ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

DEC 11 1973

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

Dear Dr. Ray:

At its 164th meeting, December 6-8, 1973, the Advisory Committee on Reactor Safeguards completed its review of the application by the Tennessee Valley Authority for authorization to operate Browns Ferry Nuclear Plant, Units 2 and 3. This project was considered at a Subcommittee meeting in Washington, D. C., on November 30, 1973. During its review, the Committee had the benefit of discussions with representatives of the Tennessee Valley Authority, General Electric Company, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported to the Commission on the construction of these units in its reports of March 14, 1967 (Unit 2) and May 15, 1968 (Unit 3).

The Browns Ferry Nuclear Plant is located in a sparsely populated area on the shore of Wheeler Lake approximately 30 miles west of Huntsville, Alabama. The plant is composed of three units. The units employ General Electric boiling water reactor nuclear steam supply systems of essentially identical design. Reactor core design for Units 2 and 3 has been substantially modified from that proposed at the construction permit stage, and is the same as that in use at Unit 1. Application for authorization to operate Unit 1 at power levels up to 3293 MWt was reviewed previously by the Committee and was reported on to you by letter dated September 21, 1972.

The emergency power system originally proposed has been substantially modified and improved. The changes were made primarily to assure adequate power capacity, lessen interactions between the units, and reduce the amount of operator coordination required. In the revised design, four diesel-generator sets are shared between Units 1 and 2 and four additional, similar sets are provided to serve Unit 3 with capability for manual paralleling with the first four sets for long term operation. The Committee believes this modification to be satisfactory.

Revised analyses of postulated control rod drop accidents which take into account the changes made in core design and which utilize recent estimates of rate of reactivity insertion, indicate that, for such accidents occurring during certain parts of the fuel cycle, the results would be unacceptable; i.e., the peak fuel enthalpy resulting in some fuel rods would exceed 280 calories per gram. In view of this, the applicant proposes to install a Rod Sequence Control System (RSCS) which he believes will render the probability of occurrence of such an accident adequately low. The proposed RSCS is a hard-wired system designed to limit rod movement sequences so as to preclude establishment of single-rod reactivity worths capable of generating excessive peak fuel enthalpy. The Committee finds this provision satisfactory.

The applicant, with the General Electric Company, has studied extensively the problem of fuel densification. His analyses indicate that, except in regard to peak clad temperatures in postulated loss of coolant accidents, the effects of expected densification are relatively small. To assure conformance with peak clad temperature and other limits of the Interim Acceptance Criteria, the applicant proposes to operate the Browns Ferry reactors in such manner as to maintain the maximum average planar linear heat generation rate (MAPLHGR) at all times below a specific allowable value. So-called "gamma" curves, developed by the applicant and reviewed and approved (to date, only for the first fuel cycle) by the Regulatory Staff, depict the allowable value as a function of fuel exposure. The applicant indicates that reactor power level may have to be restricted to less than the rated power level during portions of the fuel cycle in order to remain within the allowable MAPLHGR values. A limit on peak linear heat generation rate, not normally expected to be controlling, also is proposed by the applicant. This limit is 18.5 kw/ft including a densification spiking factor allowance specified as a function of axial position in the core. The Committee believes that the approach described is acceptable, but notes that the Regulatory Staff is still evaluating the accuracy of the method of determining core power distribution. The Committee recommends that the Regulatory Staff assure itself that the detailed procedures to be employed by the reactor operator to accomplish and to demonstrate compliance with the proposed limits on core conditions are adequate. Particular emphasis should be given to the procedures to be used when the computer normally used for core power distribution calculations is inoperable or unavailable. The Committee wishes to be kept informed.

The Committee recognizes that re-evaluation of core operating limits may be necessary as a result of possible changes in the acceptance criteria for emergency core cooling systems. The Committee wishes to be kept informed.

The inservice inspection program proposed for the reactor coolant system complies with Section XI of the ASME Boiler and Pressure Vessel Code to the extent permitted by the existing design. The Committee believes the program is acceptable, but recommends that the applicant continue to study means of assuring reactor vessel integrity in regions currently inaccessible for inspection.

The Committee recommends that limits on cumulative downtime of protection systems and engineered safety features be considered further in the development of the Technical Specifications.

Generic problems relating to large water reactors have been identified by the Regulatory Staff and the ACRS and discussed in the Committee's report dated December 18, 1972. Those problems and additional generic problems identified in more recent ACRS reports should be dealt with appropriately by the Regulatory Staff and the applicant.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Browns Ferry Nuclear Plant, Units 2 and 3 can be operated at power levels up to 3293 MWt without undue risk to the health and safety of the public.

Sincerely yours,

H. G. Mangelsdorf

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Chairman

References Attached.

References

1. Tennessee Valley Authority letter dated September 23, 1970 transmitting FSAR, Volumes 1 through 5 to Browns Ferry Nuclear Power Station Units 1, 2 and 3.

2. Amendments 10 through 41, 43 through 52 to the License Application for Browns Ferry Nuclear Power Station Units 1, 2 and 3.

3. Directorate of Licensing Safety Evaluation Report including Supplements 1 through 5.

4. Tennessee Valley Authority letter dated June 12, 1973 concerning the consequences of postulated pipe failures outside of containment.

5. GE Topical Report NEDM-10735 entitled "Densification Considerations in BWR Fuel Design and Performance" including Supplements 1 through 8.

6. Tennessee Valley Authority letter dated August 13, 1973 transmitting report entitled "Reactor Containment Building Integrated Leak Rate Test."

7. Tennessee Valley Authority letter dated November 8, 1973 transmitting report entitled "Concluding Report on the Effects of Postulated Pipe Failure Outside of Containment for Unit 1 of the Browns Ferry Nuclear Plant."