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

July 17, 1969

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

Subject: REPORT ON THREE MILE ISLAND NUCLEAR STATION UNIT 2

Dear Dr. Seaborg:

At its 111th meeting, July 10-12, 1969, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Metropolitan Edison Company and the Jersey Central Power and Light Company to construct Unit 2 at the Three Mile Island Nuclear Station. A Subcommittee also met to review this project on June 26, 1969. During its review, the Committee had the benefit of discussions with representatives and consultants of both applicants, the Babcock and Wilcox Company, Burns and Roe, Inc., General Public Utilities Corp., and the AEC Regulatory Staff. The Committee also had available the documents listed below.

The plant will be located adjacent to Unit 1 on Three Mile Island near the east shore of the Susquehanna River, about 10 miles southeast of Harrisburg, Pennsylvania. The nuclear steam supply system, engineered safety features, reactor building, and aircraft hardening protection are similar to those of Unit 1, noted in our January 17, 1968, and April 12, 1968, reports. Unit 2 will be operated at a power level of 2452 MWt.

Review of Unit 2 has taken into account the similarities of the Three Mile Island units, new features, updating of the research and development programs, and further evaluations of the site. The review also included matters previously identified that warrant careful consideration for all large, water-cooled power reactors; the Committee believes that resolution of these matters should apply equally to this reactor.

The estimate of probable maximum flood discharge in the Susquehanna River at the site is being revised upwards by the U. S. Army Corps of Engineers and will be larger than had been considered in the design of Unit 1. The applicant has stated that both units will be protected by measures which would assure a safe, orderly shutdown of the reactors in the event of the maximum flood.

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The applicant has conducted a test program in support of his proposal to grout the stranded tendons for the containment prestressing system. The Committee believes that adequate grouting can be attained through proper and careful execution of the procedures developed in this program. The applicant has proposed a program of periodic proof testing at 115% of design pressure to monitor the integrity of the containment, which has been designed conservatively to obviate any adverse effects of repeated proof testing at this high pressure. The Committee believes that such a program, involving measurement of deformations and thorough inspection for cracking of the concrete during each proof test, will provide reasonable assurance of the continued integrity of the containment.

Further review is necessary of the research and development being completed for the alkaline sodium thiosulfate spray additive to determine whether the spray systems as proposed need augmentation to achieve required performance in postulated accidents. Provisions will be incorporated in the design of the containment system to permit equipment additions if necessary to ensure limiting the radiological consequences of a loss-of-coolant accident to doses significantly below the 10 CFR 100 guideline values.

The applicant has been considering a purge system to cope with potential hydrogen buildup from various sources in the unlikely event of a loss-of-coolant accident. Additional studies are needed to establish the acceptability of this system and to consider alternative approaches. These studies should include allowance for levels of zircaloy-water reaction which could occur if the effectiveness of the emergency core cooling system were significantly less than predicted. The Committee believes that this matter can be resolved during construction of the reactor.

The Committee reiterates its belief that the instrumentation design should be reviewed for common failure modes, taking into account the possibility of systematic, non-random, concurrent failures of redundant devices, not considered in the single-failure criterion. The applicant should show that the proposed interconnection of control and safety instrumentation will not adversely affect plant safety in a significant manner, considering the possibility of systematic component failure. The Committee believes that this matter can be resolved during construction of the reactor.

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. The Committee recommends that a study be made of the possible consequences of hypothesized failures of protective systems during anticipated transients, and of steps to be taken if needed. The Committee believes that this matter can be resolved during construction of the reactor.

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The Committee recommends that the applicant study possible means of in-service monitoring for vibration or for the presence of loose parts in the reactor pressure vessel as well as in other portions of the primary system, and implement such means as are found practical and appropriate.

The post-accident cooling system must retain its integrity throughout the course of an accident and the subsequent cooling period. The applicant should review the effects of coolant temperature, pH, radioactivity, corrosive materials from the core or other parts of the containment (including stored chemicals), and potentially abrasive slurries. Degeneration of components such as filters, pump impellers, and seals by any of these mechanisms should be reviewed. Particular attention should be paid to potential problems arising from the use of dissimilar metals in these systems.

The Committee recommends that details concerning the adequacy of the design, the material characteristics, quality assurance, and in-service inspection requirements of the main coolant-pump flywheels be resolved between the applicant and the Regulatory Staff. In this connection, and, in general, the Committee continues to emphasize the need and importance of quality assurance, in-service inspection and monitoring programs, as well as conservative safety margins in design.

The Advisory Committee on Reactor Safeguards believes that the items mentioned can be resolved during construction, and that, if due consideration is given to the foregoing, Unit 2 proposed for the Three Mile Island site can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Original Signed by
Stephen H. Hanauer

Stephen H. Hanauer
Chairman

References:

1. Three Mile Island Nuclear Station - Unit 2, Preliminary Safety Analysis Report, Volumes 1-4 (Amendment No. 6, Oyster Creek Nuclear Station, Unit 2, Docket No. 50-320).
2. Amendments 7 - 10 to Application for Licenses.
3. Metropolitan Edison Company letter dated July 3, 1969.

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