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Volume 3
Project Reviews Q-Z

A Compilation of Reports of **The Advisory Committee on Reactor Safeguards**

1957 - 1984

U.S. Nuclear Regulatory Commission

April 1985

ABSTRACT

This six-volume compilation contains over 1000 reports prepared by the Advisory Committee on Reactor Safeguards from September 1957 through December 1984. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and within project name by chronological order. Part 2 categorizes the reports by the most appropriate generic subject area and within subject area by chronological order.

PREFACE

This compilation has been prepared from the ACRS Report Notebooks that are kept in the ACRS Office. The notebooks are divided into two main sections, ACRS reports on specific projects, and ACRS reports on generic subjects. Normally, each report is filed in only one notebook subsection, with some cross referencing when appropriate. In one or two instances, a report is filed in more than one location to assist the notebook users. Every effort has been made to make this compilation as complete as possible, but due to the relative length of time covered by the notebooks and the variations in record keeping procedures, it is possible that some reports may have been inadvertently omitted.

This compilation does not contain ACRS reports that contain classified or other controlled information.

FOREWORD

The Advisory Committee on Reactor Safeguards (ACRS) was created in 1953 to provide advice to the Atomic Energy Commission (AEC) on the safety of reactor systems being developed at the onset of the era of civilian use of nuclear energy. In 1957, the ACRS was established as a statutory advisory body by the Atomic Energy Act of that year.

The ACRS has continued for over 30 years in this advisory role to the AEC and its successor in reactor safety regulation, the Nuclear Regulatory Commission (NRC). The Committee has played a central role in the development of safety standards and practices as nuclear power has grown from a glamorous scientific curiosity to a huge industry, beset with not only pains of spectacularly rapid initial growth, but also subsequent public disenchantment and controversy. The influence of the ACRS has been projected in a number of ways; through its direct contact with the AEC/NRC technical staff, the industry, the national laboratories, and the universities, and, since 1973, especially through its public meetings. However, its formal advice is given in the form of letter reports to the Commission it advises. These reports, expressing the collegial opinion of the 15-member ACRS have covered a wide variety of subjects, from statements of approval, often with caveats, for the licenses for every plant in the nation, to comments on significant technical issues. Some of the reports have been landmarks and have had a major influence on the development of nuclear power and of the safety of nuclear power. Most have been more mundane and served principally to help keep the regulatory system moving along a fair and responsible course. A few may have been unwise, and better forgotten or rescinded. But, we believe these reports, taken as a whole, provide an interesting view of the history of nuclear power in the United States and the rest of the world.

On the occasion of the 300th regular meeting of the ACRS, April 11-13, 1985, we have published these volumes of the Committee's collected reports. We trust they will be of value to those interested in the past and the future of the generation of electricity and other practical uses of nuclear power.

David A. Ward
Chairman, ACRS

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

December 14, 1966

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

Subject: REPORT ON QUAD-CITIES STATION, UNITS 1 AND 2

Dear Dr. Seaborg:

At its eightieth meeting, on December 8-10, 1966, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Commonwealth Edison Company to construct two boiling water reactors at its Quad-Cities site near Cordova, Illinois. Each unit would be operated at a power level of 2255 MW(t) and be substantially similar in design to the previously reviewed Dresden Units 2 and 3. The Committee had the benefit of discussion with representatives of the applicant, the General Electric Company, Sargent & Lundy, and the AEC Regulatory Staff, and of the documents listed. An ACRS Subcommittee met to review this project at Chicago, Illinois on September 16, 1966 and on November 17, 1966. The Subcommittee visited the site during the September 16 meeting.

The complex of emergency core cooling systems for Quad-Cities is similar to that proposed for Dresden 3. Each reactor includes:

1. A High Pressure Coolant Injection (HPCI) System.
2. A high-volume, low-pressure coolant injection (LPCI) system.
3. Two core spray systems.
4. A system that will make river water available to the feedwater pumps for emergency cooling.

Considerably more information is now available on these systems and they appear to be adequate for the Quad-Cities Reactors. When additional design details become available, it is recommended that the Regulatory Staff satisfy itself with respect to the analyses of system characteristics, including the analysis related to core flooding, and the effects of blowdown on the reactor internals.

As in the case of Dresden 3, the Committee notes that the applicant has made improvements in the requirements for pressure vessel inspection during fabrication.*

The applicant has outlined a general program for periodic inspection of the pressure vessels and other components in the primary systems during the lifetime of the reactors. The Quad-Cities plants have been designed to permit improved accessibility for purposes of inspection of the regions of high stress, such as nozzles and flanges. The Committee may wish to review further the frequency and extent of inspection of the pressure vessels at the time of the request for the operating license.

Steam line isolation valves are an important safeguard in the event of failure of the steam line external to the containment. The Committee recommends that the applicant develop means of testing these valves under simulated accident conditions. These tests should be discussed with the Regulatory Staff.

The Advisory Committee on Reactor Safeguards believes that the various matters mentioned can be resolved during construction and that the proposed reactors can be constructed at the Quad-Cities site with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
David Okrent
Chairman

References Attached.

* The Committee believes that the industry should continue to pursue an orderly program leading to further improvement in the quality of pressure vessels and other components of the primary system such as valves, pumps, and piping.

References - Quad-Cities

1. Quad-Cities Station, Unit 1, Plant Design Analysis, Volumes I and II, undated, received June 6, 1966.

2. Commonwealth Edison Company letter dated August 18, 1966 to AEC Division of Reactor Licensing, with attachments: Amendment No. 1.

3. Commonwealth Edison Company letter dated September 9, 1966 to AEC Division of Reactor Licensing, with attachments: Amendment No. 2.

4. Commonwealth Edison Company letter dated October 18, 1966 to AEC Division of Reactor Licensing, with attachments: Amendment No. 3.

5. General Electric Company letter dated November 1, 1966 to Mr. Brian Grimes, AEC, with enclosure.

6. Commonwealth Edison Company letter dated November 25, 1966 to Dr. Peter A. Morris, AEC Division of Reactor Licensing, with attachments: Amendment No. 4.

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

March 9, 1971

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

Subject: REPORT ON QUAD-CITIES STATION, UNITS 1 AND 2

Dear Dr. Seaborg:

At its 131st meeting, on March 4-6, 1971, the Advisory Committee on Reactor Safeguards reviewed the application by Commonwealth Edison Company and Iowa-Illinois Gas and Electric Company for authorization to operate the Quad-Cities Station Nuclear Units No. 1 and No. 2 at power levels up to 2511 MW(t); the Committee's review for construction was based on a design power of 2255 MW(t). The application was also considered at a Subcommittee meeting held at the site near Cordova, Illinois on March 1, 1971. During its review the Committee had the benefit of discussions with representatives of the applicants, the General Electric Company, Sargent and Lundy, Inc., United Engineers and Constructors, Inc., the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed below. The Committee reported to you on the construction of these units in its letter of December 14, 1966.

Units 1 and 2 of the Quad-Cities Station located near Cordova, Illinois next to the Mississippi River are identical BWRs, substantially similar in design to Dresden Units No. 2 and No. 3. The Dresden units were reviewed for operating licenses at a similar power level; these reviews were reported to you in the Committee's letters of September 10, 1969 and July 17, 1970.

The applicant has estimated that the water level would reach plant grade at the Quad-Cities site in the event of a Mississippi River flood having a discharge of about 585,000 cfs, which exceeds the flood of historic record but is about half the Probable Maximum Flood. In the event of a predicted flood level above plant grade, the applicant proposes to shut down the reactor and to flood necessary portions of the plant in order to maintain structural integrity and enable shutdown heat removal. The Regulatory Staff should assure itself as to the adequacy of the emergency plans prepared to deal with this unlikely event.

The Committee recommends that provisions be made to remove radioactivity from moderate conductivity liquid wastes and that low conductivity liquid wastes be processed for recycle to the reactor cooling system. The Committee also recommends that maximum use be made of all liquid waste treatment systems so that releases to the river are limited to very low levels with regard to both the concentration in the discharge canal and the total amount of radioactivity.

The Quad-Cities units will employ a mixture of gadolinium and uranium oxides in certain fuel rods for reactivity control during the first fuel cycle as a substitute for boron-steel curtains. Analyses by the applicant indicate that the mechanical and thermal characteristics of these rods are acceptable; a surveillance program is planned in order to follow the performance of these rods.

Further studies should be made of the possible effects of a dropped fuel cask on the integrity of the spent fuel pool. Means of reducing damage should be examined and measures taken, if necessary, to provide the needed degree of integrity. This matter should be resolved on a reasonable time scale in a manner satisfactory to the Regulatory Staff.

The Committee recommends that a confirmatory vibration test program be undertaken as part of the start-up and power ascension test program. This matter should be resolved with the Regulatory Staff. It is also recommended that consideration be given to the use, on a developmental basis, of neutron noise measurements, accelerometers, or other devices to provide information concerning the occurrence of excessive vibrations, structural damage, or loose parts. The Committee wishes to support and encourage continuing efforts by the applicant to develop improved methods of inservice pressure vessel inspection.

Conservative pressure-temperature relationships should be established to cover reactor start-up and shut-down. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The containment is penetrated by a large number of small diameter instrument lines. The Committee recommends that special attention be given to assuring the continued integrity of these lines and to a program of periodic examination and testing of the valves in these lines. The applicant should study means to reduce the rate of leakage from instrument lines, in the event of failure, to provide greater assurance that the leakage would not damage the secondary containment or bypass the building filters. The adequacy of measures taken with regard to such instrument lines should be confirmed by the Regulatory Staff.

The applicant has indicated that the biological shield surroundng the reactor vessel can withstand the internal pressure that could be developed by a failure in the region of a nozzle safe-end; in addition, analyses of the effects of possible jet forces of such leaks should be provided to assure that such forces would not lead to failure of the shield with unacceptable consequences.

Provisions have been made to avoid possible damage to the containment if a recirculation line were to fail. The Committee believes that additional analyses should be made by the applicant to show that the unlikely failure of other lines inside the containment would not lead to unacceptable consequences due to pipe whipping. These analyses should be reviewed by the Regulatory Staff.

Performance of the emergency core cooling system has been reevaluated for the effects of possible variations in heat transfer coefficients and other parameters with regard to fuel clad temperatures. Additional studies are underway by the applicant and his contractors to provide further assurance that postulated loss-of-coolant accidents, as analysed with conservative assumptions, will not lead to peak clad temperatures which exceed limits acceptable to the Regulatory Staff. The Committee believes that these studies should be expedited and the matter resolved in a manner satisfactory to the Regulatory Staff prior to routine operation at full power. The Committee wishes to be kept informed.

The Committee reiterates its previous comments, concerning the need to study further means of preventing common mode failures from negating reactor scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. The Committee wishes to be kept informed of the resolution of this matter.

The Committee has commented in previous reports on the development of systems to control the buildup of hydrogen in the containment that might follow in the unlikely event of a loss-of-coolant accident. The applicant proposes to use a purging technique after a suitable time delay subsequent to the accident. The Committee believes that purging capability should be retained, but that the primary protection in this regard may need to utilize a method of hydrogen control other than purging. The applicant should submit, on a reasonable time scale, a proposed design for hydrogen control for review by the Regulatory Staff. The Committee wishes to be kept informed of the resolution of this matter.

The Committee believes the containment should be inerted during operation of the reactor. The Committee recognizes that inerting makes inspection and repair of the primary system more difficult, and believes it acceptable to de-inert during operation just prior to a shutdown and to re-inert during startup and operation following a shutdown. It is recommended that the need for inerting be reviewed periodically as operating experience and further knowledge from current development work are obtained, and as other means of coping with the hazards from accident-generated hydrogen are found.

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 Quad-Cities Station Units 1 and 2 can be operated at power levels up to 2511 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Spencer H. Bush

Spencer H. Bush
Chairman

References Attached

References (Quad-Cities Station)

1. Commonwealth Edison Company letter, dated August 30, 1968 with Safety Analysis Report, Vols. I, II, and III, for Quad-Cities Station

2. Commonwealth Edison Company letter, dated June 16, 1970 with Revised Safety Analysis Report, Vols. I, II, and III, for Quad-Cities Station

3. Amendments 8 through 20 to Safety Analysis Report for Quad-Cities Station

4. Quad-Cities Station Environmental Report, dated November 16, 1970, Commonwealth Edison Company

August 5, 1958

Honorable John A. McCone
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: LOCKHEED RADIATION EFFECTS REACTOR (RER)

Dear Mr. McCone:

The Advisory Committee on Reactor Safeguards at its Ninth Meeting on August 4, 1958, reviewed for the first time the Lockheed Radiation Effects Reactor (RER) proposed for operation at Air Force Plant No. 67 near Gainesville, Georgia. This is a 10 Megawatt water moderated and cooled enriched uranium reactor which will be used in an unshielded position to irradiate large aircraft components with fast neutrons and gamma rays. The operators will be protected by a shielded control room. For protection of the public, reliance is placed exclusively on distance as maintained by a rigidly controlled exclusion area. The reactor will be lowered into a pool of shielding water when it is not in operation.

Since the proposed reactor is of a type that has been operated successfully elsewhere, the Committee feels that purely from the viewpoint of its operation no especially troublesome problems should be encountered. However, there are two elements of risk to the general public in the proposed operation that do not exist in the case of other reactors operating at similar power levels. The Committee is not prepared to say that the risk is large or unacceptable but it certainly is not negligible. In operating an unshielded reactor at the proposed 10 Megawatt power level a substantial quantity of Carbon 14 and other radioisotopes will be formed in the surrounding atmosphere and soil. Some of this undoubtedly will be scattered around locally, especially if high winds or a tornado should sweep the area. The second element of risk stems from the fact that if a nuclear excursion, followed by a metal-water reaction, should occur when the reactor is suspended in its unshielded location the scatter of the fission products probably would be considerably greater than would be the case for a similar reactor surrounded by a heavy concrete shield. No containment of any kind is provided in the proposed reactor operation.

If the additional risks entailed in operating a reactor at substantial power in an unshielded position are compensated by the prospect of getting important information that cannot be obtained by safer methods then, in the present instance, these additional risks may be considered acceptable. Within its limited knowledge of radiation damage effects the Committee is doubtful that the benefits of irradiating large scale aircraft components will compensate the additional risks of doing so by the proposed method, but is willing to defer to superior judgment in this regard.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman

cc: Paul F. Foster, GM
H. L. Price, DL&R

References:

1) IAC 147 - Radiation Effects Reactor Safeguards (Hazards Summary Report, 15 January 1958.)

2) Lockheed Nuclear Products Emergency Manual, AFP No. 67, May 1958.

3) Memorandum of July 16, 1958, to Lt. Col. Fisher, AEC, from Capt. John E. Lineberger, RDZNMN, Marietta, Georgia.
Subject: Transmittal of Meteorological Data.

4) Wind Rose charts (5) for REF Valley, 2 July 58, C. R. Englund, Area Monitoring Stations.

5) LAC-RER Maximum Credible Accident Calculation Method (2 pages - undated) by Georgia Division, Lockheed Aircraft Corporation.

6) RER Operating Permissives for Over Flying Aircraft (2 pages - undated) by Georgia Division, Lockheed Aircraft Corporation.

7) Argon Dose in a Finite Volume (4 pages - undated) by Georgia Division, Lockheed Aircraft Corporation.

8) The Problem of Argon Diffusion of the Radiation Effects Reactor (8 pages - undated) by Georgia Division, Lockheed Aircraft Corporation.

9) Report to ACRS by Division of Licensing and Regulation on Lockheed Radiation Effects Reactor (RER), July 25, 1958.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

May 18, 1959

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: LOCKHEED RADIATION EFFECTS REACTOR (RER)

Dear Mr. McCone:

At the request of the Atomic Energy Commission the Advisory Committee on Reactor Safeguards reconsidered the operation of the Lockheed Radiation Effects Reactor. The specific question asked was whether this reactor can be operated for a period of 100 hours at power levels not to exceed 10 Mw without undue risk to the health and safety of the public.

The Committee has been informed that this operation is divorced from consideration of any experimental programs to be eventually developed in connection with the reactor, since these are not presently proposed by the Air Force.

While the Committee does not favor the operation of any reactor in an unshielded condition it is of the opinion that the limited duration of the proposed test operation does not pose any significant hazard to the health and safety of the public provided adequate surveillance is exercised over local radiation levels.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman

References:

1. LAC 147 - Radiation Effects Reactor Safeguards (Hazards Summary Report, 15 January 1958).

2. LAC-RER Maximum Credible Accident Calculation Method (2 pages - undated) by Georgia Division, Lockheed Aircraft Corporation.

3. Report to ACRS by Division of Licensing and Regulation on Lockheed Radiation Effects Reactor (RER), July 25, 1958.

4. LAC-RER letter report of April 30, 1959 (Georgia Division) Subject: Additional Information on Safeguards Aspects of the Radiation Effects Reactor.

5. Report to ACRS by Division of Licensing and Regulation on Lockheed Radiation Effects Reactor, May 12, 1959.

6. Letter from Dr. C. K. Beck, HEB, to Dr. McCullough, ACRS dated May 15, 1959.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
United States Atomic Energy Commission
Washington 25, D. C.

July 25, 1959

Honorable John A. McCone
Chairman
U.S. Atomic Energy Commission
Washington 25, D. C.

Subject: RADIATION EFFECTS REACTOR (RER)

Dear Mr. McCone:

At its Seventeenth Meeting, July 23-25, 1959, the Advisory Committee on Reactor Safeguards considered the Radiation Effects Reactor which is to be operated by Lockheed for the Air Force. Because of the nature of the planned experimental program this reactor is essentially uncontained and unshielded. The present facility design is such that in order to permit personnel access to the operating area the reactor must be stopped and started frequently thereby increasing the risk. Nevertheless, in the opinion of the Committee, operation of the reactor at this isolated site appears to be acceptable.

We find, however, that the controls and instrumentation are not yet suitable for a 10 megawatt reactor with the above characteristics. Power levels should be restricted to 1 megawatt until this situation is remedied.

The Committee is concerned with the problem of a new and inexperienced laboratory and feels that some months of experience should be accumulated at power levels not to exceed 3 megawatts thermal before routine operation at 10 megawatts. It would be very desirable that this restriction to reduced power operation should continue at least until inspections comparable in objectives to those of the Atomic Energy Commission have been arranged.

After the above conditions have been met, the Committee believes that the reactor may be operated at 10 megawatts thermal without undue risk to the health and safety of the public.

Sincerely yours,

C. Rogers McCullough
Chairman

cc: A.R.Luedecke, GM
H.L.Price, DL&R

<u>References:</u>

1) LAC 147 - Radiation Effects Reactor Safeguards (Hazards Summary Report, 15 January 1958)

2) LAC Memorandum dated 5 April 1959 - "Result Summary of Air, Dust, and Soil Measurements made during One Megawatt Operation of the Radiation Effects Reactor".

3) LAC Memorandum (ANP 59-596) with Enclosures "A" through "E" - received by ACRS 30 June 1959:

   Enclosure A - Maximum Credible Accident Re-Evaluation
   " B - Safety Features of the Lockheed Radiation Effects Reactor
   " C - Consequences of a Power Overshoot in RER
   " D - Operational Summary - Radiation Effects Reactor
   " E - GNL Test Articles

   <u>Note:</u> Changes to Enclosure "B" received by ACRS 9 July 1959 - Reactivity Insertion Rate in RER.

4) LAC Memorandum of 1 July 1959 - "Transmittal of Safeguards Information on Lockheed Radiation Effects Reactor" received by ACRS 9 July 1959.

5) DLR Report to ACRS on the Lockheed Radiation Effects Reactor, July 8, 1959.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

December 10, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON RADIATION EFFECTS REACTOR (RER)

Dear Mr. McCone:

At its thirtieth meeting on December 7-10, 1960, the Advisory Committee on Reactor Safeguards considered the Lockheed Radiation Effects Reactor located near Dawsonville, Georgia. The Committee had available to it the documents referenced below as well as discussion with representatives of the AEC staff, U. S. Air Force, and the Lockheed Aircraft Corporation. The Committee has commented to you regarding operation of this reactor in letters dated August 5, 1958, May 18, 1959, and July 25, 1959.

At this meeting, the Committee considered the limited operating performance of this facility over the past seventeen months. A revised contractual arrangement was described within which Lockheed will lease RER from the Air Force, rather than operating it under an Air Force R&D contract. Military requirements have diminished to the point where the contractor proposes to operate the reactor commercially, providing bulk radiation testing for other governmental organizations and private manufacturers. The Lockheed Corporation has applied for a facility license and will obtain AEC licenses for its operators as a result of this change.

In its letter of July 25, 1959 to you the Committee expressed concern regarding the instrumentation and lack of operator experience at this facility. These areas appear now to have been satisfactorily resolved.

Previously this reactor was considered under section 91b as a facility to be operated for military purposes. At the present time the Committee is being asked to consider this reactor for a civilian license so that it can be operated for commercial purposes. It is the opinion of the Advisory Committee on Reactor Safeguards that continued operation of this reactor can only be justified for work essential to the national defense.

Sincerely yours,

Sgd/LESLIE SILVERMAN

Leslie Silverman
Chairman

cc: A. R. Luedecke, GM
W. F. Finan, AGMRS
H. L. Price, Dir., DL&R

References:

1. NR 103 Radiation Effects Reactor High Power Test Program, dated Sept. 1960.

2. Application for Facility License with enclosures a, b, and e thru n, dated Sept. 20, 1960.

3. Addendum to Application for Facility License with enclosures (a) thru (i), dated November 25, 1960.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

March 4, 1961

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

Subject: REPORT ON LOCKHEED RADIATION EFFECTS REACTOR (RER)

Dear Dr. Seaborg:

At its thirty-second meeting on March 2, 3 and 4, 1961, the Advisory Committee on Reactor Safeguards again reviewed the Lockheed Radiation Effects Reactor at the request of the Division of Licensing and Regulation, letter from R. L. Kirk to T. J. Thompson, February 28, 1961. The Committee received comments from representatives of the Lockheed Aircraft Corporation, the Aircraft Nuclear Propulsion Office of the Division of Reactor Development, and the Division of Licensing and Regulation. At this time the ACRS sees no reason to change its opinion, previously expressed in a letter to the Chairman of the Commission, Dec. 10, 1960, that continued operation of this reactor at 10 MW thermal power in its present design form can only be justified for work essential to the national defense. The Committee believes that the hazards to the public from operating this reactor at its designed power and in its present unshielded and uncontained condition are greater than those generally acceptable for licensed reactor facilities. Thus there must be compelling reasons for assuming this additional risk.

During the discussion, the applicant indicated his desire to operate the reactor under restricted conditions. Capability for such limited operation appears to be needed in order to retain a competent operating crew. He intends to re-examine whether it would be worthwhile to him to add sufficient confinement and shielding to reduce the hazard at full power operation to acceptable levels. The ACRS concludes that there is reasonable assurance that this reactor can be operated without undue risk to the health and safety of the public provided (1) the power is limited to one megawatt thermal at which level it is highly unlikely that the fuel will melt even if the coolant is lost, and (2) the excess reactivity be limited to the minimum required for operation at this stated power level. It is, of course, assumed that the present procedural safeguards and environmental surveillance will continue.

Sincerely yours,

/s/

T. J. Thompson
Chairman

Reference:
1. Letter from R. L. Kirk (DL&R) to T. J. Thompson (ACRS), dated February 28, 1961.

cc: A. R. Luedecke, GM
W. F. Finan, ACMRS
H. L. Price, Dir., DL&R

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

July 18, 1963

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

SUBJECT: REPORT ON LOCKHEED RADIATION EFFECTS REACTOR (RER)

Dear Dr. Seaborg:

At its forty-eighth meeting, at Los Alamos, New Mexico on July 11-13, 1963, the Advisory Committee on Reactor Safeguards considered the request of the Lockheed-Georgia Company to increase the power of the Radiation Effects Reactor from one megawatt to three megawatts thermal. The Committee previously considered the one megawatt operation of this reactor and reported to the Commission following its thirty-second meeting. In the present review, the Committee had the benefit of the documents listed below and discussions with representatives of the Lockheed-Georgia Company and the AEC Staff.

This reactor can be operated either immersed in a deep pool of water or as an unshielded reactor above the pool surface. Operation above the pool surface permits the neutron and gamma irradiation of large samples which can be moved to the unshielded reactor on a movable platform. The licensee does not intend to irradiate samples within the core.

Representatives of the Lockheed-Georgia Company have stated that, to date, there has been no evidence of attempts by the general public to enter the exclusion area. Operations to date have caused no overexposure of operating personnel and no excessive radiation levels have been observed at the inner exclusion fence. The operating group has stated their intention to carry out refueling operations with the reactor immersed in the pool at a depth of approximately twenty feet to minimize the consequences of any postulated refueling accident. The licensee does not now propose to irradiate explosive materials. The Committee has been assured that the licensee will review with the AEC Regulatory Staff any proposal to irradiate potentially explosive materials.

The proposed increase in power to three megawatts places additional emphasis on the reliability of the cooling water supply to the core. Experiments elsewhere have shown that it is highly unlikely that the type of fuel used in this reactor will melt even if all water coolant is lost immediately after steady operation at one megawatt. Some melting may occur if coolant is suddenly lost immediately after three megawatt operation. The Committee suggests that due attention be given to the reliability and adequacy of coolant supply to the core under all conditions of operation. In addition, the Committee suggests that the available excess reactivity be limited to that required for three megawatt operation and that continuing attention be given to procedural safeguards and environmental surveillance.

With proper consideration given to the comments above, the Committee believes that the licensee can operate the facility at powers up to three megawatts thermal as proposed without undue risk to the health and safety of the general public.

Sincerely yours,

/s/ D. B. Hall

D. B. Hall
Chairman

References:
1. LNP/10331, Amendment No. 6 to License R-86, dated February 8, 1963.

2. LGD/162926, Supplement to Amendment Request to License R-86, dated June 17, 1963.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

July 15, 1964

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

Subject: REPORT ON RADIATION EFFECTS REACTOR

Dear Dr. Seaborg:

At its fifty-sixth meeting, July 9-11, 1964, at Brookhaven National Laboratory, the Advisory Committee on Reactor Safeguards reviewed a proposal by the Lockheed-Georgia Company that the Radiation Effects Reactor be used in irradiation tests of liquid hydrogen cooled materials. The Committee review included discussions with representatives of the Lockheed-Georgia Company and of the AEC Staff, and made use of the documents referenced below.

The Radiation Effects Reactor was previously considered at the Committee's thirty-second meeting in March 1961, and at its forty-eighth meeting in July 1963. These reviews led to letters recommending approval of operation at maximum reactor powers of 1 MW and 3 MW, respectively, provided that significant amounts of potentially explosive material not be irradiated without separate consideration. The present proposal is submitted in accordance with that recommendation.

The experimental design proposed involves the use of up to 1000 gallons of liquid hydrogen at a time, in close proximity to the Radiation Effects Reactor. While this is a substantial amount of potentially explosive material, such quantities have been used in other applications such as rocket propulsion. Design features that have been developed for the safe handling of liquid hydrogen are being used by the applicant. These features include double and triple containment of the liquid hydrogen bearing components, with barriers of vacuum and inert atmosphere between hydrogen and air, and an abundance of instruments to warn of the onset of potentially hazardous situations such as fire, excessive pressure buildup, and liquid hydrogen leaks. Those situations that could conceivably injure the reactor if allowed to persist lead to shutting down the reactor automatically and submerging it in its pool.

In addition, the applicant has considered the result of unlikely accidents such as massive hydrogen leaks into the reactor building, with resultant explosion while the reactor is still exposed, and has concluded that, even in this event, the damage to the reactor would not be serious.

The applicant has stated that, for the present, the tests planned in accordance with this proposal will consist of irradiations of stationary samples such as insulating material. Proposals for similar tests involving vibration or rotary motion, and irradiations of materials that could possibly liberate appreciable amounts of oxidizing agents will be submitted separately to the AEC for review.

The Committee concludes that, if planned tests of capsule integrity under large hydrogen leaks from the test tank are successful, the liquid hydrogen cooled irradiations can be performed as proposed without undue hazard to the health and safety of the public.

Sincerely yours,

/s/
Herbert Kouts
Chairman

References:

1. LGD/173219, "Amendment Request No. 9 to License R-86 - Use of Liquid Hydrogen in Radiation Effects Reactor, Docket No. 50-172", dated December 17, 1963, with enclosures (a) through (c).
2. LGD/182962, "Use of Liquid Hydrogen in Radiation Effects Reactor, Docket No. 50-172", dated May 28, 1964, with enclosures.
3. LGD/183397, "Use of Liquid Hydrogen at Radiation Effects Reactor, Docket No. 50-172", dated June 4, 1964, with enclosure.
4. LGD/184106, "Use of Liquid Hydrogen at Radiation Effects Reactor, Docket No. 50-172", dated June 16, 1964, with enclosure.
5. LGD/184511, "Use of Liquid Hydrogen at Radiation Effects Reactor, Docket No. 50-172", dated June 22, 1964.

