UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 14, 1976

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Dr. Dade W. Moeller
Chairman, Advisory Committee
on Reactor Safeguards
1016 - H Street
Washington, D.C. 20555

RE: CONSUMERS POWER COMPANY (MIDLAND PLANT, UNITS 1 & 2), DOCKET NOS. 50-329/330

Dear Dr. Moeller:

The U.S. Court of Appeals for the District of Columbia Circuit in Aeschliman v. NRC, Appeal Nos. 73-1776 and 73-1867 (July 21, 1976), ruled that your Committee's report on the Midland facility should be returned to the ACRS for clarification, in particular for further elaboration on the reference to "other problems".

This Atomic Safety and Licensing Board has been reconvened by the Commission to conduct the reopened proceedings required by the above-identified Court decision. This reopened hearing includes the issue of clarification of the ACRS report. As required by the Court, we are hereby returning the ACRS report of June 18, 1970, with its supplement of September 23, 1970, to you for clarification. Would you advise us of what action your Committee is taking or plans to take with regard to Midland in response to the Court order. We would also appreciate an estimate of the time that will be required for the clarification called for by the Court.

A prompt reply would be helpful to the Board in assessing scheduling requirements for the reopened proceeding.

Very truly yours,

Daniel M. Head

Daniel M. Head, Chairman
Atomic Safety and Licensing Board

Enclosure: ACRS report

cc w/o encl: Harold L. Reis, Esquire
Myron M. Cherry, Esquire
Jane A. Axelrad, Esquire
James N. O'Connor, Esquire

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

June 18, 1970

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

Subject: REPORT ON MIDLAND PLANT UNITS 1 & 2

Dear Dr. Seaborg:

During its 122nd meeting, June 11-13, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by the Consumers Power Company for a permit to construct the Midland Plant Units 1 and 2. During this review, the project also was considered at Subcommittee meetings held on January 22, 1969, at the plant site, on April 24, 1970, at Chicago, Illinois, on February 4, 1969, March 24, 1970, and June 10, 1970, at Washington, D. C. and at the ACRS meetings of February 6, 1969, April 9, and May 8, 1970, in Washington, D. C. In the course of these meetings, the Committee had the benefit of discussions with representatives and consultants of the Consumers Power Company, Babcock and Wilcox Company, Bechtel Corporation, Dow Chemical Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Midland Plant site is on the south bank of the Tittabawassee River adjacent to the southern city limits of Midland, Michigan. The main industrial complex of the Dow Chemical Company lies within the city limits directly across the river from the site and provides an area of controlled access about two miles wide between the reactor site and the Midland business and residential districts. The exclusion area of the plant site has a radius of 0.31 miles and includes a small segment of the Dow plant; no Dow employees are permanently assigned in this segment, and the applicant has the right to remove any persons from this segment if conditions warrant. The low population zone has a radius of 1.0 miles and contains 38 permanent residents and about 2,000 industrial workers, mainly employees of Dow Chemical Company. The number of permanent residents within five miles of the plant site was estimated to be 41,000 in 1968, mainly in the city of Midland and its environs.

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The applicant has established criteria for, and has begun the formulation of a comprehensive emergency evacuation plan. This plan is being coordinated with the well-established plan of the Dow Chemical Company for emergency evacuation of the Midland chemical plant and portions of the City of Misland in case of major emergencies at the chemical plant. Close coordination with appropriate municipal and state authorities is also being established.

The Midland units will each include a two-loop pressurized water reactor designed for initial core power levels up to 2452 MWt. The nuclear steam supply systems and the emergency core cooling systems of these units are essentially identical with those for the previously reviewed Oconee Units 1, 2 and 3 and Rancho Seco Unit 1 (ACRS reports of July 11, 1967 and July 19, 1968, respectively). The combined electrical output of the two units will be 1300 MW. In addition, 4,050,000 lbs per hour of secondary steam will be exported to the adjacent Dow plant to supply thermal energy for chemical processing operations.

The prestressed, post-tensioned concrete reactor containment buildings are similar to those approved for the Oconee Units 1, 2 and 3. The design will include penetrations, which can be pressurized, and isolation valve seal water systems to reduce leakage. Channels will be welded over the seam welds of the containment liner plates to permit leak testing of the seam welds.

