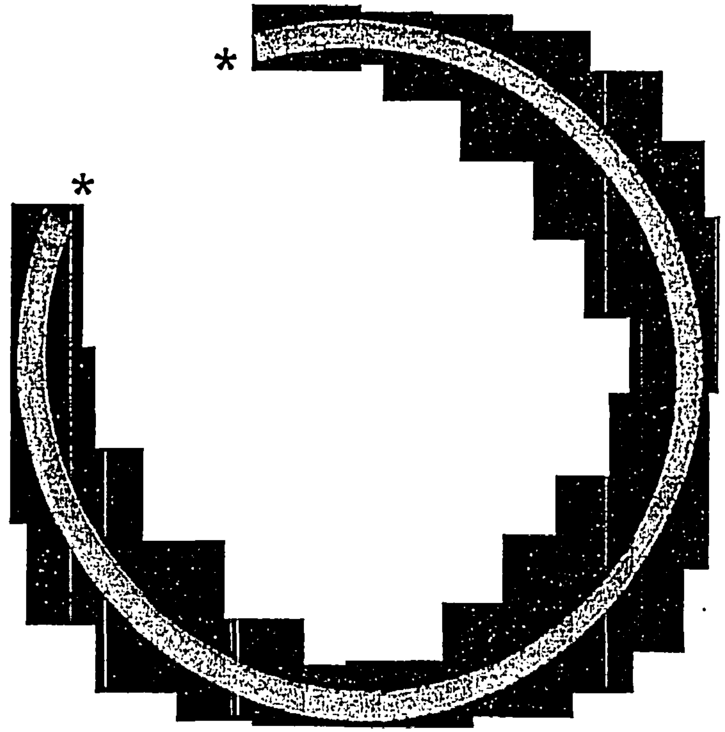
≈12 mm above burst mid-plane

Post-test Specimen B, Strain: 15% - 25%

Cross-Section for High Burnup LOCA Samples

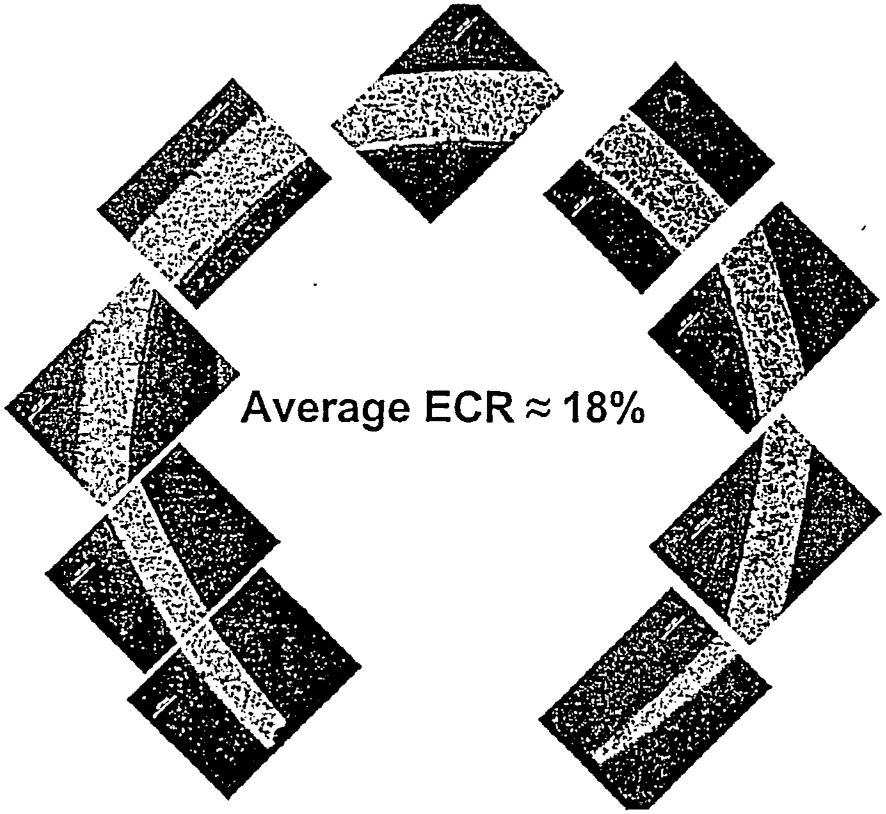


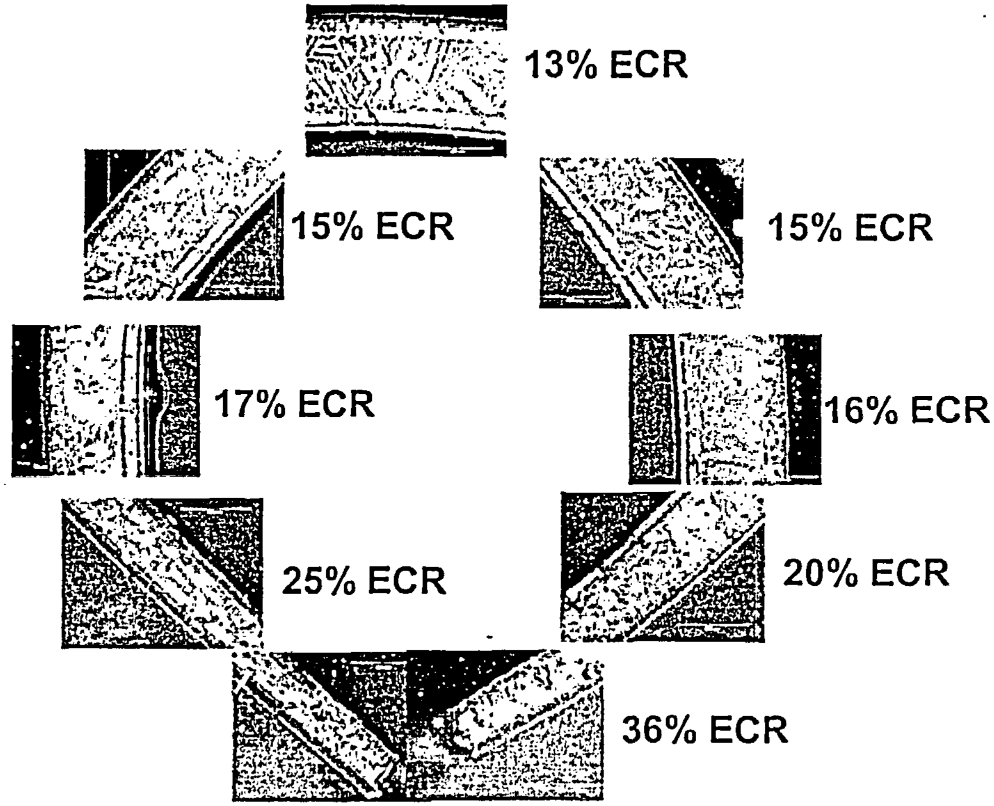
Post-test high-burnup ICL#2:
Burst mid-plane (1204°C, 5 min)
Strain: 30% - 50%
OD Oxide: 49 μm; ID Oxide: 60 μm



Post-test high-burnup ICL#3:
Burst mid-plane (1204°C, 5 min)
Strain: 36% - 52%
OD Oxide: 53 μm; ID Oxide: 56 μm
* Tip lost during sample handling

High Mag. Images of LOCA Burst Cross-Section

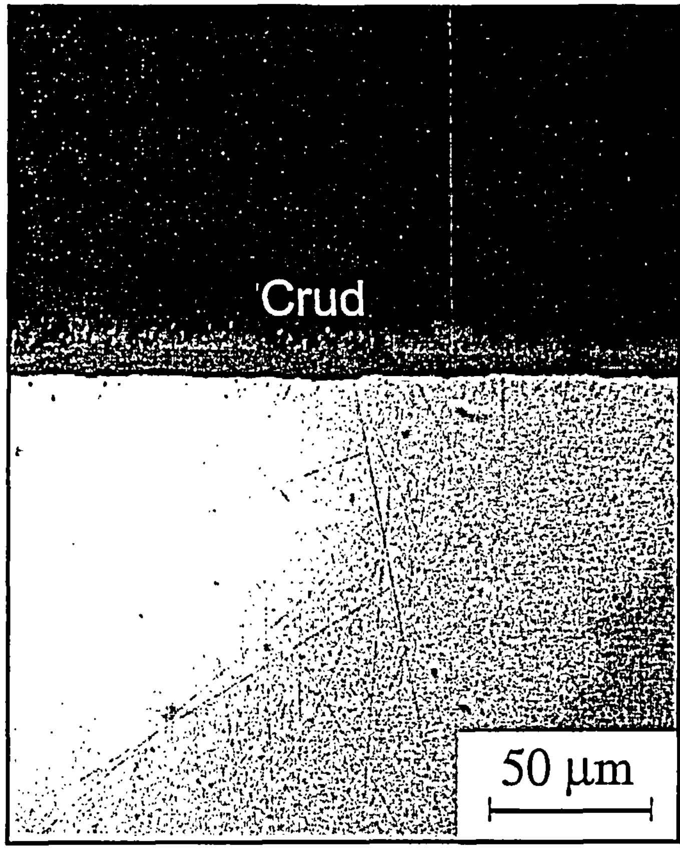




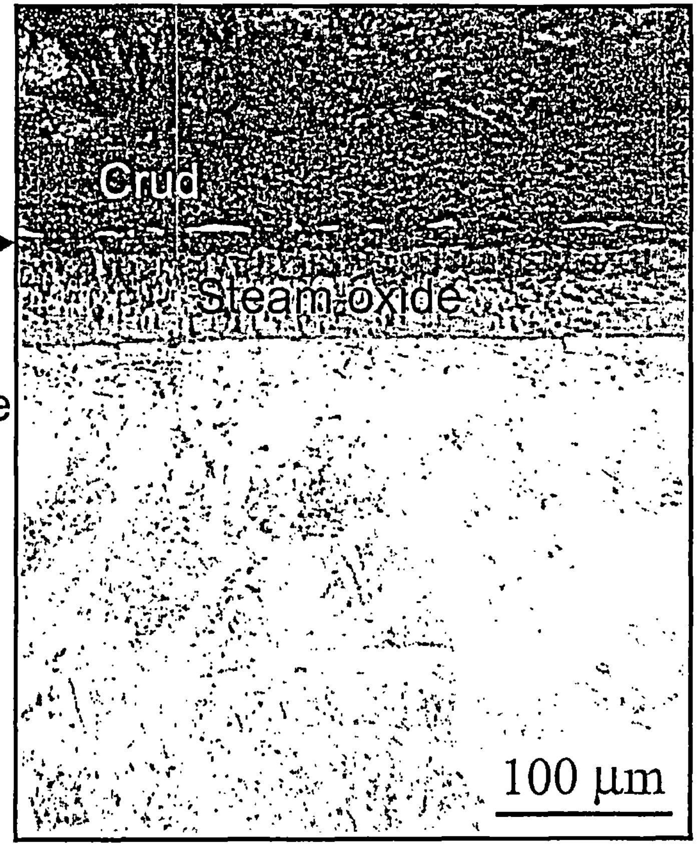
High Burnup Sample ICL#2:
Burst mid-plane (1204°C, 5 min)
Strain: 30% - 50%