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

July 19, 1968

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

Subject: REPORT ON RANCHO SECO NUCLEAR GENERATING STATION, UNIT NO. 1

Dear Dr. Seaborg:

During its ninety-ninth meeting, July 11-13, 1968, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Sacramento Municipal Utility District to construct the Rancho Seco Nuclear Generating Station, Unit No. 1. This project had been considered previously during Subcommittee meetings on April 23, 1968, at the site, and on June 28, 1968, in Washington, D. C. In the course of its review, the Committee had the benefit of discussions with representatives and consultants of the Sacramento Municipal Utility District, the Babcock and Wilcox Company, Bechtel Corporation, and the AEC Regulatory Staff. The Committee also had available the documents listed below.

This 2452 MWt pressurized water reactor will be located about 25 miles southeast of Sacramento, California, in a sparsely populated area. This region of California is seismically relatively inactive; the largest earthquake of historic record in the vicinity of the site is of Intensity VI, Modified Mercalli (MM) scale. The applicant has agreed to design for safe shutdown following an earthquake during which the maximum horizontal acceleration is 0.25 g (MM VIII), and the design will allow continued operation for an earthquake of about one-half of this acceleration. He plans to install a strong motion accelerograph.

All water needs for this plant will be supplied from the Folsom South Canal, which will pass within five miles of the site. Should completion of this canal be delayed, a separate pipeline from Lake Natoma, about 20 miles north of the site will be constructed. An on-site reservoir will have a capacity of 2500 acre-feet, sufficient for about 35 full power days of operation, and waste heat will be discharged to the atmosphere through use of cooling towers. The plant is unique in that the applicant proposes not to discharge liquid wastes to the environment. The applicant is studying methods to cope with possible build-up of tritium in the reactor coolant water.

The applicant has proposed using signals from the protection system for control and protection purposes. The Committee reiterates its belief that control and protection instrumentation should be as nearly independent of common failure modes as possible, so that the protection will not be impaired by the same failure that initiates a transient requiring protection. The applicant and the AEC Regulatory Staff should review the proposed design 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. In cases where hypothesized control or override failure could lead to the need for action by interconnected protection instrumentation, separate protection instrumentation channels should be provided or some other design approach used to provide equivalent safety.

The Committee suggests that, in view of possible uncertainties in current predictive techniques, further analyses be made of the anticipated integrated fast flux at the pressure vessel wall, and that the adequacy of the proposed pressure vessel material surveillance program be resolved between the applicant and the Regulatory Staff during construction of the station.

This reactor is similar to others designed by this vendor and reviewed previously (see, for example, the ACRS report on the Crystal River plant, May 15, 1968). The Committee continues to call attention to matters that warrant careful consideration by the manufacturers of all large, water-cooled, power reactors. These matters, referred to in the above-mentioned report, apply similarly to the Rancho Seco project.

The Advisory Committee on Reactor Safeguards believes that the items noted above can be resolved during construction, and that the proposed plant can be built at the rancho Seco site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/

Carroll W. Zabel
Chairman

References attached.

<u>References - Rancho Seco</u>

1. License Application for Construction Permit, Sacramento Municipal Utility District, dated November, 1967; Volumes I, II, III, IV of the Preliminary Safety Analysis Report for Rancho Seco Nuclear Generating Station, Unit No. 1

2. Sacramento Municipal Utility District; Amendment No. 1, dated February 2, 1968

3. Sacramento Municipal Utility District; Amendment No. 2, dated April 15, 1968

4. Sacramento Municipal Utility District; Amendment No. 3, dated May 30, 1968

5. Sacramento Municipal Utility District; Amendment No. 4, dated June 30, 1968

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

September 11, 1973

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

Subject: REPORT ON RANCHO SECO NUCLEAR GENERATING STATION, UNIT 1

Dear Dr. Ray:

During its 161st meeting, September 6-8, 1973, the Advisory Committee on Reactor Safeguards reviewed the application of the Sacramento Municipal Utility District for a license to operate the Rancho Seco Nuclear Generating Station, Unit 1, at power levels up to 2772 MW(t). This project had been considered previously during the 159th meeting of the ACRS, July 12-14, 1973, by Subcommittee meetings in Sacramento, California, on June 13 and 14, 1973, subsequent to a tour of the site, and in Washington, D. C., on August 22, 1973. In the course of its review, the Committee had the benefit of discussions with representatives and consultants of the Sacramento Municipal Utility District, the Babcock and Wilcox Company, the Bechtel Corporation, and the AEC Regulatory Staff, and of the documents listed. The Committee last reported to the Commission on the construction of this plant in its letter of July 19, 1968.

The Rancho Seco Nuclear Generating Station is located about 25 miles southeast of Sacramento, California. Water for this plant will be supplied from the Folsom South Canal. An on-site reservoir will have a capacity of 2500 acre-feet, and two spray ponds can provide cooling water for decay heat removal for about 30 days.

The Rancho Seco nuclear steam supply system employs a Babcock and Wilcox two-loop, pressurized water reactor essentially identical in design to the Oconee Nuclear Station Unit No. 1, previously reported on by the Committee. However, Rancho Seco will operate at approximately 8% higher power level and will use control of boron concentration in the core cooling water to aid in reactivity control during power maneuvering.

The application for a construction permit proposed initial operation at power levels up to 2452 MW(t), the same as the construction permit power level of the Oconee Nuclear Station, Unit 1 which employs a similar reactor. The safety analyses have been completed assuming a power of 2568 MW(t). The application for an operating license proposed power levels up to 2772 MW(t) and safety studies have been made at this power. This increase in power is accomplished by utilizing larger primary coolant pumps and by increasing the average coolant temperature rise in the core. The Committee believes that review of the operation of Oconee Nuclear Station, Unit 1 by the Regulatory Staff should be completed and satisfactory performance of Oconee Nuclear Station, Unit 1 should be demonstrated before Rancho Seco Unit 1 is operated at full power. In addition, the Committee agrees with the Regulatory Staff that it would be prudent for Rancho Seco Unit 1 to operate at power levels up to 2568 MW(t) for an appropriate time period and for the Staff and the ACRS to review this experience prior to allowing operation at full power of 2772 MW(t). Independent confirmation by the Regulatory Staff of the applicant's analyses of linear heat generation rates, operating limits, and ECCS efficacy, and submittal of a supplemental Staff Safety Evaluation Report should precede this review for operation at full power.

Fuel for the reactor has been thermally resintered with the purpose of reducing fuel densification under irradiation; furthermore, the fuel assemblies are being classified according to their maximum allowable linear heat rate and are to be loaded into the reactor according to this classification. This matter should be resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The applicant has stated that, under normal conditions, reactor produced radioactive liquid wastes will not be released to the environment. This will be accomplished primarily through processing and reuse of liquids removed from various reactor systems. The Committee believes that the effects of gradual buildup of tritium in liquids within the plant should be carefully evaluated. Factors to be assessed include potential increases in radiation exposures of operating personnel, possible difficulties in proper plant maintenance, and the possible influence of increased tritium concentrations on the consequences of unanticipated releases.

During the hot functional testing of Oconee Nuclear Station, Unit 1 which was conducted in 1972, damage occurred to some components, including reactor vessel internals. The design improvements made to Oconee Nuclear Station, Unit 1 have been made also to Rancho Seco Unit 1. The Committee believes that these changes are acceptable.

The applicant has been responsive to the Committee's recommendation that suitable instrumentation be sought to monitor for loose parts and for vibration; such instrumentation has been designed and will be utilized.

The applicant has proposed appropriate operating limitations to be applied if, at any time during operation, the moderator temperature coefficient of reactivity is positive. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Regulatory Staff has been investigating on a generic basis the problems associated with a potential reactor coolant pump overspeed in the unlikely event of a particular type of rupture at certain locations in a main coolant pipe. Some additional protective measures may be warranted, and this matter should be resolved to the satisfaction of the Regulatory Staff. The Committee wishes to be kept informed.

The Committee reiterates its previous comments on the need for further study of means for preventing common mode failures from negating reactor scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. The Committee wishes to be kept informed of the resolution of this matter.

The applicant should assure himself that instrumentation for determining the course of potentially serious accidents, on a time scale that will permit appropriate emergency action, is provided at the station and that appropriate calibration methods and calculated bases for interpreting instrument responses are available.

In view of the important role of the applicant's Management Safety Review Committee in providing continuing reviews, and in updating and implementing safety measures, the ACRS recommends that the Management Safety Review Committee include additional experienced personnel from outside the corporate structure as voting members.

The applicant has proposed measures, including alarms and administrative procedures, to prevent operating under conditions which might result in exceeding acceptable fuel limits established from accident studies and other considerations. The current review has been confined to the first fuel cycle, and the analyses have been based on the as-built fuel. The ACRS recommends that the Regulatory Staff establish suitable criteria for these measures and provide suitable bases for evaluating future loadings. The Committee wishes to be kept informed.

The Committee recognizes that re-evaluation of operating limits may be necessary as a result of possible changes in the acceptance criteria for emergency core cooling systems. The Committee wishes to be kept informed.

Other problems relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous reports should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

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 preoperational testing, there is reasonable assurance that Rancho Seco Nuclear Generating Station, Unit 1 can be operated at power levels up to 2772 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

Attachment: List of References

References

1. Sacramento Municipal Utility District (SMUD) Safety Analysis Report for Rancho Seco Nuclear Generating Station, Unit 1, Vols. I-V, May, 1971 and Vol. VI, June, 1972

2. Amendments 6 through 23 to SMUD License Application for Rancho Seco

3. Letter from E. K. Davis, SMUD, to A. Giambusso, L, dated March 23, 1973, "Final Report on Minor Imperfections Found in Pipe Welds at the Rancho Seco Nuclear Generating Station"

4. Letter from E. K. Davis, SMUD to A. Giambusso, L, dated April 3, 1973, "Interim Report on Fuel Densification"

5. Letter from E. K. Davis, SMUD, to A. Giambusso, L, dated May 1, 1973, "Interim Report on Effects of Piping Break Outside Containment"

6. Letter from E. K. Davis, SMUD, to A. Schwencer, L, dated May 3, 1973, "Review of Control Circuits"

7. Directorate of Licensing Safety Evaluation, June 8, 1973

8. Letter from H. W. Ibser, Professor of Physics, California State University to M. Libarkin, ACRS, dated June 18, 1973, concerning temperature inversions at Rancho Seco

9. Babcock and Wilcox Proprietary Report, BAW-1393, "Rancho Seco Unit 1 Fuel Densification Report," June, 1973 with supplemental information containing as-built data forwarded by letter from E. K. Davis, SMUD, to A. Giambusso, L, dated July 23, 1973

10. Report, "Rancho Seco Nuclear Service Spray Ponds Performance Evaluation," dated June 29, 1973 by the Waste Heat Management Research Project, University of California, Berkeley

11. Directorate of Licensing Technical Report on Densification of B&W Reactor Fuel, dated July 6, 1973

12. Letter from E. K. Davis, SMUD, to A. Giambusso, L, dated August 2, 1973, submitting changes to the FSAR, and the control scheme for emergency diesel engines.

July 16, 1975

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

Subject: REPORT ON RANCHO SECO NUCLEAR GENERATING STATION, UNIT 1

Dear Mr. Anders:

During its 183rd meeting, July 10-12, 1975, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Sacramento Municipal Utility District to increase the power level of the Rancho Seco Nuclear Generating Station, Unit 1, from 2568 MW(t) to 2772 MW(t). This proposal was also considered at a Subcommittee meeting in Washington, D. C. on June 24, 1975. In the course of its review, the Committee had the benefit of discussions with representatives and consultants of the Sacramento Municipal Utility District, the Babcock and Wilcox Company (B&W), and the NRC Staff, and of the documents listed.

The Committee last reported on the operation of this plant in its letter of September 11, 1973. At that time, the Committee recommended that three conditions be satisfied before the plant was permitted to operate at its design power level of 2772 MW(t).

a. Operation of Unit 1 of the Oconee Nuclear Station, the prototype for Rancho Seco Unit 1, should be reviewed and found satisfactory by the Regulatory Staff.

b. Operation of Rancho Seco Unit 1 at 2568 MW(t) should be reviewed by the Regulatory Staff and the ACRS and determined to be satisfactory.

c. The Regulatory Staff should perform an independent confirmation of the licensee's linear heat generation rates, operating limits and ECCS efficacy, and should submit a supplemental Safety Evaluation Report.

These three conditions have been satisfied.

The applicant is using control of the boron concentration in the core cooling water as an aid for reactivity control during power maneuvering. During the past several months, plant operators have applied this system in a variety of power maneuvers. Operating experience with the system to date has been acceptable.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to those items mentioned in its report of September 11, 1973, there is reasonable assurance that Rancho Seco Nuclear Generating Station, Unit 1, can be operated at design power (2772 MW(t)) without undue risk to the health and safety of the public.

Sincerely yours,

/s/ W. Kerr

W. Kerr
Chairman

References:

1. Supplement No. 1 to the Safety Evaluation by the Directorate of Licensing, U. S. Atomic Energy Commission in the matter of Sacramento Municipal Utility District, Rancho Seco Nuclear Generating Station, Unit No. 1, Docket No. 50-312, Nov. 28, 1973.

2. Supplement No. 2 to the Safety Evaluation by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission in the matter of Sacramento Minicipal Utility District, Rancho Seco Nuclear Generating Station Unit No. 1, Docket No. 50-312, June 10, 1975.

3. Safety Evaluation Report by the Directorate of Licensing, U. S. Atomic Energy Commission in the matter of Scramento Municipal Utility District, Rancho Seco Nuclear Generating Station Unit No. 1, Docket No. 50-312, December 27, 1974.

4. Letter from E. K. Davis, General Manager, Sacramento Municipal Utility District of March 12, 1975 to Mr. Angelo Giambusso, Director, Division of Reactor Licensing, NRC, Re: NRC Docket No. 50-312, Proposed Technical Specification Change No. 1, Rancho Seco Nuclear Generating Station, Unit No. 1.

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

January 14, 1975

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

Subject: REPORT ON RIVER BEND STATION, UNITS 1 AND 2

Dear Dr. Ray:

At its 177th meeting on January 9-11, 1975, the Advisory Committee on Reactor Safeguards completed its review of the application of the Gulf States Utilities Company for a permit to construct River Bend Station, Units 1 and 2. The Committee also considered this application during its 174th meeting on October 10-12, 1974. Members of the Committee visited the site on September 20, 1974, and Subcommittee meetings were held on September 21, 1974 in St. Francisville, Louisiana and on January 6, 1975 in Washington, D. C. In its review the Committee had the benefit of discussions with the AEC Regulatory Staff, representatives and consultants of the Applicant, General Electric Company and Stone and Webster Engineering Corporation. The Committee also had the benefit of the documents listed below.

The River Bend Station will be located at a 3292 acre site on the east bank of the Mississippi River approximately 24 miles north-northeast of Baton Rouge, Louisiana, which has been identified as the nearest population center. In 1970, the population of Baton Rouge was 165,963 and the population within 50 miles of the site was 626,373.

The River Bend Station consists of two nuclear units, each using a General Electric BWR/6 nuclear steam supply system having a rated power level of 2894 MW(t) and containing 592 fuel assemblies. 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. Unit 1 will be the first BWR/6 having a pressure vessel with an internal diameter of 218 inches, and as such will undergo an extensive, preoperational vibration testing program. The Committee wishes to be kept informed of the results of the tests and any significant problems encountered.

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 its consultant be used to evaluate the sensitivity of key design parameters, and to elucidate 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.

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.

Doses resulting from Design Basis Accidents are being evaluated by the Applicant and by the Regulatory Staff using the results of onsite meteorological measurements. The method of application of these results to the calculational model must be resolved before it can be determined that the doses are within construction permit guidelines. This matter should be resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

A Regulatory Staff requirement, which has become a generic issue, pertains to designing the radioactive offgas system, including the absorption 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 Committe 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 justification 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 acknowledges that the Regulatory Staff is reviewing 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 (RHR) 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 the overall generic problem has not been made, including both RHR availability and the potential loss of isolation between high and low pressure systems. 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 River Bend units meets the Interim Acceptance Criteria of June 1971. In addition, the Applicants' 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 5) suggests a need for the use of three-dimensional calculations to better 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.

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 River Bend 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.

Sincerely yours,

/s/ W. Kerr

W. Kerr
Chairman

References:
Listed on page 4

References:

1. Preliminary Safety Analysis Report, Volumes 1-9, for River Bend Station

2. Amendments 1-10 to PSAR

3. Gulf States Utilities Company letters and reports

    a. "LOCTVS Vent Clearing Model for Horizontal Vent Vapor Suppression Containment," March 3, 1974

    b. April 4, 1974 letter pertaining to offsite dose calculations

    c. April 23, 1974 letter relating to Mark III containment

    d. April 26, 1974 letter concerning soils investigation

    e. May 3, 1974 letter providing additional information on cloud depletion

    f. May 20, 1974 letter submitting a description of the proposed ground level tracer test program

    g. June 28, 1974 letter submitting a revised description of the proposed ground level tracer test program

    h. July 7, 1974 letter presenting and discussing logs and other records of petroleum exploration on the plant site area

    i. August 20, 1974 letter transmitting containment analysis information: "GE and Test Comparisons"

    j. October 21, 1974 letter and formal test report on ground level tracer tests at the site

4. AEC Directorate of Licensing reports and letters

    a. April 2, 1974 letter relating to Mark III containment

    b. "Safety Evaluation," September 24, 1974

    c. Supplement No. 1 to "Safety Evaluation," December 9, 1974

5. "Comparison of Two- and Three-Dimensional Calculations of Super Prompt Critical Excursions," A. Birkhofer, A. Schmidt, and W. Werner, Nuclear Technology, v24, p. 7-12, October 1974

References: (cont.)

6. Written Statement by Lola H. Broadbent, dated September 21, 1974 w/attachment dated January 10, 1974

7. Written Statement by Robert B. Fisher, Jr., undated (received September 21, 1974)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 17, 1984

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS INTERIM REPORT ON RIVER BEND STATION

During its 291st meeting, July 12-14, 1984, the Advisory Committee on Reactor Safeguards reviewed the application of Gulf States Utilities Company (Applicant), acting on behalf of itself and as agent for the Cajun Electric Power Cooperative, for a license to operate the River Bend Station. A tour of the facilities was made by members of the Subcommittee on the morning of June 7, 1984, and a Subcommittee meeting was held in Baton Rouge, Louisiana on June 7 and 8, 1984 to consider the application. During our review, we had the benefit of discussions with representatives of the Applicant, the NRC Staff, and members of the public. We also had the benefit of the documents referenced. The Committee commented on the construction permit application for this Station in its report dated January 14, 1975.

The River Bend Station is located in West Feliciana Parish, Louisiana on the east side of the Mississippi River approximately 24 miles north-northwest of Baton Rouge. Originally the River Bend Station was to consist of two units. Unit 2 was cancelled on January 5, 1984. Unit 1 is approximately 90% complete, with an estimated fuel load date of April 1985.

The River Bend Station uses a General Electric BWR-6 nuclear steam supply system (NSSS) with a rated core thermal power of 2894 MWt and a Mark III pressure suppression containment system with a design pressure of 15 psig.

The Applicant has structured its organization, and has provided for continuity from project initiation up to and including operation, in a notable manner. This structuring is along project team lines and appears to have provided good control and interfacing among the utility, the general contractor-architect engineer, and the NSSS designer. Further, it appears this structuring has provided this first time nuclear utility with good personnel development for the utility's overall nuclear plant responsibilities. In addition to this, the

Applicant has practiced aggressive recruiting and careful selection of qualified people and has phased them into the project in a timely manner.

The dedicated diesel generator that drives the high pressure core spray pump currently depends on cooling water supplied by pumps powered by the other two diesel generators during loss of offsite power conditions. We recommend that the merit of removing this dependency be examined.

The Applicant stated that they plan to conduct a limited probabilistic risk assessment (PRA) for the River Bend Station. We support the proposal to perform a plant-specific PRA and recommend that it include seismic- and fire-induced accident scenarios.

Although River Bend is in a relatively quiet seismic portion of the country, NRC contractor estimates of the recurrence interval for the safe shutdown earthquake are similar to those for most eastern sites. We recommend that the Applicant review, in detail, the seismic capability of the emergency AC power supplies, the DC power supplies, and small components such as actuators, relays, and instrument lines that are part of the decay heat removal system.

The Applicant has proposed to include in the River Bend Emergency Procedures a procedure for venting the containment under certain accident conditions. The bases for the decision to take this action are not yet clear. The NRC Staff has not completed its review of this proposal. We wish to be advised when the NRC Staff has reached a position on this matter and to have an opportunity to comment generically or specifically.

The NRC Staff has identified a number of license conditions and confirmatory matters, and several outstanding issues which remain to be resolved. Except for the matter of hydrogen control, we are satisfied with progress on the other topics and believe that they should be resolved in a manner satisfactory to the NRC Staff. We have not completed our review of hydrogen control for the River Bend Station, particularly as it may be impacted by differences in containment design features between River Bend and Mark III BWRs previously reviewed.

The Committee will complete its review of the full power operating license when the NRC Staff and the Applicant have made sufficient additional progress in resolving the matter of hydrogen control. In the interim, we believe that if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, the River Bend Station can be operated at power levels up to 5% of full power without undue risk to the health and safety of the public.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

References:
1. Gulf States Utilities Company, "Final Safety Analysis Report, River Bend Station," Volumes 1-18 and Amendments 1-11
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of River Bend Station," NUREG-0989, dated May 1984

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

February 17, 1967

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

Subject: REPORT ON H. B. ROBINSON UNIT NO. 2

Dear Dr. Seaborg:

At its eighty-first meeting, on January 12-14, and its eighty-second meeting, on February 8-11, 1967, the Advisory Committee on Reactor Safeguards completed its review of the application of the Carolina Power and Light Company to construct H. B. Robinson Unit No. 2 near Hartsville, South Carolina. An ACRS Subcommittee met to review this project on December 13, 1966 at Hartsville, S.C., and on February 1, 1967 in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives of the applicant, the Westinghouse Electric Corporation, Ebasco Services, Inc., consultants to these groups, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The unit includes a pressurized water reactor to be operated at 2094 MWt. It will be constructed at the H. B. Robinson Station adjacent to Unit No. 1, an existing coalfired plant. The H. B. Robinson Station is in Darlington County, approximately five miles from Hartsville and 30 miles from Florence, South Carolina. The plant is located on the shore of Lake Robinson just above the dam that impounds the water of Black Creek.

The containment is a cylindrical steel-lined concrete structure with a spherical dome and a flat base-slab. The design will permit pressurization of the containment for test purposes as may be required throughout the life of the plant. The cylindrical wall is prestressed vertically and reinforced circumferentially; the dome and base are reinforced. The tendons will be grouted in place and will not be accessible for surveillance. The applicant plans to prepare samples of similarly prestressed and grouted tendons, and to expose them to the same general environmental conditions as those experienced by the containment tendons. Samples will be available for investigation, as needed, throughout the life of the plant.

The site is in a region of moderate seismic activity and the plant is being designed accordingly. In the event that an earthquake produces significant ground acceleration at this site in the future, an on-site measurement of the shock intensity should provide data valuable in assessing the possibility of hidden structural damage to vital portions of the facility. A strong-motion accelerograph will be installed.*

The Committee notes that the entire primary system will be inspected to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class A. The emergency core cooling systems consist of a high head safety injection system, a low head residual heat removal system, and an accumulator injection system. The applicant states that, for all sizes of pipe ruptures of the primary system, a conservative evaluation of the functioning of engineered safeguards indicates that there will be no clad melting and less than one per cent of clad-water reaction. The emergency containment cooling systems consist of a spray system and a circulating air cooling system. The water injected into the core or containment is borated; in addition, sodium thiosulfate is injected into the containment spray to reduce the iodine content of the atmosphere in the unlikely event of a primary system rupture. Tests are planned to establish the performance and reliability of the sodium thiosulfate injection sub-system. The Committee recommends that the AEC Regulatory Staff review details of the test data and the design of the emergency systems as they become available.

The Committee notes that under certain highly improbable but credible accident conditions the isolation valves in the steam lines may be an important factor in preventing escape of radioactivity. The Committee is of the opinion that a special effort should be made to assure that these valves, and other valves that must be depended on for containment, are effectively tight under accident conditions.

Calculations by the applicant show that the reactor has a positive moderator coefficient at some period of the core life. The applicant is continuing his analysis of all consequences of the positive coefficient and, if necessary, will adjust the core composition to assure safety. The Committee recommends that the Regulatory Staff follow the applicant's studies and conclusions in this respect.

The Committee believes the applicant should store sufficient diesel fuel to permit operation of the emergency diesels as required by accident conditions for a minimum of one week.

*The Committee believes that the installation of a strong-motion accelerograph may be appropriate for most large power reactors, including those located in zones of relative seismic quiesence.

The Advisory Committee on Reactor Safeguards believes that the items mentioned can be resolved during construction and that the proposed reactor can be built at the H. B. Robinson site 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

References:

1. Shaw, Pittman, Potts, Trowbridge & Madden letter dated July 25, 1966 to AEC Division of Reactor Licensing transmitting "Preliminary Facility Description and Safety Analysis Report", Volumes 1-3.
2. Shaw, Pittman, Potts, Trowbridge & Madden letter dated October 10, 1966 to AEC Division of Reactor Licensing transmitting Amendment No. 1.
3. "First Supplement to Preliminary Facility Description and Safety Analysis Report" (Amendment No. 2), dated November 28, 1966.
4. "Second Supplement to Preliminary Facility Description and Safety Analysis Report" (Amendment No. 3), dated December 1, 1966.
5. "Third Supplement to Preliminary Facility Description and Safety Analysis Report" (Amendment No. 4), dated December 1, 1966.
6. "Fourth Supplement to Preliminary Facility Description and Safety Analysis Report" (Amendment No. 5), dated December 1, 1966.
7. Amendment No. 6, "Fifth Supplement to Preliminary Facility Description and Safety Analysis Report", dated January 27, 1967.
8. Amendment No. 7, "Sixth Supplement to Preliminary Facility Description and Safety Analysis Report", dated February 6, 1967.

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

April 16, 1970

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

Subject: REPORT ON H. B. ROBINSON UNIT NO. 2

Dear Dr. Seaborg:

During its 120th meeting, April 9-11, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by the Carolina Power and Light Company for a license to operate the H. B. Robinson Unit No. 2 at power levels up to 2200 MWt. During this review the project was considered at Subcommittee meetings held on January 21, 1970 at the plant site and on March 26, 1970 in Washington, D. C. In the course of these meetings, the Committee had the benefit of discussion with representatives and consultants of the Carolina Power and Light Company, Westinghouse Electric Corporation, Ebasco Services Incorporated, and the AEC Regulatory Staff. 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 February 17, 1967.

The H. B. Robinson site is in northeastern South Carolina about 56 miles from Columbia, South Carolina and consists of more than 5,000 acres including Lake Robinson. The minimum exclusion radius is 1400 feet and the nearest population center with more than 25,000 residents is Florence, South Carolina, approximately 25 miles to the southeast.

The nuclear steam supply system for the H. B. Robinson Unit No. 2 is the first of the three-loop Westinghouse line to be reviewed for operation. The design features are similar to those of the Ginna plant, previously discussed in the Committee's report to you dated May 15, 1969.

The applicant is reviewing his seismic design calculations. The results of this analysis and any corrective actions required should be reviewed by the Regulatory Staff prior to operation above 5 MWt.

Further study is required of the bases and means whereby decisions concerning reactor operation will be made in the event of an earthquake in the region of the site. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant proposes to operate Robinson Unit No. 1 (coal-fired), and Robinson Unit No. 2 (nuclear) from one control room with a crew of five, consisting of a foreman (licensed senior operator), a licensed operator at the nuclear unit console, an unlicensed operator at the coal-fired unit console, and two auxiliary operators, one (licensed) responsible for the nuclear unit and the other for the coal-fired unit. It is the opinion of the Committee that the crew size proposed by the applicant for the nuclear unit is insufficient for safety during initial operation but might be found sufficient after an adequate period of satisfactory operation and a careful assessment of the crew size required for emergencies.

The applicant is using a partial loading of helium "pre-pressurized" fuel rods. The Committee believes that some surveillance of the Robinson fuel at high burnup is appropriate, with regard to assuring the ability of fuel elements to maintain their integrity while undergoing anticipated operational transients near the end-of-life.

The applicant plans to conduct containment proof testing and leak rate testing, prior to initial operation. Subsequently, he proposes leak rate testing only of each seam and penetration of the containment. The Committee believes that periodic integrated leak rate tests should be performed until the Regulatory Staff is satisfied that the methods provided by the applicant assure the required leak tightness of the containment. The Committee recommends that further study be made of possible means to assure the continued structural integrity of the containment throughout the life of the reactor.

The applicant is currently studying the consequences of plant operation with less than three loops in service. Until it can be shown that no design limits are exceeded or that trip points will be reliably reset by automatic action, power operation with less than three loops in service should be prohibited.

The applicant stated that he would provide a second completely independent turbine speed control system designed to meet nuclear protection system criteria of redundancy, separation, and reliability to reduce the probability of an overspeed condition. In addition, protection is to be provided in appropriate areas against damage in the unlikely event of large missiles arising from failure of the turbine rotor or discs. This matter should be resolved in a manner satisfactory to the Regulatory Staff prior to or early in the operation of this plant.

