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

January 27, 1970

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

Subject: REPORT ON PALISADES PLANT

Dear Dr. Seaborg:

At a Special Meeting, January 23-24, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by Consumers Power Company for authorization to operate the Palisades Plant at power levels up to 2200 MWt. This project was also considered at the 113th ACRS meeting, September 4-6, 1969, the 115th ACRS meeting, November 6-8, 1969, and the 116th ACRS meeting, December 11-13, 1969. Subcommittee meetings were held on July 31, 1969, at the site, and on October 29, 1969, December 3, 1969, and January 22, 1970, in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives of Consumers Power Company, Combustion Engineering, Inc., Bechtel Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported to you on the construction of this plant in its letter dated January 18, 1967.

The site for the Palisades Plant consists of 487 acres on the eastern shore of Lake Michigan in Covert Township, approximately four and one-half miles south of South Haven, Michigan. The minimum exclusion radius for the site is 2300 feet and the nearest population center of more than 25,000 residents consists of the cities of Benton Harbor and St. Joseph, Michigan, which are approximately 16 miles south of the site.

The nuclear steam supply system for the Palisades Plant is the first of the Combustion Engineering line currently licensed for construction. A feature of the Palisades reactor is the omission of the thermal shield. Studies were made by the applicant to show that omission of the shield would not adversely affect the flow characteristics within the reactor vessel or alter the thermal stresses in the walls of the vessel in a manner detrimental to safe operation of the plant. Surveillance specimens in the vessel will be used to monitor the radiation damage during the life of the plant. If these specimens reveal changes that affect the safety of the plant, the reactor vessel will be annealed to reduce radiation damage effects. The results of annealing will be confirmed by tests on additional surveillance specimens provided for this purpose. Prior to accumulation of a peak fluence of 10^{19} nvt (> 1 Mev) on the reactor vessel wall, the Regulatory Staff should reevaluate the continued suitability of the currently proposed startup, cooldown, and operating conditions.

The secondary containment is a reinforced concrete structure consisting of a cylindrical portion prestressed in both the vertical and circumferential directions, a dome roof prestressed in three directions, and a flat non-prestressed base. Before operation, it will be pressurized and extensive measurements will be made of gross deformations and of strains in the linear, reinforcement, and concrete, and the pattern and size of cracks in the concrete will be observed and measured. The applicant has proposed suitable acceptance criteria for the pressure test, and the ACRS recommends that the Regulatory Staff review and assess the results of this test prior to operation at significant power.

The prestressing tendons in the containment consist of ninety, one-quarter-inch diameter wires. They are not grouted or bonded, and are protected from corrosion by grease pumped into the tendon sheaths. The applicant has proposed that selected tendons be inspected periodically for broken wires, loss of prestress, and corrosion. If degradation is detected, the inspection can be extended to the remaining tendons, all of which are accessible. The applicant is performing studies to determine the appropriate number and interval for tendon inspection. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The core is calculated to have a slightly negative moderator coefficient at full power operation at beginning-of-life, but uncertainties in the calculations are such that the existence of a positive moderator coefficient cannot be precluded. The applicant has stated that the moderator coefficient will not exceed $+0.5 \times 10^{-4}$ Δ k/k/°F at beginning-of-life, computed from start-up test data on a conservative basis. The applicant also plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses required to establish the value of the moderator coefficient.

The meteorological observation program conducted at the site subsequent to the Committee's report to you on January 18, 1967, indicated the need for the addition of iodine removal equipment to the containment for use in the unlikely event of a loss-of-coolant accident. The applicant proposed to install means for adding sodium hydroxide to the water in the containment spray system. However, because of uncertainties regarding the generation of hydrogen and the effects of other materials resulting from the reaction of this alkaline solution with the relatively large amounts of aluminum in the containment, this spray additive will not be used unless it can be shown by further studies that the use of sodium hydroxide is clearly acceptable. In addition, the applicant will carry out studies of iodine removal by borated water sprays without sodium hydroxide. If the results of these studies are not acceptable, a different iodine removal system satisfactory to the Regulatory Staff will be installed at the first refueling outage. A report on the applicant's plans will be submitted to the AEC within six months following issuance of a provisional operation license. The Committee believes that this procedure is satisfactory for operation at power levels not exceeding 2200 MWt.

The applicant has stated that if fewer than four primary coolant pumps are operating, the reactor overpower trip settings will be reduced such that the safety of the reactor is assured in the absence of automatic changes in the thermal margin trip settings.

The Committee believes that, for transients having a high probability of occurrence, and for which action of a protective system or other engineered safety feature is vital to the public health and safety, an exceedingly high probability of successful action is needed. Common failure modes must be considered in ascertaining an acceptable level of protection. Studies are to be made on further means of preventing common failure modes from negating scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant should consider the results of such studies and incorporate appropriate provisions in the Palisades Plant.

The Committee recommends that attention be given to the long-term ability of vital components, such as electrical equipment and cables, to withstand the environment of the containment in the unlikely event of a loss-of-coolant accident. This matter is applicable to all large, water-cooled power reactors.

Continuing research and engineering studies are expected to lead to enhancement of the safety of water-cooled reactors in other areas than those mentioned: for example, by determination of the extent of the generation of hydrogen by radiolysis and from other sources, and development of means to control the concentration of hydrogen in the containment, in the unlikely event of a loss-of-coolant accident; by development of instrumentation for inservice monitoring of the pressure vessel and other parts of the primary system for vibration and detection of loose parts in the system; and by evaluation of the consequences of water contamination by structural materials and coatings in a loss-of-coolant accident. As solutions to these problems develop and are evaluated by the Regulatory Staff, appropriate action should be taken by the applicant on a reasonable time scale.

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 Palisades Plant can be operated at power levels up to 2200 MWt without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References:

1. Final Safety Analysis Report for the Palisades Plant

2. Amendments No. 9-19 to license application