Cooling water for the Midland reactors is supplied from a diked pond with a capacity of 12,600 acre-feet. Make-up water is taken from the Tittabawassee River. The cooling water supply is sufficient for 100 days of full power operation without make-up during periods of low river flow. In the unlikely event of a gross leak through the dikes of the cooling pond, a supplemental source of water will be available. The supplemental source is provided within the main pond by excavating a 24 acre area to a depth of six feet below the bottom of the main pond. This source can supply shut-down cooling capability for 30 days without make-up.

The applicant will conduct an on-site meteorological monitoring program to verify the applicability of the meteorological models used for accident evaluation and routine release limits as well as to determine any meteorological effect of the cooling pond. This program should be completed during construction.

Midland is the first duel purpose reactor plant to be licensed for construction. The export steam originates from the secondary side of the steam generators and may contain traces of radioactive leakage from the primary system. The demineralized condensate from 60 to 75 percent of the export steam is returned by Dow to the feed water supply of the reactor plant. The condensate from the remaining steam is either chemically contaminated or cannot practically be returned to the nuclear plant. It is collected in the Dow waste treatment system for dilution and processing with other streams before eventual discharge to the river. Thus, the unreturned portion of the condensate represents an effluent from the reactor plant to which the requirements of 10 CFR Part 20 must apply.

This matter may be considered in two parts: (1) the steps taken by the applicant to ensure that any radioactivity in the export steam is within the limits set by 10 CFR Part 20 and as low as practicable and (2) the measures taken by the Dow Chemical Company to ensure that the export steam can be used in chemical operations without product contamination and that the unreturned steam condensate is properly managed for safe disposal. In connection with item (1), the applicant proposes to monitor and control radioactivity in the export steam. A representative, continuous sample of the export steam will be condensed for monitoring and laboratory analysis. The gamma activity of this flowing sample will be continuously monitored by on-line analyzers and an alarm actuated if the activity exceeds an appropriate limiting value. The alarm will serve to indicate any change in the integrity of the steam generators or fuel cladding. Samples of this condensate stream will be analyzed at appropriate intervals by sensitive low-level beta counting for determination of gross beta activity and concentration of selected radionuclides. The applicant agrees to limit, by maintaining high integrity of the steam generators and fuel cladding, the yearly average gross beta activity in the export steam to one-tenth or less of the limits specified by 10 CFR Part 20 for the selected radionuclides. The yearly average will include any periods of short duration when the concentrations may approach but not exceed the 10 CFR Part 20 limits. The applicant states that in his judgment it is practical to operate the plant within these limits. If these limits are exceeded, corrective measures will be taken in the plant or the delivery of export steam to Dow will be terminated. He also agrees to demonstrate the analytical equipment and procedures in development programs to be carried forward and completed during construction of the Midland Plant. In connection with item (2), Dow has stated that they will apply for a 10 CFR Part 30 Materials License to receive, possess, and use the export (secondary) steam as a source of thermal and mechanical energy. No export steam or condensate will be intentionally introduced into any product. Isolation of the export steam from contact with products will be accomplished by the use of heat exchange devices which will provide suitable physical barriers. Programs will be established to provide for detection of leaks in the heat exchange devices by analyses, monitors, and other means; for repair of leaks when detected; and for appropriate administrative control of the programs.

Dow has stated that accumulation of radioactivity from the export steam and release of radioactive materials in the effluent will be in accordance with 10 CFR Part 20. The unreturned condensate will represent less than 10% of the total liquid effluent disposed of through the Dow waste treatment plant and the annual average concentration in the total effluent is expected to be less than 1% of the 10 CFR Part 20 limits.

The Committee believes that the criteria proposed by the applicant and Dow for the control of radioactivity in the export steam are necessary and adequate. The detailed procedures for implementation should be developed during construction in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

To minimize the likelihood of subsidence at the site, the applicant and Dow have agreed to prohibit future salt mining operations within one-half mile from the center of the reactor plant. No new wells will be drilled within this distance and all existing wells will be abandoned and plugged. The Committee believes these arrangements are satisfactory.