As methods for continuous monitoring of boron concentration and a more definitive determination of gross failure of a fuel element are developed, consideration should be given to their implementation in this plant.

Studies by the applicant are underway on the following problems identified in previous reports of the Committee:

(a) A study of means of preventing common failure modes from negating scram action and of design features to make tolerable the consequence of failures to scram during anticipated transients.

(b) Review of development of systems to control the buildup of hydrogen in the containment and of instrumentation to monitor the course of events in the unlikely event of a loss-of-coolant accident.

As solutions to these problems develop and are evaluated by the Regulatory Staff, appropriate action should be proposed and taken by the applicant on a reasonable time scale. The proposed action should be reviewed by the ACRS.

Other problems relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

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 H. B. Robinson Unit No. 2 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
Chairman

References attached

References

1) Carolina Power & Light Company letter dated July 16, 1968 tsmtg Report on Incidence of Corrosion on Prestressing Steel Tendons
2) Carolina Power & Light Company letter dated December 8, 1969 tsmtg Containment Design Report
3) Carolina Power & Light Company letter dated February 18, 1970 - Responding to Fish and Wildlife Service comments on Proposed Environmental Monitoring Program
4) Carolina Power & Light Company letter dated April 6, 1970 - Identifying the Program to develop and document the additional seismic analysis for Class I equipment and piping
5) Amendment No. 8 to License Application (Final Safety Analysis Report-Volumes 1, 2 and 3) dated November 20, 1968
6) Amendment No. 9 to License Application (designated FSAR Amendment No. 1) dated September 4, 1969
7) Amendment No. 10 to License Application (designated FSAR Amendment No. 2) dated October 27, 1969
8) Amendment No. 11 to License Application (designated FSAR Amendment No. 3) dated December 2, 1969
9) Amendment No. 12 to License Application (designated FSAR Amendment No. 4) dated December 15, 1969
10) Amendment No. 13 to License Application (designated FSAR Amendment No. 5) dated December 15, 1969
11) Amendment No. 14 to License Application (designated FSAR Amendment No. 6) dated January 23, 1970
12) Amendment No. 15 to License Application (designated FSAR Amendment No. 7) dated February 6, 1970
13) Amendment No. 17 to License Application (designated FSAR Amendment No. 8) dated February 24, 1970
14) Amendment No. 18 to License Application (designated FSAR Amendment No. 9) dated February 27, 1970
15) Amendment No. 19 to License Application (designated FSAR Amendment No. 10) dated March 18, 1970
16) Amendment No. 20 to License Application (designated FSAR Amendment No. 11) March 24, 1970

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

June 11, 1974

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

Subject: REPORT ON H. B. ROBINSON UNIT NO. 2

Dear Dr. Ray:

During its 170th meeting, June 6-8, 1974, the Advisory Committee on Reactor Safeguards reviewed the request by the Carolina Power and Light Company for an amendment to License No. DPR-23 to permit an increase in the steady-state power level of the H. B. Robinson Unit No. 2 from 2200 MWt to 2300 MWt. During this review the requested power increase and the operating experience of the H. B. Robinson Unit No. 2 were considered at a Subcommittee meeting on May 21, 1974, in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives of the Applicant, the Westinghouse Electric Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below. The Committee reported on the construction of this plant on February 17, 1967, and on its operation on April 16, 1970.

The H. B. Robinson Unit No. 2 achieved criticality on September 20, 1970. The licensed full power of 2200 MWt was reached on February 23, 1971, and commercial operation started on March 14, 1971. Robinson-2 has operated successfully for two fuel cycles. Examination of data from startup testing and power operation by the Directorates of Licensing and Regulatory Operations have shown that design predictions were confirmed in most areas initially and in the remaining areas after modifications.

Although Robinson-2 was designed for operation at 2300 MWt, initial operation has been limited to 2200 MWt. The proposed increase in maximum power is based on favorable operating experience, use of prepressurized high density fuel, and on the application of thermal-hydraulic and ECCS performance evaluation models currently approved for use for Westinghouse pressurized water reactors. On the basis of analyses, the Interim Acceptance Criteria for Emergency Core Cooling Systems in Light Water

Reactors, including consideration of the effects of fuel densification, can be met for the fuel loading proposed for Fuel Cycle 3 if the linear power generation in the fuel is limited to 15.8 kw/ft. Based on this limit, operation up to power levels of 2300 MWt is acceptable, providing the total peaking factor (F_q^T) is no greater than 2.65. The Applicant intends to use excore radiation detection instrumentation to monitor the axial offset limits required to meet this peaking factor restriction.

Re-evaluation of operating limits will be necessary as a result of the recently promulgated 10 CFR Part 50.46. The Committee wishes to be kept informed.

During Fuel Cycle 2, Robinson-2 was the first nuclear power plant to depend upon the Westinghouse Axial Power Density Monitoring System (APDMS) as a means for monitoring limiting linear power generation rates in order to operate at full power. The operation of the system was generally successful and enabled safe operation with peaking factors below those which can be adequately monitored using excore instrumentation alone. This Applicant does not expect to use the APDMS system in Fuel Cycle 3 under the Interim Acceptance Criteria. However, the system may be proposed for use in this and other Westinghouse plants in the future. Consequently the Committee recommends that the use of APDMS be reviewed, giving attention to the experience in Robinson-2 and to the evaluation of possible sources of uncertainties in using APDMS to monitor peaking factors whose magnitudes are below those which can be monitored using excore surveillance techniques. The Committee wishes to be kept informed.

The Applicant has installed a strong motion recorder to monitor horizontal and vertical ground accelerations and has established the inspection and corrective actions required in the event of a seismic alarm. The Committee concurs with the Regulatory Staff that the reactor be required to be shut down if the operating basis earthquake is exceeded and remain shut down until inspection shows that no damage has been incurred which would jeopardize safe operation of the facility, or until such damage is repaired. This matter should be resolved to the satisfaction of the Regulatory Staff.

The Committee recommends that the Applicant and the Regulatory Staff review the design of the redundant turbine overspeed control system to assure proper functioning under all fault conditions. This matter should be resolved to the satisfaction of the Regulatory Staff.

The Committee believes the Applicant and the Regulatory Staff should review possible sources of debris, such as particles of loose insulation in the containment, as well as the possible effect of such debris on the functioning of engineered safeguards systems.

The Committee recommends that the Technical Specifications for H. B. Robinson-2 specify heatup and cooldown pressure-temperature limits that can be shown to be as conservative as practical with respect to 10 CFR Part 50, Appendix G.

Other generic problems relating to large water reactors identified by the Regulatory Staff and the ACRS 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 Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above and in its previous reports, there is reasonable assurance that the H. B. Robinson Unit No. 2 can be operated at power levels up to 2300 MWt without undue risk to the health and safety of the public.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References:

1. Safety Evaluation by the Directorate of Licensing, USAEC (DRL), H. B. Robinson Steam Electric Plant Unit No. 2, Power Increase, dated May 20, 1974

2. WCAP-8243, "H. B. Robinson Unit 2 - Justification of Operation at 2300 MWt", dated December 1973

3. Application by Carolina Power & Light Company (CP&L) dated February 1, 1974, requesting amendment No. DPR-23 to permit operation at steady-state power levels not in excess of 2300 MWt

4. Letter dated March 12, 1974, CP&L to DRL, submitting additional information pertinent to 2300 MWt operation

5. Letter dated April 12, 1974, CP&L to DRL, submitting additional information pertinent to 2300 MWT operation

6. Letter dated April 29, 1974, CP&L to DRL, submitting additional information pertinent to 2300 MWt operation

7. Letter dated September 7, 1973, V. Stello (DRL) to D. Skovholt (DRL) concerning use of $\overline{R}$ technique

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

November 18, 1971

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON ROME POINT NUCLEAR GENERATING STATION

Dear Dr. Schlesinger:

At its 139th meeting, November 11-13, 1971, the Advisory Committee on Reactor Safeguards completed a limited review of the suitability of the Rome Point, Rhode Island, site as a location for two light-water nuclear power reactors with an approximate capacity of 900 MWe each. As requested by the Yankee Atomic Electric Company, the review was limited to consideration of the proximity of the Naval Air Station, Quonset Point, Rhode Island, and its influence on the suitability of the site for nuclear power generation. This matter was considered by the Committee at its 137th meeting, September 9-11, 1971, and at Subcommittee meetings on August 28, 1971, at the site and on November 8, 1971, in Washington, D. C. During its review the Committee had the benefit of discussions with representatives of the Yankee Atomic Electric Company, the Narragansett Electric Company, the AEC Regulatory Staff and its consultants from the Naval Ordnance Laboratory, and representatives of the United States Navy. The Committee also had the benefit of the documents listed below.

The Rome Point site is 16 miles south of Providence, Rhode Island, on the west shore of the west passage of Narragansett Bay. The Naval Air Station, Quonset Point, is 3.5 miles north-northeast of the center of the site.

The Naval Air Station, Quonset Point, is primarily an anti-submarine training facility. The aircraft using the base are predominantly the S-2 (a twin-engine propeller-driven patrol craft), which accounts for about 77% of the usage, and the SH-3 (a patrol helicopter) which, with other helicopters, account for about 14% of the usage. Approximately 3% of the usage is by other twin-engined propeller aircraft, and another 3% is by small propeller aircraft. Interceptor attack aircraft, medium transports (150,000 lbs. or less), and heavy transports (150,000 - 280,000 lbs.) each account for about 1% of the operations. Operations (takeoffs or landings) are currently at the rate of about 100,000 per year. The Navy has indicated that it has no plans for new runways or changes in the mission of the facility.

In the mid-1970s the Navy plans to replace the S-2 aircraft with the S-3, a twin-engine turbo-jet aircraft. The maximum loaded weights of the S-2, S-3, and SH-3 are 29,150 lbs., 41,000 lbs. (estimated), and 20,500 lbs., respectively.

The S-2, S-3, and SH-3 aircraft have heavy ordnance carrying capability for bombs, mines, torpedoes, guided missiles, rockets, and fire bombs. Individual items contain up to 250 lbs. of high explosive, and the maximum projected total weapon load for any of the above aircraft (the S-3) is 7400 lbs., including the hardware. Heavy ordnance is carried in one percent or less of takeoffs, and the ordnance is normally expended before the aircraft returns to the station.

The Naval Air Station, Quonset Point, has three runways and a helicopter pad. Approximately 6% of the flight operations take off or land on runway 5-23 in the direction which would cause them to pass near the site; the distance from the near end of this runway to the center of the site is 3.5 miles and the distance from the closest point of approach of the extension of the centerline of this runway is 1.1 miles from the center of the site. Approximately 2% of flight operations take off or land on runway 1-19 in the direction which would cause them to pass near the site; distances from the near end and closest point of approach to the center of the site are 3.6 miles and 1.4 miles, respectively. Approximately 41% of flight operations take off or land on runway 16-34 in the direction which would cause them to pass near the site; distances from the near end and closest point of approach to the center of the site are 3.6 miles and 3.1 miles, respectively. All four-engine propeller aircraft and all jet transport aircraft use the 16-34 runway, which keeps them further from the site than if it were possible for them to use the shorter runways.

The normal traffic pattern around the airport does not extend over the center of the site, but traffic regulations do permit aircraft to fly over the site. Only limited relief of this condition is practical; for example, it would be possible to avoid direct overflight below 500 feet and to reroute helicopter traffic so that it would not pass over the site. Existing regulations prohibit aircraft carrying heavy ordnance from overland flight except in the takeoff or landing operations. The Committee recommends that the applicant continue to maintain liaison with the U. S. Navy for mutual exchange of information on planning, and that the Navy be requested, to the extent that operations permit, to give cognizance to the Rome Point Plant in the control of traffic, particularly large planes and planes carrying heavy ordnance.

Although the calculated probability of an aircraft striking the proposed plant is very small, the Yankee Atomic Electric Company will provide protection against the consequences of a crash of an S-2, S-3, or SH-3 aircraft not carrying heavy ordnance. The protection will cover impact forces, forces caused by explosion or ignition of signal and marker devices, and effects of a fuel fire. The Committee believes that the protection should be such as to permit safe shutdown under all circumstances following such a crash, and that, in anticipation of possible future developments, a margin should be allowed for further increases in aircraft weight and speed.

The probability of a strike by an aircraft carrying heavy ordnance is much smaller than for other S-2, S-3, or SH-3 operations, both because of the very small percentage of operations involved and because of the special restrictions on flight paths. Based on available statistics, the applicant and the Regulatory Staff have independently reviewed the probability that an aircraft carrying heavy ordnance may strike the plant. It is concluded that the probability of such a strike is only a small fraction of the background probability of an accidental strike by a large commercial aircraft using established airways. This conclusion takes into account the small number of aircraft carrying ordnance, but gives no credit for regulations that require the aircraft to remain over water after takeoff. The applicant finds that it would be impractical to provide additional protection to cover these very low probability strikes, and he does not propose to provide such protection.

Operations by larger aircraft account for about 3 percent of the total, and involve only runway 16-34, which does not normally bring the aircraft as close to the plant as do runways 1-19 and 5-23. The applicant, therefore, does not plan to provide additional protection against a strike by large planes. Studies by the applicant and the Regulatory Staff show that the probability of a strike by a large plane using the Naval Air Station, Quonset Point, is only a fraction of background.

The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the items mentioned above, the proximity of the Naval Air Station, Quonset Point, does not of itself render the Rome

Point site unacceptable, with respect to the health and safety of the public, for a nuclear plant utilizing two light-water reactors of conventional design and of the power level proposed.

Sincerely yours,

Spencer H. Bush

Spencer H. Bush
Chairman

References:

1. Yankee Atomic Electric Company letter dated February 16, 1971, submitting "Rome Point Nuclear Generating Station Preliminary Site Evaluation"

2. Yankee Atomic Electric Company letter dated July 26, 1971, "Supplemental Information Regarding Rome Point Preliminary Site Evaluation"

3. Yankee Atomic Electric Company letter dated October 19, 1971

4. Yankee Atomic Electric Company letter dated November 10, 1971

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

December 12, 1974

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

Subject: REPORT ON ST. LUCIE PLANT UNIT NO. 2

Dear Dr. Ray:

At its 176th meeting, December 5-7, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application of the Florida Power and Light Company for authorization to construct a second nuclear power unit at its Hutchinson Island site in St. Lucie County, Florida. Members of the Committee visited the site on May 19, 1974; and a Subcommittee meeting was held in West Palm Beach, Florida, on that date. A second Subcommittee meeting was held in Washington, D. C. on November 13, 1974. 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 Committee reported on the construction permit application of St. Lucie 1 (Hutchinson Island) on March 12, 1970.

The St. Lucie Plant Unit No. 2 will be located next to St. Lucie Unit No. 1 on a tract of land of approximately 1100 acres, about half way between the towns of 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 to the north. However, some buildup of population on the island is probable in the coming years, and the plant and its engineered safety features will be designed on the basis of a low population zone distance of 1 mile.

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 liquifaction, the area of most seismic Category I structures was dewatered, excavated to minus 60 feet (MSL), and filled with compacted soils to form a 30-foot-thick base.

Earthquake-induced liquefaction of banks of the cooling water canals or of the soils under a non-seismic Class 1 structure such as the St. Lucie Unit 1 switchyard represents a potential problem for the continued reliability of shutdown cooling. One important aspect of this matter relates to the potential for blockage of the inlets for the cooling water system and possibly to the presence of turbidity and particles in the cooling water. The Applicant and the Staff concur that a practical engineering solution exists for any regions which appear to be subject to liquefaction after the current tests are completed and evaluated. The Committee recommends that a conservative approach be taken in assuring integrity of the ultimate heat removal capability. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The proposed pressurized water reactor has a design power level of 2570 MW(t). The St. Lucie Plant Unit No. 2 design duplicates most of the principal features of Unit No. 1; the use of 16x16 fuel in Unit 2 is a principal difference between the two units. The containment system consists of a steel vessel enclosed within a reinforced concrete building, with the annular space maintained at a slightly 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 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 National Oceanic and Atmospheric Administration and Corps of Engineers methodology.

The St. Lucie Plant Unit No. 2 is the first to propose use of the Combustion Engineering (CE) 16x16 fuel assembly at the construction permit stage. However, some previously reviewed plants employing CE nuclear steam-supply systems are converting from 14x14 fuel to 16x16 fuel during the construction stage and should operate prior to St. Lucie Unit No. 2. Mechanical tests, fuel tests and other research and development are underway. Neither the Regulatory Staff nor the ACRS have completed their review of the new core design. The Committee wishes to be kept informed concerning the results of the various on-going experimental and analytical programs and of any design changes which may be proposed in the future.

An evaluation of the compliance of St. Lucie 2 with 10 CFR 50.46 remains to be performed; however, calculated peak clad temperatures well below the limit are anticipated by the Applicant and the Regulatory Staff.

The ATWS evaluation, including any need for design modifications, remains to be submitted by the Applicant and evaluated by the Regulatory Staff. The Committee wishes to be kept informed.

St. Lucie Unit No. 2 has some reactor vessel and core design features different from other Combustion Engineering reactors. The Regulatory Staff plans to require an instrumented reactor internals vibration program appropriate to a prototype plant unless the Applicant can provide test results for other plants which clearly substantiate the St. Lucie Unit No. 2 analytical vibration response model. The Committee concurs.

The adequacy of protection against flooding of the ECCS pump room is under study. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

Means of qualification of the electric cables from the diesel generators for operation under conditions of temporary tunnel flooding are under review. A different design approach represents a possible alternative for this important function. The Committee recommends that the Applicant and the Staff continue to study this matter.

The Regulatory Staff has proposed that the Applicant upgrade specific pressure systems to seismic Category I and Quality Group C in accordance with interpretations of Regulatory Guides 1.26 and 1.29. Included systems are the letdown loop of the chemical and volume control system, the component cooling lines which service the letdown heat exchanger and the reactor coolant pumps, and the fuel pool makeup system. The Applicant believes that alternate flow paths exist where a safety function must be met and that there is no requirement to upgrade to seismic Category I and Quality Group C in components not necessary to safety. The Committee recommends that the safety significance of these systems be reassessed by the Applicant and by the Staff and the matter resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The matter of the generation of turbine missiles and their probable effects on reactor safety is under review, including the possible need of design features to reduce the probability or mitigate the consequences. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

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

The Committee believes that the above items can be resolved during construction and that, if due consideration is given to these items, St. Lucie Unit No. 2 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/

W. R. Stratton
Chairman

References attached

<u>References</u>

1. St. Lucie Plant, Unit 2, Preliminary Safety Analysis Report, Volumes 1-8 (including Amendments 1-6, 8-16, 18-22).

2. Safety Evaluation of the St. Lucie Plant, Unit No. 2 (Directorate of Licensing Report), November 7, 1974.

3. FP&L letter dated December 31, 1973 furnishing information related to ATWS.

4. Directorate of Licensing Safety Evaluation Report, October 1974.

   a. Supplement to SER dated November 7, 1974.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 10, 1975

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

SUBJECT: REPORT ON ST. LUCIE PLANT, UNIT No. 1

Dear Mr. Anders:

At its 182nd meeting, June 5-7, 1975, the Advisory Committee on Reactor Safeguards completed its review of the application of the Florida Power and Light Company for authorization to operate the St. Lucie Plant, Unit No. 1. The project was previously considered at Subcommittee meetings at West Palm Beach, Florida on May 16, 1974; in Washington, D. C. on November 12-13, 1974, and on June 4, 1975. The facility was toured on May 16, 1974. In its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, Combustion Engineering, Inc., Ebasco Services, Inc. and the NRC Staff. The Committee reported on the construction permit application of St. Lucie Plant, Unit No. 1 (Hutchinson Island), on March 12, 1970, and on the construction permit application of St. Lucie Plant, Unit No. 2, on December 12, 1974.

The St. Lucie Plant, Unit No. 1, is located on Hutchinson Island 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 originally proposed low population zone (LPZ). The minimum exclusion distance is 5100 feet. The nearest population center is Fort Pierce (1970 population about 30,000), which is eight miles to the northwest. However, some buildup of population on the island is probable in the coming years, and the plant and its engineered safety features are being modified to meet an LPZ radius of 1 mile.

The plant site is underlain by sand to a depth of several hundred feet. To provide satisfactory bearing and settlement characteristics and resistance to liquefaction, the area of most seismic Category I structures was dewatered, excavated to minus 60 feet (MSL), and filled with compacted soils to form a 30-foot-thick base.

Earthquake-induced liquefaction of the banks of the cooling water canals or under the dam to Big Mud Creek, which provides a seismic Class 1 source of water for the ultimate heat sink, represents a potential problem for the continued reliability of shutdown cooling. The Applicant and the NRC Staff differ in their conclusions regarding a prudent interpretation of the existing data with regard to the potential for liquefaction. The Committee agrees with the Staff that unless additional information by the Applicant establishes that unacceptable soil movements cannot occur, appropriate remedial measures should be taken. This matter should be resolved in a manner satisfactory to the NRC Staff.

Questions related to the potential effects of a stalled hurricane on the integrity of safety features are currently under review. This matter should be resolved in a manner satisfactory to the NRC Staff.

Additional information and evaluation thereof is required with regard to the potential effects of tornado-induced missiles on some engineered safety features. This matter should be resolved in a manner satisfactory to the NRC Staff.

The St. Lucie Plant, Unit No. 1, includes a pressurized water reactor similar to that currently employed at the Calvert Cliffs and Millstone 2 plants. The current application requests an operating license of 2560 MWt; the power level requested in the construction permit application was 2440 MWt.

Several changes have been made in the Combustion Engineering ECCS evaluation model to bring it into conformance with the Commission Criteria per 10 CFR 50, Appendix K. A partial analysis (a break in the pump discharge leg) has been made using the new model; hot leg and suction leg analyses remain to be evaluated, but the Applicant and the NRC Staff expect the pump discharge leg break to be limiting. This analysis leads to a maximum permitted linear heat generation rate of 14.6 kw/ft. A relatively low peaking factor is required to achieve this limit and the Applicant proposes to use both in-core and ex-core instrumentation in order to assure adequate accuracy of measurement of core power distributions.

The Committee believes that the proposed monitoring methods may be acceptable, but that an augmented startup program be employed, and that satisfactory experience at steady state, 100% power and during transients at less than full power should be obtained, reviewed, and evaluated by the NRC Staff prior to operating at full power in a system-load-follow mode.

A question has arisen concerning loads on the vessel support structure for certain postulated loss-of-coolant accidents in pressurized water reactors. This matter should be resolved in a manner satisfactory to the NRC Staff.

Potentially damaging water hammer has been observed in the feed water inlet piping of some PWR steam generators. Corrective measures are planned upon completion of studies and experimental investigation of the phenomenon. The adequacy of the corrective measures should be experimentally verified to the satisfaction of the NRC Staff. The Committee wishes to be kept informed.

The analysis of Anticipated Transients Without Scram is incomplete for the St. Lucie Plant, Unit No. 1. The Committee recommends that a schedule for submission of information and for any modifications, if necessary, be prepared, and that this matter be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

Some questions remain with respect to the handling of heavy loads over the fuel storage pool. This matter should be resolved in a manner satisfactory to the NRC Staff.

Means of qualification of the electric cables from the diesel generators for operation under various environmental conditions are still under review. This matter should be resolved in a manner satisfactory to the NRC Staff.

Suitable instrumentation to follow the course of an accident has been generically identified as an important feature needed to assist operating personnel in diagnosing unexpected events. The NRC Staff should initiate prompt action to clarify the essential requirements for this instrumentation including information to be monitored, environmental conditions under which it must operate, location and type of display, relationship to normally used instrumentation and methods of assuring functional effectiveness at the time of need. Arrangements should be made to incorporate the required instrumentation in all plants licensed for construction. Where possible the necessary equipment should also be provided on licensed operating power plants. The Committee wishes to be kept informed.

The Applicant is making progress in arrangements for emergency procedures to be followed in case of an accidental release of radioactive materials from the plant. Yet to be confirmed, however, are plans of the state agencies whose actions would be essential in dealing with the population in case of some such events. The Committee recommends that the applicant and the NRC Staff continue to collaborate with the State in moving ahead to complete development of an emergency response plan and that the adequacy of arrangements for implementing such a plan be confirmed prior to initial operation of the plant.

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 St. Lucie Plant, Unit No. 1, can be operated at power levels up to 2560 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

WKerr

W. Kerr
Chairman

References

1) Final Safety Analysis Report (FSAR) with Amendments 12 through 44

2) Safety Evaluation Report by the Directorate of Licensing (DL), dated November 8, 1975

3) Supplement No. 1 to Safety Evaluation Report by DL, dated May 9, 1975

4) Letter, dated March 31, 1975, Florida Power and Light (FP&L) to DL concerning analysis of ATWT

5) Letter, dated April 9, 1975, FP&L to DL, concerning ECCS analysis

6) Letter, dated September 13, 1974, FP&L to DL, concerning design features to ensure that guideline doses of 10 CFR 100 are not exceeded

7) Letter, dated September 1, 1974, FP&L to DL, concerning the emergency plan

8) Letter, dated December 31, 1973, FP&L to DL, concerning information regarding ATWT

9) Letter, dated October 27, 1972, FP&L to DL, concerning failure of any non-Category I (seismic) equipment which could cause degradation of safety-related equipment

10) Letter, dated May 27, 1975, Conservation Alliance of St. Lucie County to ACRS, concerning emergency planning, quality control and training

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 17, 1981

The Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON ST. LUCIE PLANT UNIT NO. 2

Dear Dr. Palladino:

During its 259th meeting, November 12-14, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of the Florida Power and Light Company (the Applicant) for authorization to operate the St. Lucie Plant Unit No. 2. The project was considered at a Subcommittee meeting in West Palm Beach, Florida on October 30-31, 1981 and members of the Committee toured the facility on October 30, 1981. In its review the Committee had the benefit of discussions with representatives of the Applicant, Combustion Engineering, Inc., Ebasco Services, Inc., the NRC Staff, and members of the public. The Committee also had the benefit of the documents listed. The Committee commented on the construction permit application for St. Lucie Plant Unit No. 2 in a report dated December 12, 1974 to AEC Chairman Dixie Lee Ray.

St. Lucie Plant Unit No. 2 is located on Hutchinson Island adjacent to Unit No. 1, which went into commercial operation in December 1976. Both units use Combustion Engineering nuclear steam supply systems with a rated core power of 2560 MWt. The two units are nearly identical.

A number of items have been identified as Outstanding Issues, Confirmatory Issues, and License Conditions in the NRC Staff's Safety Evaluation Report dated October 1981. These include some TMI-2 Action Plan requirements. We believe these issues can be resolved in a manner satisfactory to the NRC Staff. We also recommend resolution of concerns on instrumentation for detection of inadequate core cooling expressed in the ACRS letter to the Executive Director for Operations dated June 9, 1981.

Discussion with the Florida Power and Light Company Staff indicated that emergency operating procedures for dealing with off-normal plant behavior that might develop during the operation of St. Lucie Plant Unit No. 2 are incomplete. We recommend that a concentrated effort be made by the Florida Power and Light Company staff to complete emergency operating procedures which take advantage of new information and approaches developed during the past two years. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

At the time this site was initially approved, the population density was relatively low, and the projected increase during the life of the plant was not unusually large. Since that time, the growth in population has been much more rapid than predicted, and current estimates predict continued growth at relatively high rates. Although the present population and that predicted for the next several years are not a cause for concern, it now seems possible that the population density in portions of the surrounding area could reach a level, during the lifetime of the St. Lucie Plant, that might then warrant additional measures. We recommend that the Applicant and the NRC Staff periodically review the actual and projected population growth. If required as a result of these reviews, plans for appropriate preventive or remedial measures could then be made in a considered but timely manner.

We recommend that the Staff give due regard to the special nature of this site in evaluating the final emergency plan.

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, staffing, and preoperational testing, there is reasonable assurance that the St. Lucie Plant Unit No. 2 can be operated at core power levels up to 2560 MWt without undue risk to the health and safety of the public.

Additional comments by Members H. W. Lewis and M. S. Plesset are presented below.

Sincerely yours,

J. Carson Mark

J. Carson Mark
Chairman

Additional Comments by Members H. W. Lewis and M. S. Plesset

In the aftermath of the accident at Three Mile Island Unit 2, which dramatically emphasized the importance of instrumentation to follow the course of an accident, the NRC Staff has required applicants for an Operating License to demonstrate specific capability to detect the onset of inadequate core cooling. For PWRs this has come to mean in practice the provision, inter alia, of an instrument which can be called a water-level indicator for the pressure vessel. (Although the NRC Action Plan allows for alternatives, none appear to have been seriously contemplated.) A number of such devices have been accepted and/or proposed, some of which measure differential pressure, some average void fraction in a part of the pressure vessel, some cooling rate at a number of places in the vessel. All can give spurious response because of dynamic effects.

Many of these views have been previously expressed in the Committee letter of June 9, 1981.