OCL#11 Sample(Unirrad. Zry-2)
Burst mid-plane (1204°C, 5 min)
Strain: 37% - 56%

Steam Oxide Layer of ICL#2 Specimen

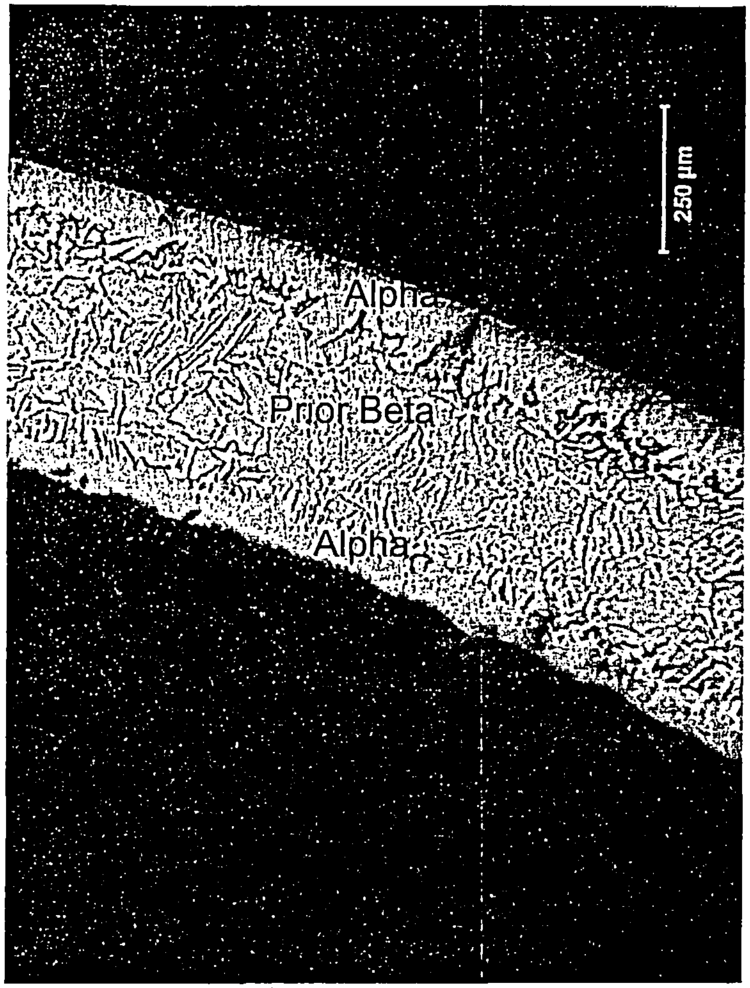


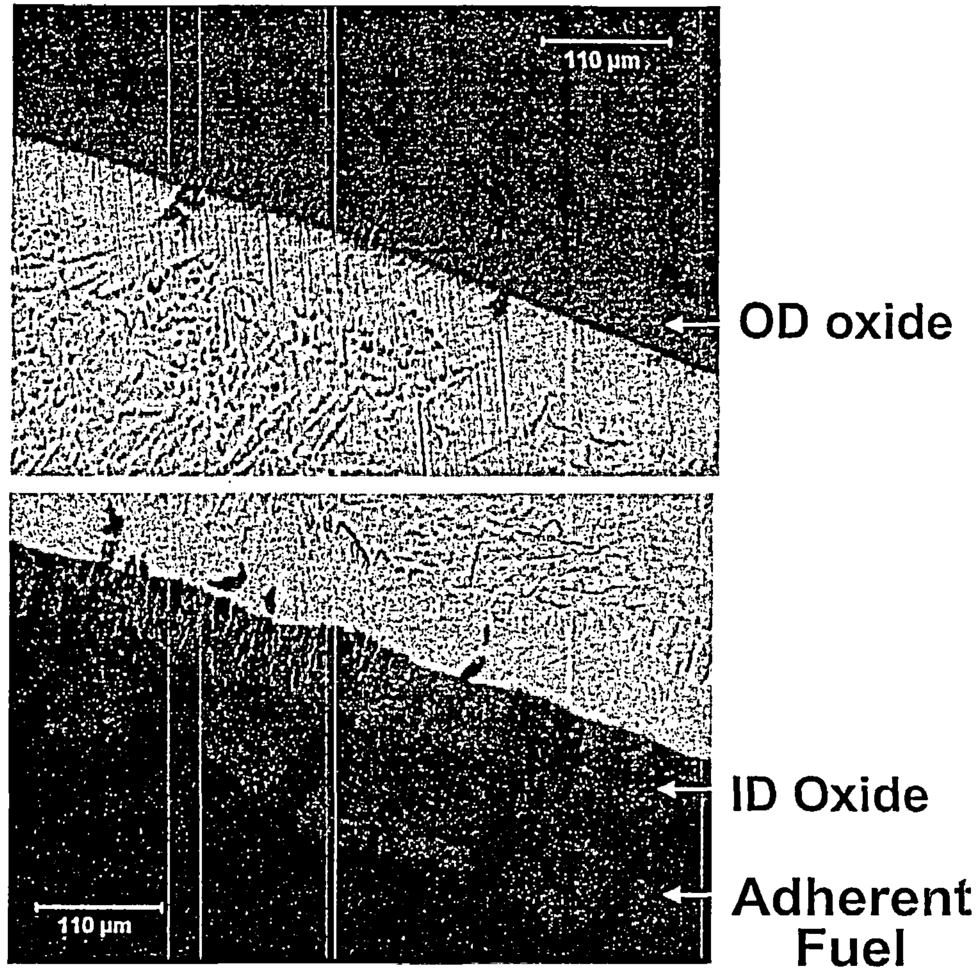
Pre-test Specimen



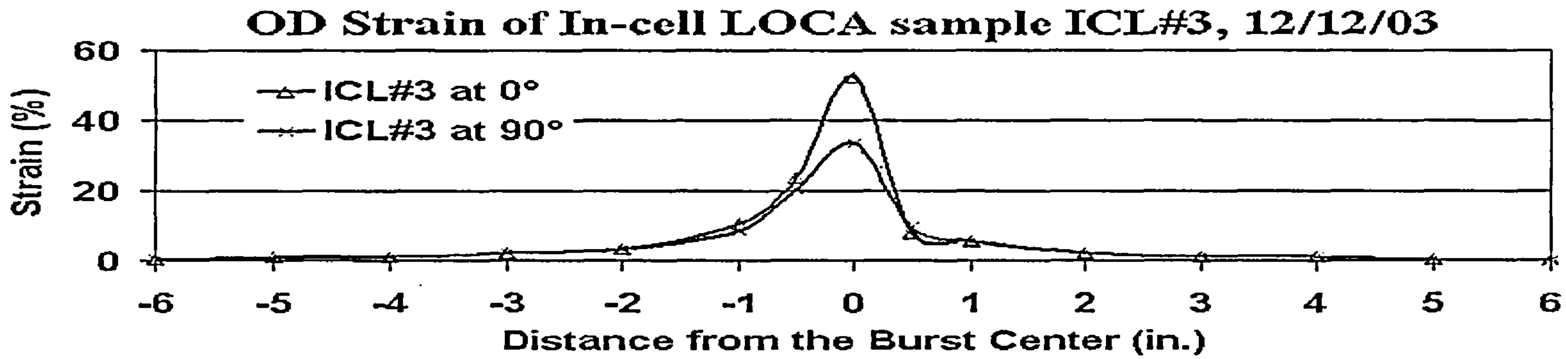
Post-test Specimen

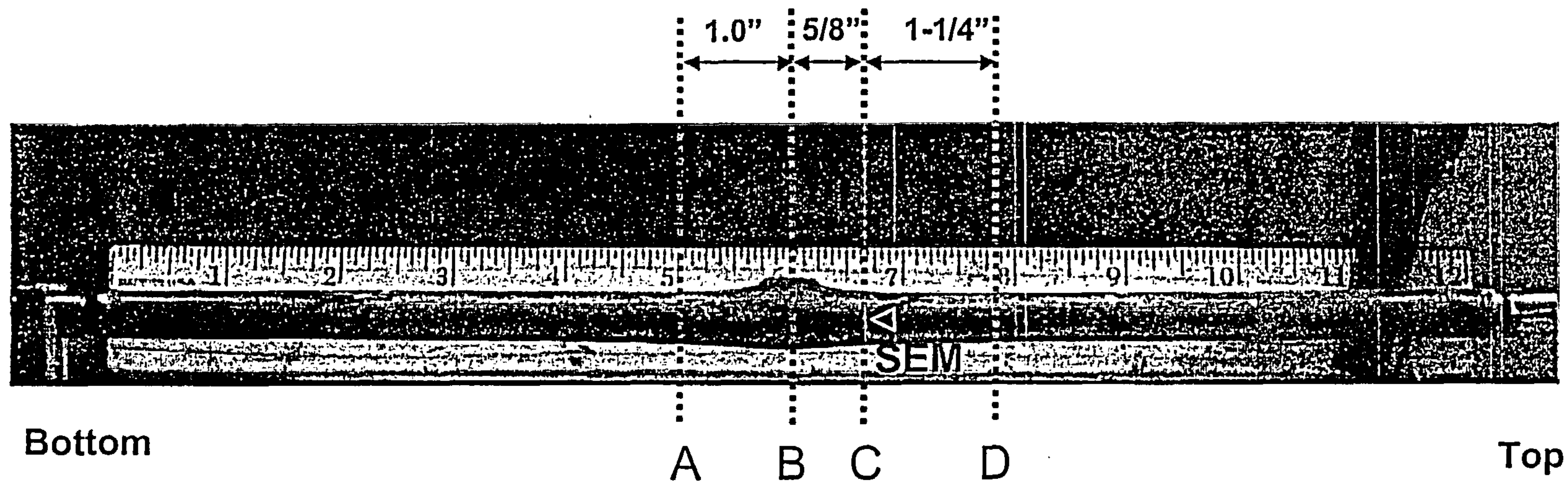
Cladding Cross-sections of ICL#3 Burst Opening





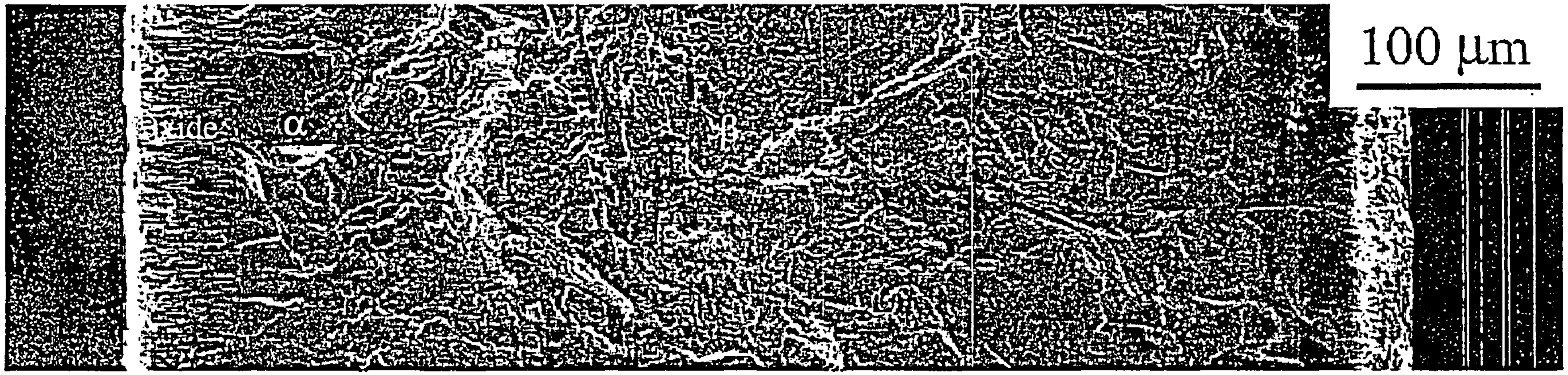
Post-test Characterization for ICL#3 Specimen

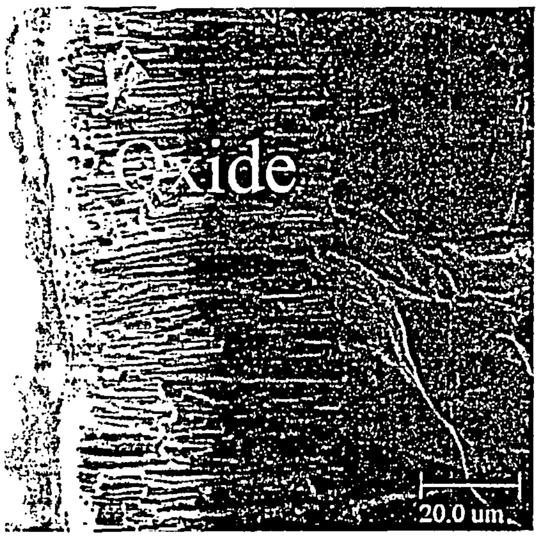




Sample ICL#3 was broken at locations A, B and C during the sample handling before the sectioning was performed at location D.

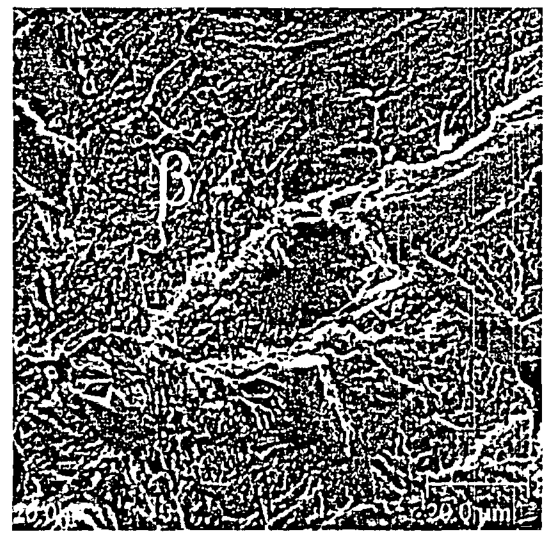
Fractography of ICL#3 Sample (20 – 30 mm to Mid-plane)





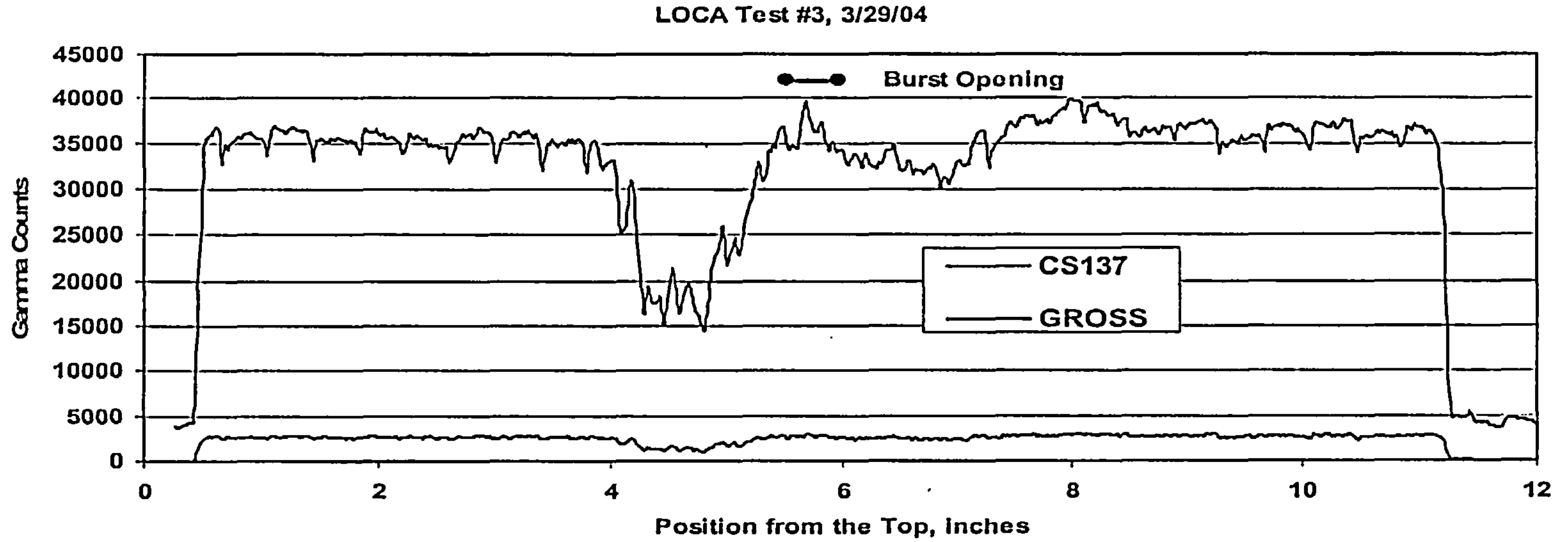
Oxide

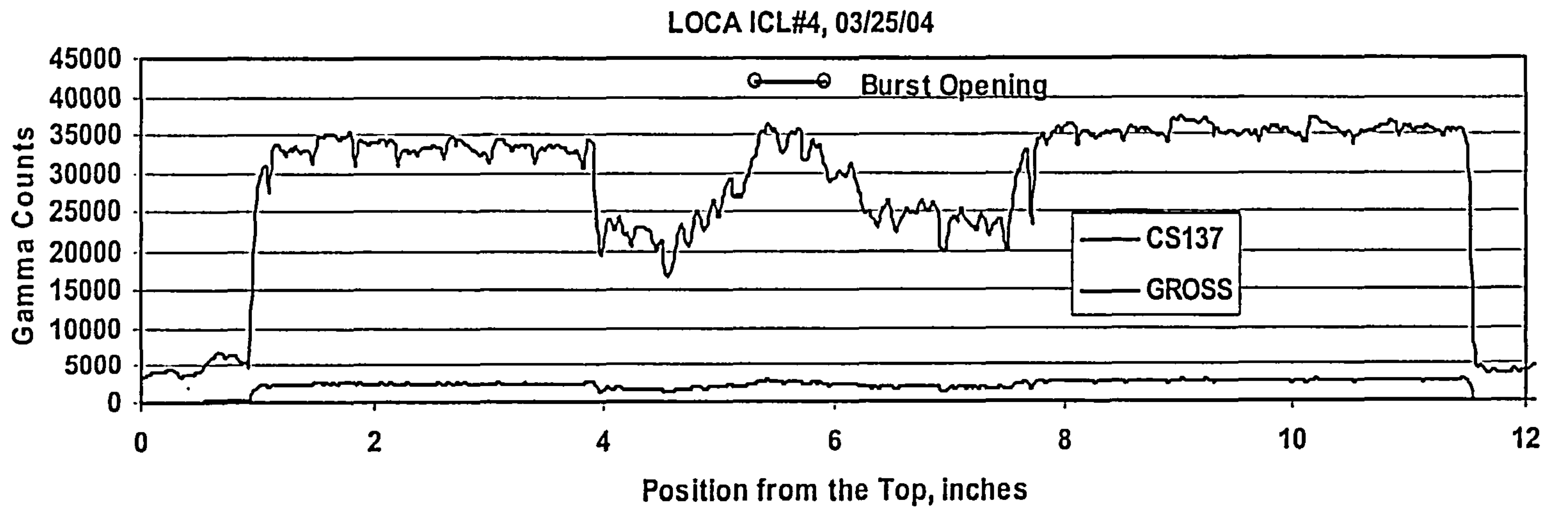
Alpha



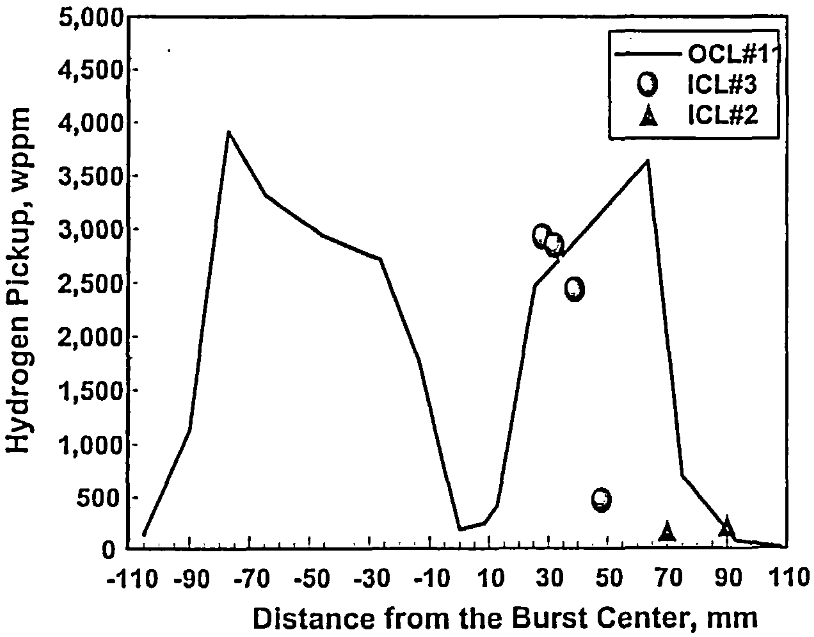
Prior Beta

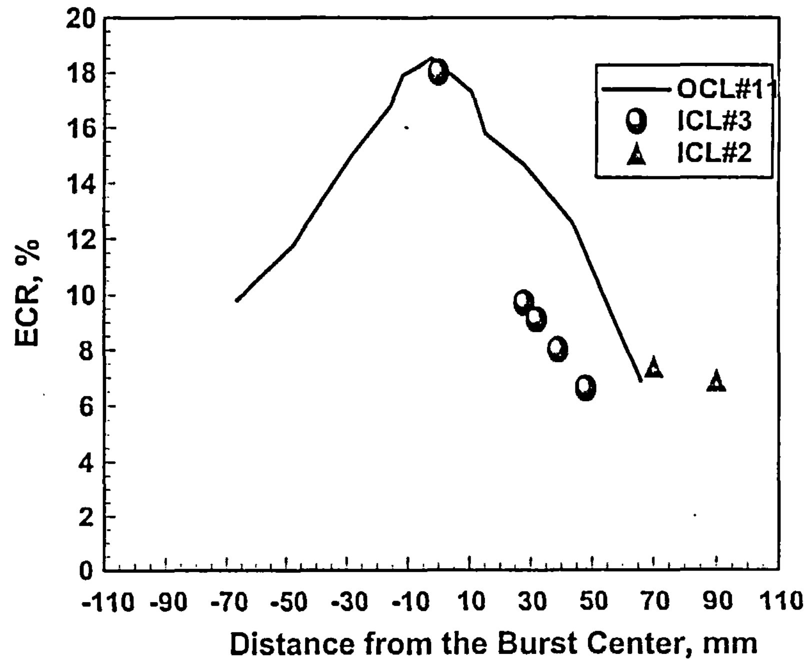
Gamma Scan Profiles for ICL#3 and ICL#4 Specimens