A large volume of liquid chlorine is maintained in a refrigerated storage vessel about one mile from the Midland plant control room. The applicant is continuing his study of the consequences of a major accidental release of chlorine from this vessel. He has included in his criteria for the design of the control room the objective of finding a practical method of maintaining the concentration of chlorine in the control room atmosphere below the eight hour threshold limiting value (TLV) of 1 ppm for the most serious conceivable chlorine accident. The Committee believes that adequate air purification facilities should be provided in the control room ventilation system to reduce chlorine concentration to the eight hour TLV of 1 ppm so that operators can work without respiratory equipment during an extended chlorine emergency. This matter should be resolved during construction in a manner satisfactory to the Regulatory Staff.

The reactor vessel cavity will be designed to withstand mechanical forces and pressure transients comparable to those considered in the design of the Zion and Indian Point-3 plants.

The applicant has stated that he will provide additional evidence obtained by improved multi-node analytical techniques to assure that the emergency core cooling system is capable of limiting core temperatures to the limits established at present. He will also make appropriate plant changes if the further analysis demonstrates that such changes are required. This matter should be resolved during construction in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The safety injection system for the Midland plant is actuated by either low reactor pressure or high containment pressure signals. However, of these two, the reactor is tripped only by the low reactor pressure signal. The Committee believes that provision also should be made to trip the reactor by the high containment pressure signal.

The applicant plans to develop more detailed criteria for the installation of protection and emergency power systems together with appropriate procedures to maintain the physical and electrical independence of the redundant portions of these systems. The Committee believes that these criteria and procedures should be reviewed and approved by the Staff prior to actual installation.

The applicant considers the possibility of melting and subsequent disintegration of a portion of a fuel assembly because of flow starvation, gross enrichment error, or from other causes to be remote. However, the resulting effects in terms of local high temperature or pressure and possible initiation of failure in adjacent fuel elements are not well known. Appropriate studies should be made to show that such an incident will not lead to unacceptable conditions.

The Committee believes that consideration should be given to the utilization of instrumentation for prompt detection of gross failure of a fuel element.

The Committee has commented in previous reports on the development of systems to control the buildup of hydrogen in the containment which might follow in the unlikely event of a major accident. The applicant proposes to make use of a technique of purging through filters after a suitable time delay subsequent to the accident. However, the Committee recommends that the primary protection in this regard should utilize a hydrogen control method which keeps the hydrogen concentration within safe limits by means other than purging. The capability for purging should also be provided. The hydrogen control system and provisions for containment atmosphere mixing and sampling should have redundancy and instrumentation suitable for an engineered safety feature. The Committee wishes to be kept informed of the resolution of this matter.

The Committee recommends that the applicant accelerate the study of 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 stated that the engineering design would maintain flexibility with regard to relief capacity of the primary system and to a diverse means of reducing reactivity. This matter should be resolved in a manner satisfactory to the Regulatory Staff during construction. The Committee wishes to be kept informed.

Other problems related to large water reactors have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports. The Committee believes that resolution of these items should apply equally to the Midland Plant Units 1 & 2.

The Committee believes that the above items can be resolved during construction and that, if due consideration is given to these items, the nuclear units proposed for the Midland Plant can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
Joseph M. Hendrie
Chairman

References

1) Amendments 1 - 12 to License Application

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

March 12, 1970

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

Subject: REPORT ON HUTCHINSON ISLAND PLANT UNIT NO. 1

Dear Dr. Seaborg:

At its 119th meeting, March 5-7, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application of the Florida Power and Light Company for authorization to construct a nuclear power plant at its Hutchinson Island site in St. Lucie County, Florida. A Subcommittee visited the site on January 5, 1970; a second Subcommittee meeting was held in Chicago on February 21, 1970. During its review, the Committee had the benefit of discussions with the applicant, Combustion Engineering, Inc., Ebasco Services, Inc., the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed.

The Hutchinson Island Plant Unit No. 1 will be located on a tract of land of approximately 1100 acres, about half way between Fort Pierce and Stuart on the east coast of Florida. About 1000 people live within a five mile radius of the site. The nearest population center is Fort Pierce (population about 34,000), which is eight miles away.

The plant site on Hutchinson Island is underlain by sand to a depth of several hundred feet. To provide satisfactory bearing and settlement characteristics and resistance to liquefaction, the first sixty feet of loose send is being removed and the excavation refilled to foundation depth with granular material compacted to a relative density of 85 percent.