We are concerned that, in the commendable eagerness to avoid a repetition of TMI, the NRC Staff is requiring ill-defined instrumentation without any clear picture of the contribution of that instrumentation to the prevention or mitigation of accidents - considerations which must necessarily be scenario dependent. If it were really true that core water level were the important parameter, then differential pressure indicators would appear to be preferable, provided the coolant is quiescent. If instead cooling capacity is important, then some form of heated wire or thermocouple would appear to be preferable. Since either may be acceptable, we are left with the inference that the NRC Staff has not really clarified the role of this instrumentation.

We believe that, before, not after requiring these instruments for all the new plants, the NRC Staff should develop a position regarding their utility. This position, which should be based upon accident analysis and risk assessment, would lead to a much clearer understanding of just what instrumentation, if any, is needed.

REFERENCES:
1. Florida Power and Light Company, "St. Lucie Plant, Unit No. 2 Final Safety Analysis Report," with Amendments 1 through 6.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of St. Lucie Plant, Unit No. 2," Docket No. 50-389, USNRC Report NUREG-0843, dated October 1981.
3. Letter from Betty Lou Wells to the Chairman of the Advisory Committee on Reactor Safeguards, dated October 28, 1981.
4. Written statement by Joette Lorian, Research Director for the Center for Nuclear Responsibility.

June 21, 1968

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

Subject: REPORT ON SALEM NUCLEAR GENERATING STATION

Dear Dr. Seaborg:

At its ninety-eighth meeting, June 5-8, 1968, the Advisory Committee on Reactor Safeguards completed its review of the application of Public Service Electric and Gas Company for authorization to construct Salem Nuclear Generating Station, a two-unit nuclear power plant. The project was previously considered at a Subcommittee meeting and site visit on May 24, 1968. During its review, the Committee had the benefit of discussions with representatives of Public Service Electric and Gas Company, Westinghouse Electric Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Salem units are to be located on a 700-acre site in Salem County, New Jersey, on the southern part of Artificial Island on the east bank of the Delaware River about 18 miles south of Wilmington, in a sparsely populated region. The nearest population center of about 25,000 people is Bridgeton, New Jersey, 15.5 miles east of the site.

The two units are to be identical. Each includes a four-loop, pressurized water reactor designed for a power output of 3250 MWt. Reactor system design and power rating are virtually the same as for the Diablo Canyon plant previously reviewed by the Committee. Containment building design is similar to that for Indian Point Nuclear Generating Unit No. 2, also previously reviewed; each building is steel-lined, of reinforced concrete, with a flat slab base.

Although referred to as an island, the site is actually connected to the mainland by a strip of tideland. This strip and the site itself, which once was a natural bar in the Delaware River, have been used in the past as a disposal area for material dredged from the river. The site now is mantled by approximately 35 feet of such fill. Below this is a layer of loose clays, silts with sand, and gravel. Underlying the latter is the Vincentown Formation, composed of unconsolidated sand and silty sand, which will be used as the load bearing stratum for the plant. The Committee believes that the foundation design features proposed by the applicant are satisfactory.

The applicant has proposed using signals from the protection system for control and override purposes. The Committee reiterates its belief that control and protection instrumentation should be as nearly independent of common failure modes as possible, so that the protection will not be impaired by the same fault that initiates a transient requiring protection. The applicant and the AEC Regulatory Staff should review the proposed design 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. In cases where hypothesized control or override failure could lead to the need for action by interconnected protection instrumentation, separate protection instrumentation channels should be provided or some other design approach used to provide equivalent safety.

The Committee has previously called attention to certain matters of significance that warrant careful consideration for all large water cooled reactors of high power density. If developments in any of these areas, particularly fuel behavior, should fail to confirm adequately the designer's expectations, system modification or restrictions on operation of the Salem Station reactors may be appropriate.

The Advisory Committee on Reactor Safeguards believes that the items mentioned can be resolved during construction and that the proposed units can be built at the Salem Nuclear Generating Station site with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
Carroll W. Zabel
Chairman

References Attached.

References - Salem Nuclear Generating Station

1. Public Service Electric and Gas Company letter, dated January 22, 1968; Amendment No. 3 to Application for Licenses, Salem Nuclear Generating Station, consisting of revised Preliminary Facility Description and Safety Analysis Report, Part B, Volumes 1, 2, 3, and 4.

2. Public Service Electric and Gas Company letter, dated February 12, 1968; Amendment No. 4 to Application for Licenses.

3. Public Service Electric and Gas Company letter, dated March 14, 1968; Amendment No. 5 to Application for Licenses.

4. Public Service Electric and Gas Company letter, dated April 15, 1968; Amendment No. 6 to Application for Licenses.

5. Public Service Electric and Gas Company letter, dated May 6, 1968; Amendment No. 8 to Application for Licenses.

6. Public Service Electric and Gas Company letter, dated May 21, 1968; Amendment No. 9 to Application for Licenses.

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

February 12, 1974

L. Manning Muntzing
Director of Regulation

EMERGENCY PLANNING FOR THE SALEM-HOPE CREEK NUCLEAR GENERATING STATIONS

During a recent Subcommittee visit to the Salem Nuclear Generating Station and the proposed site for the Hope Creek Station, Committee members noted that egress from the site did not appear to be adequate for an orderly and speedy evacuation of construction forces for the Hope Creek Station in the event it is required after the nuclear units at the Salem Station have begun operating.

The Committee recommends that the emergency plans for the Salem Nuclear Station be examined to assure that construction workers on the Salem and Hope Creek sites can be quickly and safely evacuated, by alternate paths if necessary, in the event it is required after Salem Station Unit 1 is in operation.

/s/ W. R. Stratton

W. R. Stratton
Chairman

cc: P. Bender, SECY
J. F. O'Leary, DL
A. Giambusso, DL

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 14, 1975

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

SUBJECT: REPORT ON SALEM NUCLEAR GENERATING STATION, UNIT 1

Dear Mr. Anders:

At its 178th meeting, February 6-8, 1975, the Advisory Committee on Reactor Safeguards completed its review of the application of the Public Service Electric and Gas Company, the Philadelphia Electric Company, the Delmarva Power and Light Company, and the Atlantic City Electric Company for authorization to operate the Salem Nuclear Generating Station, Units 1 and 2. The project was previously considered at a Subcommittee meeting in Washington, D. C., on November 7, 1974, and a tour of the facility was made by Subcommittee members on January 22, 1974. Certain generic aspects of the nuclear steam supply system and the new Westinghouse 17x17 fuel rod assembly were reviewed by the Committee at its 175th meeting and in connection with its review of the Trojan Nuclear Plant, which was reported on in the Committee's letter of November 20, 1974. During its review, the Committee had the benefit of discussions with representatives and consultants of the Public Service Electric and Gas Company, the Westinghouse Electric Corporation, and the Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for the Salem Nuclear Generating Station in its letter of June 21, 1968.

Because the expected fuel loading date for Unit 2 is still some distance in the future (estimated to be December, 1978), the Committee believes that its report on Unit 2 should be deferred until a time somewhat closer to the expected start of operations.

The plant is located on a 700-acre site and is adjacent to the proposed Hope Creek Generating Station on the southern part of land that is referred to as Artificial Island in Salem County, New Jersey. The site is on the east bank of the Delaware River, about 18 miles south of Wilmington, Delaware.

In connection with the construction permit review of the Hope Creek Generating Station, the Applicant is making a study to determine the probability of an accident involving waterborne traffic on the Delaware River that is of such a nature as to affect the safety of the plants. The study includes, among other things, barge collison with the service water intake structure, spills of oil or of LNG and possible fires, clouds of LNG resulting from a ship collision, and explosions of ship cargoes. The Committee believes that, if the probability of such an accident affecting the safety of the plant is not acceptably low, design changes to provide suitable protection should be required on a timely basis for the Salem units. This matter should be resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The Regulatory Staff has initiated discussions with the Federal Power Commission and other agencies concerning the potential adverse effects on the safety of the nuclear reactors of ongoing or projected installations or operations under the control or surveillance of such agences. The Applicant stated that special procedures were being instituted at other ports in connection with the transport of LNG and that they anticipated that the Captain of the Port at Philadelphia will develop similar procedures. The Committee recommends that the Regulatory Staff review the Port Plan with regard to control of hazardous shipments within the Delaware Bay and on the Delaware River. The Committee also recommends that interagency arrangements be formalized whereby the NRC is automatically informed of potential impacts on nuclear power plant safety of matters under review by other agencies.

The two units at the Salem Station are essentially identical. Each includes a four-loop Westinghouse nuclear steam supply system similar in most respects to that for the Trojan Nuclear Plant. The design core power level for Unit 1 is 3338 Mwt.

The Salem plant is scheduled to be one of the first to go into operation using a full core of 17x17 fuel. While many of the various required verification programs have been completed and reviewed by the Regulatory Staff, other tests and analyses are still to be completed and documented. These include: DNB tests with non-uniform heat flux, single-rod burst tests, fuel assembly flow tests, guide tube tests, and the effect of bowing on DNB. The results of such tests and analyses should be evaluated fully by the Regulatory Staff and resolved to their satisfaction prior to the full core use of 17x17 fuel to produce power. Four prototype 17x17 fuel rod assemblies are to be loaded into other operating pressurized water reactors in the near future; the results of these irradiations should be followed closely. The Committee wishes to be kept informed concerning the results of the various ongoing 17x17 test and analytical programs, and any design changes which may be proposed in the future.

Following each cycle of operation, 17x17 fuel assemblies will be examined for fuel rod integrity, fuel rod and assembly dimension and alignment, and surface deposits. In view of the fact that the 17x17 fuel array is a new design and that no prototype irradiations are planned for 17x17 fuel containing eight spacer-grids (which will be employed only in full - core operation), the results of surveillance programs for this type fuel should be followed closely. The Committee wishes to be kept informed.

The recently proposed method of constant axial offset control will be used for in-core power distribution monitoring and control. The Regulatory Staff should review carefully the effectiveness of this method of control in protecting against adverse consequences of postulated reactor transients and accidents. The Committee wishes to be kept informed.

Several changes are to be made in the Westinghouse ECCS evaluation model to bring it into conformance with the Commission Criteria as given in 10 CFR 50.46. The performance of the emergency core cooling systems will be re-evaluated with the approved evaluation model, and appropriate operating limits and procedures for ensuring monitoring of the power distribution are to be incorporated in the Technical Specifications. The Committee wishes to be kept informed.

The evaluation of Anticipated Transients Without Scram (ATWS) has been made generically for Westinghouse plants, and the Applicant has made comparisons indicating that the results obtained are applicable to the Salem Plant. Regulatory review should be completed and this matter resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed

Salem Unit 1 may be one of the first reactors of its type to operate with a rated power as high as 3338 Mwt. Because there is limited operating experience with very large, high-power density reactors, the ACRS believes that a more cautious than normal approach to full power is prudent, with longer periods of operation at power levels in the range of 70 to 90% of full power, and with additional monitoring of core and systems performance throughout the life of the first core. The Committee recommends that the Regulatory Staff evaluate the overall operating experience prior to sustained operation at full power.

Generic problems relating to large water reactors have been identified by the Regulatory Staff and the ACRS and discussed in the Committee's report, dated February 13, 1974. These problems should be dealt with appropriately by the Regulatory Staff and the Applicant as suitable approaches are developed.

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 Salem Nuclear Generating Station, Unit 1 can be operated at power levels up to 3338 Mwt without undue risk to the health and safety of the public.

Sincerely yours,

/s/ W. Kerr

W. Kerr, Chairman

References:

1. Final Safety Analysis Report (FSAR) for the Salem Nuclear Generating Station, Units 1 and 2 (Amendment No. 10 to the Salem Application)

2. Amendments Nos. (12-24) and (26-32) to the Salem Application

3. Safety Evaluation of the Salem Nuclear Generating Station, Units 1 and 2, dated October 11, 1974, by the U.S. Atomic Energy Commission, Directorate of Licensing (DL)

4. Letter, received February 7, 1975, Mrs. Richard Horner, Hancock's Bridge, New Jersey, to ACRS, concerning "AEC Inspection No. 50-272/74-16"

5. RO Inspection Report No. 50-272/74-16, dated December 13, 1974, concerning Unusual Occurrence - Flooding in Turbine Building and Auxiliary Building, reported December 2, 1974

6. Letter, dated October 25, 1974, Public Service Electric and Gas Company of New Jersey (PSE&G) to DL, concerning safeguards equipment control system

7. Letter, dated September 30, 1974, PSE&G to DL, concerning ATWS analysis for Salem 1 and 2

8. Letter, dated April 23, 1974, PSE&G to DL, concerning review of safety related circuity

9. Letter, dated March 7, 1974, PSE&G to DL, concerning operator requalification

10. Letter, dated March 4, 1974, DL to ACRS, concerning emergency planning for Salem-Hope Creek Stations

11. Letter, dated February 21, 1974, PSE&G to DL, concerning Quality Assurance organization

12. Letter, dated November 2, 1972, PSE&G to DL, concerning failure of Non-Class I (Seismic) equipment

13. Letter, dated January 2, 1975, PSE&G to DL, concerning ECCS analyses

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 15, 1979

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: REPORT ON SALEM NUCLEAR GENERATING STATION UNIT 2

Dear Dr. Hendrie:

During its 226th meeting, February 8-10, 1979, the Advisory Committee on Reactor Safeguards completed its review of the application of the Public Service Electric and Gas Company, et al for authorization to operate the Salem Nuclear Generating Station Unit 2. This project was initially considered in connection with the review of Salem Unit 1 and at a Subcommittee meeting in Washington, D. C. on January 24, 1979. A tour of the facility was made by Committee members on January 25, 1979. During its review the Committee had the benefit of discussions with representatives and consultants of the Public Service Electric and Gas Company, the Westinghouse Electric Corporation, and the Nuclear Regulatory Commission (NRC) Staff, as well as comments from members of the public. The Committee also had the benefit of the documents listed.

The Committee reported on the application for a construction permit for the Salem Nuclear Generating Station Units 1 and 2 in its letter of June 21, 1968. The Committee reported on the application for an operating license for Unit 1 in its letter of February 14, 1975, at which time it deferred its operating license review of Unit 2 until a time somewhat closer to the expected start of operations.

In January 1978, the NRC Staff began a re-review of Salem Unit 2 to consider changes in NRC regulations or requirements, changes in the design of the plant, and operating experience with Salem Unit 1. One phase of this re-review has included current generic matters such as fire protection, industrial security, emergency planning, and ATWS. For these matters, the NRC Staff is reviewing both Units 1 and 2, and it is expected that the resolution will be substantially the same for both units.

The other phase of the re-review has addressed the degree to which Salem Unit 2 conforms to the provisions of Regulatory Guides and Branch Technical Positions that have been adopted since the operating license review was made for Salem Unit 1. These items include those classified by the

Regulatory Requirements Review Committee as Category 2 (backfit on a case-by-case basis) and as Category 3 (backfit on all plants). A comparable review of Salem Unit 1 (which initially was identical to Unit 2) is being carried out by the Division of Operating Reactors on a different time scale. The NRC Staff has stated that the reviews for Units 1 and 2 are, or will be, coordinated to provide consistency between the two units.

The NRC Staff's re-review of Salem Unit 2 is essentially complete and will be completed before an operating license is issued. There are four outstanding issues still under review or for which complete documentation has not yet been received. There are also six items for which the NRC Staff requires only confirmatory documentation regarding their resolution. The Committee believes that all of these outstanding issues and confirmatory items can and should be resolved to the satisfaction of the NRC Staff.

In its review of Salem Unit 1 and of the Hope Creek units at the same site, the Committee expressed its concern about the possibilities of accidents involving waterborne traffic on the Delaware River that might be of such a nature as to affect the safety of the plants. This question has been addressed by the NRC Staff and the Applicant on a probabilistic basis in connection with the reviews of both the Salem and Hope Creek plants. The Committee believes that the results of these studies provide a reasonable basis for assuming that the probabilities, and thus the risks, of such accidents are sufficiently low as not to provide an undue risk to the health and safety of the public. The Committee, however, continues to be concerned about accidents of this nature and believes that the potential hazards should continue to be reviewed from time to time as the local conditions may change and as the extent and reliability of the data base may be increased.

The Committee recommends that the NRC Staff establish criteria for the implementation of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," as soon as practicable. The Committee believes that Position C.3 of this Guide should be implemented on Salem Unit 2 to the extent practicable.

With regard to the generic items cited in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977, those items considered relevant to Salem Unit 2 are: II-2, 3, 5B, 6, 7, 9, 10; IIA-2, 3, 4; IIB-2; IIC-1, 2, 3A, 3B, 4, 5, 6; IID-1, 2; IIE-1. These matters should be dealt with by the NRC Staff and the Applicant, as appropriate, when solutions are found.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the matters mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the Salem Nuclear Generating Station Unit 2 can be operated at power levels up to 3411 Mwt without undue risk to the health and safety of the public.

Mr. J. J. Ray did not participate in the Committee's review of this project.

Sincerely,

Max W. Carbon
Chairman

References

1. Salem Nuclear Generating Station, Units 1 and 2, Final Safety Analysis Report, with amendments 1 through 43.

2. Safety Evaluation Report, Supplement No. 3, by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission in the matter of Public Service Electric and Gas Company, et al, Salem Nuclear Generating Station, Unit 2, NUREG-0492, dated December 29, 1978.

3. Letter to O. D. Parr, U. S. Nuclear Regulatory Commission, Light Water Reactors Branch 3, from R. L. Mittl, Public Service Electric and Gas Company, concerning additional information on single failure criteria related to pump seal for RCP, dated January 4, 1979.

4. Letter to O. D. Parr, U. S. Nuclear Regulatory Commission, Light Water Reactors Branch 3, from R. L. Mittl, Public Service Electric and Gas Company, concerning additional information on emergency action levels, dated January 8, 1979.

5. Letters from members of the Public:

    a. Letter to E. G. Igne, ACRS Staff, from Phyllis Zitzer, of the Committee for Application of Nuremberg Principles to U. S. Nuclear Power Production, dated January 18, 1979.

    b. Letter to E. G. Igne, ACRS Staff, from Joseph Blotnick, dated January 25, 1979.

    c. Letter to E. G. Igne, ACRS Staff, from Jill Higgins, of the Delaware Safe Energy Coalition, dated January 25, 1979.

Page Revised: 2/22/79

d. Letter to E. G. Igne, ACRS Staff, from Nanci L. Reynolds, dated January 26, 1979.

e. Letter to E. G. Igne, ACRS Staff, from Roy Money, dated January 29, 1979.

f. Letter to E. G. Igne, ACRS Staff, from Frieda Berryhill, of Coalition for Nuclear Power Plant Postponement, dated January 30, 1979.

g. Letter to E. G. Igne, ACRS Staff, from Mary Lesser, dated February 4, 1979.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

April 10, 1961

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

Subject: REPORT ON SANDIA PULSED REACTOR FACILITY

Dear Dr. Seaborg:

At its thirty-third meeting, April 6-8, 1961, the Advisory Committee on Reactor Safeguards met with the AEC staff and the applicant to consider the proposal by Sandia Corporation to operate the Sandia Pulsed Reactor Facility.

The Sandia Corporation proposes to operate this facility (SPRF) within its controlled area approximately 4 miles from the City of Albuquerque. The assembly, patterned after Godiva II currently in operation at the Los Alamos Scientific Laboratory, is made of fully enriched uranium fabricated in several pieces. This assembly will normally not be operated in the same manner as a conventional reactor. The absence of heat removal provisions limits the average power to very low levels and results in a correspondingly low inventory of fission products. This reactor is located in an area separated from other installations and is provided with adequate protection.

It is the opinion of the Committee that there is reasonable assurance that the operation of the SPRF will not endanger the health and safety of the general public.

However, in view of the unusual character of pulsed supercritical facilities, the Committee would like to stress that this type of device presents peculiar hazards to the local operators. The Committee recommends that approvals for this class of reacting devices be limited to those situations where the demands for the unique services of pulsed supercritical assemblies are clearly evident. Because of the unusual local risk involved in the operation of a pulsed supercritical assembly, a careful evaluation of the qualifications of the operating staff is necessary and frequent inspections are desirable.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

Reference:

SC-4357-A(RR) - Hazards Evaluation of the Sandia Pulsed Reactor Facility (SPRF), dated February 1961.

cc: A. R. Luedecke, GM
H. L. Price, Acting Dir., Regulation
R. Lowenstein, Acting Dir., DL&R

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 14, 1977

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON PARTIAL REVIEW OF THE SITE FOR THE SAN JOAQUIN PROJECT

Dear Dr. Hendrie:

During its 209th meeting, September 8-10, 1977, the Advisory Committee on Reactor Safeguards completed a partial review of the suitability of a site on which the Los Angeles Department of Water and Power (Applicant) proposes to construct two or more nuclear power plants. Members of the ACRS Subcommittee visited the site on August 15, 1974. Subcommittee meetings were held on August 15, 1974 and June 24, 1977 at Bakersfield, California. On September 1, 1977, in conjunction with the Seismic Activity Subcommittee, a meeting was held in San Francisco, California. During its review of the San Joaquin site, the Committee had the benefit of discussions with members of the Nuclear Regulatory Commission (NRC) Staff and the United States Geological Survey (USGS), and with representatives of the Applicant and its consultants. The Committee also had the benefit of the documents listed.

The San Joaquin site review was limited in scope as permitted by 10 CFR Part 50, Appendix Q. Specifically, the scope was limited to evaluating the suitability of the site with respect to: (1) hydrology; (2) geology and seismology, including seismic input criteria; and (3) stability of subsurface materials as to the potential for subsidence.

The San Joaquin site is located in Kern County, California, in the southern San Joaquin Valley, approximately 10 miles northwest of Wasco and approximately 33 miles northwest of Bakersfield, the nearest population center, which had a 1970 population of 69,515.

The site and its environs consist primarily of unimproved and improved farm lands located on the floor of the San Joaquin Valley. The total area of the site is approximately 2500 acres.

The maximum probable flood, including wave run-up, is estimated to result in a water level of 9 ft. above the existing site level. The ACRS agrees with the Staff's position that a nuclear plant can be designed to protect against this water level. Further NRC review will be necessary at the Construction Permit stage to validate plant design for this condition.

A substantial depletion of the underlying aquifer, primarily for agricultural use, has caused approximately four feet of subsidence at the site. This subsidence is anticipated to continue at a rate of about 0.1 ft. per year. The monitoring program suggested by the Applicant is considered adequate for measuring subsidence, nonuniform settling or surface fissuring. With proper plant design such subsidence is not considered to be a problem.

The Applicant, the NRC Staff, and the USGS have agreed that horizontal ground accelerations of 0.45g and 0.225g at the site are appropriate design values for the safe shutdown earthquake (SSE) and the operating basis earthquake, respectively. The SSE value was based on a postulated 8.5 magnitude (Richter) earthquake on the San Andreas Fault at a distance of 35 miles. For the Pond-Poso Creek Fault, the Applicant postulated an earthquake of magnitude 7.0 at a location 11 miles from the site. The NRC Staff, the USGS and the Applicant believe this magnitude is extremely conservative, based on further reviews of fault length and echeloned configuration. The ACRS agrees that a maximum seismic event on this fault should lead to less than 0.45g. Questions have arisen concerning the Greeley Fault. The ACRS agrees with the position of the NRC Staff and the the USGS that this fault is not capable.

The NRC Staff has underway a program of review and reevaluation of several generic matters related to soil-structure interaction and the appropriate response spectrum for use at foundation levels of nuclear power plants. Completion of this reevaluation may result in some change in the development of the appropriate design response. The Committee believes this matter can be resolved prior to completion of the review for a construction permit for use at this site.

The Committee believes that the San Joaquin site is acceptable under the guidelines of 10 CFR Part 100 with respect to the specific site-related items noted above.

Sincerely,

M. Bender

M. Bender
Chairman

REFERENCES:

1. Los Angeles Department of Water and Power: "San Joaquin Nuclear Project, Early Site Review Report" (April 1974) with Amendments 1 through 19.
2. U.S. Nuclear Regulatory Commission: "Limited Site Review by the Office of Nuclear Reactor Regulation for the San Joaquin Nuclear Project, Project No. 499," NUREG-0284, June 1977.

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

September 12, 1963

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

Subject: REPORT ON SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 1

Dear Dr. Seaborg:

At its forty-eighth and forty-ninth meetings, July 11 to 13, and September 5 and 6, 1963, the Advisory Committee on Reactor Safeguards considered the application of Southern California Edison Company, San Diego Gas and Electric Company, Bechtel Corporation and Westinghouse Electric Corporation, for a construction permit for the San Onofre Nuclear Generating Station, Unit No. 1. The Committee had the benefit of site visits, discussions with representatives of the applicants, the AEC Regulatory Staff and consultants, and the documents listed.

The applicants propose construction of this unit by Bechtel and Westinghouse who, as co-contractors, will demonstrate full power operation prior to delivery on a turn-key basis to the owners, Southern California Edison Company and San Diego Gas and Electric Company. The reactor will be operated by Southern California Edison Company thereafter.

Unit No. 1 will be a 1210 Mw(t) pressurized light water reactor located on the Pacific coast near the northern boundary of Camp Pendleton, California. The reactor will be constructed on a 90-acre site, about two and one-half miles from the nearest boundary of San Clemente, a town of approximately 10,000 people. The site is within the Camp Pendleton Reservation and fronts on the Pacific Ocean. U. S. Highway 101 and the Atchison, Topeka and Santa Fe Railway pass through Camp Pendleton approximately one-eighth mile from the reactor.

The applicants propose to contain the reactor in a spherical steel structure designed for a maximum leakage rate of 0.1% per day at pressure and with critical penetrations designed to permit frequent leak testing. Additional engineered safeguards are required for this site. Such safeguards proposed include a multiple, borated-water injection system to prevent extensive core meltdown in the unlikely event of a major break in the primary water system, a containment spray system, and an internal air cleanup system.

A meteorological factor favorable to the proposed reactor location is the fact that air movement from the site toward San Clemente occurs at most only a few percent of the time.

Extensive study of seismology in the area had been undertaken and earthquake resistant designs using conservative factors are proposed and are to be documented by the applicants.

The ACRS has emphasized that the engineered safeguards must be designed and reviewed with great care for both adequacy and reliability. Special attention should be directed to the safety injection system which must perform as proposed to validate the applicants' assumption of low release of radioactivity to the containment under accident conditions. A halogen removal system may be required. Design details of the holdup system for reactor off-gases resulting from routine operation will also require careful attention. The ACRS has recommended study of the consequences of rainout following an accident; the results of this study should be taken into account in the final design of the engineered safeguards.

In view of the favorable prevailing wind direction, conservative seismic design approach, and with engineered safeguards of the type proposed, it is the Committee's opinion that a pressurized water reactor of the type and power level proposed can be designed, constructed and operated at the site without undue hazard to the health and safety of the public.

Sincerely yours,

/s/D. B. Hall

D. B. Hall
Chairman

References attached.

References: San Onofre

1. Part B -- Preliminary Hazards Summary Report - Southern California Edison Company-Nuclear Station at Camp Pendleton, California - Unit No. 1, dated January 1963.

2. Amendment No. 1 to Application for Construction Permit and for License-Southern California Edison Company - San Onofre Nuclear Generating Station Unit No. 1, dated May 8, 1963.

3. Amendment No. 2 - Application for Construction Permit and for License - Southern California Edison Company, San Diego Gas & Electric Company, Bechtel Corporation, Westinghouse Electric Corporation - San Onofre Nuclear Generating Station Unit No. 1, dated July 2, 1963.

4. Amendment No. 3 - Application for Construction Permit and for License - Southern California Edison Company, San Diego Gas & Electric Company, Bechtel Corporation, Westinghouse Electric Corporation, - San Onofre Nuclear Generating Station Unit No. 1, dated August 22, 1963.

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

October 8, 1966

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

Subject: REPORT ON SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1

Dear Dr. Seaborg:

At its seventy-second, seventy-seventh, and seventy-eighth meetings, the Advisory Committee on Reactor Safeguards reviewed the proposed operation of San Onofre Nuclear Generating Station, Unit 1, at power levels up to 1347 MW(t) under a provisional operating license. This project was previously discussed by the Committee in its report of September 12, 1963. In its current review, the Committee had the benefit of discussions with representatives of the Southern California Edison Company, Westinghouse Electric Corporation, Bechtel Corporation, Southwest Research Institute, and the AEC Regulatory Staff, and of the documents listed. A Subcommittee of the ACRS met on five occasions to review proposed San Onofre operation. One of these meetings included a visit to the plant site.

The San Onofre plant is the first of a new generation of power reactors to be reviewed for an operating license, and represents an increase by a factor of more than two in power level over licensed pressurized water reactors now operating. As described in the Committee's previous report, Unit 1 is a pressurized, light water reactor located on an approximately 84-acrs site on the Pacific Coast, within and near the northern boundary of the Camp Pendleton Marine Corps Base, at a point 2.5 miles from the edge of the community of San Clemente, California. The applicant has provided a 28-foot high sea wall for protection against tsunamis.

The plant is contained within a 140-foot diameter steel sphere, designed to withstand an internal pressure of approximately 46 psig, and to meet a 0.1% per day initial leakage rate. A schedule of periodic containment integrated leak rate and containment penetration testing has been established.