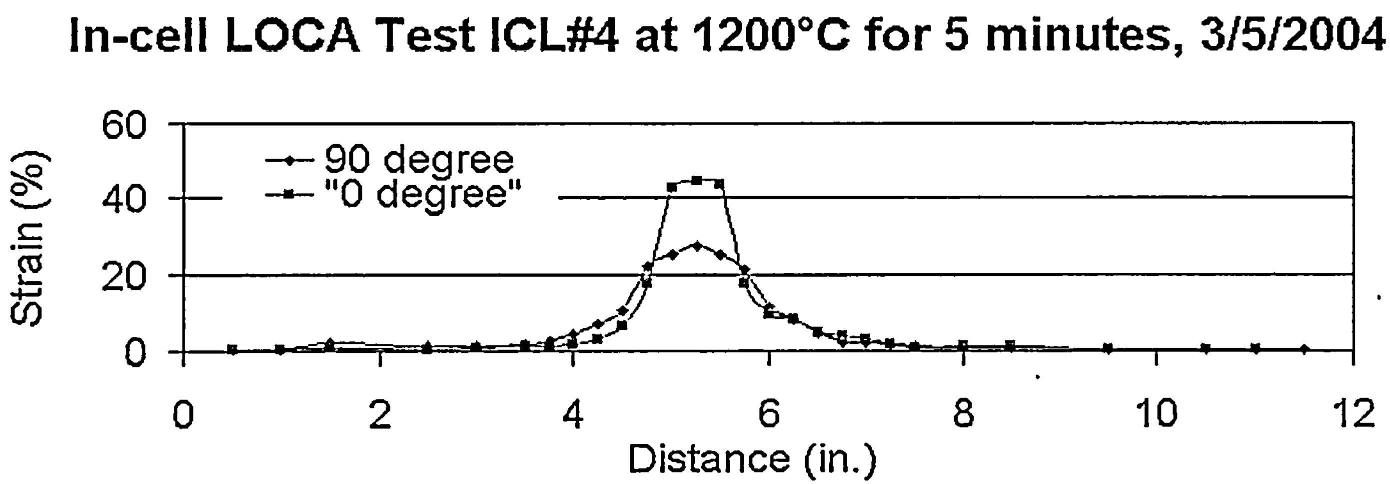


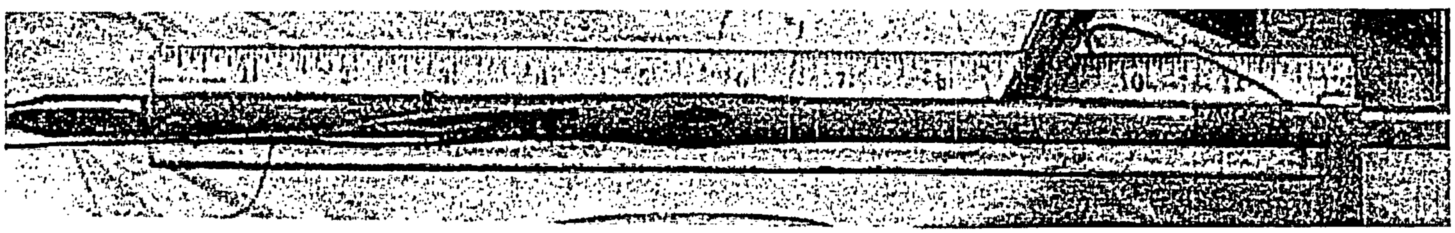
H and O Analyses of ICL#2 and ICL#3





Post-test Characterization for ICL#4 Specimen





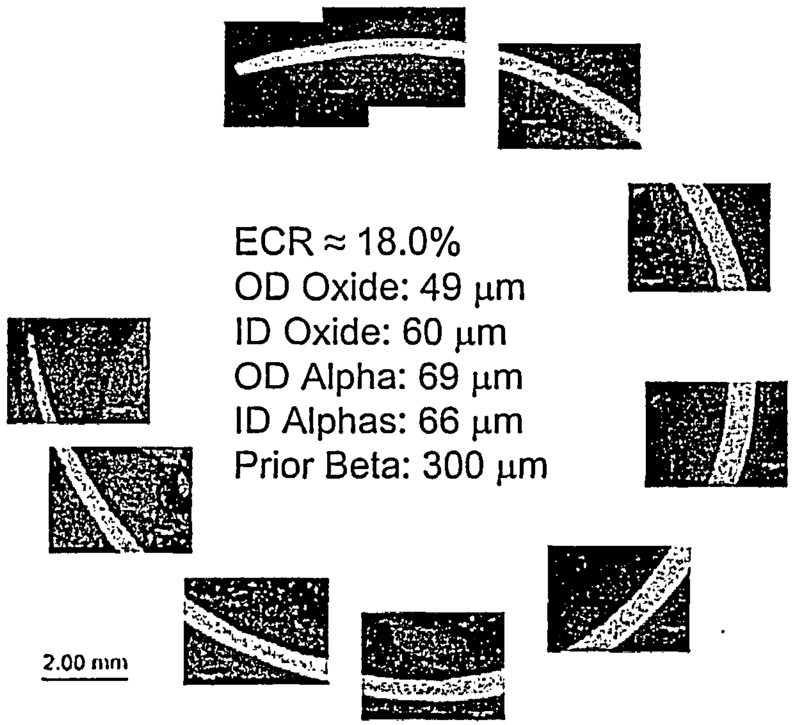
Summary of In-Cell LOCA Integral Tests with High-Burnup Limerick BWR Fueled Specimens

Parameter	ICL#1	ICL#2	ICL#3	ICL#4	OCL#11
Hold Time, minutes	0	5	5	5	5
Max. Pressure, MPa	8.96	8.87	9.0	8.86	8.61
Burst Pressure, MPa	≤8.61	≤8.01	8.6	8.0	≤7.93
Burst Temperature, °C	≈755	≈750	≈730	≈790	≈ 750
Burst Shape	Oval	Oval	Oval	Oval	Dog Bone
Burst Length, mm	13	14	11	15	11
Max. Burst Width, mm	3	3.5	4.6	5.1	1
Length of Balloon, mm	70	90	90	76	140
$(\Delta D/Do)_{max}$, %	38±9	39±10	43±9	36±9	43±10
Max. Calculated ECR, %	0	≈20	≈21	≈20	≈21
Max. ΔH, wppm	...	> 220	≥ 2900	TBD	3900

Summary

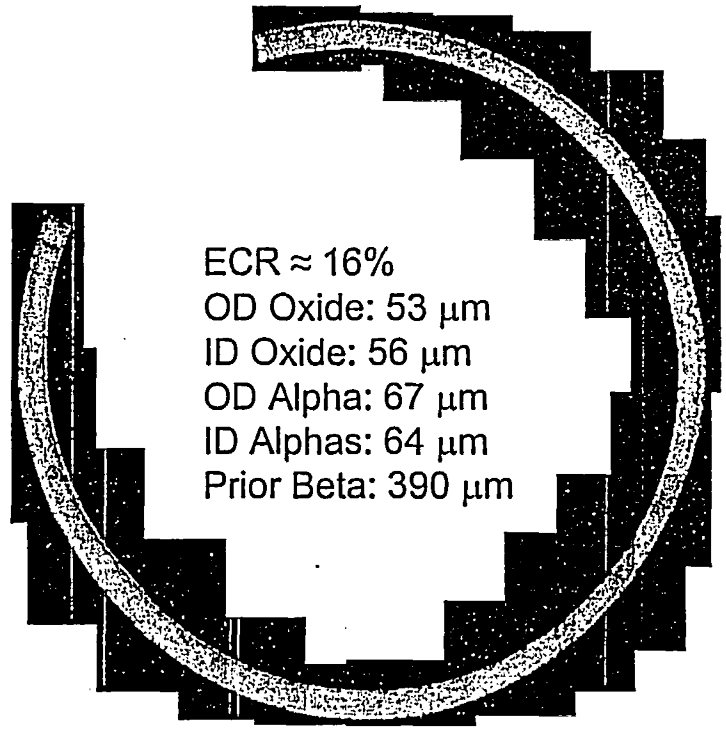
- **High-burnup BWR LOCA Test Matrix: Completed**
  - ICL#1: Ramp-to-burst test conducted in argon
  - ICL#2: LOCA sequence with 5-min. oxidation at 1204°C and slow cooling
  - ICL#3 and ICL#4: Full LOCA sequence with quench
- **Future Work with High-Burnup Samples**
  - Perform bend test on post-quench fueled samples
  - Conduct ring compression tests on defueled samples
  - Initiate Robinson PWR oxidation and LOCA tests

Cross-Section for High Burnup LOCA Samples



ECR ≈ 18.0%
OD Oxide: 49 μm
ID Oxide: 60 μm
OD Alpha: 69 μm
ID Alphas: 66 μm
Prior Beta: 300 μm

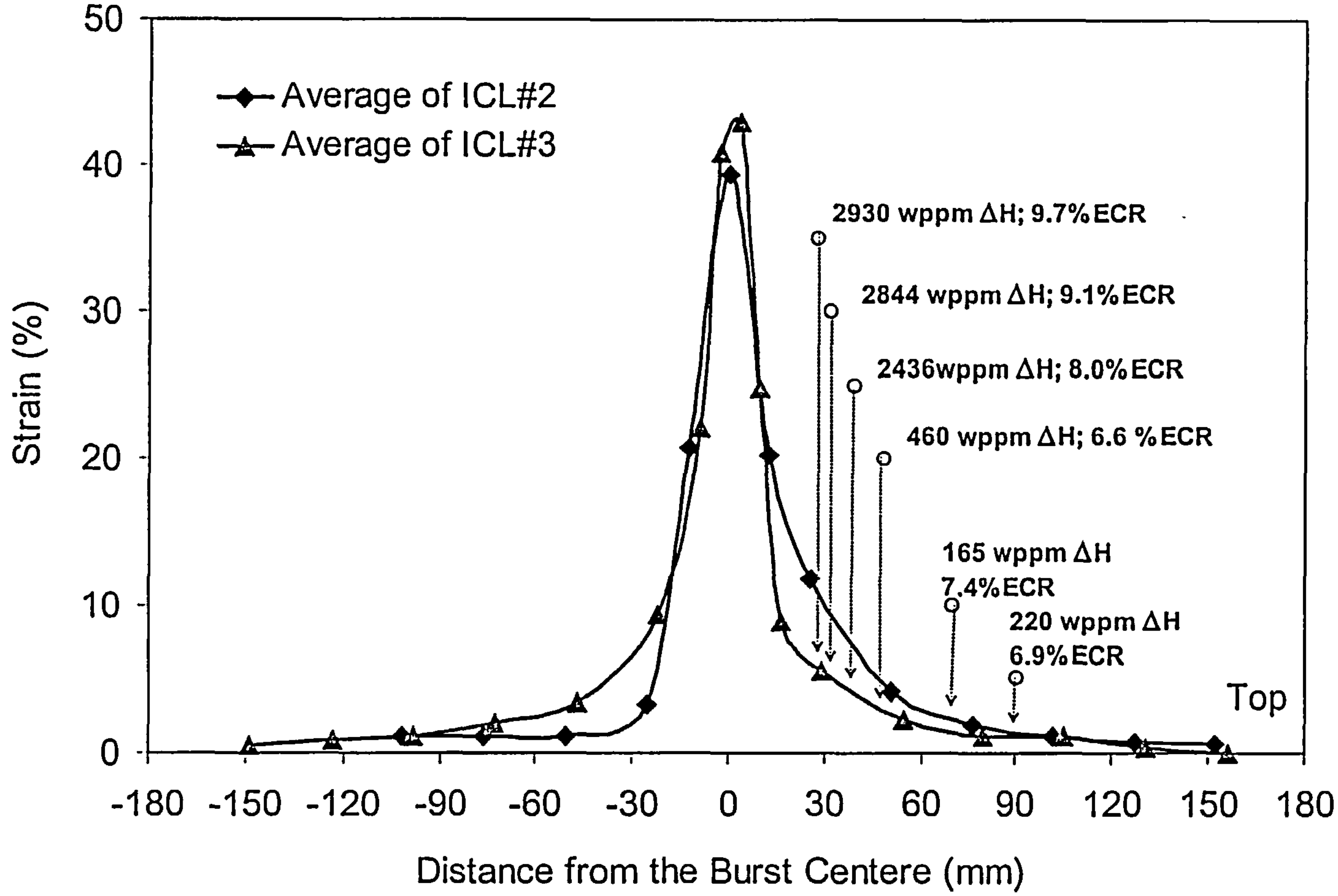
Post-test high burnup ICL#2:
Burst mid-plane (1204°C, 5 min)
Max. Strain: 30% - 50%



ECR ≈ 16%
OD Oxide: 53 μm
ID Oxide: 56 μm
OD Alpha: 67 μm
ID Alphas: 64 μm
Prior Beta: 390 μm

Post-test high burnup ICL#3:
Burst mid-plane (1204°C, 5 min)
Max. Strain: 36% - 52%

Secondary Hydriding of ICL#2 and ICL#3



Does M5™ balloon more than Zircaloy-4 under LOCA conditions ?

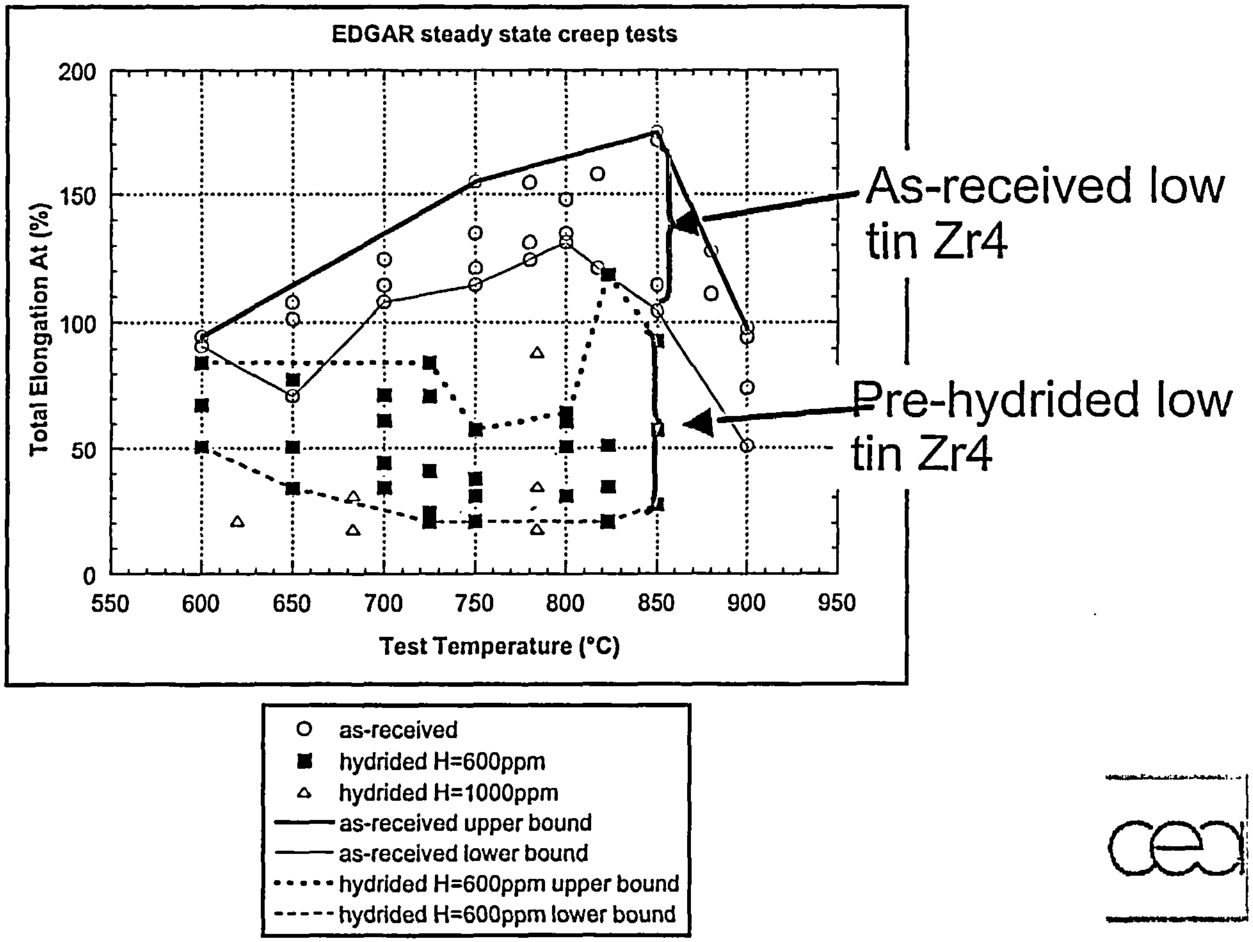
Nicolas WAECKEL (EDF)
Jean-Paul MARDON (FRA-ANP)
Laurence Portier (CEA)
Anne Lesbros (EDF)

May 25-27, 2004

BACKGROUND

- At the PHEBUS-STLOC meetings in Washington DC (Oct 2003) and Madrid (Nov 2003), IRSN asserted M5 cladding may exhibit bigger balloons than Zy4 under LOCA conditions
  - Higher risk of flow blockage for M5 ?
- IRSN's statement is based on isothermal **creep tests** performed on Nb based alloys (published by CEA, EDF and FRA-ANP in Toronto and Annecy ASTM Meetings)
  - At high fluence with a low hydrogen content (150 ppm) M5 exhibits higher ductility than Zy4, with a higher hydrogen content (600ppm)

