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MEMORANDUM FOR: Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

FROM: L. S. Rubenstein, Assistant Director
for Core and Containment Systems
Division of Systems Integration

SUBJECT: DESIGN BASIS EVENT FOR THE UNIVERSITY OF MICHIGAN REACTOR (TACS 12341)

This memorandum defines a Design Basis Event (DBE) applicable to the University of Michigan reactor. The NRC has made a commitment for such a definition during the review (currently under way) of the High Enrichment Uranium (HEU) fuel to Low Enrichment Uranium (LEU) fuel loading conversion of the Ford Nuclear Research Reactor. The University of Michigan HEU to LEU conversion has been sponsored by DOE as a lead case for other research reactor conversions. The conversion was prompted by the provisions of the nonproliferation treaty which the United States is committed to.

The DBE is defined as the total loss of coolant.

The enclosure deals with the historical, technical and legal aspects of this definition, which is the work of the Reactor Physics Section of the Core Performance Branch (CPB), DSI.

In this work, CPB benefited from discussions with individuals in the Division of Licensing (J. Miller, H. Bernard, and R. Carter), S. Weiss formerly of DOR, R. Schemel formerly of DSS, M. Wohl of the Accident Evaluation Branch, DSI, D. Skovholt formerly in charge of research reactor licensing in DPM (currently with DST) and J. Scinto and J. Laverty of the NRC legal staff. Finally the staff benefited from the review of the accident analysis work by the Reduced Enrichment Research and Test Reactor (RERTR) program at ANL and discussions with individuals of the program.

With this definition of the DBE the University of Michigan should:
(a) show that the proposed LEU loading will not result in significant release of radioactivity under the DBE conditions for the 2 MW power

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We shall examine the technical and legal aspects of this definition, existing precedents and its relation to power reactors; specifically we shall deal with:

- Essential Differences Between Power and Research Reactors
- Research Reactor Design and Power Level
- Some Legal Precedents

(a) Essential Differences Between Power and Research Reactors

The power level and the lack of containment, pressure vessel and engineered safety features are the basic differences between a research and a power reactor. In addition, one usually finds a research reactor located on a university campus with high population density. Some research reactors are housed in low leakage buildings which may be equipped with emergency air filters.

Design basis events for power reactors are those which determine the radiation source for an analytic evaluation of the site. Extensive standards have been promulgated for power reactors but in regulatory practice the research reactors have been dealt with on a case-by-case basis. The fundamental differences mentioned above do not allow either the power reactor design basis event nor the design basis to be applied to research reactors.

The staff recommendation for a DBE and its present definition, suggest that criteria for the acceptance of the results of their analysis exist. Similarly research reactor review and licensing as practiced by the staff, on a case-by-case basis, implies the existence of acceptance criteria. Such criteria would facilitate the review of a DBE analysis, but due to the diversity in design, operation, instrumentation and power level of research reactors, they have never been officially promulgated. The setting of such criteria is beyond the scope of this effort. The results of a DBE analysis will continue to be reviewed on a case-by-case basis. The justification of safe operation in the SER of past licensing actions usually was based on a relative comparison to 10 CFR parts 100 and 20 but never were the limits officially stated.

(b) Research Reactor Design and Power Level

Generally research reactors operate at a low power level, mostly below 2 MWt. The University of Michigan reactor is an open pool type light water moderated with a heavy water reflector on one side and licensed for operation at a maximum power level of 2 MWt. This reactor is one of 9 similar reactors in the U.S. with power levels up to 2 MW. These reactors differ in many design features but all have flat plate type fuel elements which contain UAlx alloy with aluminum cladding. The plates are 60 cm long, 6 cm wide and about .15 cm thick and are loaded in an aluminum structure with coolant space between the plates. The coolant flow is vertical and axial.

The total number of fuel elements and fuel plates in a core is variable. The Michigan reactor has 44 fuel elements with 167 gm of U-235 loading per element, .50% U-235 proposed burnup and 2.5×10^{20} fission/cc peak fission density. The cladding is .015 inches thick, 6061 Al and the core structure 5014 Al. The Michigan reactor has the highest reported power level and the longest operating period of the reactors in its class, with a typical cycle consisting of 25 days of around the clock operation followed by 3 days of shutdown/maintenance.

The design and the characteristics of the research reactors are such that reactivity insertion accidents can be accommodated without serious damage; however, the accidents which could empty the pool and remove the cooling water pose the greatest threat because the fuel elements may then be damaged due to decay heat. In the University of Michigan reactor there are 8 inch beam tubes and a pool outlet header with a diameter of 12 inches. This outlet header could empty the pool in less than 10 min (Ref. 3 & 4). Analysis of the heat transfer in the emptied core of a 2 MWt pool reactor concluded that the MTR type fuel must be cooled for at least 15 min before the decay heat is so low that it will not raise the temperature of the elements to the 650°C melting temperature. However, a recent analysis of a 2 MWt reactor contributed by ANL to the IAEA guidebook (Ref. 5) concluded that the log time (i.e., the time between reactor shutdown and core uncovering) must be greater than 30 min. for the cladding not to reach the melting point. (Note that the melting points assumed in the above studies are different, i.e., 580°C and 650°C respectively.) However, the University of Michigan has shown by analysis supported by some simulation experiments that even if the core is drained in 10 minutes the core will not melt (Ref. 4). We conclude that differences in analysis and lack of conclusive experimental data do not allow one to decide whether a loss of coolant accident for a 2 MWt aluminum plate type reactor will result in the release of a significant amount of fission products. If fission products are assumed to be released from such a fuel element, Jiacoletti, et. al. (Ref. 3) surveyed the literature and concluded that a conservative choice for fission product release of a failed fuel element in air would be: 100% Noble gases, 10% Halogens, 1% Tellurium and .1% Strontium and Cesium.