The core consists of a cylindrical configuration of stainless steel clad, enriched uranium dioxide pellet, fuel assemblies. Control is achieved by means of 45 absorber-rod cluster assemblies which are operated in groups, and by boron dissolved in the primary coolant. The applicant's analysis indicates an adequate safety margin in core thermal design, although the core is more advanced in this respect than those of other reactors of this type now operating.

A borated water safety injection system, supplied from a 240,000-gallon tank, has been provided to protect the core in the unlikely event of a major loss-of-coolant accident. In addition, high pressure pumps provide protection against smaller primary system breaks. These installed systems will be tested prior to core loading. Off-site power is required to operate the feedwater pumps in the safety injection trains. The applicant has stated that the reliability and continuing availability of off-site power sources will be carefully tested at suitable intervals. Two auxiliary diesel generators will be provided, to assure the availability of power for shutdown heat removal and for the containment spray system. These diesels also supply power to an electrically-driven pump which is being provided as back-up to the steam-driven feedwater pump.

The reactor pressure vessel, which is 37 feet high and about 160 inches in diameter, is fabricated from stainless steel clad, low alloy steels. The maximum fast neutron dose is calculated to be about 6×10^{19} nvt over a 30-year vessel life. A surveillance program is planned, employing eight capsules containing specimens of vessel materials, located between the thermal shield and the reactor vessel. The Committee recommends that the Regulatory Staff assess the results of this program very carefully.

A program for periodic inspection of the primary coolant system has been proposed by the applicant. The Committee believes that this represents a generally sound approach but may wish to review aspects of this program, particularly the frequency and extent of inspections, at the time of final licensing review.

The applicant will provide separate cabinets for the pressurizer level and pressure control transmitters to decrease the likelihood of simultaneous failure of these channels.

It is the opinion of the ACRS that this reactor can be operated as proposed under a provisional operating license without undue hazard to the health and safety of the public.

Sincerely yours,

/s/ D. Okrent

David Okrent
Chairman

References attached.

References - San Onofre.

1. "Final Engineering Report, San Onofre Nuclear Generating Station, Unit No. 1; Section 4, Paragraph 4.2; Section 11, Paragraph 11.1", with errata sheet, undated, received November 10, 1965.

2. Amendment No. 8, Application for Provisional Operating License, dated November 12, 1965 transmitting Final Engineering Report and Safety Analysis, Volumes I-III.

3. Southern California Edison Company letter dated September 24, 1965 to AEC Division of Reactor Licensing transmitting Marine Advisors Report, "Examination of Tsunami Potential at the San Onofre Nuclear Generating Station".

4. Southern California Edison Company letter dated November 1, 1965 to AEC Division of Reactor Licensing, with enclosures.

5. Supplement No. 1 to the Final Engineering Report and Safety Analysis, dated March 23, 1966.

6. Westinghouse Heat Transfer Apparatus Report, "Evaluation of Damage, San Onofre Steam Generator", dated January 1966 - Revised April 1966.

7. Supplement No. 2 to the Final Engineering Report and Safety Analysis, transmitted by Amendment No. 11 dated May 20, 1966.

8. Technical Specifications, undated, received June 24, 1966.

9. Southern California Edison Company letter dated August 8, 1966 to AEC Division of Reactor Licensing, with enclosures.

10. Amendment No. 13, dated August 12, 1966, with attached Supplement No. 3 to the Final Engineering Report and Safety Analysis.

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

July 12, 1971

H. L. Price, Director of Regulation

ACRS COMMENTS ON SAN ONOFRE NUCLEAR POWER STATION UNITS 2 AND 3

Based on the available evidence the ACRS believes the acceleration (g) value of 0.5 proposed by the applicant for San Onofre Units 2 and 3 is not sufficiently conservative.

/s/ R. F. Fraley

R. F. Fraley
Executive Secretary

cc: ACRS Members

July 21, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3

Dear Dr. Schlesinger:

At its 147th meeting, July 13-15, 1972, the Advisory Committee on Reactor Safeguards completed its review of the application of the Southern California Edison Company and the San Diego Gas and Electric Company to construct San Onofre Nuclear Generating Station Units 2 and 3. This project also was considered at Subcommittee meetings on January 19, 1971, at the site; July 6, 1971, in Washington, D. C.; June 19-20, 1972, in Menlo Park; and at the 135th Committee meeting, July 8-10, 1971. During its review the Committee had the benefit of discussions with representatives and consultants of the Southern California Edison Company, the San Diego Gas and Electric Company, Combustion Engineering, the Bechtel Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

San Onofre Units 2 and 3 will be located adjacent to Unit 1 on the Pacific Coast near the northern boundary of Camp Pendleton, California. The reactors will be constructed on an 84-acre site about two and one-half miles from the nearest boundary of San Clemente, a town of approximately 18,000 people; the nearest city is Oceanside (population 39,000), 17 miles southeast. The site is within the Camp Pendleton Reservation and fronts on the Pacific Ocean. U. S. Highway 101 and the Atcheson, Topeka and Santa Fe Railway pass through Camp Pendleton approximately one-eighth mile from the reactors.

The proposed pressurized water reactors for Units 2 and 3 have design power levels of 3390 MW(t) each, 23 percent greater than Arkansas Nuclear One Unit 2 (ANO-2). Coolant average mass velocities, average linear power and flux peaking factors are the same as those for ANO-2, previously reviewed and reported in the Committee's letter of February 10, 1972. The Committee reiterates that adequate confirmation of the designer's expectations must be obtained to justify the higher power densities of these reactors and ANO-2.

The emergency core cooling systems (ECCS) for these reactors have been evaluated by the applicant using the approved Combustion Engineering evaluation model for use with the "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors." The applicant has agreed to design the San Onofre 2 and 3 ECCS in accordance with studies similar to those being conducted by Combustion Engineering for the ANO-2 facility. The final design should be reviewed by the Regulatory Staff and the ACRS prior to fabrication and installation of major components.

The applicant intends to use prepressurized fuel, citing benefits in lower fuel temperatures and control of cladding creep. The Committee reserves judgment on the benefits of prepressurized fuel under normal and possible accident conditions. The Regulatory Staff should complete its analyses of prepressurized fuel. The Committee wishes to be kept informed.

The applicant has agreed to meet the Regulatory Staff requirements for a Safe Shutdown Earthquake producing ground accelerations of 2/3g with occasional peaks up to 3/4g, and has proposed suitably amplified response spectra for use in design. Since the peaks above 2/3g are expected to be nonperiodic, they have no significant effect on the design spectra. The design includes lowered vertical profiles and separate foundation mats for structures such as containment, fuel pools, and the control room. The Regulatory Staff should review the design of these structures. The applicant is reviewing the effect of a severe offshore earthquake on the height of a tsunami and will provide appropriate protection against wave runup.

The applicant has indicated that he is considering studies similar to those made at San Onofre Unit 1 to determine the vibration characteristics of the major reactor components and the response of safety instrumentation to seismic loadings. The Committee encourages such experimental verification of the anticipated behavior of important components and instruments during earthquakes. The Committee suggests that a program be developed prior to reactor operation to guide or implement decisions concerning reactor operation in the event of a large earthquake in the region of the site.

The applicant has concluded that the likelihood of seismic surface displacement at the plant site is negligible. In addition, the applicant intends to establish the absence of seismically induced branch fault indications at the site during construction excavation. The Committee agrees that if, as anticipated, no such faults are found, the probability of ground displacement at the plant location is negligible.

The applicant has agreed to meet the Regulatory Staff requirements pertaining to appropriate tornado design criteria for the plant at this location. The Committee agrees that San Onofre 2 and 3 should meet suitable criteria and wishes to be kept informed concerning the resolution of this matter.

The Committee understands that the Regulatory Staff is reviewing the adequacy of the proposed design pressure for the reactor containment buildings. The Committee wishes to be kept informed.

The Committee recommends that the applicant give careful attention to the possible use of instrumentation capable of providing continuing quantitative information on the local performance characteristics in the high-power-density cores.

The applicant's quality assurance program appears generally satisfactory. However, the Committee suggests that the responsibility of the Chief Quality Assurance Engineer with respect to the functions of the Quality Control Engineer be precisely defined, since the two functions currently report to different organizational groups.

The Committee reiterates its previous comments concerning the need to study further means of preventing common mode failures from negating reactor scram action, and design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. The Committee wishes to be kept informed of the resolution of this matter.

Other problems relating to large water reactors, which have been identified by the Regulatory Staff and the ACRS and cited in previous reports, should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during construction and that, if due consideration is given to the foregoing, San Onofre Nuclear Generating Station Units 2 and 3 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

C.P. Siess

C. P. Siess
Chairman

Attachment:
References

References

1. Southern California Edison Company letters, May 28, 1970, and February 24, 1971, forwarding Vols. 1 through 4 and Vol. 5, respectively, Preliminary Safety Analysis Report for San Onofre Nuclear Generating Station Units 2 and 3

2. Amendments 1 through 13 to the License Application

FOR SEPTEMBER 11, 1973 LTR TO DIXY LEE RAY, TRANSMITTING MANGELSDORF MEMO TO MUNTZING RE FORKED RIVER, SAN ONOFRE 2&3, AND WATERFORD 3 ECCS DESIGNS, SEE PAGES 613-615 UNDER "FORKED RIVER".

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 10, 1981

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3 SEISMOLOGY AND GEOLOGY

Dear Dr. Ahearne:

During its 250th meeting, February 5-7, 1981, the Advisory Committee on Reactor Safeguards completed its review of seismic and geologic issues as part of its review of the application of Southern California Edison Company, et al, to operate San Onofre Nuclear Generating Station Units 2 and 3. These matters had been considered previously during a Subcommittee meeting in Inglewood, California on January 31, 1981. A tour of the site was conducted on January 30, 1981. The Committee commented previously on these matters in its report of July 21, 1972 on the application to construct these units. During the current review, the Committee had the benefit of discussions with representatives and consultants of Southern California Edison Company, the Nuclear Regulatory Commission (NRC) Staff, and the U.S. Geological Survey (USGS), as well as comments from members of the public. The Committee also had the benefit of the documents listed.

The San Onofre site is located on the coast of southern California in San Diego County approximately 62 miles southeast of Los Angeles, and within the boundaries of Camp Pendleton United States Marine Corps Base.

The geology and seismology of the site were reviewed in detail prior to issuance of construction permits for San Onofre 2 and 3 by the staff of the U.S. Atomic Energy Commission and its geological and seismological advisors, the U.S. Geological Survey and the National Oceanic and Atmospheric Administration, and by the Committee.

Extensive additional investigations were made after the issuance of construction permits for San Onofre 2 and 3. Included were detailed examinations of excavations along the Cristianitos fault and of the sea cliff exposures, geologic mapping, field examinations, offshore seismic reflection profiles, and analyses of recent seismic data. The geologic information and data from this work and other sources have amplified the knowledge of the hypothesized Offshore Zone of Deformation (OZD). The OZD lies about five miles offshore from the San Onofre site, and extends from the Newport-Inglewood fault zone south to the Rose Canyon fault zone. The OZD is considered potentially active and is the controlling geologic feature on which the seismicity of the San Onofre site is determined.

Although the site is located within one mile of the Cristianitos fault, investigations show that the 120,000 year old overlying terrace deposits have not been disturbed by fault activity. This and other available evidence indicate that the Cristianitos fault is "noncapable."

Offshore from the site is a region of faulting that has been termed the Cristianitos Zone of Deformation (CZD). The CZD lies oblique to the OZD and extends to within one mile of the OZD. Investigations have shown that the CZD should be treated as "noncapable."

A number of different methods were used to assess earthquake potential of the OZD, including the following:

- Historical seismicity
- Slip-rate
- Fault-length
- Fault area

Determination of potential earthquake magnitude using the various methods noted above, indicates that a surface wave magnitude of $M_S 7$ represents a reasonable and conservative interpretation of the available geological and seismological information. Potential ground motion at the plant site was evaluated assuming that an $M_S 7$ earthquake could occur along the OZD. Both empirical data and theoretical models were utilized.

Based on our review of the information which has become available since the Committee's construction permit review, we agree that the San Onofre 2 and 3 safe shutdown earthquake high frequency acceleration anchor point (0.67g) and design spectrum are acceptable.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

REFERENCES
1. Final Safety Analysis Report (FSAR) for San Onofre Nuclear Generating Station, Units 2 and 3, Vols. 1-23
2. "Safety Evaluation Report (Geology and Seismology) Related to the Operation of San Onofre Nuclear Generating Station, Units 2 and 3" - NUREG-0712, dated December 1980.
3. Letter from Richard Wharton, Attorney for Intervenors, to Richard Savio regarding San Onofre site seismology and geology, dated February 2, 1981.
4. Letter from H. William Menard, U.S. Department of the Interior, Geological Survey, to Harold Denton regarding USGS review of San Onofre site seismology and geology, dated November 26, 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 17, 1981

The Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON THE SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3

Dear Dr. Hendrie:

During its 251st meeting, March 12-14, 1981, the Advisory Committee on Reactor Safeguards completed its review of the application of Southern California Edison Company, et al, for licenses to operate the San Onofre Nuclear Generating Station Units 2 and 3 (SONGS 2 and 3). The Committee considered related seismic and geologic issues during its 250th meeting, February 5-7, 1981, and reported on these matters in its letter of February 10, 1981. Plant features were considered during Subcommittee meetings in Washington, DC on February 18, 1981 and March 11, 1981. During its review, the Committee had the benefit of discussions with the Applicant, Combustion Engineering, Inc. (CE), Bechtel Power Corporation, and the Nuclear Regulatory Commission Staff. The Committee also had the benefit of the documents listed.

SONGS Units 2 and 3 utilize CE Nuclear Steam Supply Systems with design power levels of 3410 MWt each. Control of both units will be accomplished from separate facilities within a shared control room. SONGS Unit 2 is the second CE plant to utilize 16x16 fuel. The containment buildings are prestressed concrete with a design pressure of 60 psig and a volume of 2.3 million cubic feet.

SONGS Unit 2 is the second CE-designed nuclear plant to use a digital computer as part of the reactor protection system. The computerized portion of the system was reviewed extensively by the NRC Staff and by the Committee during the review of Arkansas Nuclear One Unit 2 (ANO-2). The operating experience at ANO-2 and modifications to the software since the ANO-2 review were the subject of a Subcommittee meeting held on February 24, 1981. The ACRS believes the operating experience to date has been favorable. A data tie between the plant safety computer and the plant process computer has been provided, and its safety value is under review by the NRC Staff. The ACRS believes this feature is an asset to safety and recommends that it be retained on a permanent basis.

The Applicant described the organization of the plant staff, including the number of individuals engaged in the startup program, maintenance, engineering, operations, and health-physics. The compositions, duties, and interrelationships of the Safety Review Groups were reviewed. Training programs were also discussed. The Committee believes the Applicant is emphasizing plant staffing and personnel training, but that extensive further effort will be required to have staffing completed in accord with the Applicant's proposed operating schedule. The Committee further notes that the NRC criteria for staffing and training of operational support personnel are inadequately defined. The Committee recommends that the NRC Staff develop improved bases for judging the adequacy of the qualifications, training, and organizational structure for support personnel, especially in the areas of maintenance and water chemistry control.

The Applicant presented information on operating procedures for plant accidents. The procedures are organized by logic diagrams to aid the operators in diagnosing the accident and in providing instructions for corrective actions. The Committee notes that the SONGS Units 2 and 3 procedures represent a significant improvement over previous standard practice, but the Committee encourages continuing efforts to improve further the manner in which guidance is provided to operators in emergencies. We also recommend that the Applicant review procedures and training provided to deal with the occurrence of an earthquake to confirm that the guidance provided is adequate. We recommend that the NRC Staff include this matter in its reviews of emergency procedures.

NUREG-0737, "Clarification of TMI Action Plan Requirements," requires an unambiguous, easy to interpret indication of inadequate core cooling in nuclear plants. Core exit thermocouples and heated junction thermocouples located at discrete axial locations are part of the system proposed to meet this requirement. The proposed method looks promising and should be given appropriate attention by the NRC Staff. The Committee will review this proposal, along with other proposals, on a generic basis.

The Applicant is still engaged in preparation and submittal of emergency plans to the surrounding communities. When all the final plans are available, they will be reviewed by the Federal Emergency Management Agency. A test exercise is planned to evaluate the plans' effectiveness. Some questions exist concerning the ability of certain systems to function after a major seismic event. These include emergency alarm features to alert the public to an accident in the plant, meteorological and field radiation monitoring, communications, and emergency evacuation.

The ACRS has previously recommended that probabilistic safety analyses be performed for all plants in operation or under construction. The Committee believes that this recommendation is applicable to SONGS Units 2 and 3, but that such studies need not be performed prior to licensing of the plant.

The plants are still being reviewed for conformance with NUREG-0737. The resolution of four items remains open. The Committee believes these items should be resolved in a manner acceptable to the NRC Staff. The Committee wishes to be kept informed.

The Committee recommends that SONGS Units 2 and 3 employ a seismic scram such as is installed at Diablo Canyon, set to actuate at 50% to 60% of the safe shutdown earthquake acceleration.

The ACRS believes that, if due consideration is given to the recommendations above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that San Onofre Nuclear Generating Station, Units 2 and 3 can each be operated at power levels up to 3410 MWt without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. Southern California Edison Company, et al, "San Onfore Nuclear Generating Station, Units 2 and 3 Final Safety Analysis Report," Vols. 1-23, with Amendments 1 through 22.
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of San Onofre Nuclear Generating Station, Units 2 and 3, Docket Nos. 50-361 and 50-362," USNRC Report NUREG-0712, February, 1981.
3. U. S. Nuclear Regulatory Commission, "Supplement No. 1 to the Safety Evaluation Report Related to the Operation of San Onofre Nuclear Generating Station, Units 2 and 3, Docket Nos. 50-361 and 50-362," USNRC Report NUREG-0712, February, 1981.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

March 16, 1959

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: NUCLEAR MERCHANT SHIP REACTOR PROJECT (N. S. SAVANNAH)

Dear Mr McCone:

At its fourteenth meeting (March 12-14, 1959) the Advisory Committee on Reactor Safeguards continued its review of the Nuclear Merchant Ship (N. S. Savannah). Representatives were present from the Division of Reactor Development, Babcock & Wilcox Company, New York Shipbuilding Corporation, Geo. G. Sharpe Co., States Marine Corporation, and the United States Coast Guard. The ACRS has also had the benefits of thorough review of the ship by the Hazards Evaluation Branch and by Oak Ridge personnel. The pertinent documents are listed at the end of this letter.

Inasmuch as a final hazards review for the N. S. Savannah has not yet been submitted, and considerable information is still outstanding, the ACRS is not in a position to make a final recommendation to the Commission concerning the overall safety of this nuclear ship and the possible restrictions which may be required for the adequate protection of the public. However, the Committee has been asked to make an interim report. The ACRS has focused its attention primarily upon the nuclear propulsion system for which a large part of the information is available.

Pressurizer In the N. S. Savannah, the pressurizer not only maintains the primary system pressure, but also provides a heat source and sink for reducing pressure transients which occur during changes in load demand. The ACRS believes that the pressurizer can be made to work without jeopardizing the safety of the ship. The ACRS is somewhat uncertain as to the adequacy of the detailed design of the pressurizer

Subject: N. S. Savannah, cont.

inasmuch as the detailed information of transient response of the pressurizer has not been submitted to us. Under any circumstances, the actual performance of the system in addition to analog simulation, must be available before complete confidence in the adequacy of the system can be assured.

Rod Control and Scram System The design of this combined electrical rod drive and hydraulic scram system is new. The adequacy of this system should be demonstrated by extensive testing of prototypes and selected production units under all credible conditions of life, presence of solids in the water, misalignment, angle of roll and tilt, etc.

Containment The design of the containment vessel seems adequate provided the numerous penetrations do not themselves provide a channel through which fission products escape. The ACRS does not yet have sufficient information to decide whether the valving on these penetrations is adequate.

Interlocks on Loop Pumps A cold water accident initiated by the starting of an idle, cold-loop pump might create a serious nuclear excursion even if the scrams work. Therefore, it is essential that reliable and multiple interlocks be used to prevent this possible accident.

Miscellaneous Comments The considerations of the shock-loading under collisions appear to be adequate. However, ACRS has not fully completed its review in this area.

In view of the new design and unproven features associated with the reactor used in the N. S. Savannah, the Committee is of the opinion that the extension shakedown and testing required should not be carried out over the full power range at the dockside location. The Committee advises that severe limits must be placed on the operations during the dockside testing and understands that the necessity for restrictions has been recognized.

Subject: N. S. Savannah, cont.

The meteorological problems at this site (and also in general for rivers, estuaries, ports and at sea) have not been adequately resolved.

The required maneuverability of the ship results in a large amount of thermal cycling of the UO_2 fuel elements. This places an increased burden on the testing required to prove adequacy of fuel elements. It is obvious that a steam bypass around the turbine would reduce the problems associated with the thermal cycling.

The Committee is also not yet convinced that the operator will have a sufficiently informed staff to execute its overall responsibility for the safety of the nuclear ship. It is aware that crew members are in training. However, it urges that States Marine Corporation quickly acquire individuals who can understand and partake in the hazards analyses which are now under way.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman

cc: A. R. Luedecke, GM
H. L. Price, DLR

References:

1) Preliminary Safeguards Report - Babcock & Wilcox Company, BAW-1117 - Volume I - Revised December 22, 1958
Volume II - Revised November 3, 1958

2) DLR Comments to ACRS on Nuclear Merchant Ship Reactor Project (N. S. Savannah), November 4, 1958

3) DLR Report to ACRS on N. S. Savannah Control System, January 5, 1959

4) DLR Report to ACRS on(N.S. Savannah) Nuclear Merchant Ship Reactor, February 24, 1959

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

January 21, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: NUCLEAR MERCHANT SHIP (N. S. SAVANNAH)

Dear Mr. McCone:

The Advisory Committee on Reactor Safeguards Subcommittee on the N. S. Savannah met on January 7, 1960, with members of the Hazards Evaluation Branch and the Division of Reactor Development and representatives of the contractors on this project. During this meeting the Subcommittee was informed of the proposed augmentation program to deal with possible difficulties and to extend the testing period.

The ACRS is planning to continue its review of the N. S. Savannah at its meeting January 28-30, 1960. The Committee is anxious to complete its study and give advice to the Commission as soon as possible. A number of questions were resolved at the Subcommittee meeting. There are still some matters of design, testing, startup location, startup procedures, and operation outstanding which have not yet been adequately studied and resolved. Further, the details of the startup and testing programs have not been submitted. In view of this situation it seems unlikely that all of the questions will be resolved at this January meeting of the Committee. However, the review of the project will be advanced as far as possible and the Committee's views thereon will be furnished.

The details of the startup program may not necessarily determine the suitability of the Camden site from a safety point of view. Therefore, if possible, it seems especially opportune at this time to advise upon the suitability of the Camden site for startup since the necessary information for such a judgment may now be in.

Sincerely yours,

/s/
Leslie Silverman
Chairman

February 1, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: NUCLEAR MERCHANT SHIP (N. S. SAVANNAH)

Dear Mr. McCone:

At its twenty-third meeting, January 28-30, 1960, the Advisory Committee on Reactor Safeguards continued its review of the N. S. Savannah. The Committee had been asked by the Division of Licensing and Regulation to make an interim report especially regarding current conclusions as to the general design features of the reactor system and to include what other statements the Committee could make at this time relative to any other features of the reactor complex. The Committee was presented with additional information from the New York Shipbuilding Corporation, The Babcock & Wilcox Company, the Maritime Reactors Branch of the Division of Reactor Development, and the Hazards Evaluation Branch. The documents so far available to the ACRS are listed below.

The Committee agrees with the HEB that, in general, the design of the nuclear power system and its containment appears to be adequate for a nuclear propelled merchant ship subject to proof of component integrity in an extensive test program. However, there still remains an uncertainty as to the achievable filter efficiency of the iodine removal system. Since this filter is an important item affecting the adequacy of the overall containment, this problem needs further clarification.

Because containment is such an important factor in the free movement of this ship, the Committee believes that more consideration should be given to this item than is now planned:

1) The strength of the containment vessel has been confirmed by a hydrostatic test at static design pressure rather than at overpressure as is customary for a code vessel. Additional information should be supplied which will indicate the maximum stresses in the containment vessel and its structural supports under full loading of shielding and under the dynamic stresses of roll and pitch.

2) Careful consideration should be given to identify and tie down all components which could penetrate the containment vessel as missiles.

3) Periodic checks to establish the reliability of the filters in the ventilating system should be installed.

4) More consideration should be given to the feasibility of a device to monitor continuously the leakage through the containment vessel.

It is to be expected that the testing program will indicate that minor changes in design should be made. With the information now available, the Committee believes that the changes indicated by the testing program are not likely to require major design modifications in order to insure that the operating ship will not jeopardize the health and safety of the public.

Because of the prototype nature of the reactor during initial startup and early power operation, an unforeseen event might occur which would cause a major release of radioactivity at a time when the containment system is inoperative. Furthermore, the Committee believes that it is unwise to expose the public to radiation resulting from an accident in the experimental phases of the startup program when the number of individuals so exposed can be greatly reduced by moving this inherently mobile reactor. The Committee therefore recommends that extensive nuclear testing be carried out only at a site which has a far smaller population density than does the Camden site. The problem of setting the upper limit to the nuclear operation which may be carried out at Camden is complex. The Committee therefore recommends that a thorough study be made to resolve this question of upper limit and to analyze the problems of a specific alternate site.

The Committee wants to emphasize that its recommendation for an alternate startup site does not necessarily indicate, as of now, a belief that the N. S. Savannah should not enter highly populated ports after the characteristics of the reactor system are fully known. The entry into ports will be reviewed at a future time.

Sincerely yours,

/s/

Leslie Silverman
Chairman

References

1) BMI-B&W-634 - Simulation of the Heat-Transfer Characteristics of the Fuel Pins in a Nuclear Reactor, September 27, 1957.

2) BMI-B&W-639 - Simulation of a Control System for a Merchant-Ship Pressurized-Water Reactor, January 14, 1958.

3) BAW-1117 (Revision 1, December 22, 1958) Volume I - Nuclear Merchant Ship Reactor Project Preliminary Safeguards Report, September 15, 1958.

4) BAW-1117, Volume II, November 3, 1958 - Nuclear Merchant Ship Reactor Preliminary Safeguards Report.

5) BAW-1150 - Nuclear Merchant Ship Reactor Project Supplementary Information on Reactor Safeguards, June 1, 1959.

6) BAW-1154 - Nuclear Merchant Ship Reactor Project Control Rod Dependability Study, June 22, 1959.

7) BAW-1176, C-81, AEC R&D Report - Nuclear Merchant Ship Reactor Control Rod Driveline Tests, November 1959.

8) BMI-B&W-650 - Investigation of the Effect of a Steam-ByPass System on Control of the NMSR Plant, October 14, 1959.

9) ORNL CF 59-9-9 - Environmental Analysis of NS Savannah Operation at Camden, November 6, 1959.

10) N. S. Savannah Preliminary Safeguards Report Test, Start-up and Trials, prepared by New York Shipbuilding Corporation, Camden, N. J., November 23, 1959.

11) DL&R Report to the ACRS on the N. S. Savannah, November 4, 1958.

12) U. S. Weather Bureau Comments on BAW-1117, September 15, 1958.

13) DL&R Report to the ACRS on the N. S. Savannah with letter of transmittal to C. Rogers McCullough from H. L. Price, dated January 6, 1959.

14) DL&R Report to the ACRS on the N. S. Savannah, February 24, 1959.

References (continued)

15) DL&R Report to the ACRS on the N. S. Savannah, June 30, 1959.

16) U.S. Weather Bureau Comments on BAW-1150, July 9, 1959.

17) DL&R Report to the ACRS on the N. S. Savannah, November 25, 1959.

18) US Weather Bureau Comments on "N. S. Savannah Preliminary Safeguards Report; Test, Start-up and Trials", December 1, 1959.

19) U.S. Weather Bureau Comments on ORNL 59-9-9 (rev.), "Environmental Analysis of N.S. Savannah Operation at Camden", December 3, 1959.

20) Office of Health and Safety Comments on "Environmental Analysis of NS Savannah Operation at Camden", December 7, 1959.

21) DL&R Report to the ACRS on N. S. Savannah, January 12, 1960.

22) Report of the "N.S. Savannah" Review Committee to Dr. Frank K. Pittman, Director, Division of Reactor Development, October 1959.

23) Main Condenser Design Integrity with Regard to Prevention of Condensate Contamination with Seawater, by The Babcock & Wilcox Company, received October 1959.

24) ORNL-CF-59-6-55 - Application of Electroless-Nickel Brazing to Tubular Fuel Elements for the N. S. Savannah, June 2, 1959.

25) BAW - N. S. Savannah - Reactor Safeguards Information, Sections I, II, and III, December 3, 1959.

26) Main Condenser Isolation Support Plate Temperature Gradient During Steam Dump, N. S. Savannah, by The Babcock & Wilcox Company, January 6, 1959.

27) Memorandum from Frank K. Pittman to A. R. Luedecke, "Design Reviews of the N. S. Savannah", December 28, 1959.

28) Reactor Safeguards Information, The Babcock & Wilcox Company, January 27, 1960.