ISOTHERMAL CREEP tests RESULTS

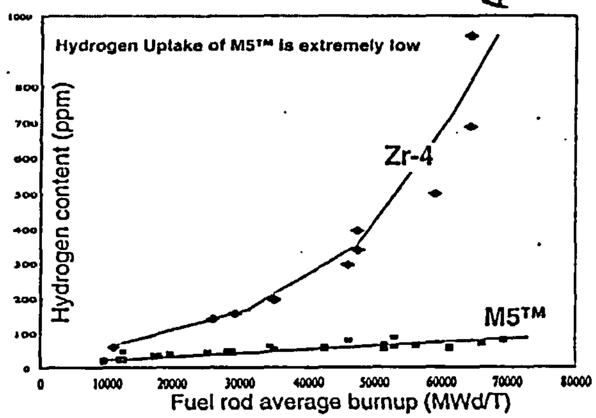


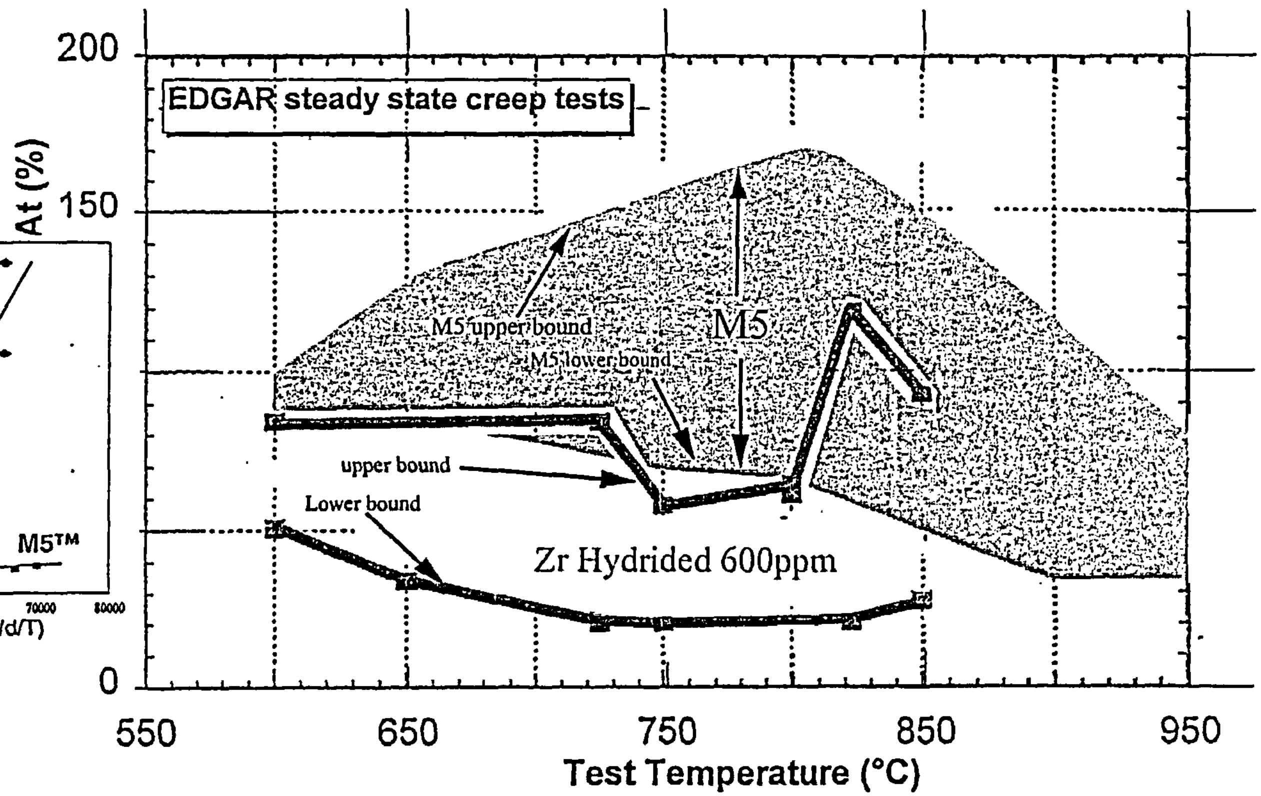
Pending Issues : **BUNDLE BLOCKAGE GEOMETRY**

Corrosion and hydrogen uptakes are lowered for modern clad alloys : ductility is better kept. Lower the corrosion, lower the associated H uptake, better the ductility, bigger the balloons

higher blockage ratio will be likely for modern alloys

▪See references 7 to 9





Creep tests not relevant to determine the balloon sizes

- **Isothermal creep tests** are relevant to determine the cladding creep laws NOT the cladding strains at failure under LOCA conditions
- The international community is using the **thermal ramp test** results to assess the cladding strains at failure under LOCA conditions

LOCA design approach

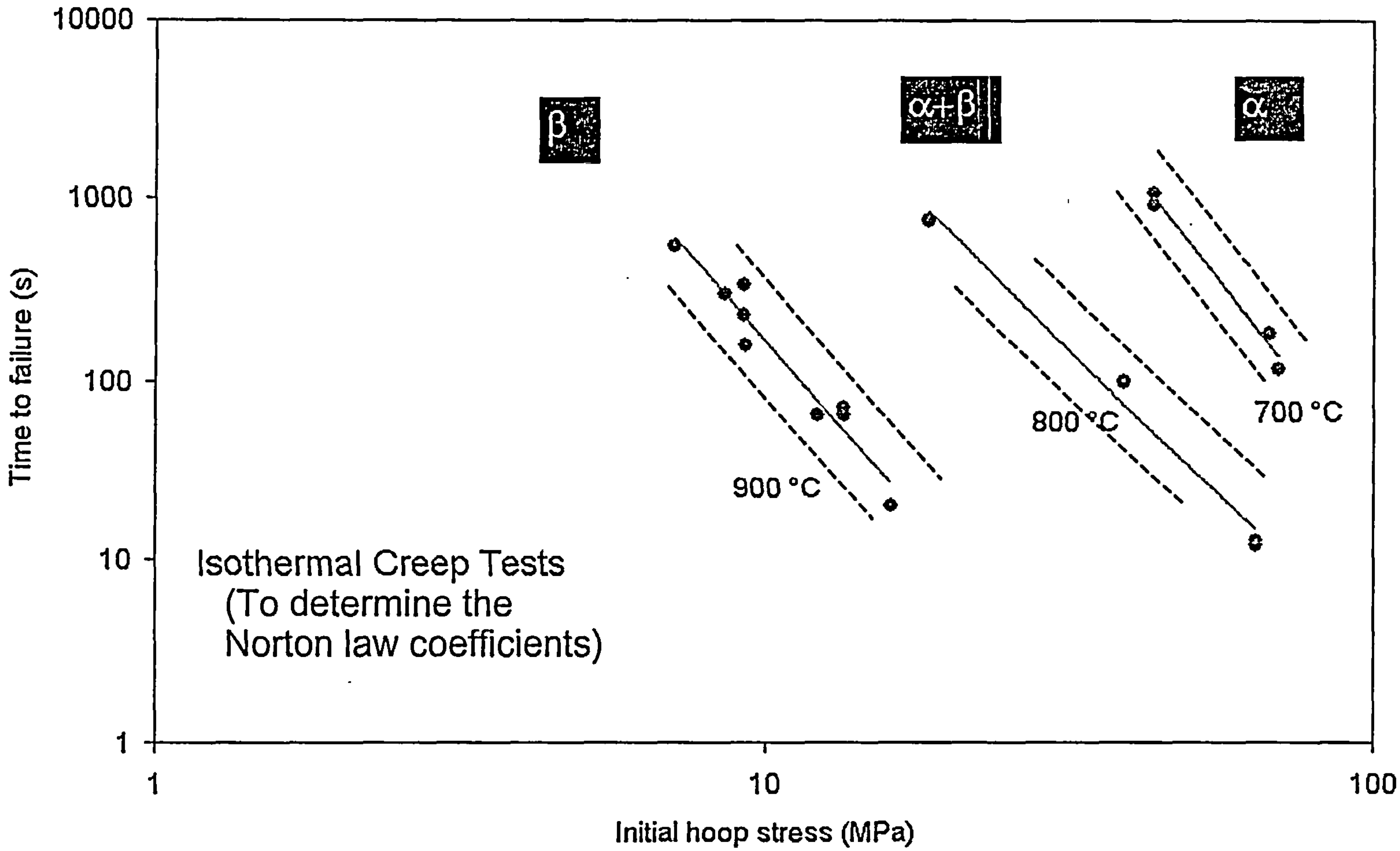
- To simulate and analyze the clad ballooning behavior during the blowdown phase of a LOCA transient one needs :
  - To know (at each time of the transient) the metallurgical state of the cladding (α and β phases distribution),
  - To identify the creep laws for the 3 temperature domains : α, $\alpha+\beta$ et β,
  - To characterize the cladding strains at failure

- *The approach is global : the impact of one parameter has to be investigated throughout the entire process.*

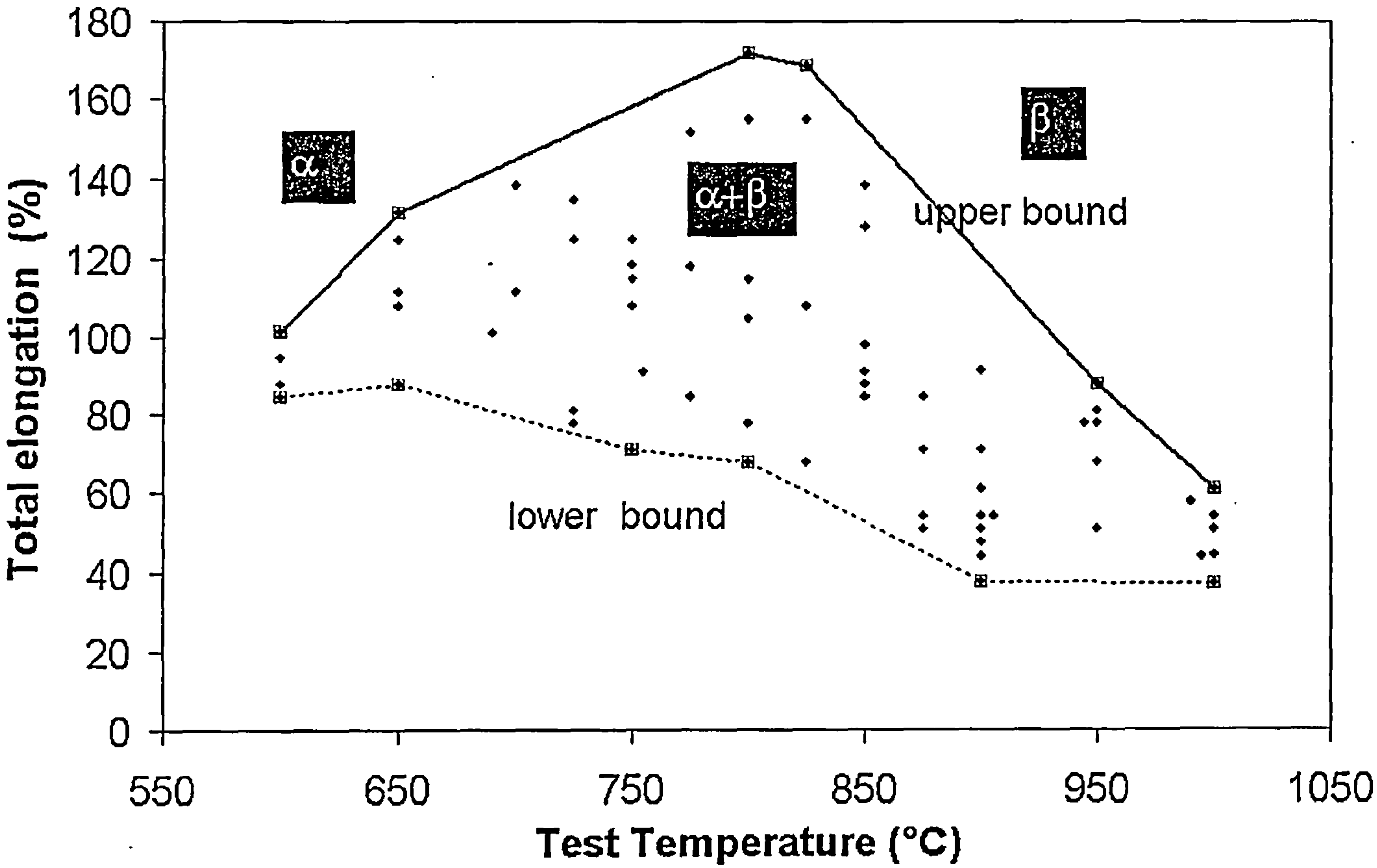
Separate effect tests

- Separate effect tests were performed on as received and pre-hydrided conditions to provide the requested data :
  - Metallurgical tests to determine the transition temperatures and the phase transformation kinetics
    - » At equilibrium and in heating and cooling conditions
  - Isothermal creep tests (rod internal pressure is maintained constant) to determine the mechanical laws
    - » Isothermal conditions
  - Thermal Ramp tests to determine the strains at failure
    - » Prototypical of a LOCA thermal transient

