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

DEC 12 1974

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

Subject: REPORT ON ALLENS CREEK NUCLEAR GENERATING STATION, UNITS 1 AND 2

Dear Dr. Ray:

At its 176th meeting on December 5-7, 1974 the Advisory Committee on Reactor Safeguards completed its review of the application of the Houston Lighting and Power Company for a permit to construct the Allens Creek Nuclear Generating Station, Units 1 and 2. The application was also reviewed in Subcommittee meetings held at Wallis, Texas on November 19, 1974, and in Washington, D. C., on November 23, 1974. The site for the proposed station was visited by Committee members on November 19, 1974. During its review the Committee had the benefit of discussions with representatives of the applicant, his consultants and contractors, and representatives of the Regulatory Staff and of the documents listed.

The site of the Allens Creek Nuclear Generating Station covers approximately 11,000 acres of land in Austin County, Texas. The nearest population center with more than 25,000 persons is Houston, Texas, which is 45 miles east of the site.

The Allens Creek Nuclear Generating Station consists of two nuclear units, each using a General Electric BWR/6 nuclear steam supply system having a design power level of 3579 MW(t) and containing 732 fuel assemblies in a pressure vessel with an internal diameter of 238 inches. The Committee reported on the BWR/6 system on September 21, 1972. Each unit will be provided with a Mark III containment system which includes a free-standing steel shell as the primary containment structure; the Committee reported on the Mark III containment concept in a letter dated January 17, 1973 and again in its report on the Grand Gulf Nuclear Station, Units 1 and 2, dated May 15, 1974.

Because of pumping of ground water, subsidence has occurred in the region of the site for several years. The applicant has evaluated potential subsidence at the site during the life of the plant based on the drawdown of ground water in Houston and at the site. This evaluation is being reviewed by the Regulatory Staff. The applicant has committed to install a system to monitor subsidence in the general area of the site. The details of this system are being formulated. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The General Electric Company is conducting an analytical and experimental program intended to provide more detailed knowledge of the behavior of the Mark III containment system. Among the phenomena for which further information is needed are vent-clearing, vent interaction, pool swell, pool stratification, and dynamic and asymmetric loads on suppression pool and other containment structures. This program is of importance to the completion of the validation of the Mark III concept. The Committee emphasizes the importance of directing the test and analytical programs toward providing not only empirical design correlations but also toward more detailed evaluations of the relevant two-phase phenomena in order to enable the better application of a specific set of scaled tests to a range of actual reactor conditions. Further, the Committee recommends that the independent models developed by the Regulatory Staff and their consultants be used to evaluate the sensitivity of key design parameters, including additional effects noted in the experimental programs, such as oscillatory phenomena. The Committee urges that the R&D program be expedited so that all design-related issues are fully resolved prior to completion of construction of affected portions of the plant. Should any results indicate a significant deviation from current predictions of the designer, the Committee wishes to be informed promptly.

In the Mark III containment the proper functioning of the pressure suppression system during a LOCA depends upon the drywell to divert the steam released to the suppression pool. The applicant has been requested by the Regulatory Staff to subject the drywell to full-design-pressure strength and leak rate tests. The Committee concurs with the Regulatory Staff.

The applicant has proposed and the Regulatory Staff has accepted, a combustible gas control system designed on the basis of an assumed one percent metal-water reaction. The system contains hydrogen recombiners and a controlled purging system for the drywell. The Committee notes that appropriate attention should be given to gas mixing in the drywell.

A Regulatory Staff requirement, which has become a generic issue, pertains to designing the radioactive offgas system, including the adsorption beds to Seismic Category I to meet item C.1.p. of Regulatory Guide 1.29. This Guide requires that the offgas system meet the seismic requirements if potential offsite doses exceed 0.5 rem. The Committee recognizes that the offsite dose will be a function of the total source term, the assumptions relating to the rate of release of the source, and the assumed meteorology. The Committee believes that appropriate conservatisms should be used in determining the dose in the unlikely event of a seismically induced failure of the offgas system. However, the Committee questions the validity of multiplicative conservatisms when the source of radioactivity is relatively limited. The Committee recognizes that the application of Regulatory Guide 1.29 has major design implications to several auxiliary systems in addition to the offgas system. The Committee urges that the applicant and the Regulatory Staff arrange to have additional research conducted to better define quantitatively the key factors necessary for evaluating this type of accident situation. The Committee also requests that the Regulatory Staff review the conservatisms in the source term and in the meteorological model to establish whether all of the required conservatisms are appropriate. The Committee wishes to be kept informed.

In the view of the Regulatory Staff, the proposed design of the residual heat removal system has not been demonstrated to be capable of functioning assuming the most restrictive single failure as required by General Design Criterion 34. The Committee believes that an adequate system analysis of this generic problem has not been made which takes into account the complete system and all modes of behavior. The Committee recommends that additional study be made. The Committee wishes to be kept informed.

The Regulatory Staff has determined that the ECCS performance evaluation of the Allens Creek units meets the Interim Acceptance Criteria of June, 1971. In addition, the applicant's ECCS performance evaluation, using an approved General Electric model to show compliance with the Final Acceptance Criteria of 10 CFR 50.46, must be submitted and then reviewed and approved by the Regulatory Staff. The Committee wishes to be kept informed.