29) N. S. Savannah - Contract No. 529 - Report of Endurance Test - Main Steam Expansion Joints, January 21, 1960.

30) N. S. Savannah Summary of Iodine Removal Factors, received January 29, 1960 (undated).

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

July 25, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: N. S. SAVANNAH

Dear Mr. McCone:

At its twenty-seventh meeting, July 20-22, 1960, the Advisory Committee on Reactor Safeguards again reviewed the status of N. S. SAVANNAH program. Its previous review was summarized in our letter of February 1, 1960. The final hazards summary report has not yet been submitted in its entirety. The present review was held at this time, at the request of the Division of Licensing and Regulation, in order to provide a forum for discussion among the applicants, ACRS and the AEC staff which would expedite the final review by ACRS. This procedure was considered advisable in view of the tight schedule which the applicant is attempting to maintain. The ACRS limited its area of consideration to those factors which are essential for progressing through the initial start-up, not including testing at full power. Limits proposed by the applicant for power operation at Camden, New Jersey, were submitted too late to be evaluated. The Committee had discussions with the AEC staff, representatives from New York Ship and Babcock and Wilcox, and access to the information in the documents referenced below.

Although there has been considerable information developed by the applicant since the last ACRS meeting, the conclusion of the ACRS, at this time, remains much the same as it was in our letter of February 1, 1960. The reason for this is that all of the necessary information is not yet in and much of what has been submitted was too late to be read. However, the current review has been successful in providing all parties with a better identification of the areas of uncertainty.

Progress has been made relative to the questions concerning the containment vessel which were raised in the February 1, 1960 letter. Stress calculations have been submitted by the applicant. The

Maritime Reactors Branch has promised to obtain from the U. S. Coast Guard a statement of their comprehensive approval of the mechanical integrity of the containment vessel -- not including the leak test. New York Ship has stated that they will install a small fan and filter parallel to the main fan and filter through which the ventilation of the lower void is exhausted to the stack. This small fan and filter system would be available, in case of loss of the main fan anf filter system, to insure that a small negative pressure will exist in the upper and lower void space and that the exhaust from this area will pass through a filter should activity be released inside of the containment vessel. The design and efficacy of either filter system is yet to be established. Attention must be given to the problem of fire in the charcoal component. The ACRS recommends that the leakage of the containment vessel be checked by continuous monitoring in a manner which will insure that the leak rate will not exceed the design leak rate.

The major areas of concern with this reactor are now associated with component, system and critical testing, the start-up procedures and the adequacy of the start-up crew. Because of the lack of a prototype for this reactor which contains several significant components of new design, because of the desire to start nuclear operation at Camden, and because of the fact that the applicant has not been responsible for the start-up of other reactors in the past, the Committee believes that special attention must be given to these items. It does not appear that these areas will be adequately developed by the time now scheduled for fuel loading -- September-October 1960. The Committee urges that an unusually cautious schedule be adopted which will guarantee ample time to shape the operating staff and its technical back-up group into a smooth running, thoroughly informed, cooperative team.

Sincerely yours,

/s/

Leslie Silverman
Chairman

References:

1. BAW-1164 - Nuclear Merchant Ship Reactor Final Safeguards Report, Volume IV, Organization and Management of Operations, dated January 1960.
2. BAW-1164 - Nuclear Merchant Ship Reactor Final Safeguards Report, Volume VII, Power Plant Accidents, dated March 1960
3. Plant Operating Manual, N. S. SAVANNAH, Volumes I, II and III, dated April 1960
4. BAW-1164 - Nuclear Merchant Ship Reactor Final Safeguards Report, Volume I, Description of the N. S. SAVANNAH, dated June 1960
5. BAW-1164 - Nuclear Merchant Ship Reactor Final Safeguards Report, Volume VIII, Ship Accidents, dated June 1960
6. BAW-1164 - Nuclear Merchant Ship Reactor Final Safeguards Report, Volume III, Development of the N. S. Savannah Operating Manual, dated July 1960
7. N. S. SAVANNAH Final Safeguards Report, Test, Start-up and Trials, dated July 13, 1960
8. Site Report: York River, dated May 1960
9. N. S. SAVANNAH, Analysis of Structure of Containment Vessel, dated July 29, 1958
10. Development of Test Procedures, undated
11. Biographies of Start-up Organization, undated
12. Control Rod Drive Mechanism Testing, undated
13. N. S. SAVANNAH discussion of the Effects of Leakage, Temperature and Humidity on Containment Pressure, dated July 19, 1960
14. N. S. SAVANNAH, Summary of American Bureau of Shipping and U. S. Coast Guard Approval Action on Containment Vessel and Related Structural Plans
15. N. S. SAVANNAH, Description of Containment Vessel Purging and Reactor Space Ventilation, dated July 14, 1960
16. N. S. SAVANNAH, Locations of Radiation Monitoring Detectors, dated July 13, 1960
17. Study of Feasibility of Continuous Leakage Monitoring of the Containment Vessel, N. S. SAVANNAH, dated July 11, 1960
18. Adequacy of SAVANNAH Start-up Team, dated July 19, 1960
19. N. S. SAVANNAH, Proposed Sequence of Events for Start-up and Trial Program, dated July 1960

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

November 4, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Dear Mr. McCone:

The Advisory Committee on Reactor Safeguards has considered the Commission's request to convene a special one-day meeting of the full Committee during the week of November 13 to consider the Nuclear Merchant Ship N.S. SAVANNAH. We understand that additional documentation of the major points would be provided at that time.

The Committee plans to review this project during its special meeting of December 8, 1960, provided it receives the aforementioned documentation by November 18.

Sincerely yours,

/s/

Leslie Silverman
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON 25, D.C.

December 13, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON NS SAVANNAH

Dear Mr. McCone:

At its thirtieth meeting the Advisory Committee on Reactor Safeguards reviewed the NS SAVANNAH nuclear power plant through that phase identified as Sea Trials, Phase V. The Committee had the benefit of comments from members of the Division of Licensing and Regulation, New York Shipbuilding Corporation, Babcock & Wilcox Company, States Marine Lines, U. S. Coast Guard, and George C. Sharp Inc. The more recent Committee reports on this reactor system were addressed to you on February 1 and July 25, 1960. Information supplied to the Committee since that time (see the reports referenced below) enables the ACRS to remove the restrictions stated in these reports. The Committee now concludes that, with the two restrictions mentioned below, the reactor can be operated without undue hazard to the health and safety of the public.

1. The Committee believes that it would be imprudent to operate the ship reactor at more than 7 MW (10% of full power) in the start-up at Camden. At this level it is reasonable certain that the fuel elements would not release significant amounts of fission products into the containment vessel in the event of an accident. In spite of the very low probability of escape of radioactivity, it is not incredible that at higher power levels during the initial start-up operations amounts of radioactivity which could be harmful to the public might escape due to a now unforeseen maloperation of equipment or to faulty operational procedures. It must be recognized that the reactor is essentially mobile and can be transported safety to the York River site for further testing. Furthermore, from tests up to 10% of full power, sufficient information will be obtained about the behavior of the power plant to make a large shore support facility unnecessary.

It is, therefore, our belief that it is unwise to take a risk, however small, at initial start-up in the highly populated Camden area when the risk can be avoided by easily achievable means. The Committee wishes to emphasize that this opinion does not reflect a lack of confidence in the design, start-up procedures, or operation of this reactor. It merely recognizes that with any reactor which has not been extensively tested at power or as a full prototype, there exists a remote possibility of errors, in design or in operating procedures, which might be hazardous to the public.

2. Before the return of the ship to Camden with the primary system pressurized or before the start of Phase VI testing -- Extended Sea Trials -- the ACRS recommends that the performance of the NS SAVANNAH be documented for formal review by the staff and the ACRS. It is highly desirable to establish as soon as possible what limitations, if any, should be placed, because of reasons of safety to the public, upon subsequent operation of the ship.

The ACRS also recommends that the following items be followed up by AEC and settled to their satisfaction without further review by ACRS. ACRS, however, would like to be informed of the results of the AEC's effort as soon as possible.

a. The results of the start-up test program should be obtained through the 10% power level before the reactor is operated at higher power levels. The ACRS commends the good judgment of the New York Shipbuilding Corporation in proposing a review at this level.

b. A review should be made of the pertinence of recent failures of 17-4 PH steel to be used in the NS SAVANNAH control rod design, and to work out a satisfactory alternative design if the AEC deems it to be necessary.

c. A study should be made of the chance that undesirable events might result from the flooding of the reactor compartment, which has recently been proposed as a "last resort" protective procedure. In particular, the AEC should determine that no adverse stresses will be induced by the possible contact of cold water with the hot containment vessel, especially in the regions of the structural and attachment welds.

d. A review should be made of the program aimed at developing a nondestructive monitoring procedure which is capable of being used for frequent checking of the integrity of the filters for iodine and particulates. It is desirable that this test be available before the ACRS review mentioned under restriction (2) above takes place.

Sincerely yours,

/s/

Leslie Silverman
Chairman

References:

1. Supplementary Information Concerning the NS SAVANNAH: Status of Ebasco Services Design Review dated Nov. 18, 1960; NS SAVANNAH Containment Vessel, Stress Calculations for Dynamic Loading dated Nov. 3, 1960; ACRS Information dated Nov. 17, 1960.
2. Resume of Filter Characteristics - NS SAVANNAH dated Nov. 18, 1960.
3. SAVANNAH Supplement dated Nov. 30, 1960.
4. Reactor Systems Coordinated Tests, undated, received Dec. 2, 1960.
5. Supplement to Vol. I, Safeguards Report dated Sept. 16, 1960.
6. Nuclear Merchant Ship, Final Safeguards Report - Test, Start-up and Trials dated Sept. 16, 1960.
7. BAW 1164, Nuclear Merchant Ship, Final Safeguards Report, Vol. V, Crew Training dated Sept. 1960.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON 25, D.C.

April 4, 1962

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

Subject: N. S. SAVANNAH - INTERIM PHASE V ORGANIZATION

Dear Dr. Seaborg:

At its fortieth meeting on March 29-31, 1962, the Advisory Committee on Reactor Safeguards reviewed plans for the transfer of the responsibility for interim operation of the N. S. SAVANNAH prior to Phase VI from the New York Shipbuilding Corporation to the States Marine Lines, Inc. The Committee had the benefit of comments from members of the Division of Licensing and Regulation, the Division of Reactor Development, States Marine Lines, Maritime Commission, the Babcock and Wilcox Company, and references as listed below.

The N. S. SAVANNAH is currently based at Yorktown, Virginia and is undergoing sea trials offshore from that port. It is now in Phase V (initial sea trials) of its planned construction and operation schedule. It is intended to continue on an interim restricted basis of operation in the present area with no entry to any port other than Yorktown, Virginia until authorization to commence Phase VI is received.

The Maritime-AEC Joint Group has requested that the Commission approve the organization that the General Agent (States Marine Lines, Inc.) intends to use when it accepts responsibility for the operation of the ship. The States Marine Lines intends to employ this organization during interim operation and into Phase VI. A part of this interim organization is a nuclear reactor support staff aboard ship under the direction of States Marine Lines and including a nuclear engineer and other personnel from Babcock and Wilcox, instrumentation personnel from Todd Shipbuilding Corporation, and radio-chemistry personnel from outside contractors.

The responsibilities of the proposed organization, the experience of the ship's officers and crew, the inclusion of a nuclear reactor advisory staff, and the provision for the preservation of operational continuity have been clearly defined. The Committee believes that the transfer of the operational responsibility for the N. S. SAVANNAH from the New York Shipbuilding Corporation to the States Marine Lines, Inc. will not adversely affect its previous conclusions with regard to the operation of this reactor.

Sincerely yours,

/s/

F. A. Gifford, Jr.
Chairman

References:

1. NS SAVANNAH Operation (The Developmental Period) Safeguards Report, dated December 1961.
2. NS SAVANNAH Operation (Interim Prior to Phase VI), dated March 22, 1962.
3. Errata Sheet NS SAVANNAH Interim Operations, undated, recd March 28, 1962.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

July 27, 1962

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

Subject: REPORT ON PROPOSED PORT VISITS BY NS SAVANNAH - SAVANNAH, GEORGIA THROUGH GALVESTON, TEXAS

Dear Dr. Seaborg:

The Advisory Committee on Reactor Safeguards has reviewed the general plans for the operation of the NS SAVANNAH during the period up to and including the initial entrance of the ship into the Port of Galveston early in the winter of 1962-63. No detailed port reviews have been considered by the Committee. The Committee has considered the documents referenced below and has held discussions with representatives of the AEC-Maritime Joint Group, the Division of Reactor Development, the Regulatory Staff, and States Marine Lines, at one subcommittee meeting and three full Committee meetings. One of these meetings, held June 11 and 12, 1962 aboard the operating ship, was principally devoted to a full-scale review of the operating history.

Except for the containment leakage rate, the performance history to date appears to be in accordance with design expectations. No serious operational problems have arisen. The Committee recognizes that the NS SAVANNAH is the first ship of its kind and in many ways is a pioneering effort.

The AEC-Maritime Joint Group has been able to develop a plan and a set of interim factors of analysis for port entry for the initial phases of the operation. The proposed interim factors of analysis appear to be basically consistent with those utilized to judge other reactor installations. The ship is mobile and, even if the reactor is inoperative, the ship may be moved by available nonnuclear auxiliary power or towed away by properly equipped tugs. This mobility adds a new element to reactor safety considerations. On the one hand, it introduces the possibilities of collisions and groundings, while on the other hand it permits movement of the reactor itself away from populated areas. The collision statistics presented as a part of the

NS SAVANNAH Hazards Summary Report, and the design consideration given to making the reactor compartment as collision proof as possible, are reassuring. The Committee is in general agreement that mobility does provide an additional safety factor in that the reactor can be moved away from the public rather than vice versa. It should be pointed out that the use of ship mobility as a part of safety planning will require strict assurance as to the prompt availability of properly trained shipboard personnel, competently manned tug boats, and carefully laid safety plans. The interim analysis factors proposed, which make use of ship mobility to modify time-distance site requirements, would appear to be in general agreement with the Committee's views. However, the Committee believes that the proposal to assume an evacuation exposure time of one hour while the ship is moving in or out of a harbor, as compared with the assumed two-hour exposure time when the ship must be moved away from its dockside location, requires careful study by the Regulatory Staff.

The double containment, with halogen and particulate removal systems, provides engineered safety additional to that present in most other reactors and is considered in the interim factors of analysis. If full advantage is to be taken of this additional safety, requirements must be established and enforced to protect the integrity of the double containment. In particular, the early detection and plugging of leaks, the continuing efficiency checks of the halogen and particulate filters, and strict control of entry to the containment become very important.

The Committee recognizes that the continuing analysis of each port situation is a very difficult task. It therefore believes that the effort made by the AEC-Maritime Joint Group to develop interim port entry analysis factors has been a necessary one. It is our understanding that each port report and decision up to and including Galveston will be reviewed and approved by the Regulatory Staff.

It is clear that, during its initial operations, the NS SAVANNAH will be the object of much public attention; and its primary purpose will be that of exhibition. It will, therefore, attract large crowds of visitors. In view of the fact that the ship is a prototype, and knowledge that comes from experience remains to be acquired, the ACRS recommends that, during the period up to and including the Galveston port entry, the reactor be shut down and depressurized after berthing in a port if visitors are to be invited aboard in large numbers.

In summary, the Committee notes the following points:

(a) the interim port analysis factors are in general agreement with site criteria for other reactors;

(b) the availability of auxiliary power will be frequently checked and the ship mobility will always be positively assured;

(c) the containment leakage including halogen and particulate removal filters will be monitored frequently;

(d) the power history of the reactor will be controlled before port entry;

(e) to date the ship has encountered favorable operating experience.

Consequently, the Committee believes that the NS SAVANNAH can be operated at sea, and in ports that are acceptable under the proposed interim factors of analysis, without undue risk to the health and safety of the public.

Sincerely yours,

/s/

F. A. Gifford, Jr.
Chairman

References attached

References:
1. NS SAVANNAH Technical Specifications, dated April 1962
2. Seattle Site Report, undated, received May 22, 1962
3. Savannah Site Report, dated May 1962
4. New York Site Report, dated May 1962
5. Panama Canal Site Report, dated May 1962
6. Port Operation in the Matter of NS SAVANNAH, Report 970/5053, dated July 18, 1962
7. Request for Authorization for Initial Operation of NS SAVANNAH, dated July 18, 1962
8. NS SAVANNAH Filter and Sorption Unit Test Program, dated May 15, 1962
9. NSS-112 - SAVANNAH Nuclear Power, Summary Test Report, Reactor Operations at Yorktown, dated May 11, 1962
10. NS SAVANNAH Phase VI Extended Experimental Operation, dated June 4, 1962
11. Memo from E. Kemper Sullivan to Lowenstein, DL&R, dated June 18, 1962, Tug Availability in NS SAVANNAH Ports of Call
12. NS SAVANNAH, Handling Visitor Crowds, May 12-13, 1962
13. NS SAVANNAH Phase VI Operation - Statement by E. K. Sullivan, undated, received June 11, 1962
14. NS SAVANNAH Phase VI Program - Statement by E. K. Sullivan, undated, received June 12, 1962
15. The Probability of Damage to a Ship from Earthquakes, dated June 8, 1962
16. Preliminary Results of Dye Dispersal Field Test in Galveston Bay and Vicinity, dated June 8, 1962
17. Tug Availability in NS SAVANNAH Ports of Call, dated June 8, 1962
18. NS SAVANNAH - Containment Vessel Leak Tests (SML-NSS-1), dated June 4, 1962
19. NS SAVANNAH Shielding Survey, May 20-25, 1962
20. Iodine Removal Efficiencies of Reactor Compartment Ventilation Systems on NS SAVANNAH, dated June 8, 1962
21. ORNL-62-6-3, NS SAVANNAH Site Evaluations, dated June 9, 1962
22. NS SAVANNAH, Presentation of Ship Operating Plans, dated June 28, 1962
23. Curve NS SAVANNAH, Maneuvering Upon Loss of Power, dated June 28, 1962

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

November 14, 1962

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: REPORT ON N.S. SAVANNAH

Dear Dr. Seaborg:

At a special meeting, November 9-10, 1962, the Advisory Committee on Reactor Safeguards reviewed the operational experience of the N.S. SAVANNAH since its departure from Yorktown on August 20, 1962. The Committee had the benefit of the documents referenced below and of discussions with the AEC staff and with the representatives of the AEC/MA Joint Group, and the States Marine Lines.

The Committee is encouraged by the progress that has been made in solving initial shakedown problems and in developing a new administrative organization. Vigorous programs of rectifying operational difficulties, and improvement of procedures, training, and maintenance have been instituted. In general, the Committee believes that the short range program of interim operations interspersed by frequent shutdowns and inspections, can be carried out without undue hazard to the health and safety of the general public. At the same time the Committee reaffirms its belief that, when the vessel is in populated areas, every precaution should be exercised to reduce the effects of accidents which, although they may have a lower probability of occurrence, might be more severe than those postulated. With these facts in mind, the Committee recognizes certain long range problems upon which it believes actions should be initiated immediately. These will be outlined below.

Experience with the N.S. SAVANNAH has shown that it is very important to have available an auxiliary electrical power source in port. It is even more important to have a reliable auxiliary power source at sea or during port entry in event of a scram or other unusual circumstances. The Committee urges early action to provide a reliable auxiliary source of power, with propulsion reversal capability, adequate for safe handling of the ship in ports and at sea.

The Committee would like to point out that the use of inert gas within the containment, a procedure necessary to diminish the fire hazard, has reduced drastically the possibility of carrying out maintenance within the containment and the possibility of making observations of the primary system performance at pressure and temperature. Experience at other reactors has indicated that this is an important facet of reactor safety. The Committee urges that all possible speed be employed to change the present control rod system to one which will avoid the problems observed in the present system and which will permit adequate observation and maintenance within the containment.

The Committee notes that the containment leakage and filter efficiencies are being checked frequently and recommends continued vigilance since performance has not always met specifications. The Committee believes that the proposed scheme for modification of the reactor compartment ventilation, cooling, and filtration system will be a desirable improvement to the effectiveness and safety of this system and urges its early completion.

The ACRS was informed of the proposed plans to visit the ports of San Francisco, Long Beach, Los Angeles, Honolulu, Portland, San Diego, and Balboa before proceeding to Galveston. The N.S. SAVANNAH still has not had a long operational period and therefore decisions as to the ports visited and the docks selected should be made conservatively. In particular, where more than one suitable docking site is available in the same general area, the Committee believes that the more conservative should be chosen regardless of other considerations.

In summary, the Committee believes that by taking all practicable precautions the N.S. SAVANNAH, prior to the Galveston overhaul, can enter ports found suitable under a conservative application of the Interim Port Analysis Factors without undue hazard to the health and safety of the public. However, the Committee recommends that only after the long term problems identified above have been rectified, should consideration be given to any modification of the existing Interim Port Analysis Factors except to clarify them or make them more conservative.

Sincerely yours,

/s/

F. A. Gifford, Jr.
Chairman

References Attached

References:

1. Memo frm Robb to Price, dated 8/30/62 - 970/4608, Subject: Memorandum & Authorization dated August 3, 1962 in matter of N.S. SAVANNAH; Proposed Significant Change No. 13.
2. Memo frm Robb to McCool, dated 10/2/62 - 970/4870, Subject: Memorandum & Authorization dated August 3, 1962 for Operation of the N.S. SAVANNAH.
3. Memo frm Robb to Price, dated 10/9/62 - 970/4918, Subject: Memorandum & Authorization dated August 3, 1962 in matter of N.S. SAVANNAH; Proposed Significant Change No. 17.
4. Memo frm Robb to Price, dated 10/15/62 - 970/4955, Subject: Memorandum & Authorization dated August 3, 1962 in matter of N.S. SAVANNAH; Proposed Significant Change No. 17, Amendment No. 1
5. Memo frm Robb to Price, dated 10/19/62 - 970/4967, Subject: Memorandum & Authorization dated August 3, 1962, in matter of N.S. SAVANNAH.
6. Memo frm Robb to Price, dated 10/24/62 - 970/4984, Subject: Memorandum & Authorization dated August 3, 1962 in matter of N.S. SAVANNAH; Significant Change No. 17, Amendment No. 2.
7. Memo frm Robb to Price, dated 10/23/62 - 970/4990, Subject: Memorandum & Authorization dated August 3, 1962 in matter of N.S. SAVANNAH; Significant Change No. 17, Amendment No. 2.
8. Memo frm Robb to Price, dated 11/2/62 - 970/5096, Subject: Memorandum & Authorization dated August 3, 1962 in matter of N.S. SAVANNAH; Proposed Significant Change No. 19.
9. N.S. SAVANNAH Operation Analysis for the Period - May 1, 1962 - November 1, 1962 - 970/4978 SML-NSS 3.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

February 6, 1963

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

Subject: REPORT ON N. S. SAVANNAH

Dear Dr. Seaborg:

At its 46th meeting, January 31 - February 2, 1963, the Advisory Committee on Reactor Safeguards reviewed the request submitted by the AEC-Maritime Joint Group for approval of Proposed Significant Changes No. 13 and No. 18. The Committee had the benefit of the references listed below and discussions with representatives of the AEC-Maritime Group, the Marvel Schebler Corporation, and the Regulatory Staff. Discussions on these changes were also held at a subcommittee meeting November 30, 1962, and the 45th ACRS meeting during December 1962.

It is not intended in this report to review comments made in past Committee reports on this reactor. However, the Committee still believes that a reliable auxiliary power source with adequate maneuvering and propulsion reversing capability should be installed. The Committee believes that the installation of a control rod drive system that will not require an inert gas atmosphere in the primary containment should be carried out as soon as possible.

These and a number of other overhaul items appear to be postponed to some indefinite time beyond their original proposed date at Galveston. This leads the Committee to recommend that there be a complete review of the status of the ship and its proposed post-Galveston operation before it puts to sea again.

In regard to Change No. 13, the proposed ventilation, filter, and monitoring changes should substantially decrease any radioactive releases associated with the reactor compartment. The use of spring-loaded doors and the alarm system proposed will help to assure that negative pressures will be maintained in the reactor compartment. In addition, the improved installation will reduce temperatures in the reactor compartment and permit more frequent inspections. The Committee believes that the revisions proposed for the filter and ventilation systems provide adequate assurance that they will function properly in event of an accident resulting in the release of fission products to the compartment.

Change No. 18 requests the substitution of hermetically sealed Marvel-Schebler control rod drives instead of the existing hydraulic-electric drives. In general, the Committee is impressed with the advantages to be gained with the new type of rod drive. The Committee has a reservation concerning the lack of a spring to aid in the initiation of rapid rod insertion in event a scram is required.

A new method of rod actuation and control is proposed. This system is novel and unproven. As presently conceived it would give no read-out of individual control rod position in the control room and would operate the rod groups according to a pre-set plan. The Committee can see no valid reason for not indicating the rod position in the control room and recommends that this be done. The Joint Group has stated that it is feasible to do this. In many ways the proposed method of control may prove in the long run to be safer than others. However, this reactor is a first of its kind with no prototype, no hot critical flux measurements, and no incore instrumentation. Therefore, the Committee believes that introduction of this new system should be carried out carefully and prudently. The Committee recommends that the rods, even with the new system, be operated one group at a time from the control room by means of a group select switch as is currently being done with the system now in use.

The Committee believes that the proposed ventilation system changes outlined in Change No. 13 as modified by the Regulatory Staff represent a distinct improvement in safety of the N.S. SAVANNAH. The Committee believes that with the changes stated above, the Marvel-Schebler rod drive system also represents an improvement in safety. However, the Committee cannot recommend full approval of the proposed control system since the testing program has not been completed.

The Committee specifically cannot recommend any approval of operation beyond Galveston and feels that such approval must await a full review of the ship status at that time.

Sincerely yours,

/s/

D. B. Hall
Chairman

References:

1. BAW-1249, Vol. I, N.S. SAVANNAH Replacement Control Rod Drives Safeguards Report, dated October 1962.
2. BAW-1249, Vol. II, N.S. SAVANNAH Replacement Control Rod Drives Safeguards Report, dated October 1962.
3. BAW-1203, Vol. I, Nuclear Merchant Ship Reactor Project, Extended Zero Power Tests -- N.S. SAVANNAH Core 1, Final Report, dated January 1961.
4. Proposed Significant Change No. 13, Memo 970/4608, dated August 30, 1962.
5. Proposed Significant Change No. 13, Modified Reactor Space Ventilation Operation Description, Rev. 1, dated November 28, 1962.
6. Answers to Questions Concerning Significant Changes 13 and 18, undated, received January 24, 1963.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

February 6, 1963

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

Dear Dr. Seaborg:

Attached is a letter from Mr. E. Jansson, Swedish Atomic Energy Board, requesting additional information concerning the ACRS report dated November 14, 1962 on the N. S. SAVANNAH. Our interim reply to Mr. Jansson is also attached for your information.

The Committee has discussed Mr. Jansson's request and has concluded that since interpretation of opinions to persons outside the AEC, as requested, is not the function of the ACRS, this letter should be referred to you. We will, however, be happy to provide any assistance we can in formulating a reply to Mr. Jansson.

It should be noted that the information provided to Mr. Jansson in connection with the Elk River project was of a factual nature and dealt primarily with listing appropriate reports in the public domain.

Sincerely yours,

/s/
D. B. Hall
Chairman

Attachments:
1. Original ltr 12/28/62 frm Jansson to Gifford.
2. Cy of ltr 1/18/63 frm Gifford to Jansson.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

April 27, 1963

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

Subject: REPORT ON N. S. SAVANNAH

Dear Dr. Seaborg:

At its forty-seventh meeting on April 11-13, 1963, the Advisory Committee on Reactor Safeguards reviewed the status of the N. S. SAVANNAH. This project was last considered at the Committee's meeting of January 31 - February 2, 1963, as reported on February 6, 1963. The Committee, in this most recent review, had the benefit of presentations by the AEC-Maritime Joint Group on N. S. SAVANNAH, States Marine Lines, U. S. Coast Guard, ORNL, Babcock & Wilcox Company and AEC staff, and of the reports referenced.

Numerous significant changes to the ship, its propulsion system, and its methods of operation are in progress or completed. The Committee considers that most of these changes improve operability or safety of the ship. Since there have been no reported malfunctions of the present control rod drives, the Committee believes that the continued temporary use of the present drives and the use of inert gas in the containment is acceptable pending installation of the new drives.

The Committee believes that the question of pressure relief and scram settings for operation at the proposed 80 MW(t) power level can be resolved by the AEC staff and the Joint Group, so as to insure that transient pressures do not exceed design values.

Concerning the removal of the loss of flow scram, the Committee suggests that the Joint Group and the AEC staff assure themselves that there is no reasonable possibility that a loss of coolant flow to the core could occur without causing a scram in time to prevent substantial core damage. Otherwise, this change does not appear to constitute a reduction in safety.

In view of the experience gained to date in the control of visitors, the Committee sees no objection to increasing the limit on the number of visitors on board at any one time to 750.

A two-day sea trial is proposed after completion of dockside testing. The Committee urges that both the dockside testing and the sea trials be of such duration and design as to test the new or modified installations thoroughly.