Creep tests on as-received M5 : Time to failure (t_R) vs σ_0

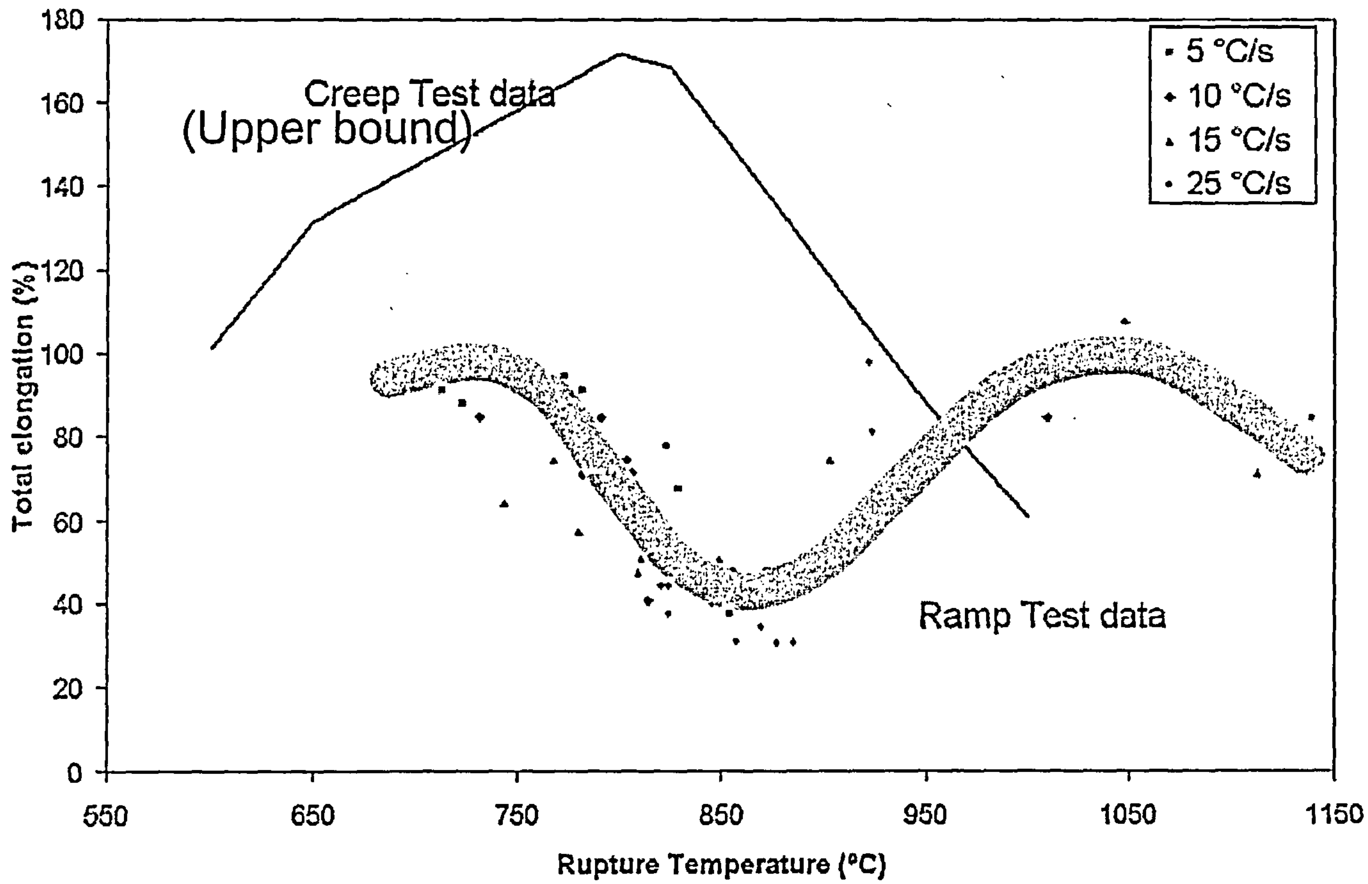


Creep tests on as-received M5: ductility



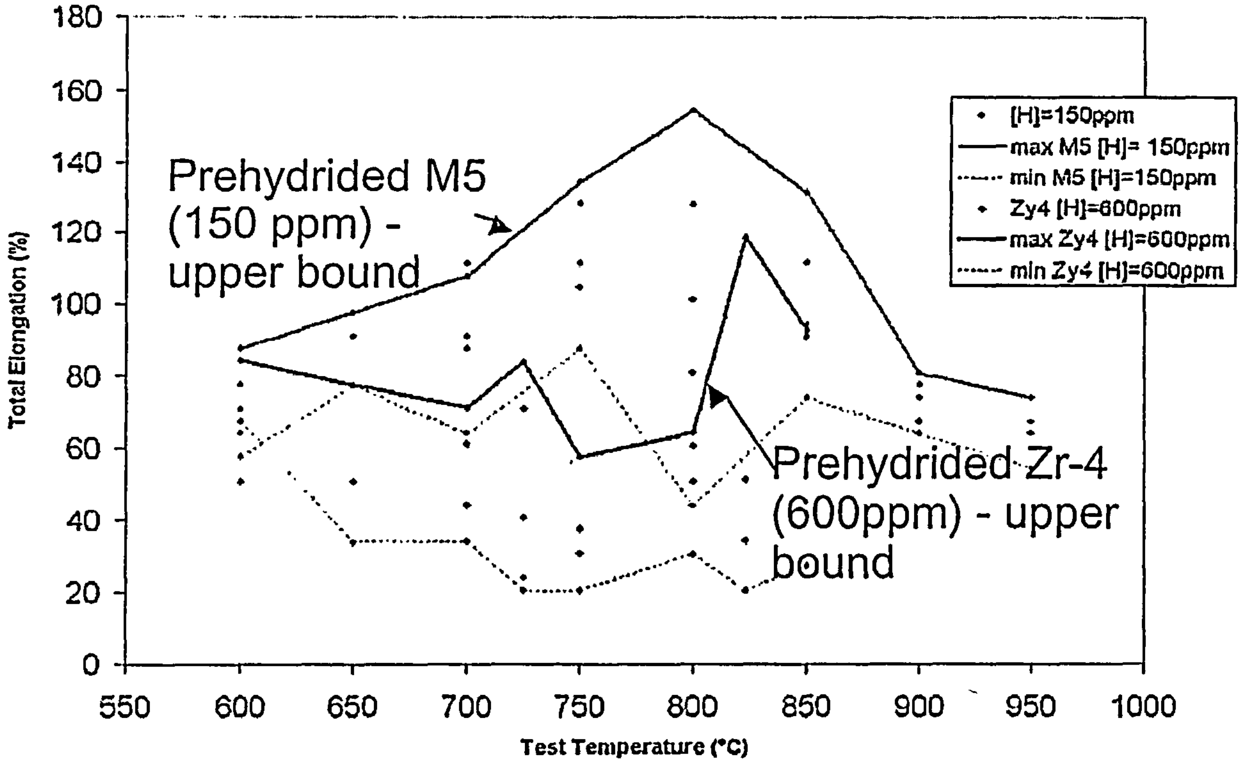
Temperature ramps on as- received M5 : ductility

Ductility decreases in the ramp test



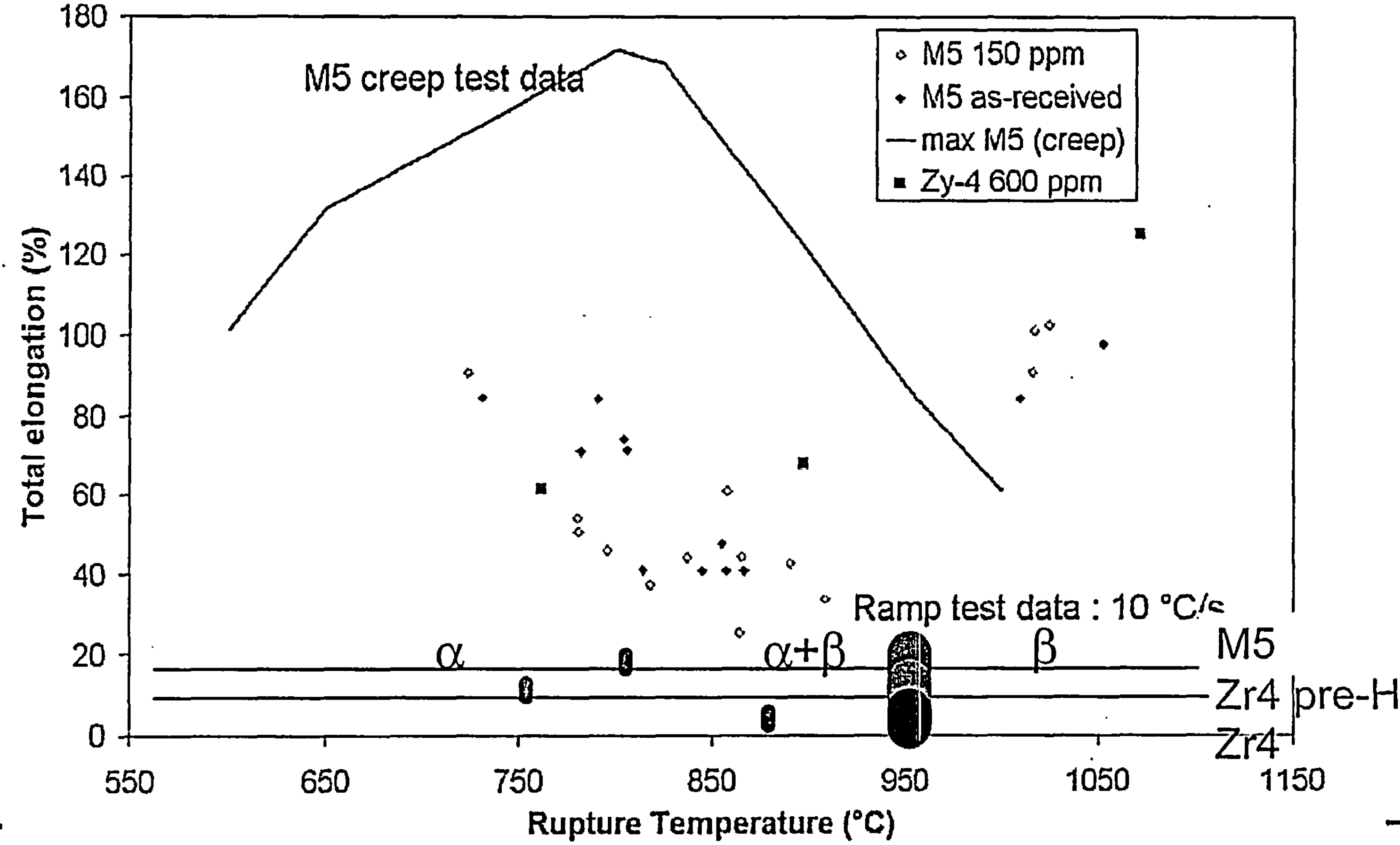
M5 (150 ppm) versus Zy-4 (600 ppm)

- Although Zy-4 600 ppm exhibits a lower ductility than M5 150 ppm in the **creep** tests.....



M5 150 ppm versus Zy-4 600 ppm

……both alloys exhibit similar ductility in the **ramp tests**



Conclusion

- Creep tests are not relevant to conclude on ballooning under LOCA conditions

- As-received and pre-hydrided M5 and prehydrided Zy4 exhibit similar behavior in term of strains at failure (according to the available data : the data base has to be completed)
  - Similar maximum strains at failure for M5 and Zy4
  - α and β humps slightly shifted for M5 : same transient may give different strains for M5 and Zy4, so that no significant change is expected in the design analysis conclusions

February 23, 2004

MEMORANDUM TO: Suzanne C. Black, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

FROM: Farouk Eltawila, Director *RA/*
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

SUBJECT: RESPONSE TO USER NEED FOR DEVELOPMENT OF RADIOLOGICAL SOURCE TERMS FOR REVIEW OF MIXED OXIDE FUEL LEAD TEST ASSEMBLIES

By memorandum dated November 27, 2003, your office requested[1] research support of NRR reviews of license amendments for the use of mixed-oxide (MOX) fuel. That memorandum supplemented previous requests from your office on the same issue[2,3].

We are aware that Duke Energy Corporation has submitted a license amendment[4] to allow the use of four lead test assemblies at Catawba, and that the licensee has requested a decision on its application by August 2004. We are also aware of NRR's intent to issue a safety evaluation of the lead test assembly amendment in March 2004. Consistent with these requirements, our schedule for responding to your request was provided by memorandum dated January 7, 2004[5]. This memorandum transmits the short-term items you requested to facilitate the lead test assembly review process. These are:

(1) For non-LOCA design basis accidents, affirm that the current regulatory positions on gap fractions in Regulatory Guide 1.183[6], Section 3.2 and Table 3, are conservative for use with weapons-grade MOX fuel such as that proposed by Duke.

(2) For fuel handling accidents, affirm that current regulatory assumptions on iodine decontamination in Regulatory Guide 1.183 are conservative for weapons-grade MOX Lead Test Assemblies. The objective of this assessment is to affirm that the currently allowable spent fuel pool rod pressure of 1300 psig is conservative for use with weapons-grade MOX fuel.

Results of our analyses, which have been discussed with your staff, are provided in Attachment A of this memorandum. The analytical models, used to account for the effects of plutonium, are described in Attachment B. The mixed-oxide model information, along with the experimental data upon which the models are based, were previously provided to your office.

Our analyses support the contention that, for nominal conditions, the requirements of Regulatory Guide 1.183 on gap inventory and rod pressure are met. Compliance under more severe conditions, which account for uncertainties, are described in Attachment A.

Our analyses are based on calculations performed under an Office of Nuclear Regulatory Research contract with Pacific Northwest National Laboratory (PNNL). For the MOX LTA review activity, this arrangement has been particularly useful for several reasons:

- PNNL provided technical support directly to NRR during the original formulation of Regulatory Guide 1.183.

- Calculating the gap inventory of radionuclides identified in Regulatory Guide 1.183 requires a departure from a more routine calculation of stable noble gases in the gap. To satisfy this requirement, PNNL proposed an analyses using two fission gas release models: Massih and ANS-5.4. This dual approach is described in Appendix A and code changes necessary to accomplish this are documented in Attachment C.

- Through a long-standing program sponsored by Office of Nuclear Regulatory Research, PNNL has been an active participant in the ANS-5.4 Working Group activities on radioactive fission gas release. Consequently, the results provided in this memorandum are particularly well-qualified.

The schedule requested by your office for delivery of these analyses has been very aggressive. However, the cooperation and technical support provided by your staff have enabled the work to be accomplished in a timely manner. We appreciate your efforts.

Should you require additional information regarding this transmittal, please contact John Voglewede of my staff on 415-7415.

Attachments: As stated

REFERENCES

1. Memorandum from James Dyer (NRR) to Ashok Thadani (RES) on *User Need Request for Development of Radiological Source Terms for Design Basis Accident Analyses in Support of Reviewing Amendments Associated with Mixed-Oxide Fuel and High Burnup Low Enrichment Uranium Fuel* dated November 27, 2003

2. Memorandum from Sam Collins (NRR) to Ashok Thadani (RES) on *Research User Need for Development of Multiple Issues to Prepare for Reviewing Amendments Associated with Mixed-Oxide Fuel* dated November 5, 1999

3. Memorandum from Sam Collins (NRR) to Ashok Thadani (RES) on *Update of Active NRR Requests for Assistance* (pp. 16-17) dated January 31, 2002

4. Letter from Duke Energy to USNRC on *Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50* dated February 27, 2003

5. Memorandum from Ashok Thadani (RES) to James Dyer (NRR) on *Response to User Need Request for Development of Radiological Source Terms for Design Basis Accident Analyses in Support of Reviewing Amendments Associated with Mixed Oxide Fuel and High Burnup Low Enrichment Uranium Fuel* dated January 7, 2004.

6. *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, U.S. Nuclear Regulatory Commission Regulatory Guide 1.183, July 2000.

ATTACHMENT A
FRAPCON3.2 Input and Release Calculations for the Duke MOX Lead Test Assemblies

Duke Power Inc. (Duke) has submitted a license request to the Nuclear Regulatory Commission (NRC) to operate 4 mixed oxide (MOX) lead test fuel assemblies (LTAs) in the Catawba PWR, in support of the joint U.S./Russia program for disposition of weapons-grade plutonium. The predicted at-shutdown bounding radioactive gas inventory in the fuel rods available for release in a spent fuel handling accident (often referred to as the "gap inventory" of radioactive isotopes) is one of the items involved in the review of this request.

This analysis has two goals:

1) For design basis accidents that do not involve core-wide fuel damage, affirm that the current regulatory positions on gap fractions in Regulatory Guide 1.183 (Ref. 1), Section 3.2 and Table 2 (reproduced as Table 1 below) are conservative for use with the weapons-grade MOX fuel such as that proposed by Duke Power.

2) In addition to the gap fractions requested above, assess the gap inventory of fission gases in the context of rod pressure of fuel stored in the spent fuel pool. Duke has stated that the pressure would be less than the currently allowable spent fuel pool rod pressure of 1300 psig. The objective of this assessment is to affirm that the current regulatory assumption regarding rod pressure of fuel stored in the spent fuel pool is conservative for use with weapons-grade MOX fuel.

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Nobel Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

Table 1: Non-LOCA Fraction of Fission Product Inventory in Gap From Regulatory Guide 1.183

It should be noted from the above table that the iodine isotopes are the only halogens and the cesium isotopes are the only alkali metals that are calculated using the ANS5.4 analysis methodology. The assignment of elements to the halogen, noble gases, and alkali metal groups, is based on similarity in chemical behavior. The elements considered in this evaluation are appropriate surrogates for the remaining elements in the groups. These later elements are not important contributors to design basis accident radiological consequences.

Code Input

In order to make this assessment, Duke was requested to provide details on MOX rod design and fabrication, and a bounding or peak-rod power history. The geometry of the rod, pellet fabrication, and reactor conditions were provided in Table 1 of the Duke Response to NRC Staff Request for Additional Information dated November 21, 2003 (Ref. 2). The following are the assumptions or changes from this table, which were made in the process of constructing FRAPCON-3.2 code (Ref. 3) input for gas release assessment.

- FRAPCON-3.2 does not consider the isotope, Pu-238, so the small fraction of the fuel that was identified as Pu-238 was modeled as Pu-242.
- The coolant flow rate given was calculated at inlet conditions for the total core cross section area. To reflect the higher power in the LTA, a higher value of 3.0×10^6 lb/ft^2-hr was used.
- FRAPCON does not have the material properties to model the zirconium-niobium cladding alloy called M5. These analyses were made using the similar material properties of the zirconium-tin cladding alloy called Zircaloy-4.
- FRAPCON does not allow the modeling of chamfered pellets. In order to give the correct volume reduction due to the dish and chamfer, the dish dimensions were increased slightly.
- The run was made using 12 axial nodes, 17 radial fuel nodes, and 45 radial fuel nodes for the gas release routine. These parameters have been found to give reasonably accurate calculations for a 12 foot rod.

The above assumptions are reasonable and should not have a significant impact on the results of the release calculations presented in this report.

Reference 2 provides a table containing effective full power days (EFPD) for each cycle, the MOX fuel lead assembly peak rod exposure at each time step and the radial peaking factor (Fdelta-H). From the values of time and burnup, and the pellet diameter and density, the linear heat generation rate (LHGR) can be calculated for each time step. The power history can also be derived from the Fdelta-H and core average LHGR values for this plant. As might be expected, power histories derived from the peak exposure and time values are very close to the Duke estimated peak exposure of 56.7 GWD/MTM, however, using the power history from the Fdelta-H values and core average power the peak exposure is calculated to be 59.81 GWD/MTM and does not match the peak pin exposure given by Duke of 56.7 GWD/MTM. The two different power histories are shown in Figure 1.

