March 17, 2005

Mr. Alexander Marion
Senior Director, Engineering
Nuclear Energy Institute
1776 I Street, NW, Suite 400
Washington, DC 20006-3708

SUBJECT: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT (EPRI) (TR) 1002865, "TOPICAL REPORT ON REACTIVITY INITIATED ACCIDENTS: BASES FOR RIA FUEL ROD FAILURES AND CORE COOLABILITY CRITERIA" (TAC NO. MB4921)

Dear Mr. Marion:

By letter dated June 12, 2002, the Nuclear Energy Institute submitted the subject TR for Nuclear Regulatory Commission (NRC) review and endorsement for application to advanced light-water reactor fuel designs intended for extended burnup use. The TR contains a set of revised acceptance criteria for reactivity initiated accidents (RIAs) and would be used by licensees to demonstrate that they meet the requirements of General Design Criterion 28. By letters dated November 12 and 25, 2002, EPRI submitted the proprietary computer program FALCON BETA-RIA, which was used in preparing the RIA criteria in TR 1002865.

The NRC staff has reviewed the TR and concludes that the proposed RIA criteria are not acceptable for endorsement. This conclusion is based on a number of technical issues that are discussed in the enclosed Safety Evaluation (SE). In addition, the NRC staff believes that these issues cannot be resolved in the near term and that endorsement of new criteria should not be delayed any further.

Therefore, the NRC staff has developed alternative RIA criteria, which are also discussed in the enclosed SE. It is requested that you review the staff's alternative criteria and provide recommendations for removal of any areas where you believe that excess conservatism exists and can be removed with justification. The NRC staff looks forward to receiving your input and requests a response to NRC's alternative criteria by mid-August 2005.

Of course, should you desire to respond in parallel to the staff's concerns with your criteria, please treat the staff's concerns like a request for additional information and provide your response to both the identified concerns with your TR and the staff's proposed RIA criteria.

Pursuant to 10 CFR 2.390, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.390.

Sincerely,

/RA/

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosure: As stated

cc w/encl: See next pages

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ELECTRIC POWER RESEARCH INSTITUTE (EPRI) TOPICAL REPORT (TR) -1002865,

"TOPICAL REPORT ON REACTIVITY INITIATED ACCIDENTS: BASES FOR

RIA FUEL ROD FAILURES AND CORE COOLABILITY CRITERIA"

PROJECT NO. 689

1.0 INTRODUCTION

By letter dated June 12, 2002, the Nuclear Energy Institute submitted EPRI TR-1002865 for Nuclear Regulatory Commission (NRC) review and endorsement. This TR contains a set of revised acceptance criteria for application to pressurized water reactor (PWR) and boiling water reactor (BWR) fuel during reactivity initiated accidents (RIAs). The revised criteria would be used by licensees to demonstrate that they meet the requirements of General Design Criterion 28 as it relates to the effects of postulated reactivity accidents.

2.0 EVALUATION

The staff has reviewed TR-1002865 and concludes that the proposed criteria are not acceptable for endorsement. This conclusion is based on a number of technical issues with the methodology proposed in the EPRI TR as follows:

Issues with Coolability Limit

Although we believe there may be a justifiable coolability limit at an enthalpy above the Research Information Letter (RIL)-0401 cladding failure threshold, we cannot accept the proposed limit because of the lack of experimental data. Assertions are made about limited amounts of fuel dispersal, small magnitude of pressure pulses, and inefficient energy conversion of non-molten particles. However, these assertions are based mostly on tests in the Nuclear Safety Research Reactor (NSRR) with medium-to-high-burnup specimens under conditions that are far from typical for a nuclear power plant. No analysis is presented that would allow one to interpret these data or to compare predicted light-water reactor pressure pulses with vessel limits. We do not believe it is adequate to assume that molten fuel is needed to produce an energetic fuel-coolant interaction and then use a burnup-dependent melt enthalpy as the limit. In addition, the FALCON code version used in the development of the coolability limit underestimates fuel temperatures at high burnups resulting in an overprediction of the energy needed to cause fuel melting when the fuel burnup is above 40 gigawatt days per metric ton uranium (GWD/MTU). This effect leads to a non-conservative melt enthalpy limit. In addition, the TR discusses the impact of gadolinia on the proposed fuel coolability limit but does not address the impact of using other poisons such as erbia.

ENCLOSURE

Issues with Failure Threshold

A best estimate fit of the mechanical failure data is used to determine critical strain energy density (CSED). This fit means that slightly less than 50 percent of the rods will fail just below the failure threshold when failure is not predicted. This is not consistent with NRC criteria that dose consequences for accidents not be underestimated. A lower bound fit of the failure data would be more appropriate to determine RIA failure and subsequent radionuclide release and dose estimates.

The REP-Na8 and 10 tests were assumed to be in a separate failure population as the result of oxide spalling. However, the cladding cracks were not associated with hydride blisters or spalled locations in these tests. We believe that spalling did not cause early failure in these tests but was simply a consequence of heavy oxidation that produced uniform hydrides which degraded ductility. Eliminating these two tests results in an artificially high CSED curve.

The single mixed-oxide (MOX) fuel failure in the French Test Reactor (CABRI) series was assumed to be different from the uranium dioxide (UO_2) failures because of inhomogenieties, and a gas expansion model was invoked for only this test. We are not convinced that gas expansion can significantly enhance the cladding stress as postulated. Thermal expansion of MOX and UO_2 are almost identical, so the loading from expansion should not depend on homogeneity.

We agree that the method of determining oxidation of the two Special Power Excursion Reactor Test tests was not accurate. We do not agree that adjusting that result for densities (Pilling Bedworth ratio) corrects the problem. Our previous analysis for the irradiation conditions of these specimens indicates that the oxide thickness should be less than 65 microns, rather than larger as was concluded in the report. Moving these points out to 130 microns produces a non-conservative acceptance limit.