The Committee must emphasize the need for a greater variety of technical skills aboard this vessel than are customary on merchant ships. It is imperative that competence in nuclear skills must not be allowed to fall below that of similar land based power reactors, and that professional standards of such personnel as electronics experts and health physicists must not be compromised. It is noted that a re-alignment of operating personnel has been instituted, which, while it is expected to increase operational dependability through additional line function officers, does reduce the actual number of persons available in certain technical areas.

Operation of the N. S. SAVANNAH in crowded metropolitan areas can only be reconciled with current views on reactor siting by the incorporation of both assured engineered safeguards and assured ability to remove the ship rapidly in event of an accident. The Committee has given considerable attention to assuring itself that the engineered safeguards and mobility of the N. S. SAVANNAH are established and demonstrable beyond any reasonable doubt at all times when the ship is in such areas.

The two principal engineered safeguards that provide protection against release of radioactivity in the unlikely event of an accident are (1) the reactor containment vessel and (2) the reactor compartment with its air cleaning system. The design of these has been reviewed by the Committee previously. The containment vessel is believed to be structurally adequate, although its leak rate has increased since initial testing. The latest test shows that containment vessel tightness is still acceptable. However, since leakage may increase with plant operation, the Committee is of the opinion that 'as-is' leakage tests must be made at the proposed regular intervals. After the detection and elimination of leaks wherever possible, the containment vessel should then be retested. The Committee wishes to point out that the present maximum practical test pressure is a small fraction of the design pressure, and that extrapolation to higher pressure is necessarily uncertain, in particular since the high pressure may open leakage paths not found at lower pressures. The Committee suggests that a test procedure at the maximum practicable pressure be developed to provide the best possible basis for extrapolation from the leak test pressure to the pressure that would exist following a severe accident.

The reactor compartment is to be maintained at below atmospheric pressure so that any leakage will always be into the compartment. The minimum differential between the inside and the outside of the compartment has been established at about 0.5 inches of water. Since the pressure on opposite sides of the ship may differ by several inches of water due to wind effects, the adequacy of the selected compartment pressure and of the monitoring installation must be assured. Otherwise, a small out-leakage could exceed the total radioactive discharge from the air cleaning system.

The air cleaning system is relied upon to remove substantially all particulate and halogen radioactivity from the air exhausted from the reactor compartment. The system is designed as duplicate, parallel units of which one will be in service, and the other, in clean tested condition, will be available in an emergency. Filters of the type used in this system have been shown to be very effective on test materials. In addition, it is reported that the previous on-board installation did not deteriorate over its service period. However, the radioactive materials that might be released in event of an accident may be removed with different efficiencies than are found with test materials. In addition, the effectiveness of the air cleaning system depends on the integrity of its structure and on its installation and maintenance.

The Committee believes that as a factor of conservatism, the assumed performance of the air cleaning system should be lower than the test values due to (1) uncertainties in the parameters controlling the air cleaning process and (2) the possibility of faults in the installation or of deterioration during operation, particularly due to vibration. Furthermore, the failure of a penetration seal or the opening of a moderate size leak in the containment in case of an accident might produce sufficient pressure to rupture the filters of the cleaning system with substantial loss of filter protection.

The Committee has suggested installation of on-board testing equipment for frequent air cleaning efficiency determinations. It believes that such measurements should be made prior to each port entry. However, until such testing equipment is installed, the existing test procedures should be used for frequent checks. The present research programs on air cleaner performance and on the nature of actual accident releases should help to determine whether restrictions may be relaxed. Laboratory and operating tests to determine the effect of vibration on air cleaner performance are desirable. Installation of a pre-filter which would prevent damaging the 'absolute' filter and blanketing of the carbon beds by condensing steam should be considered.

In addition to engineered safeguards, the N. S. SAVANNAH depends upon its mobility to provide adequate protection in populated areas. This mobility in event of an accident may be achieved either by an adequate auxiliary power system or by prompt availability of tugs. In crowded waters, a reactor scram could lead to a loss of control of ship movement and therefore could contribute to a ship accident. The Committee is still of the opinion that an adequate auxiliary propulsion system is necessary in this prototype ship, and believes that the Joint Group should continue to explore with the Coast Guard and other responsible group ways to install such a propulsion system. This statement is intended to apply only to the N. S. SAVANNAH and does not pass judgement on future nuclear ships. Under present conditions, an acceptable temporary alternative appears to be to require tugs in attendance, or on 30-minute call at such times as required under the pre-Galveston porting criteria unless the reactor is shut down and at least partially depressurized. This restriction could be removed (except possibly for the largest cities) if auxiliary power suitable for maneuvers in restricted water during emergencies, even without tugs, can be installed aboard the SAVANNAH and adequately demonstrated.

If, due to any of a variety of reasons such as fog, pier blockage, or wrecks, the mobility of the N. S. SAVANNAH cannot be assured, the Committee believes that the reactor should be shut down and depressurized when at dock, unless the site meets the guide lines of 10CFR Part 100 as modified by permissible credit for the engineered safeguards and by the recent reactor operational history. The Committee believes that, on an interim basis, the values and calculational methods used in the pre-Galveston porting criteria as modified because of the immobility of the ship should be applied in evaluating the engineered safeguards.

The Committee considers the new "Proposed Interim Operating Specifications" to be simpler and more practical than the guides used prior to the Galveston overhaul. However, because of questions that have been raised about the containment leakage rate, the efficiency of the air cleaning system, and ship mobility under conditions which may exist in case of an accident, the Committee is of the opinion that the ship should continue to use the procedures and criteria in effect prior to the Galveston overhaul.

The pre-Galverson procedures and criteria state that "While under way and accompanied to two or more tugs, a one hour exposure limitation will be assumed in determining that exposure to any member of the general public will not exceed 25 rem whole body or 300 rem thyroid". The Committee believes that the one-hour exposure limitation can also be applied if the ship is at dockside, with two or more tugs under power and in attendance at the ship, and if no external conditions prevent movement of the ship.

In summary, the Committee has reviewed the operating history of the N. S. SAVANNAH up to the present period of overhaul at Galveston. It has reviewed the significant changes being made during this overhaul, and considers that these generally represent improvement in operability and safety of the ship. The Committee believes that, subject to the points specified in the above paragraphs, the N. S. SAVANNAH can continue to be operated and visit ports under the interim criteria of August 1, 1962, without undue hazard to the health and safety of the public.

Sincerely yours,

/s/
D. B. Hall
Chairman

References:

1. TODD/SML-NSS 6, N.S. SAVANNAH Operations, May 1962-March 1963, dated March 15, 1963.
2. TODD/SML-NSS-10, N.S. SAVANNAH Technical Specifications, dated March 1963.
3. BAW-1264, CA-7, N.S. SAVANNAH Safeguards Report for 80-MW Operation, February 1963.
4. Evaluation of Radiation Damage to the N.S. SAVANNAH Reactor Vessel, dated March 1963.
5. Evaluation of Electrical Cable Operation in the N.S. SAVANNAH Containment Cupola, dated March 1963.
6. "Organization Chart N.S. SAVANNAH, dated April 8, 1963", and attached summary of changes, dated April 10, 1963.
7. N. S. SAVANNAH - A Discussion of Take Home Motor Performance and Modifications, dated March 26, 1963.
8. Letter from O. C. Rohnke, U.S. Coast Guard, to H. L. Price, AEC, dated April 9, 1963, Subject: N.S. SAVANNAH Emergency Propulsion, U.S. Coast Guard Policy Concerning Requirements.
9. Proposed Interim Operation Specifications, dated March 15, 1963.
10. Evaluation of the Consequences of the Maximum Credible Accident for the N.S. SAVANNAH, dated March 1963.
11. Proposed Interim Operating Specifications, Revised April 9, 1963.
12. Proposed Significant Change No. 20, Memo 970/5530, dated Feb. 14, 1963.
13. Proposed Significant Change No. 21, Memo 970/5632, dated Feb. 25, 1963.
14. Proposed Significant Change No. 22, Memo 970/5730, dated March 8, 1963.
15. Proposed Significant Change No. 23, Memo 970/5810, dated March 20, 1963.
16. Proposed Significant Change No. 24, Memo 970/5811, dated March 21, 1963.
17. Proposed Significant Change No. 25, Memo 970/5879, dated March 22, 1963.
18. Proposed Significant Change No. 26, Memo 970/5884, dated March 26, 1963.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON 25, D.C.

November 13, 1963

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

Subject: REPORT ON N. S. SAVANNAH

Dear Dr. Seaborg:

At its fifty-first meeting on November 7-8, 1963, the Advisory Committee on Reactor Safeguards reviewed the proposal for 30 MW(t) dockside operation of the N. S. SAVANNAH at Todd Shipyards in Galveston, Texas. The Committee at its fiftieth meeting was given a report of the ship's status, and the Committee last commented on the N. S. SAVANNAH at its forty-seventh meeting. In its present review, the Committee had the benefit of discussions with the AEC-Maritime Joint Group, Babcock and Wilcox Company, American Export Lines, and the AEC Staff, and of the reports referenced.

The N. S. SAVANNAH has been immobilized at the Todd Shipyards since February 1963. A change in the operating organization has necessitated an extensive training program which is reported to be progressing satisfactorily. Improvements have been made in the air cleaning installation for the reactor compartment both by provision of on-board testing capability and air cleaning units of improved efficiency.

The Committee was informed of the results of containment vessel leak testing. Tests made at 60 psig, one without preconditioning, have resulted in acceptably low observed leak rates of 1.25% to 1.4% per day.

Start-up after the extended period without nuclear operation will present a problem in that the source has decayed to a level below that which can be observed on installed instrumentation. The Joint Group proposes to augment the equipment with sensitive scalers and institute start-up procedures intended to preclude any undesirable power excursion. There are two requirements for neutron sources to be used at reactor startup. First, there must be enough neutrons present so that, allowing for leakage and absorption, the fission chain reaction is initiated promptly on withdrawal of control rods. Since the emission of these neutrons is a statistical matter, it is clear that it must be practically certain that a chain reaction will be initiated well before the time that the rods are withdrawn to positions beyond prompt critical. Second, in order to make absolutely sure that the control rods are not so withdrawn, neutron detectors of sufficient sensitivity should be located so as to give clear indication of the presence of neutrons at all times during rod withdrawal. This Committee believes that it is unwise to start up any reactor in which the source level of neutrons is so low that the detector is unable to distinguish the source level from background. The Committee would like to be assured that appropriate procedures are used so that neutron detection will be observed at early stages of approach to criticality. It is assumed that the procedures described in the approval of Change No. 25 will be followed.

During operation at low powers for operator training and for physics tests, such as rod calibrations, it is proposed to leave the power level trip points set at the normal value of 120% of full power (96 MW(t)). This choice has been made to avoid modifications in the safety circuitry and provide the operators with training in actual procedures. The Committee does not endorse the wide discrepancy (300%) between operating levels and scram levels, and suggests that the Regulatory Staff confer with the Joint Group in order to determine the extent of compromise in protection resulting from the decision to leave the trip circuits unchanged.

An analysis of the site has led the operator to conclude that, for the proposed limitation of reactor power to 30 MW(t), the reactor operation will conform to the guides set forth in 10CFR100 for stationary reactors. However, the Committee has been assured adequate tug service will be available as usual on short call.

On several occasions the Committee has commented on the control rod drives used in the N. S. SAVANNAH and expressed dissatisfaction with the operating provisions made necessary by leakage from the hydraulic drives. These operational difficulties will be removed by replacement with the Marvel-Schebler rod drives. The Committee was informed that the schedules for testing and operator training for these new rod drives will extend into the early Fall of 1964. At present, installation is not scheduled before 1965. The Committee regrets the delay in installing the new rod drives and urges that they be installed as soon as the testing program has demonstrated a satisfactory level of reliability. The Committee believes the use of the presently installed rod drives provides an adequate control function for reactor operations;

the objection which the Committee has expressed to the present rod drive system has arisen from the flammable nature of the hydraulic actuating fluid and the resulting requirement for maintaining an inert environment in the containment vessel.

In summary, the Committee has reviewed the changes in operating personnel, changes in the reactor facility, and the proposed dockside operation at Todd Shipyards. Subject to the points raised in the paragraphs above, the Committee believes that the N. S. SAVANNAH can be operated at the proposed site up to 30 MW(t) without undue hazard to the health and safety of the public.

Sincerely yours,

/s/
D. B. Hall
Chairman

References:

1. TODD/SML-NSS 15, "N. S. SAVANNAH Technical Specifications", dated May 1963.
2. Revised pages to TODD/SML-NSS 15, undated, received June 4, 1963.
3. U.S.A.E.C. Approval of Change No. 25, dated May 2, 1963.
4. STS-1, "N. S. SAVANNAH - Hazards Summary Report for Dockside Operation", dated September 1963.
5. STS-1, Supplement 1, "N. S. SAVANNAH - Hazards Summary Report for Dockside Operation", dated October 1963.
6. Memo 970/6991, Filter System Test Results, dated October 31, 1963.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

February 15, 1964

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

Subject: REPORT ON N.S. SAVANNAH

Dear Dr. Seaborg:

At its fifty-third meeting on February 13-15, 1964, the Advisory Committee on Reactor Safeguards reviewed the proposal for the 1964 sea trials of the N.S. SAVANNAH operating out of the Todd Shipyards in Galveston, Texas, and to be conducted in the Gulf of Mexico. The Committee had the benefit of the report of a subcommittee meeting held on the N.S. SAVANNAH at Galveston on February 6, 1964, and of discussions with the AEC-Maritime Joint Group, the Savannah Technical Staff, the American Export and Isbrandtsen Lines, and the AEC staff, and of the reports referenced.

Seven trials, totaling 16 days at sea, are planned for the period February-May 1964. The ship will operate in a triangular area, approximately 200 miles on a side, in a little traveled section of the Gulf of Mexico south of Galveston.

With certain exceptions, all operations at dockside and during sea trials will be performed using procedures and guides that have been reviewed and commented on by the ACRS in previous reports. Several changes in approved Technical Specifications are made necessary or desirable by changes in the ship and the operating agent. The Committee concurs in the following changes requested in STS-6 (Dec. 1963):

Item B.1 & B.5 - Permissible containment leakage rate shall not exceed 2.0% of the contained free volume in 24 hours at 60 psig. Tests shall be conducted quarterly at an internal pressure of 60 psig.

Item I.1.a - The N.S. SAVANNAH shall be operated for the Atomic Energy Commission/Maritime Administration Joint Group by the American Export & Isbrandtsen Lines.

Item I.1.d - Changes in organization and responsibility for dockside and sea trial operation as described.

The Committee has not reviewed the question of whether the N.S. SAVANNAH operations in port can be judged on the basis of site criteria for stationary reactors and does not wish to imply by what is stated in this letter that it has accepted the principle that it can be so judged. The Committee recommends that present Galveston mobility requirements remain unchanged insofar as availability of tugs is concerned. The Committee is of the opinion that, when the ship is at dockside during the period of the Galveston sea trials only, no significant increase in hazard is involved in making the following exceptions to Technical Specifications as stated in STS-6:

Item I.4.f - It shall not be mandatory to demonstrate the operability of the take home motor.

Item I.6 - There shall be no requirement to maintain vacuum on the main condenser.

The Committee recommends that the conditions calling for inerting the containment vessel be continued as in the past.

It is the opinion of the Committee that the proposed sea trials out of Galveston, Texas can be conducted within the limitations described above without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

<u>References</u>:

1. STS-2, "NS SAVANNAH - Galveston Outage Report", dated November 1963.
2. STS-6, "NS SAVANNAH - Hazards Summary Report for 1964 Sea Trials", dated December 1963.
3. Revised and additional pages for STS-6, undated, received January 23, 1964.
4. Revised pages for STS-6, undated, received January 30, 1964.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

April 9, 1964

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

Subject: REPORT ON N. S. SAVANNAH MARITIME ADMINISTRATION BACKUP CREW

Dear Dr. Seaborg:

At its fifty-fourth meeting on April 2-4, 1964, the Advisory Committee on Reactor Safeguards reviewed the training, qualifications, and experience of the proposed Maritime Administration (MARAD) backup crew and the operations to be carried out by this group during sea trials out of Galveston, Texas. The Committee had the benefit of a subcommittee meeting on March 27, 1964, and discussions with representatives of the AEC-Maritime Joint Group, the MARAD crew senior officers, the STS training staff, and the AEC staff. The Coommittee also had the benefit of the documents referenced.

The training program of the MARAD crew was outlined in some detail. It appears equivalent to corresponding training given to the regular American Report and Isbrandtsen (AE&IL) crew. The qualifications and specified licensed reactor operating capability of the AE&IL crew and the MARAD crew appear to be comparable, although sea trial experience and ship operation as a team will be lacking in the latter. Some experience of this kind will be gained in the proposed MARAD crew operations out of Galveston in late April.

The ACRS is aware of the desire of the Maritime Administration to train and have available a government crew with the capability of returning the ship to Galveston in the event of any contingency. The Committee believes that this crew will have the necessary competence to conduct the proposed preparations in Galveston and the sea trial. The Committee is concerned, however, that at the end of this trial, when the MARAD crew members return to their regular assignments, the majority will no longer be concerned with N. S. SAVANNAH operations except on an irregular basis. A proposed annual training period of two to four weeks was outlined for keeping this crew in a state of readiness. The Committee believes that further attention should be given to this retraining problem.

Based on evaluation of the qualifications, training and experience of the MARAD crew, the ACRS believes that the proposed limited sea trial can be conducted out of Galveston without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References - N. S. SAVANNAH
1. STS-11, "N. S. SAVANNAH Hazards Summary Report for 1964 Sea Trials - Maritime Administration Backup Crew", dated February 1964.
2. STS-11, Revision 1, dated March 16, 1964.
3. STS-11, Revision 2, dated March 20, 1964, "Figure 4-3. Ship Organization".

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

May 13, 1964

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

Subject: REPORT ON N.S. SAVANNAH

Dear Dr. Seaborg:

At its fifty-fourth meeting on April 2-4, 1964, at a special meeting on April 17, 1964, and at its fifty-fifth meeting on May 7-9, 1964 at Argonne, Illinois, the Advisory Committee on Reactor Safeguards reviewed three documents relative to various aspects of operation of the N.S. SAVANNAH after completion of the recent Galveston sea trials. These three are: STS-8, "N.S. SAVANNAH Summary Report for Domestic and Foreign Port Visitation", dated February 1964; STS-9, "N.S. SAVANNAH Technical Specifications", dated March 1964, and "Port Operation of the N.S. SAVANNAH", dated March 1964. The Committee had the benefit of a Subcommittee meeting on March 27, 1964 and, during the above meetings, discussions with representatives of the AEC-Maritime Joint Group, the Savannah Technical Staff, the U.S. Coast Guard, the American Export and Isbrandtsen Lines, and the AEC staff, and of the reports referenced.

The three primary safeguards other than those provided in the reactor system itself are: (1) the containment, (2) the reactor compartment and its air-cleaning systems, and (3) the mobility of the ship.

During the Galveston outage, the integrity of the containment was tested and improved. Continued testing following periods of operations at sea should provide an adequate basis for judging the performance of the containment. After a year's accumulation of experience and test data it would be appropriate to reconsider the leakage rate assumed in the porting criteria.

The proposed operation of the reactor compartment at a negative gauge pressure of one inch of water with the alarm set at a negative pressure of 0.3 inches of water and with the differential continuously recorded, constitute significant improvements. It is proposed to make iodine and dioctylphthalate (DOP) tests of the air-cleaning systems quarterly and to make DOP tests within one week prior to port entry. In its letter of April 27, 1963, the Committee stated that it believed that such measurements should be made prior to each port entry. The Committee still recommends that DOP testing be done within approximately one day prior to each port entry, that work on the development of a suitable iodine test be continued, and that iodine tests be made part of the preport entry testing program when a suitable test becomes available.

The AEC-Maritime Joint Group has proposed that mobility of the ship be depended on as a safeguard. Mobility is not an engineered safeguard; it introduces dependence on availability of tugs and on performance of the ship and tug organizations and crews during moving of the ship. Therefore, the Commiteee recommends that the mobility requirement proposed by the Joint Group in Reference 6 be replaced by the requirement that adequate tugs remain in attendance at the ship until such time as there is a calculated interval of one hour between an accidental loss of coolant and the first fuel-clad melting. After that time, the tugs should be on call so that, in the event of an accident, the tugs can arrive at the ship at least one-half hour before the calculated time when such melting is predicted to start. These time interval calculations should be based on conservative assumptions such as: total loss of electric power, loss of coolant as assumed in the MCA, and no emergency water injection. This method of operation will do much to assure the safety of the tug operators and ship crew as well as the safety of the general public in the unlikely event of an emergency.

If these precautions are followed, it is the opinion of the Committee that the proposed technical specifications and port visit criteria in References 5 and 6 provide an adequate basis for port analyses for the N.S. SAVANNAH and that the ship can continue to be operated without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References Attached.

References: N.S. SAVANNAH

1. STS-8, "N.S. SAVANNAH Summary Report for Domestic and Foreign Port Visitation", dated February, 1964.

2. STS-3, "N.S. SAVANNAH Containment Integrity - Its Measurement and Improvement", dated February 1964.

3. STS-4 "Status Report - N.S. SAVANNAH Ventilation System Filter Testing", dated November 1963.

4. "Tell Tale Arrangement", undated, received March 31, 1964.

5. Memorandum from D. L. Crook to H. L. Price dated March 31, 1964, 970/9460, transmitting STS-9, "N.S. SAVANNAH Technical Specifications", dated March 1964, received April 2, 1964.

6. "Port Operation of the N.S. SAVANNAH", dated March 1964, received April 2, 1964.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

May 13, 1964

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

Subject: REPORT ON N.S. SAVANNAH CHANGES

Dear Dr. Seaborg:

At its fifty-fifth meeting on May 7-9, 1964, at Argonne, Illinois, the Advisory Committee on Reactor Safeguards reviewed proposals from the AEC-Maritime Joint Group to: (1) remove accident dosimeters from the N.S. SAVANNAH, (2) operate the auxiliary propulsion system without maintaining a vacuum on the main condenser, and (3) use existing cooling coils for removing heat from the containment vessel under accident conditions as described in Change No. 30. The Committee had the benefit of discussions with representatives of the AEC-Maritime Joint Group, the Savannah Technical Staff, the American Export and Isbrandtsen Lines, and the AEC Staff, and of the reports referenced.

The Committee agrees that safety would not be adversely affected by removal of the accident dosimeters from the ship. However, it is suggested that other monitoring and instrument systems be studied and if necessary modified to assure that, in the unlikely event of an accident, they will supply the Master of the ship with sufficient information on the performance of engineered safeguards to enable him to assess the situation and take appropriate action.

Maintaining a vacuum on the main condenser in order to reduce windage in the turbine when the auxiliary propulsion motor is operating does not appear essential.

The Committee has not yet evaluated the cooling requirements in the containment under accident conditions, a problem which is being studied by the Joint Group and the AEC Staff. However, the Committee believes that the use of presently installed cooling coils for heat removal during emergency conditions as proposed in Change No. 30 provides a useful backup for other containment heat removal systems.

The Committee believes that continuing to operate the N.S. SAVANNAH while the above questions are being resolved does not present an undue hazard to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References:

1. Memorandum from D. L. Crook to M. L. Price dated March 24, 1964, 970/9494, with attached "Proposed Significant Change No. 30 - Use of Cooling Coils in Containment Vessel to Reduce Pressure in Accident Conditions".
2. STS-S&L-1, "N.S. SAVANNAH Containment Vessel Cooling to Reduce Pressure in Accident Conditions", dated April 3, 1964.
3. "Proposed Significant Change No. 30 - Use of Cooling Coils in Containment Vessel to Reduce Pressure in Accident Conditions. Revision 1 (April 13, 1964)".
4. STS-S&L-2, "Nuclear Accident Dosimeter Requirements for N.S. SAVANNAH", dated April 1964.
5. STS-S&L-3, "Main Condenser Vacuum Requirement for N.S. SAVANNAH", undated, received April 22, 1964.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

January 25, 1965

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

Subject: REPORT ON N.S. SAVANNAH

Dear Dr. Seaborg:

At its sixty-first meeting, January 14-16, 1965, the Advisory Committee on Reactor Safeguards reviewed the application of the Maritime Administration for an operating license for the N.S. SAVANNAH. The Committee had the benefit of discussions with the AEC-Maritime Joint Group, the Savannah Technical Staff, ship personnel, representatives of the American Export Isbrandtsen Lines, Todd Shipyards, and the AEC Staff. It also had available the documents listed below.

The N.S. SAVANNAH has now completed some 77,500 miles of sea travel. More than 1,250,000 people have visited the ship in many ports of the world. Its operating history, especially considering that it is a first-of-its-kind vessel and has been subject to the obvious pressures which came from making scheduled visits to many ports, has been good. The master of the ship and others have stated that there have been no serious malfunctions of the reactor. The leak rate of the containment has remained well below specification.

There are still features that are not up to the safety levels that the Committee deems generally advisable, but none of these items appear to be of a major nature. For example, the present control rod system continues to leak hydraulic oil, but in diminished quantities. The leaking flammable oil requires that the containment be filled with inert gas to avoid any possibility of fire. This fact, in turn, tends to inhibit entry and, hence, tends to reduce the number and thoroughness of inspections of the area. The applicant reports that the presence of small amounts of particulate matter in the hydraulic fluid has prevented proper operation of the valves in the system on several occasions and has led to the failure to scram of one, or at most two, individual rods. However, in every case the rods have been driven in by the rod run-in mechanism. The applicant also reports some corrosion and pitting of the buffer seal shafts, but in no case has any rod ever stuck for this reason.

The present rod system has the disadvantages of a sliding shaft seal between atmospheric pressure outside and high pressure inside, the requirements of a separate and necessary hydraulic fluid system with its attendant control valves, a separate nitrogen system to provide a driving force for the oil accumulators, and an electrical control system with many relays. Each of these features can be subject to difficulties and, in consequence, this is not a wholly satisfactory system.

The alternate Marvel-Schebler drive system also have difficulties. While the drives themselves are fully contained within the pressure housing and require no shaft seal, and while they require no hydraulic fluid or nitrogen system and appear to be much more nearly fail-safe than the present drives, the applicant has stated that the accompanying electrical control circuitry is not working correctly and that installation could not be started before July 1965. The Manager of the Joint Group and a Chief Engineer of the ship have both stated that they feel that the present control rod system provides adequate safety. In view of the good scram and run-in history of these rods, and their reported continuing improvement in operating characteristics, the Committee believes that these control units could continue to be used for operation of the reactor.

At the same time, the Committee recommends that work be continued in readying a more satisfactory control rod system for shipboard use. Such rods should be fully contained within the high pressure system, should be dependent on as few auxiliary systems as possible, and should be fail-safe.

The Committee would like to review this situation in the early summer of 1965.

The Committee would like to emphasize again the importance of maintaining properly trained and competent officers, crew, and specialists such as health physicists. In particular, the Committee believes that the nuclear advisor plays an important role, at least at this early stage, and should continue to be available on board after licensing. A thorough appreciation of the hazards of nuclear operation by all crew members is particularly important.

In its letter of May 13, 1964 the Committee recommended a tug availability criterion that:

> "... adequate tugs remain in attendance at the ship until such time as there is a calculated interval of one hour between an accidental loss of coolant and the first fuel-clad melting. After th t time, the tugs should be on call so that, in the event of an accident, the tugs can arrive at the ship at least one-half hour before the calculated time when such melting is predicted to start. These time interval calculations should be based on conservative assumptions such as: total loss of electric power, loss of coolant as assumed in the MCA, and no emergency water injection. This method of operation will do much to assure the safety of the tug operators and ship crew as well as the safety of the general public in the unlikely event of an emergency".

The Committee believes that the N.S. SAVANNAH should continue to use this criterion. This criterion provides substantial added assurance that mobility will be provided in the unlikely event of a serious nuclear accident at dockside. It will also provide an incentive for operation at lower powers in port areas in order to reduce the fission product burden and thus increase the time to melt in a postulated total loss of coolant accident. The Committee would like to point out that 10 CFR Part 100 might be applied to a shipboard reactor in the same way it is applied to land based reactors without taking any credit for the mobility of the ship. This is consistent with reactor safety practice in this country. However, if reliance is to be placed on mobility, it must be assured that mobility is indeed available and in time. The Committee believes that the "time-to-melt" criterion provides a substantial extra measure of this assurance. In addition, it provides considerable extra protection against a loss-of-coolant accident in which containment is very much less effective than expected.

At the same time, the engineered safeguards on the ship remain important. The "time-to-melt" criterion would not alone protect the public in the unlikely event of some other kinds of accidents, such as nuclear excursions. Furthermore, protection of the public, the passengers, and the crew must still be provided when the ship is in motion or when movement of the ship is not possible for weather reasons.

As mentioned in its letter of May 13, 1964, the Committee continues to believe that appropriate tests of the efficiency of the iodine adsorbers need to be devised. The Committee believes that such tests should be made routinely along with the particulate filter tests within one day of each port entry. Therefore, it recommends that the development of iodine tests be pursued vigorously.