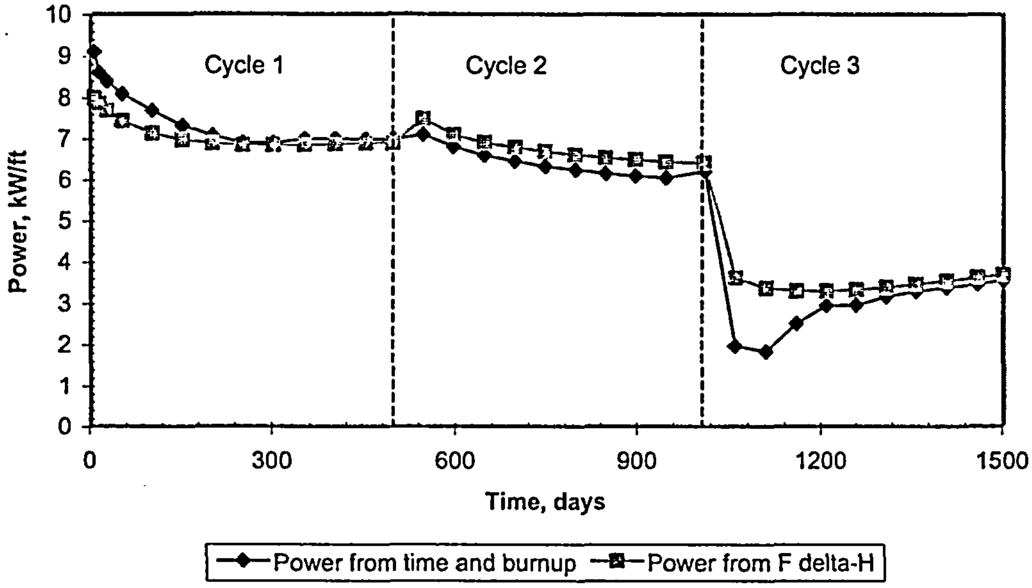


Figure 1: MOX LTA Power histories derived from time and burnup while the other is derived from radial peaking factor and core average power all supplied by Duke

The power history used for this analysis was the one derived from burnup and time; however, some release results are given for the other power history to show that it provides slightly higher release values. The difference between these two power histories can be accommodated by the power uncertainties described later in this report. Time and burnup from Reference 2 provided a power history consisting of 45 pairs of time and power that were used as input for FRAPCON-3.2. The power histories provided in Reference 2 were characterized by Duke as being conservative and bounding for the expected power histories to which the MOX LTAs may be irradiated. Reference 2 also provided 18 axial power profiles and the times that each should be used during the irradiation life of the LTA's. These profiles were input directly into FRAPCON along with the time steps each should be used.

Fission Gas Release Models

The FRAPCON-3.2 code contains two fission gas release models. The first is the ANS5.4 model. The ANS5.4 model can predict both the stable noble gas isotopes, and the radioactive isotopes based on their half-lives. The ANS5.4 model was developed with a thermal conductivity model that did not account for conductivity degradation due to burnup and also did not account for the radial power peaking at the pellet edge due to epithermal neutron buildup of plutonium while the FRAPCON-3.2 code accounts for these effects. The ANS5.4 gas release model has not been calibrated with the FRAPCON-3.2 code and, therefore, the absolute release values calculated with this model in FRAPCON-3.2 will significantly over predict fission product release fractions because the diffusion coefficients contain the effects of fuel thermal conductivity burnup degradation and radial power peaking, which are already modeled in FRAPCON-3.2. This problem can be accommodated by considering the ANS5.4 model in context of the second fission gas release model in FRAPCON-3.2.

The second fission gas release model in FRAPCON-3.2 is the Massih model, which has been calibrated with the new thermal conductivity and radial power models. Its predictions have been verified against a large number of fission gas release data from UO_2 fuel rods and a much smaller database from MOX fuel rods (Refs. 4 and 5, respectively). However, the Massih model is only capable of predicting the stable noble gas isotopes and not the radioactive isotopes. Therefore, the ANS5.4 model is needed to calculate the release values for the radioactive isotopes based on the Massih predictions for the stable isotopes.

Calculational Results

In order to model the release of radioactive products, the MOX LTA calculations were initially run with FRAPCON 3.2 using the Massih gas release model to determine the stable fission gas release. The Massih model has been shown to give good gas release predictions for the stable release for both UO_2 and MOX fuel rods. The FRAPCON-3.2 code was then run with the ANS5.4 model using the power history supplied by Duke but reduced by an amount such that the gas release value predicted by ANS5.4 would match the end of life gas release value predicted by Massih using the Duke power history. A second case was also run where the power history was reduced by a factor such that the peak gas release predicted by ANS5.4 would match the peak gas release predicted by Massih using FRAPCON-3.2 and the power history supplied by Duke. This approach normalizes the effective diffusion coefficients for the ANS 5.4 model such that the ANS5.4 model will now correctly calculate the same stable release fractions for the noble gases as are calculated with the Massih model. Figure 2 shows the gas release predictions for the Massih and ANS5.4 models and the ANS5.4 model with the effective diffusion constants modified to match the EOL and peak Massih stable FGR.

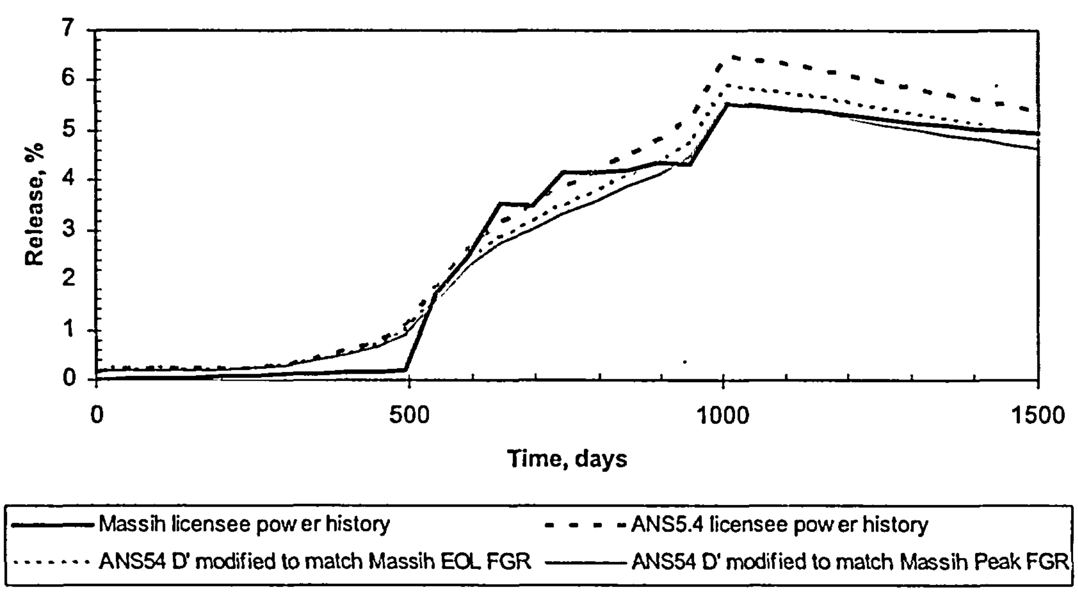


Figure 2: Gas release predictions using the Massih and ANS5.4 models and the ANS5.4 model with the effective diffusion constants modified to match the EOL and peak Massih FGR

Using these modified power histories (resulting in normalized effective diffusion coefficients) the ANS5.4 model was used to predict the release of the radioactive isotopes. For the calculation of I-131, a diffusion constant 7 times that of the noble gases was used and for cesium the diffusion constant was assumed to be 2 times that of the noble gases as recommended by ANS5.4 (Ref. 6). The ANS 5.4 standard also recommends that the release fractions of the long-lived isotopes of Kr-85, Cs-134 and Cs-137 isotopes (half-life of 10.72 years, 2.06 years, and 30.17 years, respectively) be conservatively calculated assuming that they are stable. So the Kr-85 release fraction was calculated using the stable gas routine in ANS5.4 and the Cs-134 and Cs-137 were calculated using the stable gas routine with twice the diffusion constant for noble gases. The shorter lived radioactive isotopes; Kr-87, Kr-88, Xe-133, Xe-135, and I-131 were calculated using the radioactive gas routine in ANS5.4 with 7 times the diffusion constant of noble gases for the I-131. It should be noted that FRAPCON-3.2 using the Massih model will predict best-estimate release fractions for the noble gases when best-estimate input is used, e.g. using best-estimate power histories.

The ANS5.4 standard recommends using time steps of at least two half-lives in the calculation of radioactive release during time-varying power histories. Our review concluded that, for I-131, time steps of 40 days or greater ensure high numerical accuracy for the calculation. In order to accomplish this, the original power history of 45 points was reduced to 33 steps where each time step was greater than 40 days. This power history as determined from burnup and time can be seen in Figure 1. In doing this, several of the axial power profiles that were very similar were combined into a single shape. The FRAPCON predictions using the large time steps were very similar to the predictions using the original 45 time steps.

Table 2 shows the nominal release results at the end of each cycle for the ANS5.4 calculations using the nominal power history modified to match the end of life FGR prediction by Massih. These same results are also shown in Figure 3. Since the release of the isotopes, Kr-87 and Kr-88 are very small; they are not included in the plot in Figure 3.

Table 2: Nominal release results from ANS5.4 with nominal power history modified to match Massih EOL FGR prediction

	Stable Xe & Kr and Kr-85 %	Cs-134 & Cs-137 %	Kr-87 %	Kr-88 %	Xe-133 %	Xe-135 %	I-131 %
End of Cycle 1	1.0	1.4	0.0	0.0	0.3	0.1	0.8
End of Cycle 2	5.9	8.0	0.1	0.2	1.2	0.3	3.8
End of Cycle 3	4.9	6.6	0.0	0.0	0.0	0.0	0.1

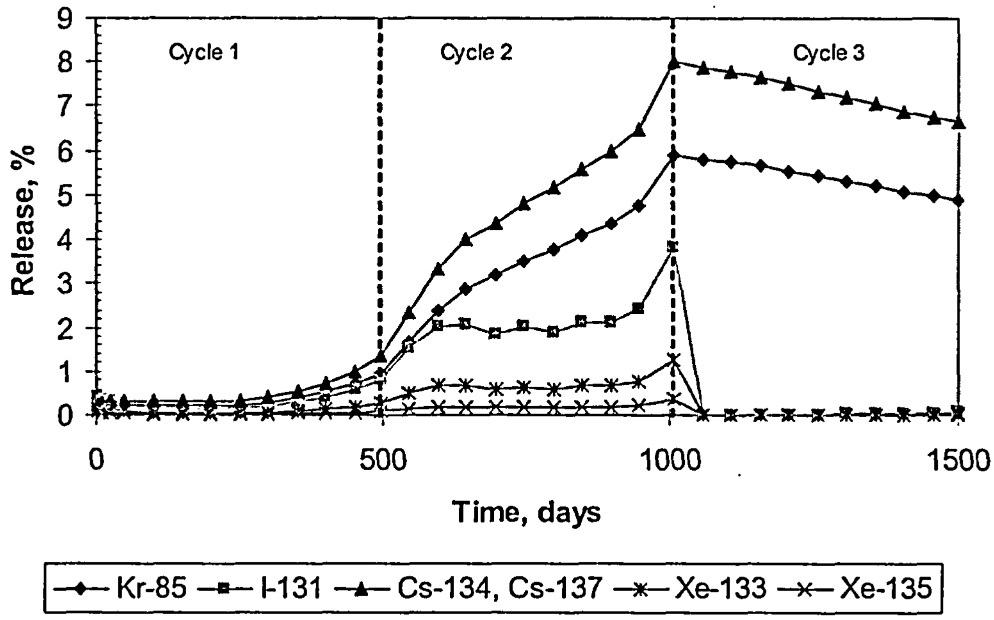


Figure 3: Nominal release results from ANS5.4 with the nominal power history modified to match Massih EOL FGR prediction

Table 3 shows the nominal release results at the end of each cycle for the ANS5.4 calculations using the power history modified to match the peak FGR prediction by Massih which occurred at the end of the second cycle. These same results are also shown in Figure 4. Since the release of the isotopes, Kr-87 and Kr-88 are very small; they are not included in the plot in Figure 4.

Table 3: Nominal release results from ANS5.4 with the nominal power history modified to match Massih peak FGR prediction

	Stable Xe & Kr and Kr-85 %	Cs-134 & Cs-137 %	Kr-87 %	Kr-88 %	Xe-133 %	Xe-135 %	I-131 %
End of Cycle 1	0.9	1.3	0.0	0.0	0.2	0.1	0.8
End of Cycle 2	5.6	7.6	0.1	0.2	1.2	0.3	3.6
End of Cycle 3	4.6	6.3	0.0	0.0	0.0	0.0	0.1

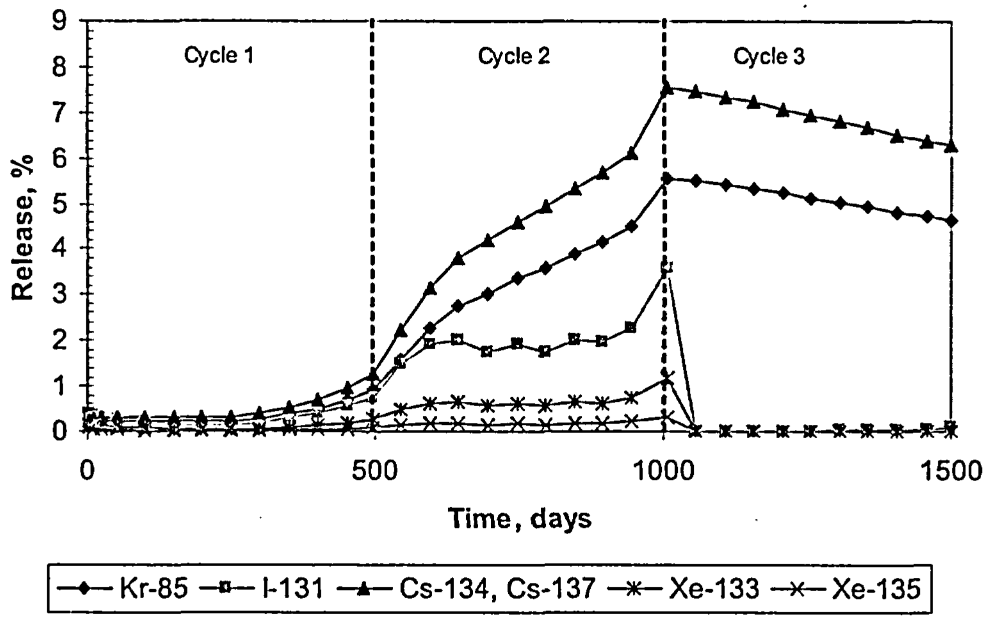


Figure 4: Nominal release results from ANS5.4 with power history modified to match Massih peak FGR prediction

The NRC-NRR request was for releases at rod-average burnups of 25, 45, 50 and 62 GWD/MTM. However, examination of Tables 2 and 3 and also Figures 3 and 4 demonstrate that all the radioactive isotope peak release values occur at EOC2 (provided in Table 3 and Figure 4). These nominal peak release values calculated with the ANS5.4 model at EOC2 are bounded by the Regulatory Guide 1.183 recommended release values for these isotopes.

<u>Limitations of Calculations</u>

The FRAPCON3.2 code has been verified against a wide variety of fuel conditions in terms of rod powers, burnup levels, fission gas release and fuel enrichment. Another important parameter for MOX fuel is stoichiometry, i.e., oxygen-to-metal ratio (O/M). For UO_2 fuel rods the code has been verified for rod powers up to 18 kW/ft, maximum fuel burnups of 65 GWd/MTU, fission gas release up to 33%, and U-235 enrichments up to 13%. For MOX fuel rods the code has been verified for fuel rod-average powers up to 9 kW/ft (steady-state) and ramped rod-average powers up to 10.7 kW/ft, maximum fuel burnups of 60 GWD/MTM, fission gas release up to 25%, percent plutonium up to 6%, and O/M ratios between 1.98 to 2.0. The fuel parameter that has the largest impact on fission product release calculations is the rod power. In relation to this limitation the calculations presented in this report do not include the impact of transients. An example of how fission gas release changes with rod powers will be provided in the next section along with the code calculational uncertainties in fission product release.

Power History and Calculational Uncertainties

The uncertainty in the FRAPCON-3.2 predicted release fractions for the radioactives can be estimated based on the standard deviation of the FRAPCON-3.2 predictions of stable noble gases to measured data from UO_2 and MOX fuel. The standard deviation for UO_2 fuel stable gas predictions of 0.026 absolute release fraction and a standard deviation for MOX of gas predictions of 0.048 absolute release fraction. The actual uncertainty in the MOX predictions is most likely much closer to those for UO_2 fuel but the calculated value is much higher because there are only 6 MOX data points making the uncertainty high because of the small MOX database. In actuality the mechanisms for release from UO_2 and MOX fuel is the same with the primary difference being in the diffusion coefficients for MOX versus UO_2 fuel such that uncertainties should be similar between these two fuel types. Therefore, the UO_2 and MOX predictions compared to data were combined to give an overall standard deviation of 0.031 for absolute release fraction.

The release fractions provided above are nominal release values based upon the power histories derived from Duke provided time and burnup values are reasonably close to nominal for the peak rod in the LTAs. In order to bound possible differences between the expected power history and the actual irradiation of the MOX LTAs, the power was increased by 5%, and the radioactive release values were calculated again with the effective diffusion constant in ANS5.4 modified to match the end of life gas release from Massih. The effective diffusion constant in ANS5.4 was also modified to match the peak gas release from Massih. Table 4 shows the results of these calculations. These results with a 5% conservatism on rod power (calculated from the Duke provided values of time and burnup multiplied by 1.05) show that the peak release values calculated with ANS5.4 model that occur at EOC2 are bounded by the Regulatory Guide 1.183 recommended release values for these isotopes. Table 4 also shows the release results in parenthesis (for the isotopes with release values > 1%) using rod powers (calculated from the Duke provided values of Fdelta-H and core average power multiplied by 1.05). These results with the Fdelta-h powers show that with a 5% increase in power the radioactive releases for Kr-85 and Cs isotopes are slightly higher than those recommended by Regulatory Guide 1.183, i.e., 10.6% versus 10% and 13.6% versus 12%, respectively.