The proposed pressurized water reactor has a design power level of 2440 MW(t) and is similar to the previously reviewed Maine Yankee and Calvert Cliffs reactors (ACRS reports dated July 19, 1968 and March 13, 1969). The containment system consists of a steel containment vessel enclosed within a reinforced concrete building, with the annular space maintained at a slight negative pressure and exhausted through filters. The applicant has stated that the containment and other structures and systems important to safety will be designed to meet the same tornado

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design criteria as have been used for other recently reviewed plants, and that protection of vital components will be provided against the probable maximum hurricane-induced flood and runup level as estimated by the Coastal Engineering Research Center.

The applicant stated that a dynamic seismic analysis will be performed on the primary system. Several other matters related to seismic design, including the spectra to be used in the design of piping and equipment, and the design procedures to be used for various types of Class 1 piping, should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant stated that the primary system will be designed so that annealing of the pressure vessel will be practical at a temperature of at least 650° F.

Pump seal and other leakage from emergency core cooling (ECCS) equipment and lines outside the containment may lead to undesirable releases of radioactivity in the unlikely event of a loss-of-coolant accident. The Committee recommends that the atmosphere around the ECCS lines and pumps outside the containment be vented through a charcoal filter system.

Further study is required with regard to potential releases of radioactivity in the unlikely event of gross damage to an irradiated subassembly during fuel handling and the possible need for a charcoal filtration system in the fuel handling building. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

All hot process lines penetrating the containment annulus will be designed with a guard pipe to direct steam flow back to the primary containment in the unlikely event of a rupture of the process pipe in the annulus region. In view of the importance of the guard pipes, the applicant will arrange for an independent review of the design.

The applicant stated that he will install a concrete wall in the containment penetration room to separate the cables and penetrations for redundant devices essential to safety. The Committee believes that the separation of redundant elements in the penetration room and elsewhere requires further study, as to both criteria and design details.

A suitable preoperational vibration testing program should be employed for the primary system. Also, attention should be given to the development and utilization of instrumentation for in-service monitoring for excessive vibration or loose parts in the primary system.

When details of the planned loads and ratings of the emergency diesel generators become available, the Regulatory Staff should assure itself that adequacy of design conservatism is realized and that sufficient testing and experience will be available prior to plant startup to prove the reliability of the emergency power system.

The Committee reiterates its interest in active participation by applicants in overall quality assurance programs to better assure the construction of safe plants. In this regard, a greater level of direct participation by the applicant in the quality assurance program of the Hutchinson Island Plant would be desirable.

Information on a number of items, identified in previous reports of the Committee, is to be provided by the applicant to the Regulatory Staff during construction. These include:

a) A study of 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.

b) Review of development of systems to control the buildup of hydrogen in the containment, including an appropriately conservative estimate of possible hydrogen sources, and of instrumentation to monitor the course of events in the unlikely event of a loss-of-coolant accident.

Other problems related to large water reactors have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports. The Committee feels that resolution of these items should apply equally to the Hutchinson Island Plant.

The Committee believes that the above items can be resolved during construction and that, if due consideration is given to these items, the nuclear plant proposed for the Hutchinson 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,

/s/

Joseph M. Hendrie
Chairman

References attached.

References - Hutchinson Island Plant Unit No. 1

1. Hutchinson Island Plant Unit No. 1, Preliminary Safety Analysis Report, Volumes 1 - 3.

2. Florida Power & Light Company letter, dated April 1, 1969.

3. Amendments 1 - 8 to License Application.

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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

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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 ($\nearrow$ 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 liner, 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 $+0.5 \times 10^{-4} \Delta$ 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,

Original Signed by
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

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

January 17, 1968

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

Subject: REPORT ON THREE MILE ISLAND NUCLEAR STATION UNIT 1

Dear Dr. Seaborg:

At its ninety-third meeting, January 11-13, 1968, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Metropolitan Edison Company to construct Three Mile Island Nuclear Station Unit 1. This project had been considered previously at Subcommittee meetings held on January 4, 1968, in Washington, D. C., and on October 19, 1967, in Hershey, Pa. During its review, the Committee had the benefit of discussions with representatives and consultants of the Metropolitan Edison Company, the Babcock and Wilcox Company, Gilbert Associates, Inc., and the AEC Regulatory Staff. The Committee also had available the documents listed below.

