ELECTRIFY THE WORLD

EPRI

September 9, 2003

Mr. Ashok C. Thadani
Director, Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Dear Mr. Thadani,

The purpose of the Argonne National Laboratory (ANL) High Burnup LOCA Test Program is to confirm the applicability of the LOCA cladding embrittlement acceptance criteria defined in 10CFR50.46 for high burnup fuel. The US nuclear fuel vendors and utilities, working through the EPRI Robust Fuel Program (RFP), have recently reviewed the technical and regulatory bases for the peak cladding temperature (2200°F) and maximum local oxidation (17%) limits defined in 10CFR50.46 paragraphs b.1 and b.2. This letter is to express our concern that the ANL High Burnup LOCA Program, as constituted, may not provide data suitable to confirm the applicability of the current cladding embrittlement criteria. This input is being provided consistent with the memoranda of understanding between the NRC and EPRI, Framatome-ANP and Westinghouse to provide advice and consultation on proposed testing conditions.

Basis of Criteria

The underlying purpose of the peak cladding temperature (PCT) and local oxidation (Equivalent Cladding Reacted or ECR) criteria is to ensure that fuel rod damage is limited such that the core remains in a coolable geometry following the LOCA event. During the ECCS rulemaking in the 1970s, this was achieved by establishing criteria for peak cladding temperature and high-temperature oxidation which guaranteed that the cladding retained sufficient ductility to survive without fragmentation from the thermal shock quench loads and the postulated post-LOCA impact loads at temperatures following quench. The limits that would assure such ductility were established through a cumulative understanding of several types of tests, including ring compression tests. The resultant peak cladding temperature limit of 2200°F and the local oxidation limit of 17% (which at the time was required to be calculated using the Baker-Just Model) were considered to contain substantial margin with respect to the conditions that cause severe cladding embrittlement.

In the mid-1980s, the NRC staff extensively reviewed the large amount of experimental and analytical research on the behavior of LOCA emergency core cooling systems (ECCS) conducted following the 1973 ECCS hearings. This review is summarized in NUREG-1230, entitled "Compendium of ECCS Research for Realistic LOCA Analysis." Although the key focus of the review was to establish a basis to use best-estimate evaluation models for LOCA analysis, the review also concluded that sufficient data was available to assess the degree of

CHARLOTTE OFFICE
1300 W T Harris Blvd | Charlotte NC 28262-8550 USA | 704.547.6100 | Fax 704.547.6168
CORPORATE HEADQUARTERS
3412 Hillview Avenue | Palo Alto CA 94304-1395 USA | 650.855.2000 | Customer Service 800.313.3774 | www.epri.com

conservatism in the PCT and ECR criteria. Based on thermal shock quench tests and post-quench impact tests, the review concluded that the cladding embrittlement criteria established in the 1970s are sufficiently conservative to avoid cladding fragmentation by both thermal shock and post-LOCA impact loads.

In coming to this conclusion, it was recognized that even though the original PCT limit was apparently based on ring compression tests, retaining the limit at 2200°F would be conservative with respect to the amount of cladding ductility required to survive both thermal shock quench and post-LOCA impact loads. Additionally, given the conservative nature of the ring compression tests, the technical basis for the ECR limit was sufficiently established by the use of thermal shock and impact test data.

Following the NRC staff review, 10CFR50.46 was revised based on the NUREG-1230 conclusions to allow the use of more realistic evaluation models combined with an uncertainty evaluation (53 Federal Register Notice pages 35996-36005 issued in 1987). Although the technical basis was expanded with newer thermal shock and impact test data, the embrittlement criteria defined in 10CFR50.46 paragraphs b.1 and b.2 remained unchanged, since these criteria still provide appropriate margin to the conditions leading to significant core damage. In issuing the 10CFR50.46 revision regulations, it was acknowledged that the use of best-estimate models (for example, Cathcart-Pawel for oxidation) including consideration of uncertainties provides a suitably conservative basis for LOCA evaluations when used in conjunction with the demonstrated safety margins in the retained cladding embrittlement criteria.

Argonne Program

The Argonne program needs to support and confirm the current U.S. regulations on cladding embrittlement during a LOCA. It is of utmost importance that the Argonne test program provide the same technical data as used in the mid-1980s evaluations to confirm the applicability of the cladding embrittlement criteria established in 1988 to high-burnup fuel, because the criteria established in 1988 are based on a more realistic assessment of the margins compared to the 1970s position. It is not evident that the program can accomplish this goal, because the current Argonne testing matrix focuses heavily on thermal shock quench tests and ring compression testing and contains no impact testing.

While the planned thermal shock quench tests are adequate to confirm the applicability of the existing 10CFR50.46 criteria for conditions during a LOCA event, it is not clear how the ring compression tests would confirm the safety margin in the current criteria for post-LOCA conditions since this safety margin was established using impact tests during the 1980s assessment. To confirm the current criteria for post LOCA conditions after reflood, the impact tests are more applicable. The program test plan should be revised to include impact tests similar

to those relied upon in the 1988 technical evaluation to support the current criteria for post-LOCA conditions.

We believe that the timing of the Argonne testing program makes this matter of urgent importance and stand willing, as an industry, to assist the NRC with its resolution. We urge you to direct your staff to initiate further discussions with Rosa Yang or Odelli Ozer of the EPRI staff at 650-855-2481 or 650-855-2089, respectively.

Sincerely,

David J. Modeen
Vice President & Chief Nuclear Officer

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cc. J. Wermiel, NRC – NRR
F. Eltawila, NRC – RES
J. Rosenthal, NRC – RES
R. Meyer, NRC – RES
A. Marion, NEI