95-0001644

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DEFENSE NUCLEAR FACILITIES SAFETY BOARD

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June 23, 1995

Dr. Terry R. Lash
Director, Office of Nuclear Energy
Department of Energy
Washington, D.C. 20585

Dear Dr. Lash:

A Defense Nuclear Facilities Safety Board (Board) staff review team visited the Advanced Test Reactor (ATR) at the Idaho National Engineering Laboratory on May 2-4, 1995. Enclosed for your information and use is a trip report prepared by our staff on safety analyses and thermal hydraulic performance of the Advanced Test Reactor.

The Board notes that the Office of Nuclear Energy is conducting a review of the recent revision to the ATR Safety Analysis Report. The Board is aware that there is substantial expertise in nuclear and thermal-hydraulic performance resident within the Westinghouse Savannah River Company that may be available if necessary. Mr. Daniel Ogg of the Defense Nuclear Facilities Safety Board staff will be available to provide any additional information you may require.

Sincerely,

John T. Conway
Chairman

c: The Honorable Tara O'Toole, EH-1
Mr. John Wilcynski, Manager, ID Operations Office
Mr. Mark Whitaker, EH-9

Enclosure

DEFENSE NUCLEAR FACILITIES SAFETY BOARD

May 23, 1995

MEMORANDUM FOR:	G. W. Cunningham, Technical Director
COPIES:	Board Members
FROM:	A. De La Paz, J. Roarty
SUBJECT:	Safety Analyses and Thermal-Hydraulic Performance, Advanced Test Reactor, Idaho National Engineering Laboratory, Report of Site Visit, May 2-4, 1995

1. **Purpose:** This trip report describes a Defense Nuclear Facilities Safety Board (Board) staff review of the Advanced Test Reactor (ATR) core thermal-hydraulic performance and primary coolant system integrity as documented in a draft safety analysis report (SAR) dated October 25, 1994. This review was conducted on May 2-4, 1995 by Board staff members A. De La Paz, D. Ogg, and J. Roarty. A principal objective of this review was an assessment of the ATR accident prevention and mitigation strategies to maximize the safety margin in reactor core thermal-hydraulic performance and primary system reliability.

2. **Summary:** The ATR was designed in the early 1960s when reactor safety criteria were in a formative stage. Modest fuel melting, especially in test reactors, was recognized as possible during limiting accident conditions. A similar situation existed in the Savannah River K-Reactor where an extensive review and outage for incorporating safety upgrades was undertaken in 1989. The ATR, in response to post Three-Mile Island and Chernobyl accident reviews and a more recent probabilistic risk assessment, has undergone a number of safety upgrades which were reviewed by the Board's staff during this visit.

   The ATR primary coolant system integrity is of concern as fuel melting could occur in loss of coolant accidents with pipe breaks greater than 3 inches. Additional inspections have been undertaken and stress mitigation has been adopted to reduce this concern. The thermal performance of the core is also of concern and a number of defense-in-depth measures have been adopted by Lockheed Idaho Technologies Company (LITCO) or are proposed by the Board's staff. An example of a proposed measure is to consider operating the ATR with three core coolant pumps instead of two to increase the core thermal margin.

   A detailed assessment of the nuclear-thermal-hydraulic performance of the ATR was not performed by the staff as additional information must be generated to facilitate such a review. LITCO personnel indicated such information could be made available in the near future. A Department of Energy (DOE) SAR review is currently underway. It may be appropriate for

DOE to utilize Westinghouse Savannah River Company (WSRC) K-Reactor expertise for review of the nuclear and thermal-hydraulic performance of the ATR.

3. **Background.** The ATR consists of 40 fuel elements arranged in a cloverleaf pattern. Each element has 19 curved fuel plates. The ATR operates with vertical down flow through each fuel element. The core has a maximum power level of 250 megawatts, an inlet pressure of 355 psia, and an inlet temperature of 125°F. The peak heat flux during reactivity addition accident conditions is about 3.0×10^6 Btu/hr-ft^2. The ATR began operation in 1969 based on design criteria that allowed the onset of clad melting during limiting accidents. Current criteria as used in the draft SAR require that the fuel plates remain in a coolable geometry even for accidents which are unlikely.