The station is located on Three Mile Island near the east shore of the Susquehanna River in Dauphin County, Pennsylvania, about 10 miles southeast of Harrisburg. Unit 1 is a pressurized-water reactor plant, rated at 2452 MWt, and is similar in design to the units already approved for construction at the Duke Power Company's Oconee Nuclear Station. Flood protection is to be provided at the site by suitable earth dikes. Two natural-draft cooling towers are to be used for condenser-water cooling.

The emergency core cooling system (ECCS) includes two core flooding tanks, two independent low-pressure systems, and two independent high-pressure systems. Two separate systems are provided for containment cooling. One system consists of three fan-cooling units, and the other consists of two spray systems. The applicant stated that suitable and periodic component and integrated system tests will be performed on these engineered safety features. To further insure low containment leak rates, a fluid block system and a containment penetration pressurization system are to be provided.

Operation of the ECCS is initiated automatically by redundant low-pressure signals from transducers actuated by pressure in the two primary loops. The Committee recommends that in the interest of diversity another method,

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different in principle from the one proposed, should be added to initiate this function. The diversity thus achieved would enhance the probability that this vital function would be initiated in the unlikely event it is needed.

The output circuit of the proposed reactor protection system consists of a single d-c circuit (bus) fed from two station batteries. Both feeders must be interrupted to de-energize the bus and drop all rods. Failure to interrupt either feeder, or any other event that prevents de-energizing the single bus, will inhibit dropping all the rods. The Committee believes this system can and should be revised to correct the deficiency. The revised design should be provided for review prior to installation of the protection system.

The applicant has proposed using certain signals from protection instruments for control purposes. The Committee believes that control and protection instrumentation should be separated to the fullest extent practicable, and recommends that the applicant explore further the possibility of making safety instrumentation more nearly independent of control functions.

Consideration should be given to the development and utilization of instrumentation for prompt detection of gross failure of a fuel element.

The applicant described the research and development work planned to confirm the final design of the plant. The Committee continues to emphasize the importance of work to assure that fuel-rod failures in loss-of-coolant accidents will not affect significantly the ability of the ECCS to prevent clad melting.

The applicant is continuing studies on the possible use of part-length rods for stabilizing potential xenon oscillations. Solid poison shims will be added to the fuel elements if necessary to make the moderator temperature coefficient more negative at the beginning of core life.

The Regulatory Staff should review the effects of blowdown forces on core internals and the development of appropriate load combinations and deformation limits. The Regulatory Staff should also review analyses of the possible effects upon pressure vessel integrity of thermal shock induced by ECCS operation.

The applicant has proposed core barrel check valves between the hot leg and the cold leg to insure proper operation of the ECCS under all circumstances. Analytical studies indicate that vibrations will not unseat these valves during normal operation. This point should be verified experimentally.

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The Advisory Committee on Reactor Safeguards believes that the various items mentioned can be resolved during construction and that the proposed reactor can be constructed at the Three Mile Island site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/ C. W. Zabel

Carroll W. Zabel
Chairman

References:

1. Metropolitan Edison Company letter, dated May 1, 1967; Application for Reactor Construction Permit and Operating License, Metropolitan Edison Company, Three Mile Island Nuclear Station Unit 1; Preliminary Safety Analysis Report, Vols. 1, 2, and 3.
2. Metropolitan Edison Company letter, dated July 21, 1967; Amendment No. 1 to application.
3. Metropolitan Edison Company letter, dated October 2, 1967; Amendment No. 2 to application, including Supplement No. 1, Safety Analysis Report, Vol. 4.
4. Metropolitan Edison Company letter, dated November 6, 1967; Amendment No. 3 to application, including Supplement No. 2.
5. Metropolitan Edison Company letter, dated December 8, 1967; Amendment No. 4 to application, including Supplement No. 3.
6. Metropolitan Edison Company letter, dated December 22, 1967; Amendment No. 5 to application, including Supplement No. 4.
7. Metropolitan Edison Company letter, dated January 8, 1968; Amendment No. 6 to application.

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

July 11, 1967

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

Subject: REPORT ON OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

Dear Dr. Seaborg:

At its eighty-sixth meeting, on June 8-10, 1967, and its eighty-seventh meeting, on July 6-8, 1967, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Duke Power Company to construct the Oconee Nuclear Station, Units 1, 2, and 3, at a site near Clemson, South Carolina. This project was reviewed by an ACRS Subcommittee on May 2, 1967, at the site and at Clemson, and on May 31 and June 23, 1967, in Washington, D. C. The Committee had the benefit of discussions with representatives of the Duke Power Company and its consultants, The Babcock and Wilcox Company, Bechtel Corporation, and the AEC Regulatory Staff, and of the documents listed.