A recent publication (See Reference 11) suggests a need for the use of three-dimensional calculations to correctly predict peak flux and temperature distributions for super-prompt-critical excursions. This may be relevant to analysis of the rod-drop accident, and both General Electric and the Regulatory Staff have initiated work to clarify the situation. This matter should be resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The Regulatory Staff is continuing to review several items that apply to the Allens Creek Nuclear Generating Station which are also generic to BWR/6 reactors and to Mark III containment systems. The Committee wishes to be kept advised of the resolution of these matters.

Additional generic problems relating to large water reactors have been identified by the Regulatory Staff and the ACRS and have been discussed in the Committee's report dated February 13, 1974. These problems should be dealt with appropriately by the Regulatory Staff and the applicant.

The ACRS believes that the above items can be resolved during construction and that, if due consideration is given to these items, the Allens Creek Nuclear Generating Station Units 1 and 2 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Dr. W. R. Stratton did not participate in the Committee's review of this project.

Sincerely yours,

Edward A. Mason

Edward A. Mason
Acting Chairman

References:

1. The Houston Lighting and Power Company (HL&P) Preliminary Safety Analysis Report (PSAR), Volumes 1-14, for the Allens Creek Nuclear Generating Station Units 1 and 2.

2. Amendments 1-16, 18-25 to Preliminary Safety Analysis Report.

3. Directorate of Licensing letter dated March 4, 1974, concerning meeting with Water Rights Commission and Site visit to Allens Creek Nuclear Generating Station.

4. Directorate of Licensing letter dated March 8, 1974, concerning transmittal of information concerning BWR/6 Reactor Design.

5. Houston Lighting and Power Company letter dated April 1, 1974, concerning Brazos River Channel.

6. Houston Lighting and Power Company letter dated June 3, 1974, transmitting hydrology summary.

7. Houston Lighting and Power Company letter dated July 10, 1974, committing to reference GESSAR review of the three new General Electric Control and Protection designs.

8. Directorate of Licensing letter transmitting "Summary Statement of Outstanding Safety-Related Issues" received November 7, 1974.

References Cont'd

9. Safety Evaluation of the Allens Creek Nuclear Generating Station Units 1 and 2 dated November 1974.

10. Directorate of Licensing letter transmitting Summary Statement of Outstanding Safety-Related Issues received December 12, 1974.

11. "Comparison of Two-and-Three Dimensional Calculations of Super Prompt Critical Excursions" by A. Birkhofer, A. Schmidt, and W. Werner, Nuclear Technology, Volume 24, pp. 7-12, October 1974.

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

October 12, 1967

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

Subject: ARGONNE ADVANCED RESEARCH REACTOR

Dear Dr. Seaborg:

At its ninetieth meeting, October 5-7, 1967, the Advisory Committee on Reactor Safeguards reviewed the proposed Argonne Advanced Research Reactor (AARR) which is to be constructed at the Argonne National Laboratory. A Subcommittee meeting was held in Washington, D. C. on September 29, 1967. During its review, the Committee had the benefit of discussions with representatives of the Argonne National Laboratory, the Division of Reactor Development and Technology, and the AEC Regulatory Staff and of the documents listed.

The proposed AARR is a light-water-cooled and moderated, beryllium-reflected, flux-trap reactor. Many of the features of the reactor are similar to those of the High Flux Isotope Reactor (HFIR), which was previously reviewed and discussed in Committee letters, dated May 9, 1960, July 25, 1960, July 15, 1965 and May 11, 1966. In particular, the fuel elements, core geometry, and control rods and drives are identical to those used in HFIR. Several features, however, differ significantly. The pressure vessel of the AARR is constructed of 304-L stainless steel. The number and size of the neutron beam tubes have been increased, and a larger number of rabbits and positions for long-term irradiations have been included in the internal thermal column. A reinforced concrete containment structure designed to withstand an internal pressure of 6 psig encloses the reactor.

The seismic design criteria are not yet completely defined. The Committee believes that questions related to these criteria should be resolved by the applicant with the Regulatory Staff before the containment base slab is poured.

The present review is the first of two reviews to be conducted at the construction phase. Although general design criteria are available, firm designs of several features, such as the containment structure, the pressure vessel, and instrumentation, will not be available for review until after contracts are let for design and construction. The ACRS will wish to review the designs as early as possible.

On the basis of the information presently available, it is the opinion of the ACRS that there is reasonable assurance that a reactor facility of the type proposed can be constructed and operated at the Argonne National Laboratory site without undue hazard to the health and safety of the public.

Dr. Herbert S. Isbin, Dr. Harry O. Monson, and Dr. David Okrent did not participate in the above review.

Sincerely yours,

/s/ N. J. Palladino

N. J. Palladino
Chairman

References:

1. Volume I, Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor, dated March 31, 1966, Revised December 9, 1966.
2. Volume II, Appendices to the Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor, dated March 31, 1966, Revised December 9, 1966.
3. Supplement 1 to the Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor, dated November 9, 1966.
4. Index to Responses to AEC Questions, dated January 23, 1967, and Corrected Figure IV-G-3 to Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor.
5. Supplement 2 to the Preliminary Safety Analysis Report on the Argonne Advanced Research Reactor, dated April 5, 1967.
6. Supplement 3, Compendium of Safety Considerations (Questions and Answers) Raised During the DRL Review of the Argonne Advanced Research Reactor PSAR, undated (received August 8, 1967).