Table 4: Release results from ANS5.4 with power history increased by 5% and the effective diffusion constant modified to match Massih EOL and peak FGR predictions

	Stable Xe & Kr and Kr-85 %	Cs-134 & Cs-137* %	Kr-87 %	Kr-88 %	Xe-133 %	Xe-135 %	I-131** %
Regulatory guide limits 1.183	10.0	12.0	5.0	5.0	5.0	5.0	8.0
ANS5.4 effective diffusion constant modified to match Massih EOL gas release							
End of Cycle 3	7.3 (8.7)	9.6 (11.2)	0.0	0.0	0.0	0.0	0.1 (0.2)
ANS5.4 effective diffusion constant modified to match Massih peak gas release							
End of Cycle 2	8.2 (10.6)	10.9 (13.6)	0.2	0.3	1.8 (2.2)	0.5	5.4 (6.6)
* Cesium is the only alkali metal calculated to be released with the ANS5.4 model ** Iodine is the only halogen calculated to be released with the ANS5.4 model							

Table 5 shows the radioactive fission product release values using the 5% greater power history (calculated from the Duke provided values of time and burnup multiplied by 1.05) plus a 1-sigma to account for code calculational uncertainties. The standard deviation for the stable noble gas release with both UO_2 and MOX data was 3.1%. The standard deviation for the radioactive isotopes was obtained by scaling the standard deviation for the stable release by the ratio of the predicted release of the radioactive isotope divided by the stable noble gas release value. Table 5 also shows the release results in parenthesis (for the isotopes with release values > 1%) using rod powers calculated from the Duke provided values of Fdelta-H and core average power multiplied by 1.05. These results show that with a 5% increase in power plus a 1-sigma due to code calculational uncertainties the radioactive releases for Kr-85 and Cs isotopes are higher than those recommended by Regulatory Guide 1.183.

Table 6 shows the radioactive fission product release values using the 5% greater power history (calculated from the Duke provided values of time and burnup multiplied by 1.05) plus a 2-sigma due to code calculational uncertainties. Table 6 also shows the release results in parenthesis (for the isotopes with release values > 1%) using rod powers calculated from the Duke provided values of Fdelta-H and core average power multiplied by 1.05. These results show that with a 5% increase in power plus a 2-sigma due to calculational uncertainties the radioactive releases for I-131, Kr-85 and Cs isotopes are higher than those recommended by Regulatory Guide 1.183. As noted previously the analyses presented in this report do not evaluate reactivity insertion accidents, this accident scenario will be addressed in a future report to NRC.

Table 5: Comparison of Regulatory Guide limits on Non-Loca release to those calculated release results for MOX LTAs using FRAPCON3.2/ANS5.4 with 5% increased power above Duke powers plus 1-sigma uncertainty on the calculation

	Stable Xe & Kr and Kr-85 %	Cs-134 & Cs-137* %	Kr-87 %	Kr-88 %	Xe-133 %	Xe-135 %	I-131** %
Regulatory guide limits 1.183	10.0	12.0	5.0	5.0	5.0	5.0	8.0
ANS5.4 effective diffusion constant modified to match Massih EOL gas release							
End of Cycle 3	10.4 (11.8)	13.7 (15.1)	0.0	0.0	0.1	0.0	0.2
ANS5.4 effective diffusion constant modified to match Massih peak gas release							
End of Cycle 2	11.3 (13.7)	15.0 (17.6)	0.3	0.4	2.5 (2.9)	0.7	7.5 (8.5)
* Cesium is the only alkali metal calculated to be released with the ANS5.4 model ** Iodine is the only halogen calculated to be released with the ANS5.4 model							

Table 6: Comparison of Regulatory Guide limits on Non-Loca release to those calculated release results for MOX LTAs using FRAPCON3.2/ANS5.4 with 5% increased power above Duke powers plus 2-sigma

	Stable Xe & Kr and Kr-85 %	Cs-134 & Cs-137* %	Kr-87 %	Kr-88 %	Xe-133 %	Xe-135 %	I-131** %
Regulatory guide limits 1.183	10.0	12.0	5.0	5.0	5.0	5.0	8.0
ANS5.4 effective diffusion constant modified to match Massih EOL gas release							
End of Cycle 3	13.5 (14.9)	17.7 (19.1)	0.0	0.0	0.1	0.0	0.2
ANS5.4 effective diffusion constant modified to match Massih peak gas release							
End of Cycle 2	14.4 (16.8)	19.1 (21.6)	0.3	0.5	3.2 (3.5)	0.9	9.5 (10.5)
* Cesium is the only alkali metal calculated to be released with the ANS5.4 model ** Iodine is the only halogen calculated to be released with the ANS5.4 model							

LTA Rod Pressures in a Spent Fuel Pool

Calculations have also been performed with FRAPCON3.2 to determine if the MOX LTAs, when stored in a spent fuel pool, will exceed 1300 psig. These calculations were performed for the four scenarios above, i.e., nominal power history provided by Duke, a 5% increase in rod powers from those provided by Duke, a 5% increase in rod powers plus a 1-sigma code calculational uncertainty, and a 5% increase in rod powers plus a 2-sigma code calculational uncertainty. The MOX LTA rod pressures were conservatively estimated (small conservatism) assuming a spent fuel temperature of 100°C at a pressure of 1atm. The results are shown in Table 7, demonstrating that even with the most conservative case of 5% power increase plus a 2-sigma uncertainty the rod pressures remain significantly below 1300 psig. Table 7 shows in parentheses the results of these calculations using the rod powers calculated from the values of Fdelta-H provided by Duke. This analysis has not assessed whether the decontamination factor of 200 is conservative at the rod pressures calculated in Table 7.

Table 7: MOX LTA calculated rod pressures in spent fuel pool

	Rod Pressure in Spent Fuel Pool psia	FGR %
Nominal Power History	621 (676)	4.9 (5.9)
Nominal Power History increased by 5%	730 (807)	7.3 (8.7)
Nominal Power History increased by 5% plus 1- sigma on FGR	873 (956)	10.4 (11.8)
Nominal Power History increased by 5% plus 2- sigma on FGR	1015 (1105)	13.5 (14.9)

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2. Canady, K.S., Letter dated December 10, 2003 to U.S. Nuclear Regulatory Commission containing Duke response to NRC Staff Request for Additional Information Dated November 21, 2003.
3. G.A.Berna, et al, *FRAPCON-3: A computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup,* NUREG/CR-6534 Vol.2 *Code Description* (PNNL-11513) 1997
4. G.A.Berna, et al, *FRAPCON-3: A computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup,* NUREG/CR-6534 Vol.3: *Integral Assessment* (PNNL-11513) 1997.
5. Lanning, D.D., et al., "FRAPCON-3 Code Updated With MOX Fuel Properties", to be presented at the *ANS/AESJ/ENS Topical Meeting on LWR Fuel Performance, Orlando Fla. September, 2004.*
6. Turner, S.E. et al, *Background and Derivation of ANS – 5.4 Standard Fission Product Release Model*, NUREG/CR-2507, 1982

ATTACHMENT B
MODIFICATIONS TO FRAPCON-3.2 FOR MOX FUEL

1.0 FUEL THERMAL CONDUCTIVITY

The major modification to FRAPCON-3.2 for application to MOX fuel was the addition of a fuel thermal conductivity model specific to MOX fuel. This was selected as a combination of the Duriez stoichiometry-dependent correlation, derived from diffusivity measurements on unirradiated fuel pellets (Reference 1), plus the burnup degradation contained in a modified version of the NFI fuel thermal conductivity model (Reference 2). The combined model was described in PNNL's paper for the Halden EHPG meeting in Storefjell, Norway (Reference 3). In that paper, code-data comparisons were made with the new model added, for three instrumented MOX fuel tests in Halden Reactor: IFA-629.1, IFA-610.2,4 and IFA-648.1. Since then, comparisons have also been made to IFA-629.3 (the ramp-test extension of IFA-648.1), and to IFA-606. All these tests and their reference documents are briefly summarized in Table 1.

Predicted-vs.-measured results for all the comparisons are shown in Figure 1. The normalized temperature differences (predicted-minus measured divided by measured minus coolant temperature) are shown as a function of LHGR in Figures 2 and 3. As can be seen, the predictions are very close to the data for IFA's 629.1, 606, and 610.2,4 (Figure 2). They deviate about 5% above the data for IFA 629.3 and 5% below the data for IFA-648.1 (Figure 3). Since the same rods and thermocouples are used in both tests, it may be that the LHGR associated with measured temperature may deviate from true values in one or both tests. Halden Project has been requested to investigate this possibility.

Overall, the addition of comparisons to the extensive raw data files from IFA's 648/629.3, and the digitized data from IFA 606, has extended the data base but provided no net incentive to change in the model from that presented at the Storefjell meeting.

2.0 FISSION GAS RELEASE

Design, operation, and FGR data provided by Halden has provided opportunity to compare code predictions to the steady-state FGR from three full-length MOX PWR rods (the 'mother rods" N06, N12, and P16 for instrumented sections tested in IFA's 610.2,4 and IFA-648.1/629.3). Comparison has also been made to end-of-ramp FGR for the power-ramp tested instrumented fuel rod sections in IFA's 629.1, 629.3, and 606. The results, with no modification to the FGR model, are shown in Figure 4. It is clear that FRAPCON-3.2 is generally under predicting the FGR for these 6 cases. Multiplying the diffusion constant by 1.75 raises the FGR to a closer overall comparison with this available data (see Figure 5), and has been incorporated for MOX into FRAPCON-3.2.

3.0 HELIUM PRODUCTION AND RELEASE

Puncture data and gas analysis was provided for two of the three mother rods, N12 and P16 (Reference 4). This permits evaluation of the change to rod helium inventory from beginning of life (BOL) to of life (EOL) The results indicate negligible change ~3% relative) in the helium inventory from beginning to end of life. These results are summarized in Table 2. This is consistent with current FRAPCON-3.2 predictions, and no change to FRAPCON-3.2 regarding helium release is recommended at this time. It should be noted that the initial fill gas pressure for these rods was relatively high at 363 psia, vs. a somewhat smaller value probable for MOX rods used in the U.S. for plutonium disposition. There is some evidence and theory that suggests higher fill gas pressure will reduce helium release.

4.0 ADJUSTMENTS FOR PLUTONIUM ISOTOPES

Input parameters have been added to signal when MOX fuel is being analyzed, and to initialize the concentrations of plutonium isotopes in the TUBRNP subcode, which calculates radial power and burnup profiles within the fuel pellets. Given this initialization, TUBRNP appears to calculate the radial profiles for LWR MOX fuel with acceptable accuracy. This was assessed by comparing code calculations to MCNP code calculations for radial power profiles (where the MCNP results were provided by ORNL). An example of this comparison is shown in Figure 7.

5.0 Xe/Kr RATIO

Fission gas is partitioned into krypton and xenon fractions within the code. Currently, the code uses Xe/Kr ratio of 5.67 in making this partition, which is appropriate for urania fuel. For MOX fuel, the majority of fissions occur in plutonium, and the xenon stable isotope yields are higher. Gas analysis data from MOX rod punctures at nominal to high burnup indicates Xe/Kr ratios of approximately 19 (Reference 4), however, Xe/Kr fission yields for plutonium indicate a value of 16 (see Reference 5 for example). The code has been altered to use the ratio of 16 when MOX fuel is being analyzed. The effects of this change are a small decrease in gas conductivity and a very small decrease in gap conductance for cases where fission gas concentration in the plenum gas becomes significant. However, the output gas species ratios now reflect a more realistic Xe/Kr ratio for MOX.

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3. Lanning, D.D., and C.E. Beyer. 2002. "Revised UO_2 Thermal Conductivity for NRC Fuel Performance Codes." *Transactions of the American Nuclear Society, 2002 Annual Meeting,* volume 86, pp 285-287. June 9-13, 2002, Hollywood, Florida.
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HPR 349/30 .. "The FIGARO Programme: The Behaviour of Irradiated MOX Fuel Tested in the IFA-606 Experiment, Description of Results and Comparison with COMETHE Calculation" L.Mertens, M. Lippens and J Alvis, March 1998.

HWR-586 "The Re-Irradiation of MIMAS MOX Fuel in IFA-629.1" Rodney J. White, March 1999.

HWR-603 "The Lift-Off Experiment with MOX Fuel Rod in IFA-610.2; Initial Results" Stephane Beguin, April 1999.

HWR-650 "The Lift-Off Experiments IFA 610.3 (UO_2) and IFA610.4 (MOX): Evaluation of In-Pile Measurement Data". Hajimi Fuji, Julien Claudel, February 2001.

HWR-651 "Results from the Burnup Accumulation Test with High Exposure (63 GWd/kg HM) MOX Fuel (IFA-648)". Julien Claudel, Francois Huet, February. 2001

HWR-664 "Summary of Pre-Irradiation Data on Fuel Segments Supplied by EdF/FRAMATOME and Tested in FA-610, 629 and 648". Masahiro Nishi, and Byung-Ho Lee, February, 2001.

HWR-714 "Ramp Tests with Two high Burnup MOX Fuel Rods in IFA-629.3". Benoit PetitPrez, June 2002

Table 1. Instrumented MOX Tests in Halden. All with re-fabricated PWR rod sections containing MIMAS MOX Fuel (Calculated FGR values are code predictions with diffusion constant multiplier = 1.0)

Reactor/Full Length Rod (Rod Diameter in mm)	Base Irradiation Cycles	Burnup, GWd/MTM (and FGR%) at end of base irradiation	Sponsor	Halden Test (IFA No.) and report (HWR No.)**	Test Type and Max. Rod-Average LHGR , kW/m	End of Test FGR % And Measurement Type
St. LaurentB1/J09 (9.35)	2	27 (low)	Halden Group	629.1 HWR-586	Ramp (35)	25% (Puncture) 26% PT[b] 17% calculated
Gravelines-4/N06 (9.35)	4	48 (4.12) (2.6% calculated)	Halden Group	610.2,4* HWR-603,650	Lift-off (10)	--
Gravelines-4/N12 (9.35)	4	50 (4.86) (3.0% calculated)	Halden Group	648* (629.3) HWR-651 (HWR-714)	SS (10) (Ramp, 25)	--
Gravelines-4/P16 (9.35)	4	47 (2.58) (1.7% calculated)	Halden Group	648* (629.3) HWR-651 (HWR-714)	SS (10) (Ramp, 25)	7% (PT) 2.3% calculated
Beznau-1 (10.7)	5	50 (low)	Belgo-Nucleaire[a] (FIGRARO)	606 (HPR-349/30)	Ramp (32)	13% (PT and puncture) 19% calculated.