A loss of coolant accident submitted by Babcock & Wilcox (Ref. 6) for their 1 MWt pool reactor indicates that the cladding temperature will not reach the melting point. There are other examples in the literature of loss of coolant accident analysis for research reactors but none are close to the design and power density of the University of Michigan reactor.

(c) Some Legal Precedents

The case of the TRIGA reactor which was built and licensed to operate on the campus of Columbia University in New York City was brought to the Atomic Safety and Licensing Appeals Board (ASLAB) in 1971-72, by residents of

Morningside Heights (Ref. 7). In these proceedings the ASLAB accepted the regulatory staff's design basis event which consisted of the "failure of the cladding of a single fuel element while exposed to air" and a release fraction of 1.5×10^{-5}. The decision stated that 10 CFR Part 100 is not applicable to research reactors and in addition, "As a general proposition, the Appeals Board does not consider it desirable to use the standards of 10 CFR Part 20 for evaluating the effects of a postulated accident in a research reactor inasmuch as they are unduly restrictive for that purpose." Finally the Appeals Board stated that for the Columbia University reactor "the 10 CFR Part 50 Appendix D, procedures are not applicable to this licensing action."

While the precedent of the Columbia University reactor cannot be ignored, the fact that the Appeals Board did not uphold the request to make Appendix D applicable implies that the Board recognizes the NRC's prerogative to decide the licensability of each research reactor on its own merits. This view was concurred in by the NRC legal staff. Therefore while the staff cannot ignore the design basis event for the Columbia University reactor it must support all review decisions on considerations of the technical merit of each case.

<u>The University of Michigan Ford Nuclear Reactor</u>

The conclusions from the preceding discussion can be summarized in the following:

(a) due to inherent design differences, the DBEs for power reactors cannot be applied to research reactors.

(b) the NRC has the right and obligation to grant or deny a research reactor license based on its particular merits.

(c) the total loss of water in the reactor pool of a research reactor can be defined as the DBE, i.e., the event for which the risk to the public health and safety is greater than that from any event that can be mechanistically postulated.

(d) the University of Michigan Ford Nuclear Reactor operating at a 2 MWt power level with plate type aluminum alloy fuel elements must be shown that for the LEU loading will not result in significant radiation release under the DBE.

It is therefore incumbent upon the University of Michigan to:

(a) show that the proposed LEU loading will not result in significant release of radioactivity under the DBE conditions for the 2 MW power level, or

(b) provide an emergency cooling system that will remove decay heat until the core can be sufficiently cooled without such emergency cooling, or

(c) establish the power level (< 2 MWt) for which there will be no significant release of radioactivity under the DBE conditions.

References

1. Safety Analysis, "Utilization of Low Enrichment Uranium (LEU) Fuel in Ford Nuclear Reactor," by the University of Michigan, Ann Arbor Michigan, October, 1979.

2. Letter, Karl Kniel to R. W. Reid, Review of the Document "Safety Analysis, Utilization of Low Enrichment Uranium (LEU) Fuel in the Ford Nuclear Reactor," dated February 13, 1980.

3. Jiacoletti, R. J., et. al., "Consequences of Sabotage of Nonpower Reactors," NUREG/CR-0843 (LA-7845-MS) dated June, 1979.

4. Bullock, J. B., "Fission Product Heating in the Ford Nuclear Reactor," Phoenix Memorial Laboratory, Memorandum Report No. 5, University of Michigan, 1963.

5. "IAEA Guidebook on the Safety and Licensing Aspects of Research Reactor Core Conversions from HEU to LEU Fuels," Appendix A, Draft No. 1, by ANL, 1980.

6. Letter to R. W. Reid, NRC from A. F. Olsen, Babcock & Wilcox, Licensing Administrator, "License Renewal, R-47 Docket 50-59," dated August 1, 1979, and "Hazards Evaluation in Support of Request for Amendment 6 License R-47 to Operate the Lynchburg Pool Reactor at 1000 kW," Report BAW-74 Supplement, April, 1962.

7. Atomic Safety and Licensing Appeal Board, Decision Docket No. 50-208, "In the Matter of Trustees of Columbia University in the City of New York" issued May 18, 1972, Atomic Energy Commission Reports, p. 849 and initial decision, April 6, 1974, AEC reports p. 594.