   The Board's staff previously conducted a review of the ATR on August 30-September 1, 1994.

4. **Discussion:**

   a. Draft SAR Accident Analysis: The adequacy of the ATR accident analysis provided in the draft SAR is difficult to evaluate as core performance is displayed in statistical terms (standard deviations to a limit) vice the actual thermal-hydraulic parameters that might exist in the limiting coolant channels during accidents. LITCO personnel outlined a future program to obtain thermal-hydraulic parameters under accident conditions. This information should be available during the review of the draft SAR, presently being conducted by DOE. The Board's staff believe that it would be beneficial for DOE to consider the use of WSRC personnel experienced with the K-Reactor power limits program to provide an independent assessment of ATR core performance, including a critique of the models and correlations used in the nuclear and thermal-hydraulic design of the core. Such an independent assessment would increase the credibility of the DOE SAR review.

   b. Primary Coolant System Integrity: The ATR primary coolant system is of particular concern since the design basis accident (DBA) is limited to a three-inch equivalent diameter pipe break. In response to a concern of the Board's staff, LITCO management indicated that, in addition to periodic walkdowns and visual inspection for leaks, a program is underway to conduct confirmatory ultrasonic testing of pipe wall thickness in areas susceptible to wall thinning. In addition, to minimize piping system stress, modifications have been made to primary loop check valves to reduce pressure surges caused by check valve slam.

   c. Primary System Coolant Flow: In 1978, the primary system coolant flow rate was reduced about 12% by going from three pump operation to two pump operation. This change was implemented to reduce the cost of electricity. The Board's staff noted that the ATR safety analysis states there is a significant increase in plate power capability with three pump versus two pump operation.

d. Reactivity Insertion Accident: Another DBA is a significant power surge resulting from voiding of an experimental in-pile tube due to the positive void coefficient of reactivity associated with these tubes. Metal-to-water ratio limits have been imposed on in-pile components to reduce the severity of this accident. In addition, more restrictive requirements have been imposed on fuel plate power limits, as well as tighter controls on reactor coolant pressure setpoints.

e. ATR Plant Upgrades: The ATR has undergone a number of reviews following the Three-Mile Island and Chernobyl accidents, primarily in areas peripheral to the core, including modifications in the canal area and the electrical switchgear room. The Board's staff has also noted a heightened sensitivity in accident prevention and mitigation since their visit in August 1994. A number of upgrade features were completed following a probabilistic risk assessment of ATR potential accident scenarios, including those involving fuel handling operations.

f. Reactivity Coefficients: The Board's staff discussed the kinetics parameters that are utilized in the core transient analysis. These kinetics parameters include the void coefficient of reactivity and the moderator temperature coefficient of reactivity. Both the experimental loop and reactor coolant temperatures and void fractions affect these parameters. The Board's staff noted that limited experimental data were available to benchmark these parameters. Specifically, several experimental cases were referenced for the void coefficient of reactivity but no cases were referenced for the moderator temperature coefficient of reactivity. Currently, no uncertainty is applied to these calculated parameters when they are used in the transient analysis. DOE and LITCO personnel should consider the feasibility of performing additional void coefficient of reactivity measurements by examining the effects of metal to water ratios in the experimental loops. This is important since the transient with the highest reactivity insertion would be a result of voiding in an experimental loop. Also, it may be possible to perform temperature coefficient of reactivity measurements during reactor startup. This would depend upon the core heatup rate as a function of one, two, or three pump operation. Such data would provide additional confidence that these coefficients of reactivity used in the transient analysis are conservative.

g. Authorization Basis: LITCO personnel were questioned regarding the availability of an Authorization Basis for ATR operation and the performance of unreviewed safety question screenings and evaluations. Although the elements of the basis are available in various documents, no formal compilation currently exists as an Authorization Basis.

5. **Future Actions:** The Board's staff intends to follow the conduct of the DOE SAR review, especially those elements discussed in this trip report.