*Note that IFA's 610.2,4 and IFA-648.1 operated in a PWR-condition loop within the HBWR, thus at a coolant temperature and pressure of 310 C and 2250 psia, instead of normal HBWR conditions (240 C, 500 psia)

**HWR-664 contains design, precharacterization, and base irradiation data for the St.Laurent and Gravelines EdF rods.

[a] Note this is proprietary data
[b] Pressure transducer

Table 2. Helium Results from Halden Test High-Burnup PWR MOX "Mother Rods"

Reactor/Full Length Rod (Rod Diameter in mm)	Base Irradiation Cycles	Burnup, GWd/MTM	BOL/EOL Helium inventory, STPcc
Gravelines-4/N12 (9.35)	4	50	449/454
Gravelines-4/P16 (9.35)	4	47	417/422

Figure 1. Predicted vs. Measured Fuel Center Temperatures for Halden MOX Tests

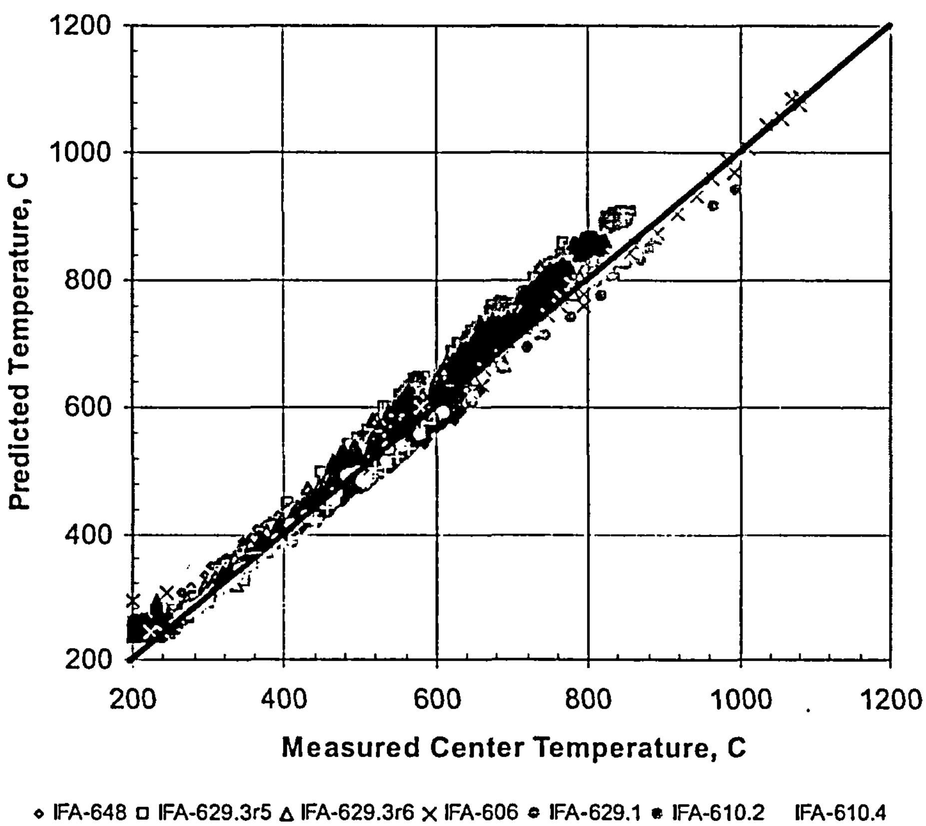


Figure 2. Normalized Predicted-minus-Measured Temperature Difference (IFA's 629.1, 610.2,4 and 606)

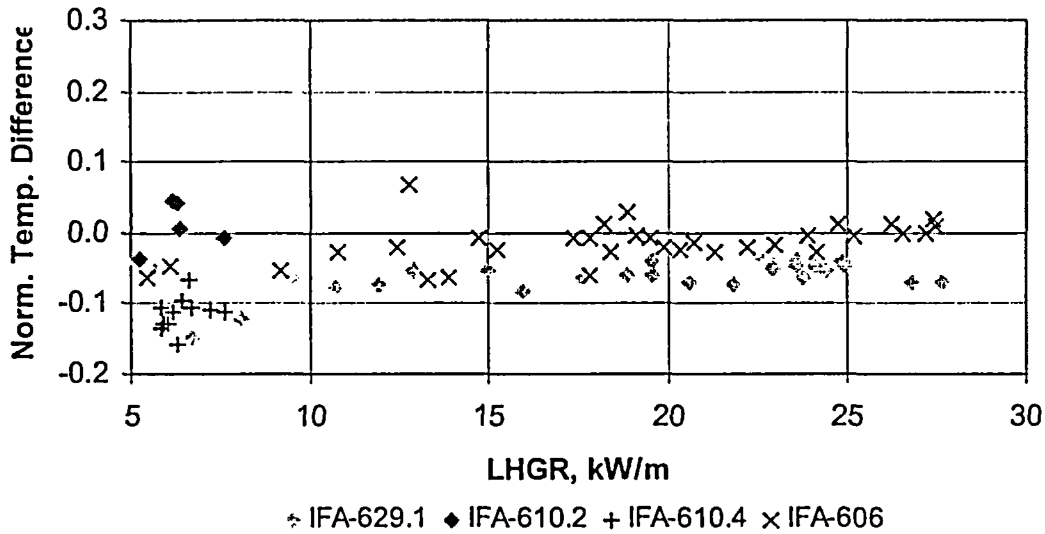


Figure 3. Normalized Predicted-minus-Measured Temperature Difference (IFA's 648.1 and 629.3 [same rods, 629.3 subsequent to 648.1])

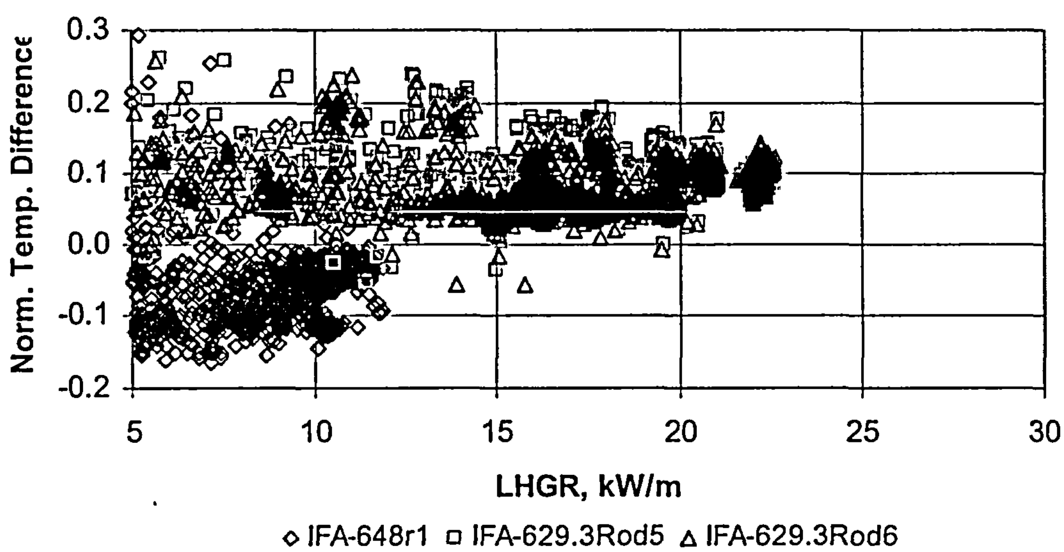


Figure 4. Predicted vs. Measured FGR, unmodified FGR model

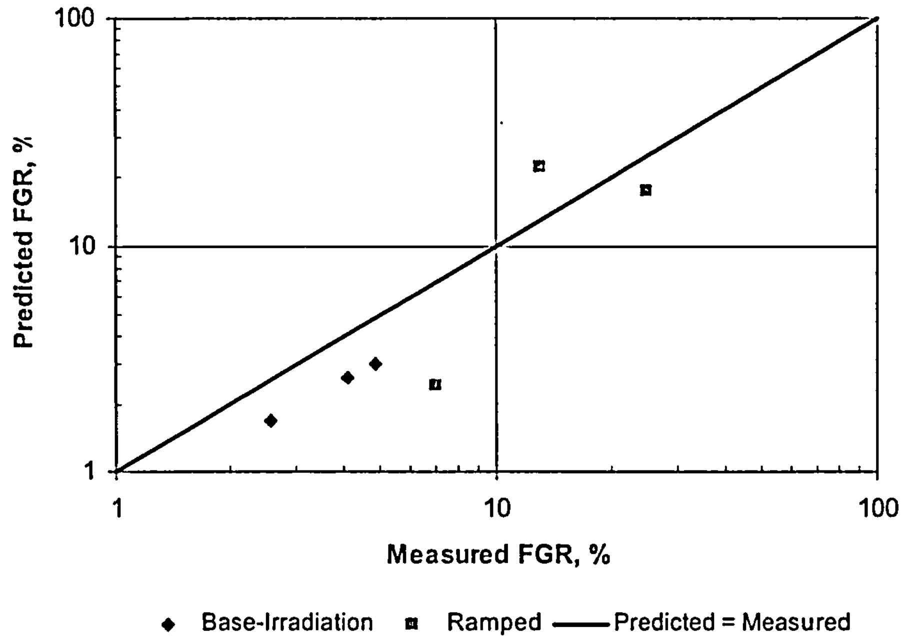


Figure 5 Predicted vs. Measured FGR, Diffusion constant x 1.75

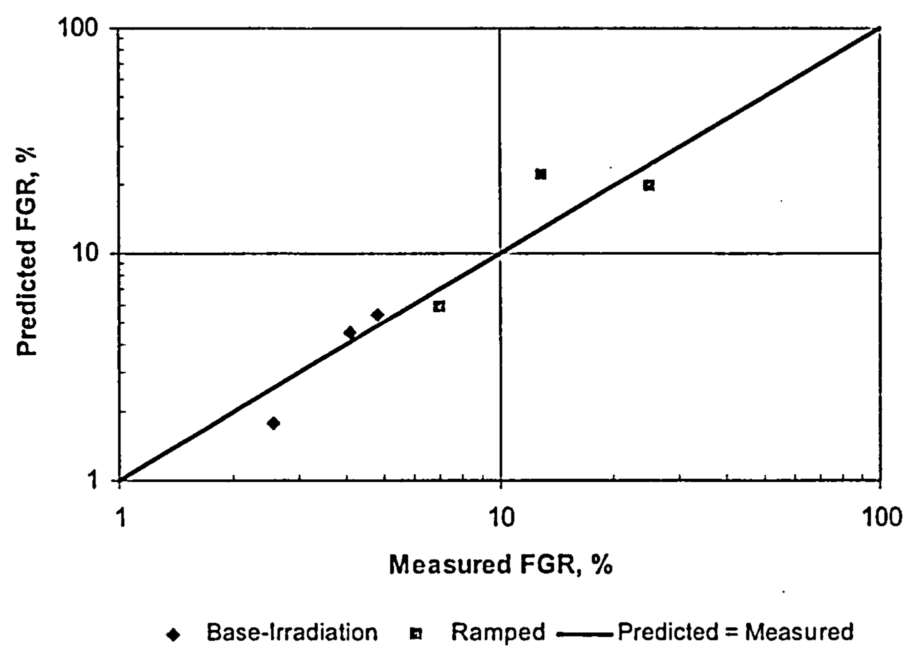
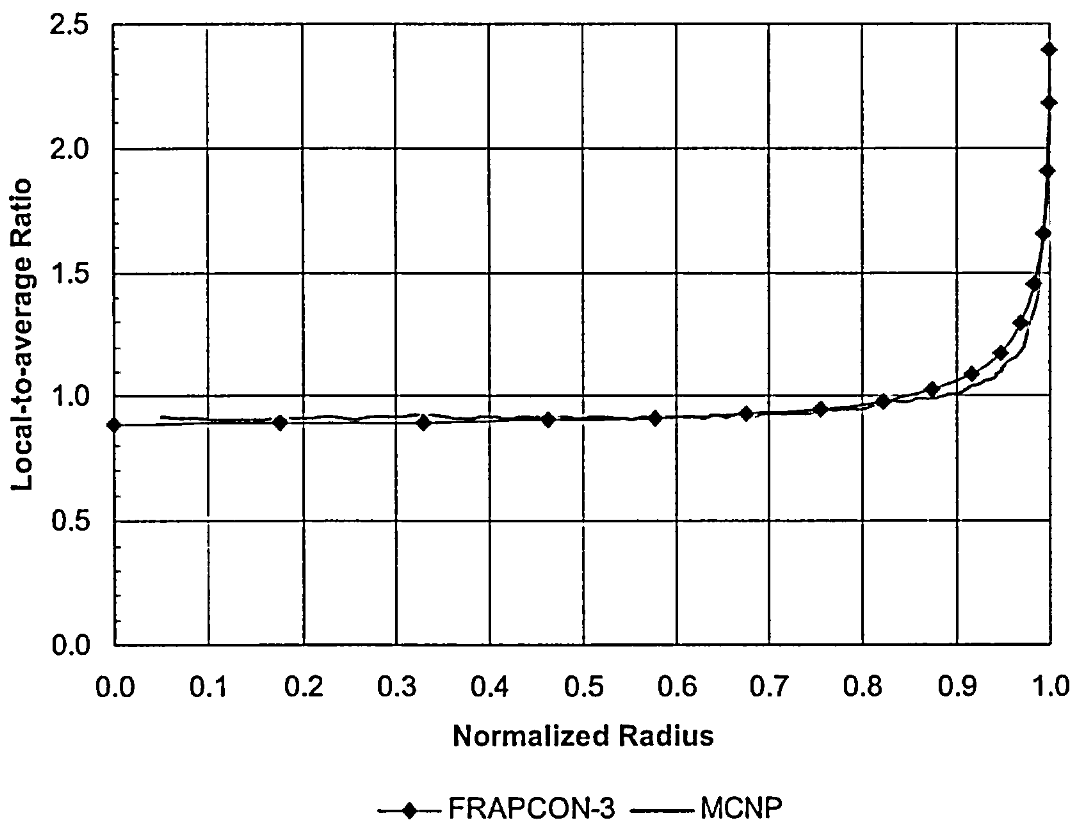


Figure 6

MCNP and FRAPCON-3 Calculations of normalized radial power profile in PWR MOX Fuel at 50 GWd/MTHM (Initial content = 5 wt.% WG Plutonia in MOX)



Attachment C

Changes made to FRAPCON to predict the radioactive fission product release from the Duke MOX lead test assembly

The following is a line by line listing of the changes made to FRAPCON in order to make the runs described in the above paper.

In the subroutine initial.f

Add the variable "ngasmod" to the namelist /frpcon/

After the line

```
igas=0
```

Add the line

```
ngasmod=1
```

In the common file comms.h

Add the variable "ngasmod" to the common /inpti/

In the subroutine frpcon.f

Add the variable "ngasmod" and "bu" to the call to fgasre

In the subroutine fgasre.f

Add the variables "ngasmod" and "bu" to the subroutine statement

Change the line

```
if(ngasr.gt.10) go to 211
```

To

```
if(ngasmod.eq.1) go to 211
```

Add the variables "gasflg" and "bu" to the call to ans54

After the call to ans54, add the lines,

```
      call ans54cs (it,brnup,dp,jpow,nt,im,rv,dv,nr,ProblemTime
    + , qaxnorm,rc,rdotcs
    + ,irl,qmpy,jst,releascs,den,dco,ngasr,crad,tfr,nr,pi,gasflg)
      if(gasflg.eq.1) then
            if(jpow.eq.1) then
                  csrel=0.0
                  bpsum=0.0
            endif
            csrel=csrel+bp*rdotcs
            bpsum=bpsum+bp
            if(jpow.eq.nt) then
                  csrel=csrel/bpsum
      write(9,209) ProblemTime(it)/86400.0, csrel*100.0
209   format(2x,'Time(days), Cs Release(%)',2x,f10.1,2x,f10.4)
            endif
      endif
```

In the subroutine ans54

Add the variables "gasflg" and "bu" to the subroutine statement

In the dimension statements add the variables "tempksub(50) and "busub(50)

In the dimension statements change

ansr(10)	to	ansr(50)
f(10)	to	f(50)
flxfac(10)	to	flxfac(50)
pf(10,21)	to	pf(50,21)

prdct(11,21,400)	to	prdct(50,21,400)
flxfacb(11)	to	flxfacb(50)
ansd(176000)	to	ansd(800000)

Change the line

```
      dimension decay(11), releas(11), half(11)
```

To

```
      dimension decay(5), releas(5), half(5), halflife(5)
      integer gasflg
c     Half-lives in seconds from Chart of Nuclides, 15th edition
c     Order is Kr-87, Kr-88, Xe-133, Xe-135, I-131
c
      data (halflife(i),i=1,5)/4572.0,10224.0,452995.0,
     & 32760.0, 692928.0/
```

After the line

```
      tempk=(tempf+459.65)/1.8
```

Add the line

```
      tempksub(i2)=tempk
```

After the line

```
      bup = burnup*prdct(i2,jpow,nt)
```

Add the line

```
      bupsub(i2)=bup
```

After the line

```
      210 continue
```

Add the line

```
      if(gasflg.eq.1) then
```

Change the lines

```
      do 250 i=1,11
      rtime = 1+(i-1)*deltim
      half(i) = 10.**rtime
      decay(i) = 1/half(i)
```

To

```
      do 250 i=1,5
      half(i)=halflife(i)
      decay(i)=0.6931472/half(i)
```

After the line

```
      j1 = i+nreg*(jpow-1)+npow*(nt-1)*nreg
```

Add the line

```
      ansdlock=ansd(j1)
```

Change the lines

```
      do 310 jx=1,11
      xmu = decay(jx)/ansd(j1)
      taut = ansdrad*ProblemTime(nt)
```

To

```
      do 310 jx=1,5
      ansdrad=ansdlock
c     D is 2 times higher for Cs and 7 times higher for I
c     NUREG/CR-2507
      if(jx.gt.4) ansdrad=ansdlock*7.0
      xmu = decay(jx)/ansdrad
      taut = ansdrad*dt(nt)
```

Change line

```
      frac=frac*wf
```

To

```
        fracl=frac*wf
```
Change line
```
        releas(jx) = releas(jx)+frac
```
To
```
        releas(jx) = releas(jx)+fracl
```
Change line
```
        do 330 i=1,11
```
To
```
        do 330 i=1,5
```
After the line
```
        330 continue
```
Add the lines
```
        timday=ProblemTime(nt)/86400.0
        if(iwrote.eq.0) write(7,1001)
        iwrote=1
1001  format(4x,'Time',7x,'Days',3x,'Rod Burnup',20x,'Gas Release',
      & 'Fractions',/,4x,'Step',15x,'(MWd/MTU)',5x,
      & 'Kr-87',6x,'Kr-88',5x,'Xe-133',5x,'Xe-135',6x,'I-131')
        write(7,1002) nt-1, timday, bu, (releas(ii),ii=1,5)
      1002  format(2x,i5,2x,2(f10.1,2x),8(e10.4,1x))
        endif
```
Add a new subroutine called ans54cs which is identical to ans54, except for the following changes

In the subroutine statement, the name should be changed from "ans54" to "ans54cs"

After the line
```
        ansd(j1) = ansd(j1)*100.**(bup/28000.)
```
add the line
```
        ansd(j1) = ansd(j1)*2.0
```
Remove everything between the lines
```
        210 continue
```
And
```
        return
```