ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 12, 1975

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: REPORT ON SUMMIT POWER STATION UNITS 1 AND 2

Dear Mr. Anders:

At its 179th meeting, March 6-8, 1975, the Advisory Committee on Reactor Safeguards completed its review of the application of the Delmarva Power and Light Company for a permit to construct the Summit Power Station, Units 1 and 2. The Committee reported previously on the Conceptual Design for a Large High Temperature Gas-Cooled Reactor (HTGR) in its letter of November 12, 1969; that design was a protoype for the Summit Power Station. Subcommittee meetings were held in Des Plaines, Illinois on April 23, 1974, in Washington, D.C. on August 7, 1974, at Newark, Delaware on September 19, 1974 in conjunction with a site visit, in Washington, D.C. on October 30 and November 13, 1974 and in Des Plaines on February 20, 1975. In addition, a General Atomic Company Subcommittee meeting was held in Denver on January 30-31, 1975. Previous consideration was given to this project during the Committee's 169th meeting May 9-11, 1974 and 175th meeting November 14-16, 1974. The Committee had the benefit of discussions with representatives and consultants of the Delmarva Power and Light Company, the General Atomic Company and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed below.

The Summit Power Station will be located on a 1,807 acre site in New Castle County, Delaware, approximately 1.2 miles south of the Chesapeake and Delaware Canal. The nearest population center is Wilmington, Delaware, approximately 15 miles north-northeast of the site. The 1970 population of Wilmington was about 80,000. The 1970 population within 50 miles of the site was about 5.5 million, which is anticipated to increase to about 6.4 million by 1980.

The Summit Power Station consists of two nuclear units, each using a General Atomic High Temperature Gas Cooled Reactor (HTGR) having a rated power level of 766 MW(e). All safety systems were analyzed and designed for 2100 MW(t). The nuclear steam supply system for each unit will be housed inside a conventional steel-lined reinforced concrete containment structure. The HTGR is located in a prestressed concrete reactor vessel (PCRV) generally of the same general type as that of the Fort St. Vrain HTGR plant. The entire primary system, including helium circulators and steam generators, is contained within the PCRV cavities. This four loop plant utilizes a helium-cooled graphite-moderated thorium-uranium fuel cycle to produce superheated steam for use in a conventional reheat, steam-turbine cycle. Since this plant will be the prototype for four-loop HTGRs, an appropriate testing program to confirm design and operating features will be required. The Committee wishes to be kept informed of progress in research and development and testing of components critical to safety such as primary circulators, primary valves, core auxiliary cooling systems, insulation, verification of PCRV design, and steam generator performance.

The Committee recognizes that the Summit Power Station represents a new design so that many of the proposed systems and components are relatively untested at this time. This aspect is apparent in the NRC Staff Safety Evaluation Report (SER) where several items are unresolved or resolution is to be deferred until the post-construction permit period. The Committee urges the resolution of these outstanding items well before equipment is installed.

A significant number of outstanding items remain in the field of nuclear instrumentation, moisture monitors and various electrical systems. Particular attention should be given to the environmental qualification of vital instruments prior to installation. These items should be resolved to the satisfaction of the NRC Staff. The Committee wishes to be kept informed.

Further information is being developed by the applicant and his contractors with regard to the subject of anticipated transients without scram. This matter should be resolved in a manner satisfactory to the NRC Staff and the ACRS.

The NRC Staff is gaining an independent capability for accident analysis of HTGRs. The Committee believes this is an appropriate step. The Committee recommends that the NRC Staff also assure that appropriate independent confirmation of the adequacy of actual design exists for the PCRV, core structural supports, and other vital structures for this prototype reactor.

Substantial information concerning performance of vital materials and components such as behavior of fuel, graphite moderator and structural members, insulation, liner, instrumentation, valves, circulators, steam generators, and PCRV should be developed during power ascension and operation of the Fort St. Vrain Reactor.

The NRC Staff should reconfirm the adequacy of performance criteria for graphite used in structural components, including such factors as permissible level of impurities, mechanical behavior, acceptable flaw sizes, and dimensional changes due to neutron irradiation.

The Committee reiterates its interest in construction to high quality standards and in the development of well-conceived surveillance and inspection programs for vital components. Current progress on the ASME Section XI Division 2 Code for Inservice Inspection is an acceptable beginning. Continued effort is required to develop inspection criteria for vital components such as insulation, graphite structures, circulators and steam generators. Similar programs are required for the PCRV tendons. These programs should cover both the integrity of vital components and their operational reliability. A necessary aspect of the surveillance testing of this prototype plant is a well conceived vibration testing program acceptable to both Staff and ACRS.

Since this is the first HTGR incorporating a pressure containment, attention should be given to a confirmation of the containment design including the validity and conservatisms in the various design basis accidents as they affect containment response.

The Committee recommends that the NRC Staff and the Applicant review the plant designs and layout for potential enhancement of physical security, particularly the protection of the fuel.

The ACRS believes it advisable to review the various outstanding items cited in this report and the SER in approximately 12-18 months.

The Advisory Committee on Reactor Safeguards believes that the above items can be resolved by the applicant and the NRC Staff during construction. Subject to the satisfactory resolution of these items the Committee believes that the Summit Power Station can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

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William Kerr
Chairman

References Attached

References

1. Preliminary Safety Analysis Report for the Summit Power Station, Volumes 1-7.

2. Amendments 1-32 to the PSAR.

3. Delmarva Power and Light Company letters and reports:

   a. July 5, 1974 letter response to six Regulatory Staff items.

   b. August 9, 1974 revised letter response to six Regulatory Staff items.

   c. August 28, 1974 letter incorporating ASME Code Case 1637 into the application.

   d. December 10, 1974 letter requesting approval of ASME Code Case 1637.

   e. December 12, 1974 letter regarding main loop valve position indication.

   f. December 27, 1974 letter regarding peak containment design pressure.

4. AEC/NRC Licensing Staff reports and letters:

   a. March 29, 1974 Interim Status Report.

   b. April 5, 1974 Errata sheet.

   c. September 6, 1974 Advanced draft of Chapter 2 of Safety Evaluation Report (SER).

   d. February 6, 1975 Safety Evaluation Report and Summary Tabulation of Outstanding Items.

5. Allis Chalmers Power Systems Inc. Proprietary Engineering Report E402 "Turbine Missile Analysis for 3600 RPM Steam Turbine Generators with 35 inch Last Stage Blades for High Temperature Gas Cooled Reactor Applications" dated January 20, 1974.

6. Allis Chalmers Power Systems Inc. Proprietary Topical Report "Overspeed Prevention of 3600 RPM Steam Turbine Generators for High Temperature Gas Cooled Reactor Applications" E110873' Rev a. dated November 27, 1973.

7. Letter from Dr. Wallace F. Walters, Assistant Professor, University of Delaware dated November 6, 1974.