It is known that stress state has a significant impact on strain-to-failure and will also impact CSED. The mechanical test data presented are based on three stress states: hoop tensile (uniaxial ring tests), axial tensile, and biaxial (burst tests). The EPRI CSED model development includes the ring (tensile) strain-to-failure data and appears to fit this data better than the burst or axial tension data. Based on round-robin tests performed by Argonne National Laboratory, Commisssariat a l'Energie Atomique, and the Russians, it is known that the ring tensile tests are not a measure of material property in terms of strain-to-failure but, instead, ring test failure strains are a function of specimen size and test apparatus. The burst data which are the most relevant to RIA and the axial tensile data appear to have lower strain energy densities (SEDs) at failure on average than the ring test data. This suggests that the proposed CSED correlation may seriously under-predict the probability of failure leading to a non-conservative failure limit.

It appears that only PWR data are used to develop the CSED correlation as a function of oxide/cladding thickness ratio. However, the hydrogen pickup fraction is considerably higher for BWR cladding than for PWRs. Section 3.3.2 of the TR suggests that mechanical tests provided by Wisner and Adamson (1998) demonstrate that BWR Zircaloy-2 cladding will have equal or greater ductility than PWR Zircaloy-4 cladding. Examination of this reference appears to support this assertion up to a burnup of 45 GWD/MTU but does not support it above this burnup level. In addition, recent RIA tests performed on BWR rods in NSRR suggest that the BWR rod ductility drops significantly at burnups above 57 GWD/MTU and hydrogen levels above 150 – 200 parts-per-million. Therefore, the proposed limit provides non-conservative results.

The argument is made that the SED/CSED model is derived directly from the J-integral methodology used for fracture mechanics analysis. This argument may be challenged by many fracture mechanics experts; however, there is no argument with regard to the application of J-integral fracture toughness with respect to the American Society for Testing and Materials (ASTM) standards for determining fracture toughness. A specific test method is specified for determining the J-integral values as stated in ASTM 813 and ASTM 1820 and this ASTM test method is not the same as the test measurement methods used to develop the SED/CSED methodology.

Furthermore, ASTM 1820 states that J-integral test should not be applied to brittle materials. Zircaloy uniform elongation strains of less than 1.0 to 1.5 percent strain (applicable to high burnup cladding) with a high hydrogen content appear to display significant brittle behavior. This suggests that the use of SED/CSED may not be applicable to high burnup cladding. This lack of applicability for the J-integral tests is further demonstrated by examining the available open literature J-integral data from irradiated Zircaloy with high hydrogen content. A plot of the J-integral data from irradiated Zircaloy versus hydrogen content does not show a discernable dependence, while a plot of the plane strain fracture toughness (K_{IC}) test data demonstrates a definitive dependence with a significant decrease in K_{IC} with increasing hydrogen content, which is expected due to hydrogen embrittlement. This suggests that the J-integral method may not be applicable in determining hydrogen embrittlement in high burnup Zircaloy.

The staff attempted to duplicate the SEDs calculated with FALCON by programing Equation 2-14 into the FRAPTRAN code and applying the same methodology. The studies performed with FRAPTRAN were unable to duplicate the SED values for those CABRI and NSRR test rods with plastic strains at or above 1 percent. This indicates that the SED values may be code-dependent.

3.0 NRC STAFF PROPOSAL AND CONCLUSION

Although the staff can not accept the EPRI proposal in TR-1002865, it recognizes that the current Regulatory Guide 1.77 criteria may not be conservative and believes that the endorsement of new criteria should not be delayed further since both NSRR and CABRI are undergoing enhancements and will not be available to provide additional data in the near term. Therefore, the staff is proposing the following criteria for hot zero power conditions and licensees would have the opportunity to select any one of the following as their RIA acceptance criteria:

1. Rod worths should be maintained less than $2.2 with cladding oxidation up to 70 microns and less than $1.7 for oxidation greater than 70 microns. (These rod worth limits were developed by the Office of Nuclear Regulatory Research (RES) and documented in RIL-0401.)

2. Reactivity excursions should not exceed the cladding failure threshold curve in the RES RIL-0401 (Figure 1 of cover memo).
3. Reactivity excursions exceeding the cladding failure threshold curve in the RES RIL-0401 would require a dose consequence analysis and a separate coolability

limit would be derived using validated mechanistic models to calculate the pressure pulse that would result from fuel dispersal. The coolability limit would demonstrate that the maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and that resulting system damage on other fuel assemblies is within allowable limits. The mechanistic models for fuel-coolant interaction would have to consider coolant conditions, particle size and temperature distribution, mixing, amount of fuel dispersed, timing, and coherency. Appropriate uncertainties would be included in the methodology to account for the lack of data.

It is the staff's opinion that options 2 and 3 of the RIA acceptance criteria options will require the use of three-dimensional analysis methods. The first option would not require analysis of the RIA event. The second option does not provide a separate coolability limit but the use of a cladding-specific oxidation correlation may provide additional operational margin. The acceptance criteria for at-power RIA analysis would include both the above acceptance criteria and a departure from nucleate boiling (DNB) analysis with clad failure assumed when DNB occurs as currently specified in the staff's acceptance criteria.

The staff recognizes that there is likely excess conservatism in the RIL-0401 cladding failure threshold and invites all stakeholders to identify areas of excess conservatism that can safely be removed with justification. The discussion of each excess conservatism should include a technical basis that demonstrates the available margin based on data. The staff will take future recommendations under advisement and consider removal of the identified conservatism. The staff requests the industry to provide the input on the identified areas of excess conservatism by mid-August 2005.

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Date: March 17, 2005

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