Each unit of the Oconee Station includes a pressurized-water reactor rated at 2452 MWt. Each unit is to be provided with an emergency core cooling system (ECCS), including two core flooding tanks, three high-pressure injection pumps, and three low-pressure injection and recirculation pumps. The applicant proposes not to operate a unit with a core flooding tank valved off. The Committee recommends that the Regulatory Staff review the detailed design of the ECCS and the analysis of its performance for the entire spectrum of break sizes, as soon as this information is available. In this respect:

1. The Regulatory Staff should review analyses of possible effects, upon pressure-vessel integrity, arising from thermal shock induced by ECCS operation.*

2. The effects of blowdown forces on core and other primary system components should be analyzed more fully as detailed design proceeds.*

3. Further evidence should be obtained to show that fuel-rod failure in loss-of-coolant accidents will not affect significantly the ability of the ECCS to prevent clad melting.*

Dupe of

4. The applicant has proposed adding swing-check valves in the core barrel to ensure obtaining adequate height of cooling water in the core under all circumstances of ECCS operation. This feature should be further reviewed to ensure that no new problems are introduced.

5. The applicant will explore further possibilities for improvement, particularly by diversification, of the instrumentation that initiates ECCS action.

Emergency power sources for the ECCS and other safeguards are: (a) the other Oconee units (each unit can withstand and will be tested to withstand instantaneous loss of load without a reactor trip or a turbine trip); (b) two hydroelectric units at Keowee station less than one mile away, with independent overhead and underground transmission lines; and (c) a gas-turbine unit thirty miles away with independent transmission line, transformer, and switchyard -- all in addition to the usual multiple ties to the power transmission grid. The applicant stated that switching and sequencing of sources, buses, and loads would be such that no single failure would impair system availability.

The applicant stated that the entire primary system of each unit, including the inside and outside of the reactor vessel, will be accessible for inspection over the life of the plant.

The Committee continues to emphasize the importance of quality assurance in fabrication of the primary system as well as inspection during service life, and recommends that the applicant implement those improvements in primary system quality that are practical with current technology.*

The moderator coefficient of reactivity is calculated to be positive at the beginning of core life, for the first core. The applicant is making detailed studies of the effect of this coefficient on the course of postulated accidents; if necessary, the coefficient will be made more negative by the addition of solid poison shims to the core.

Further evidence should be obtained concerning the ability of the fuel to withstand expected transients at the end of its anticipated lifetime.*

The applicant is investigating further the stability margin for xenon oscillations.

The containment structures are similar to those for the Turkey Point reactors previously reviewed. Consideration should be given to improved inspection of welds in the steel liner of such containments, because an acceptance pressurization test does not stress the liner to postulated accident conditions.

Power for the reactor protection systems and the safeguards protection systems for all three units is provided by a system of six batteries, static inverters, and six buses. The same batteries, via other inverters and buses, provide power to the control systems for all three units. The Committee urges the applicant to review the design of these systems with respect to independence of each unit from troubles in the others.

The applicant proposes to construct a submerged earthen weir in the intake canal to assure a heat sink in the event Keowee Reservoir is drawn down excessively. The Committee believes that careful attention is necessary in the design and construction of this weir to avoid hydraulic erosion and soil instability, particularly in case of rapid drawdown.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved by the applicant and the Regulatory Staff during construction of the reactors. On the basis of the foregoing comments, the Committee believes that the proposed Oconee Nuclear 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,

/s/
N. J. Palladino
Chairman

*The Committee believes that these matters are significant for all large water-cooled power reactors, and warrant careful attention.

References:

1. Duke Power Company, Oconee Nuclear Station, Units 1 and 2, Preliminary Safety Analysis Report, Volumes I and II, undated, received December 5, 1966.
2. Amendment No. 1, dated April 1, 1967
3. Amendment No. 2, dated April 18, 1967.
4. Amendment No. 3, dated April 29, 1967.
5. Amendment No. 4, dated May 25, 1967.
6. Amendment No. 5, dated June 16, 1967.

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