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

February 11, 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON SUITABILITY OF THE SCOTTSVILLE, PENNSYLVANIA SITE FOR A SODIUM-COOLED, FAST-NEUTRON REACTOR

Dear Dr. Seaborg:

At its 109th meeting, May 8-10, 1969, and its 118th meeting, February 5-7, 1970, the Advisory Committee on Reactor Safeguards reviewed the conceptual design proposed by General Public Utilities and Atomics International Division of the North American Rockwell Corporation for a fast-neutron, sodium-cooled power reactor to be located near Scottsville, Pennsylvania. A Subcommittee considered the proposal in Washington, D. C., on April 9, 1969, and June 4, 1969, visited the site on July 18, 1969, and reviewed the facilities, research and development programs of Atomics International at their laboratories near Canoga Park, California, on September 18 and 19, 1969. During this review, the Committee had the benefit of discussions with representatives of General Public Utilities, the Pennsylvania Electric Company, Atomics International, the AEC Regulatory Staff, and their consultants, and of the documents listed below.

The site is in a seismically quiet, rural area of Pennsylvania on the Susquehanna River about 25 miles northwest of Scranton (population approximately 111,000). Approximately 5000 people live within five miles of the site, which consists of about 1300 acres with an exclusion radius of 2000 ft. Geological studies indicate that a suitable foundation for the plant exists. Meteorological studies of the valley site are underway. Cooling towers are to be utilized for waste heat rejection, with make-up water from the river.

The proposed 1250 MW(t) reactor employs mixed uranium-plutonium oxide, stainless steel clad fuel in a cylindrical core and a loop-type primary system with mechanical pumps. The three primary loops are contained within metal-lined, shielded vaults having an inert atmosphere. Each primary loop, together with its non-radioactive, secondary sodium loop and steam generator and associated equipment, is an independent heat transfer circuit. Each circuit is capable of removing all the reactor decay heat by natural convection. In view of the importance of this function, the Committee recommends consideration be given to an additional diverse system for this purpose.

The reactor vessel is contained within a guard vessel. In addition, the free volume of the reactor vessel cavity is limited so that the sodium level would remain above the top of the core, even if both the reactor vessel and the guard vessel lost their integrity.

The primary system piping and non-reactor components are contained within enclosures of restricted volume so that with the proper shutdown of pumps, a pipe leak would not lead to uncovering the core or an interruption of the decay heat removal path. The Committee believes that further consideration should be given to possible means for assuring such protection even in the event of failure of pumps to shut down as intended.

Design of the core fuel assemblies and the core support structure is underway. Consideration is being given to several alternative designs directed toward the accommodation of swelling of the stainless steel in the core, which is exposed to high neutron fluence. Because of the incomplete status of the core mechanical design, the Committee did not review the acceptability of this aspect of the reactor.

Multiple flow inlets of each fuel assembly and other design features are employed to reduce the likelihood of flow blockage. Thermocouples for each core fuel assembly exit, pressure pulse sensors, and fuel failure detection equipment are proposed to protect the integrity of the core and to prevent or limit accidents. Space will be provided above each assembly for appropriate instrumentation as may be developed.

The proposed reactivity control and safety system consists of fifteen poison rods which, in addition to scram under gravity, can be rapidly inserted by motor drives. In order to provide additional protection for an emergency situation, the applicant is urged to continue his search for diversity in the reactivity shutdown system.

Containment and control of radioactivity in the unlikely event of a serious accident are provided by a reactor building featuring an inner and outer containment barrier. The inner barrier is defined by the metal-lined, reinforced concrete vaults surrounding the reactor and the primary system; it includes a containment dome above the refueling plugs and will be inerted during reactor operation. The design basis for the inner containment is that it withstand the consequences of postulated, low probability accidents including that involving a loss of coolant flow followed by a failure to scram. A post accident heat removal system is provided in the reactor vessel cavity for removal of fission product decay heat from the disrupted core in the unlikely event of a severe accident.

The outer containment barrier is a reinforced concrete, steel-lined building with a design leak rate of 0.5% per day at 10 psig. The design pressure was chosen to provide protection in the unlikely event of a major sodium fire occurring while the primary system vaults are open for maintenance with the reactor shut down.

The Committee believes that the accidents chosen for evaluation by the applicant provide an acceptable basis for definition of the design requirements of the primary and secondary containment barriers and the engineered safety systems. Confirmation by analyses and experiments will be needed as to the capability of the structure and engineered safety systems to meet the requirements.

The proposed reactor design is in a conceptual stage. Specific criteria remain to be formulated in several areas, and research and development is required in order to confirm the safety of the plant with regard to several matters, including protection against sodium-steam reactions in the secondary system; assessment of the probability, detection and protection requirements for fuel failure propagation; studies of common failure modes in systems important to safety; and protection against potential missiles.

Assuming satisfactory resolution of matters such as those discussed above, the Advisory Committee on Reactor Safeguards believes that the proposed site is acceptable for a reactor of the general type and power level proposed.

Sincerely yours,

/s/

Joseph M. Hendrie
Chairman

References attached.

References - Scottsville, Pennsylvania Site

1. Atomics International Letter dated May 16, 1968, Request for Pre-Application Site Review; AI-67-MEMO-150, Volume 1 - Safety Evaluation of a 500 MWe FBR Demonstration Plant - Safety Analysis; Volume II - Safety Evaluation of a 500 MWe FBR Demonstration Plant - Site and Environment, and AI-67-MEMO-149, 500 MWe FBR Demonstration Plant Description (all Company Official).

2. Atomics International Letter dated February 7, 1969, Revision of AI-67-MEMO-149 and AI-67-MEMO-150, Volume II; Replacement Pages for Volume I of same (all Company Official).

3. Atomics International; AI-AEC-MEMO-12761 (Rev. 1); An Evaluation of the Behavior of Aerosols Produced by LMFBR DBA, dated May 15, 1969.

4. Atomics International Letter dated October 16, 1969; AI-AEC-12767; Studies of Boiling Initiation for Sodium Flowing in a Heated Channel, dated September 30, 1969.

5. Atomics International Letter, dated October 21, 1969; Report on Impact of Steel Swelling on FBR Core Design, dated October 1, 1969 (Proprietary).

6. Final Draft of Responses to AEC/DRL Questions on GPU/AI 500 MWe FBR Demonstration Plant, Safety Evaluation Report, October 8, 1969 (Company Official).

7. Supplement 1 to AI-67-MEMO-150, Volume I, Answers to DRL Questions, dated October 24, 1969 (Company Official).

8. Answers to Informal AEC Questions (January, 1970).