In its letter of May 13, 1964, the Committee also suggested "that other monitoring and instrument systems be studied and if necessary modified to assure that, in the unlikely event of an accident, they will supply the master of the ship with sufficient information on the performance of engineered safeguards to enable him to assess the situation and take appropriate action." The Committee believes that this suggestion should also be pursued vigorously.

In summary, the Committee believes that proposed solutions to the problems regarding iodine adsorber tests and information availability for the master of the ship should be reviewed by the Staff of the Division of Reactor Licensing and implemented before the license is issued. The Committee recommends that the present "time-to-melt" criterion be retained in determining requirements for tug availability. Subject to these conditions, the Committee believes that the N.S. SAVANNAH has demonstrated that it can be operated satisfactorily as proposed by the Maritime Administration without undue hazard to the general public.

Sincerely yours,

/s/

W. D. Manly
Chairman

References Attached.

References:

1. Memorandum from U. M. Staebler, DRD, to R. E. Hollingsworth, General Manager, dated November 30, 1964, Subject: N.S. SAVANNAH - Marvel-Schebler Drive Program.

2. Memorandum from D. L. Crook, MA-AEC Joint Group to U. M. Staebler, dated November 24, 1964, 970/10857, Subject: N.S. SAVANNAH - Marvel-Schebler Drive Program.

3. Letter from John E. Bone, American Export Isbrandtsen Lines, Inc. to D. L. Crook, A.E.C./MarAd Joint Group dated November 19, 1964, Subject: N.S. SAVANNAH Control Rod Drive System.

4. Maritime Administration, U.S. Department of Commerce, N.S. SAVANNAH License Application, 970/10867, dated December 8, 1964.

5. STS-60, N.S. SAVANNAH Summary Report for Licensed Operations, dated November 1964.

6. STS-10, Port Operation of the N.S. SAVANNAH, dated November 1964.

7. STS-50, N.S. SAVANNAH Annual Operations Report, May 1963-April 1964, dated November 1964.

8. Memorandum from D. L. Crook, MA-AEC Joint Group to R. L. Doan, Division of Reactor Licensing dated November 19, 1964, 970/10819, transmitting STS-51, Quarterly Report, N.S. SAVANNAH Operations, May 1 - August 1, 1964, undated, received November 23 and December 15, 1964.

9. STS-59, An Evaluation of the Practice of Retaining Tugs on the Basis of Time to Melt, dated November 1964.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

May 17, 1965

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

Subject: REPORT ON N.S. SAVANNAH

Dear Dr. Seaborg:

At its sixty-third meeting on May 13-15, 1965, the Advisory Committee on Reactor Safeguards reviewed the application of First Atomic Ship Transport, Inc. (FAST), a subsidiary of American Export Isbrandtsen Lines, Inc. (AEIL) for a three-year license to operate the N.S. SAVANNAH under bare-boat charter from the Maritime Administration.

The Committee last reported in a letter of January 25, 1965, when the application for an operating license by the Maritime Administration was considered. The Committee had the benefit of the reports listed and of discussion with representatives of FAST, AEIL, Babcock and Wilcox Company (B&W), the U. S. Coast Guard and the AEC Staff.

N.S. SAVANNAH has been operated by the AEIL for the Maritime Administration from the time of the overhaul at Todd Shipyards in Galveston, Texas in May 1964 until the overhaul presently being completed at the same yard. During this period, the ship has been in practically continuous service to and from foreign ports experiencing, among other things, two violent storms. Performance has been satisfactory.

It is proposed to operate the ship only as a cargo vessel, putting the passenger facilities in standby. As a cargo ship, large scale public visits will be discontinued, and only small numbers of invited guests will visit the ship at one time.

Relief from responsibility for the safety of passengers simplifies ship operations. The Committee believes that the applicant should remain alert to the need for protection of shore population in case of an unlikely serious accident.

The applicant has a new management and operating organization which contains a considerable number of personnel previously associated with this project.

Certain specialized services are to be secured by contract, e.g., safety and inspection committee services from Nuclear Utility Services, assignment of nuclear advisors and technical services from B&W, water treatment control from Bull and Roberts, Inc., construction and maintenance from Todd Shipyards, and others as needed. The Committee considers that the FAST operating personnel, although of high ability, have had a minimum of training and experience in their present jobs and as a team. It recommends continued training and effort to maintain continuity, particularly in supervisory positions.

The safety criteria for operation, as described in the license application and reports FAST-1 and FAST-2, will be substantially the same as in the past. The applicant will apply the "time-to-melt criterion" previously recommended by the Committee. The applicant has stated that positive steps are being taken to implement as soon as possible other previous Committee recommendations concerning (a) a check of the iodine removal system efficiency before each port entry, and (b) provision of adequate information to the master concerning the course of an emergency involving the reactor.

The applicant stated that an experienced nuclear advisor will be included in the ship's complement while at sea. Criteria are being prepared governing operation of the reactor with anomalous control rod patterns. The Committee still wishes to review the status of the Marvel-Schebler control rod drive system when testing is completed and to be kept informed of progress on the other topics mentioned above.

The applicant has requested that tug availability not be required when the mooring location and the reactor operational history is such that the guidelines of 10 CFR Part 100 are met, after taking credit for the containment leak rate and filter efficiencies listed in report FAST-2. The Committee believes that such a porting criterion may be acceptable at certain mooring locations. It recommends that these arrangements be made on an individual basis with the Regulatory Staff.

The Committee believes that, with due attention to the considerations indicated above, the applicant, First Atomic Ship Transport, Inc., can operate the N.S. SAVANNAH in the mode described in its application without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

W. D. Manly
Chairman

References Attached

References:

1. N.S. SAVANNAH License Application, dated April 1965, First Atomic Ship Transport, Inc.

2. FAST-1, N.S. SAVANNAH Technical Specifications, dated April 1965, First Atomic Ship Transport, Inc.

3. FAST-2, N.S. SAVANNAH Port Operation Criteria, dated April 1965, First Atomic Ship Transport, Inc.

4. Annual Report 1964, American Export Isbrandtsen Lines, Inc.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
United States Atomic Energy Commission
Washington 25, D. C.

September 14, 1959

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D.C.

Subject: THE SAXTON REACTOR

Dear Mr. McCone:

At its nineteenth meeting, September 10-12, 1959, the Advisory Committee on Reactor Safeguards considered the Saxton Reactor for a construction permit. The Preliminary Hazards Summary Report was the basis of discussion with the Hazards Evaluation Branch and the licensee.

This is to be a 20 MW (thermal) light water moderated and cooled, pressurized water (2000 psi) reactor operated primarily for research and development with steam supplied, incidentally, to an existing 10 MW turbine generator. It is proposed to conduct the experimental program over a five-year period, culminating in a series of nuclear superheat experiments.

The site of 150 acres is located in south central Pennsylvania, halfway between Pittsburgh and Harrisburg, about three-fourths of a mile from the Borough of Saxton. The Saxton station of the Pennsylvania Electric Company, a 50 MW (e) coal fired steam generating plant (one 30 MW, two 10 MW turbine generators) is now located on the site. Because of favorable meteorology, topography, geology, hydrology and low population density, it is the Committee's opinion there is reasonable assurance that a reactor of this general type can be constructed and operated at this site without undue hazard to the health and safety of the public.

A preliminary presentation of the planned research program was made. Evaluation of this program must be deferred pending more detailed studies of all phases of design and operation.

Sincerely yours,

C. Rogers McCullough
Chairman

cc: A.R.Luedecke, GM
H.L.Price, DL&R

References

1) Saxton Nuclear Experimental Corporation Application for Reactor Construction Permit and Operating License -- Part B - Preliminary Hazards Summary Report (received by AEC July 1959).

2) Division of Licensing and Regulation Report to ACRS on The Saxton Reactor, August 20, 1959.

3) U. S. Weather Bureau Comments on Part B, "Preliminary Hazards Summary Report, Saxton Nuclear Experimental Corporation, August 19, 1959.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

September 26, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: SAXTON NUCLEAR EXPERIMENTAL CORPORATION

Dear Mr. McCone:

The Saxton Nuclear Experimental Corporation's application was considered by the Advisory Committee on Reactor Safeguards at the nineteenth meeting and reported to you in a letter dated September 14, 1959. In March 1960 it was proposed that the use of a multi-layer pressure vessel be considered. This proposal was supplemented with technical information covering the design, construction and test of this type of vessel.

An Advisory Committee on Reactor Safeguards subcommittee reviewed the documents listed and held a meeting with representatives of Saxton, Westinghouse, A. O. Smith Corporation, and the AEC staff. It was determined the history of multi-wall pressure vessels has been good, the technique of manufacture is well established, and satisfactory methods for ultrasonic testing of the vessel and the welds have been developed.

In the opinion of the ACRS the change to a multi-layer pressure vessel designed and constructed specifically for the Saxton reactor will not introduce any additional hazard to the health and safety of the public.

Sincerely yours,

/s/ Leslie Silverman

Leslie Silverman
Chairman

References:

1. Amendment #3, dated March 11, 1960.
2. Multi-layer Construction for the Saxton Reactor Vessel, (WCAP-1391), dated March 1, 1960.
3. Amendment #3, Supplement #1, dated Aug. 22, 1960.
4. Supplementary Technical Information on the Saxton Reactor Vessel, (WCAP-1620), dated Aug. 17, 1960.

cc: A. R. Luedecke, Gen. Mgr.
W. F. Finan, AGM for R&S
H. L. Price, Dir., DL&R

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

July 8, 1961

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

Subject: REPORT ON SAXTON NUCLEAR EXPERIMENTAL PROGRAM

Dear Dr. Seaborg:

At its thirty-fifth meeting on July 6-8, 1961, the Advisory Committee on Reactor Safeguards considered the Saxton Reactor Facility on the basis of the documents referenced below, and discussion with representatives of Westinghouse Electric Corporation, Pennsylvania Electric Company, Metropolitan Edison Company, Jersey Central Power and Light Company, Gilbert Associates Inc., and the staff of the AEC. Prior to this an ACRS Subcommittee visited the site of the Saxton Reactor Facility on June 23, 1961. The Saxton Reactor Facility was also the subject of letters from the Committee dated September 14, 1959, and September 26, 1960.

The Saxton Reactor is a relatively small light-water moderated and cooled pressurized water reactor located on an acceptable site. In many respects the Saxton Reactor is similar to the Yankee and Belgian reactors, also designed by Westinghouse. At this time, an initial operating license is requested covering only the start-up program including operation up to the rated power level of 20 MW(t), but not covering the planned five-year post-construction research and development program.

The Committee notes that the minimum calculated burn-out safety factor for steady-state condition is somewhat above two and for all transients likely to occur is not significantly below two.

Since a large amount of reactivity has to be controlled by a small number of control rods, each rod has to be of high worth and there is the possibility that the reactor can be made critical by complete withdrawal of a single control rod. A system involving manually set limit switches is provided to prevent excessive rod withdrawal and to provide an adequate shutdown margin.

The Committee concluded that the Saxton Reactor Facility can be operated, through its start-up program delineated above, without undue hazard to the health and safety of the public.

Dr. Leslie Silverman did not participate in the reviews or discussions of this project.

Sincerely yours,

/s/

T. J. Thompson
Chairman

References:

Final Safeguards Report, undated, received April 26, 1961.

Answers to Questions asked by ACRS July 7, 1961, on Saxton Reactor Plant, undated, received July 7, 1961.

Amendment #6 to License Application, dated June 9, 1961.

Amendment #7 to License Application, dated June 30, 1961.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

May 12, 1962

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

Subject: REPORT ON SAXTON NUCLEAR EXPERIMENTAL CORPORATION -- PHASE I OF RESEARCH AND DEVELOPMENT PROGRAM

Dear Dr. Seaborg:

At its forty-first meeting, May 10-11, 1962, the Advisory Committee on Reactor Safeguards considered, at the request of the Division of Licensing and Regulation, Phase I of Saxton Nuclear Experimental Corporation's five-year research and development program including applicable proposed changes in Technical Specifications, and a small reduction in the minimum burnout safety factor. The Committee had the benefit of a report from its subcommittee and the documents referenced as well as discussions with representatives of Saxton Nuclear Experimental Corporation, Westinghouse Electric Corporation, and the AEC staff.

The Phase I program has two major objectives: (1) to investigate the use of soluble neutron poison, boric acid, for chemical shim and (2) to investigate the feasibility of raising the specific power of fuel rods to 16 kw/ft.

The proposal to reduce the minimum steady state burnout safety factor from 2.4 to 2.2, in connection with a change in correlation resulting from a re-evaluation of available burnout data, does not materially affect the safety of the reactor.

It should be emphasized that this reactor is primarily intended for experimental studies and that the production of power is only incidental. Experiments are to be conducted in a step-wise fashion permitting examination of the results of each step before proceeding to the next. The limits set on unexplained reactivity loss when operating with boron; results of reanalysis of accidents covered in the final safeguards report; and the limited number of test fuel assemblies, give additional assurance that the Phase I program can be conducted in the Saxton reactor without undue risk to the health and safety of the public.

Subj:Saxton

Dr. Leslie Silverman did not participate in the Committee's consideration of this project.

Sincerely yours,

/s/ F. A. Gifford

F. A. Gifford, Jr.
Chairman

References:

1. Amendment No. 10 to license application transmitting
   (a) Safeguards Report for Phase 1 of Saxton Nuclear Experimenta Corporation's Five-Year Research and Development Program, dated December 22, 1961.

   (b) Proposed Changes in Technical Specifications Applicable to Conduct of Phase 1 of the Saxton Five-Year R&D Program, undated.
2. Supplement No. 1 to Amendment No. 10, dated April 10, 1962.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

September 12, 1963

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

Subject: REPORT ON SAXTON NUCLEAR EXPERIMENTAL CORPORATION

Dear Dr. Seaborg:

At its forty-ninth meeting on September 5 and 6, 1963, the Advisory Committee on Reactor Safeguards considered the application of the Saxton Nuclear Experimental Corporation for a full-term operating license through June 30, 1969 to replace the present provisional license which will expire on September 30, 1963. The Committee had the benefit of referenced documents and discussions with representatives of Saxton Nuclear Experimental Corporation, Westinghouse Electric Corporation and the AEC Regulatory Staff.

The Committee has previously reported on the application for a construction permit, proposed use of a multi-layer pressure vessel, request for operating license and Phase I of the Research and Development Program in letters dated September 14, 1959, September 26, 1960, July 8, 1961, and May 12, 1962.

This reactor has operated satisfactorily throughout the period of its provisional operating license. It is the opinion of the ACRS that the Saxton Nuclear Experimental Corporation can continue to operate this reactor at power levels up to 23.5 Mw(t) under existing limitations without undue hazard to the health and safety of the public.

Dr. Leslie Silverman did not participate in the review of this project.

Sincerely yours,

/s/
D. B. Hall
Chairman

References:
1. Amendment No. 12, dated May 28, 1963.
2. Reactor Plant Operating Experience Report - April 1962 to April 1963, dated May 29, 1963.

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

July 19, 1965

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

Subject: REPORT ON SAXTON NUCLEAR EXPERIMENTAL CORPORATION

Dear Dr. Seaborg:

At its sixty-fourth meeting, held July 8-10, 1965, the Advisory Committee on Reactor Safeguards considered the application of the Saxton Nuclear Experimental Corporation for the use of a partial plutonium core in the second core loading of the Saxton reactor. The Committee had the benefit of a Subcommittee meeting held on May 4, 1965, of the referenced documents, and of discussions with representatives of Saxton Nuclear Experimental Corporation, Westinghouse Electric Corporation and the AEC Regulatory Staff.

The Committee has previously reported on the application for a construction permit, on the proposed use of a multi-layer pressure vessel, on the request for an operating license, on the Phase I Research and Development Program, and on the application for a full-term operating license in letters dated September 14, 1959, September 26, 1960, July 8, 1961, May 12, 1962, and September 12, 1963.

To date, Saxton has operated its first core loading to an average burnup of more than 8500 MWD/MTU. Since late 1962, the Saxton reactor has used boric acid in the coolant to meet some reactivity control requirements. Operation is reported to have been satisfactory, and reactivity anomalies which may have been attributable to boron hideout have been kept within 0.002 Δ k/k during a variety of experimental studies on coolant pH, nucleate boiling, and deposits on fuel elements.

The applicant has suggested that the detailed reactivity-follow program and the requirement that unexplained reactivity not exceed 0.003 Δ k/k are no longer needed. The Committee agrees and recommends that the applicant and the Regulatory Staff select new appropriate limits to reactivity anomalies beyond those attributable to discrepancies between prediction and observations of long term reactivity effects due to burnup.

In the proposed second core loading, nine assemblies are fueled with UO_2 spiked with 6.6 w/o PuO_2, while the remaining twelve assemblies will contain enriched uranium, as at present. The basic thermal design criteria for Core II are the same as for Core I. The new assemblies are expected to have improved mechanical features. The applicant reports that nuclear characteristics have been confirmed with critical experiments and that they lead to a dynamic reactor behavior generally similar to that of Core I.

Analyses by the applicant indicate that, in the unlikely event of a serious accident, the consequences to the health and safety of the public are not significantly affected by the use of plutonium oxide fuel in the second core loading.

With the establishment of an appropriate limit on reactivity anomalies, the ACRS believes that the Saxton reactor can be operated with the partial plutonium loading of Core II, as proposed, without undue hazard to the health and safety of the public.

Dr. N. J. Palladino did not participate in the review of this project.

Sincerely yours,

/s/
W. D. Manly
Chairman

References:

1. The Saxton Chemical Shim Experiment, dated August, 1964.
2. Safeguards Report for the Saxton Reactor Partial Plutonium Core II, dated March, 1965.
3. Supplement No. 1 to Safeguards Report for the Saxton Reactor Partial Plutonium Core II, dated May, 1965.

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

August 17, 1967

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

Subject: REPORT ON SAXTON REACTOR

Dear Dr. Seaborg:

At its eighty-eighth meeting, August 10-12, 1967, the Advisory Committee on Reactor Safeguards considered the application by Saxton Nuclear Experimental Corporation for authorization to increase the maximum power level of the Saxton reactor from 23.5 MWt to 35 MWt for a limited time during Core II operation. Operation of Core II at 23.5 MWt was reviewed by the ACRS at its sixty-fourth meeting, July 8-10, 1965, and discussed in a Committee letter dated July 19, 1965. During consideration of the proposed power increase, the Committee has had the benefit of discussions with representatives of Saxton Nuclear Experimental Corporation, Westinghouse Electric Corporation and the AEC Regulatory Staff, and of the documents listed below. A Subcommittee of the ACRS met to review the power increase on August 9, 1967.

The higher power level will permit operation of fuel elements at linear power ratings similar to those of large scale pressurized water reactors now in the design and construction stages. The higher power level operation under consideration is to be limited to the remaining life of Core II. It is estimated that there will be sufficient reactivity to operate about 10 weeks at the 35 MWt level.

The power increase will be accomplished in a number of separate steps, allowing sufficient time between steps to evaluate reactor performance in comparison with predictions for each step. A local linear heat generation rate of 19.1 KW/ft of fuel rod will not be exceeded.

The ACRS believes there is reasonable assurance that the Saxton reactor can be operated at the increased power level, as proposed, without undue hazard to the health and safety of the public.

Mr. N. J. Palladino did not participate in the review of this project.

Sincerely yours,

/s/

C. W. Zabel
Acting Chairman

References:

1. Letter from Saxton Nuclear Experimental Corporation, dated January 18, 1967; Change Request No. 25 to the Saxton Technical Specifications; Application for Amendment No. 3 to Operating License; and Safeguards Report for the Saxton Reactor Operating at 35 MWt, dated December, 1966.

2. Letter from Saxton Nuclear Experimental Corporation, dated June 27, 1967, and Amendment No. 1 to Change Request No. 25.

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

December 10, 1974

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

Subject: REPORT ON SEABROOK STATION, UNITS 1 and 2

Dear Dr. Ray:

At its 176th Meeting, December 5-7, 1974, the Advisory Committee on Reactor Safeguards reviewed the application of the Public Service Company of New Hampshire, et al, for permits to construct Seabrook Station, Units 1 and 2. This project had been considered previously during a Subcommittee meeting in Hampton, New Hampshire, on August 21-22, 1974, subsequent to a tour of the site by members of the Committee on August 21, 1974; at the 173rd Meeting of the Committee, September 5-7, 1974; during a Subcommittee meeting in Washington, D. C., October 9, 1974; at the Special Meeting of the Committee, October 31- November 2, 1974; and during a Subcommittee meeting, December 4, 1974. During its review, the Committee had the benefit of discussions with the AEC Regulatory Staff and representatives and consultants of the applicant, the Westinghouse Electric Corporation, and United Engineers and Constructors, Inc. The Committee also had the benefit of the documents listed below and of comments and presentations from members of the public.

The site for the station is a 750-acre tract located near the town of Seabrook, New Hampshire. The site is approximately 12 miles south-southwest of Portsmouth, New Hampshire and 40 miles north-northeast of Boston, Massachusetts. Portsmouth is the nearest population center with 1970 population of about 26,000. Due to the beach areas of Seabrook and Hampton, New Hampshire, there is a large summertime increase in population within a few miles of the site.

The Seabrook Station will utilize two, four-loop pressurized water reactor nuclear steam supply systems each having a power level of 3411 MW(t) and a design similar to that of the Catawba Nuclear Station units previously reviewed by the Committee and reported upon in its letter of November 13, 1973.

The Regulatory Staff has determined that the ECCS performance evaluation for the Seabrook Station units meets the Interim Acceptance Criteria of June 1971. In addition, the applicant's ECCS performance evaluation, using an approved Westinghouse model, to show compliance with the Final Acceptance Criteria of 10 CFR 50.46 must be reviewed and approved by the Regulatory Staff.

The Committee recommended in its report of September 10, 1973, on acceptance criteria for ECCS, that significantly improved ECCS capability should be provided for reactors filing for construction permits after January 7, 1972. The Seabrook Station units are in this category. These units will use 17x17 fuel assemblies similar to those to be used in Catawba Units 1 and 2. Although calculated peak clad temperatures in the unlikely event of a LOCA are less for 17x17 assemblies than for a 15x15 array, the Committee believes that the applicant should continue studies responsive to the Committee's September 10, 1973 report. If studies establish that significant further ECCS improvements can be achieved, consideration should be given to incorporating them into this plant.

Although many details of the proposed 17x17 fuel design are available, complete analyses of the performance of this fuel arrangement are not yet available from the applicant, and the AEC Regulatory Staff has not completed its review. The Committee will review and address questions relating to the proposed 17x17 fuel design within the next few months in connection with operating license applications for other nuclear units employing similar fuel.

The applicant proposes a horizontal ground acceleration of 0.25g on bedrock at foundation as a seismic design basis for safe shutdown. Extensive consideration by the ACRS and its consultants of the site, of the foundation structure, and of the relationship of the site to the tectonic province in which it is located has led the Committee to conclude that the proposed acceleration is acceptable for this site.

Field and laboratory investigations by the applicant indicate that there are no known geologic features in the vicinity of the site that are likely to localize seismicity. Nevertheless, the Committee believes that all site excavations should be carefully mapped and any unusual features reviewed by geology and seismology experts of the applicant and the Regulatory Staff prior to being covered over or severely weathered.

One aspect of the engineered safety features in this plant which warrants further examination is the necessity of a cooling system for the charcoal adsorption beds in case of a major accidental release of airborne radioactive material within containment or the fuel storage building. To assist in resolving this issue, the Committee recommends that a parametric study be conducted to define an upper limit of the source term, to estimate quantitatively the resulting radionuclide loading on the beds, and to calculate the subsequent temperature increase as a function of time within adsorption beds of various configurations. If the heat load is not too large, such steps as increased air flow through the beds, cooling of the gas prior to entry into the bed, and rearrangement of the charcoal configuration within the beds may be adequate. The Committee wishes to be kept informed.

The Seabrook Station Units 1 and 2 will be the first commercial nuclear power plant in the State of New Hampshire. For this reason, the Committee recommends that the applicant and Regulatory Staff give particular attention to assuring proper coordination with appropriate state and regional agencies in the development of effective emergency plans for this facility. Because of the proximity of the Seabrook Station to the beaches on the coast and because of the nature of the road network serving the beaches, the applicant has given early attention to the problems of evacuation. The Committee believes, however, that further attention needs to be given to evacuation of residents and transients in the vicinity even though they may be outside the LPZ.

Several unresolved issues, such as appropriate capacity of the containment ventilation system and the containment enclosure transient pressure analysis following a postulated pipe break outside of containment, should be resolved in a manner satisfactory to the Regulatory Staff.

Generic problems relating to large water reactors have been identified by the Regulatory Staff and the ACRS and 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 Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during construction and that, if due consideration is given to the foregoing, the Seabrook Station Units 1 and 2 can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Additional remarks by Dr. D. Okrent are attached.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References attached

Additional comments by D. Okrent

The Seabrook Station site is near what is generally recognized as the Cape Ann-to-Ottawa Trend. Mechanisms for earthquake generation in the New England area are not well understood, and expert opinion differs concerning the potential for and probability of relatively large earthquakes at or near the site.

The Regulatory Staff have ultimately based their judgment as to an acceptable safe shutdown earthquake on the application of 10 CFR Part 100, Appendix A, rather than a probabilistic estimate of earthquake size versus recurrence interval. It is of interest to note that Appendix A provides only general guidance; furthermore, it specifically refers to the possible choice of a safe shutdown earthquake larger than that found in the historical record for a tectonic structure or province.

During the ACRS review the Regulatory Staff did state that the seismicity of the tectonic region applicable to the Seabrook site could be interpreted to be about an order of magnitude larger than other tectonic provinces having a similar maximum historical seismic event. Furthermore, a member of the Regulatory Staff stated that his estimate of the probability per year of occurrence of an earthquake of intensity MM VIII at the Seabrook site is about 10^{-4}, and the Staff did not rule out the possibility of a larger earthquake occurring within the region under consideration. They stated that conservatisms in analysis, stress limits, and other factors decrease the overall probability of failure of seismic Class 1 structures and piping by a few orders of magnitude and hence, the overall probability of a seismically induced accident exceeding 10 CFR Part 100 would be acceptably low. However, earthquakes are almost unique in their ability to fail each and every structure, system, component, or instrument important or vital to safety, and, in my opinion, the Staff evaluation of additional margin available from stress limits, methods of analysis, etc., did not consider all such systems, e.g., D.C. power or emergency A.C. power.

It is clear that the capability of a reactor to achieve safe shutdown, assuming its SSE occurs, cannot be fully demonstrated by test. Those limited, detailed independent audits of seismic design of actual plants that have been published indicate that some inadequacies in design and construction exist. Equally or more important, it appears to be unlikely that the plant could survive safely, with a high degree of assurance, a larger earthquake having one or two orders of magnitude lower probability than the proposed SSE.

Additional comments by D. Okrent (continued)

Given this background, and recognizing the substantial surrounding year-round population density and the very high nearby population during the summer months at Seabrook, I am left uneasy and believe it would be prudent to augment the proposed SSE acceleration of 0.25g.

I also wish to reiterate my conclusion previously stated in connection with the review of Grand Gulf Units 1 and 2, namely that it would be prudent to provide some additional margin in the seismic design bases for most future nuclear plants sited east of the Rockies.

References:

1. Public Service Company of New Hampshire Application for a Construction Permit for the Seabrook Station with Preliminary Safety Analysis Report (PSAR), Volumes 1 through 7.

2. Amendments 1-13, 15-19, and 21-26 to the PSAR.

3. Directorate of Licensing's Safety Evaluation of the Seabrook Station, Units 1 and 2, dated August 14, 1974; Supplement 1, dated August 20, 1974; and Supplement 2, dated October 8, 1974.

4. Directorate of Licensing's Summaries of Outstanding Safety-Related Issues for the Seabrook Station, Units 1 and 2, dated August 16, 1974; October 9, 1974; and October 31, 1974, respectively.

5. Public Service Company of New Hampshire letters:

   a. October 23, 1973, concerning transient beach population.
   b. December 21, 1973, concerning waste processing system.
   c. December 26, 1973, concerning geology-regional fault investigations.
   d. October 1, 1974, concerning anticipated transients without scram and reactor protection system.

6. New England Coalition on Nuclear Pollution letters:

   a. August 15, 1974, concerning seismic issues and population density and evacuation.
   b. October 25, 1974, concerning site characteristics, geology and seismology.
   c. December 2, 1974, concerning seismology.
   d. December 5, 1974, concerning seismology.

7. Elizabeth H. Weinhold letters:

   a. August 13, 1974, concerning seismology, geology, and evacuation.
   b. Undated (received October 3, 1974) concerning safe shutdown earthquake design value.
   c. October 21, 1974, concerning earthquake intensities.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 19, 1983

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON LOW POWER OPERATION OF THE SEABROOK STATION, UNITS 1 AND 2

During its 276th meeting, April 14-16, 1983, the Advisory Committee on Reactor Safeguards reviewed the application of the Public Service Company of New Hampshire, acting as agent for and on behalf of the Seabrook Owners Group (the Applicant), for an operating license for the Seabrook Station, Units 1 and 2. The station is to be operated by the Public Service Company of New Hampshire. This application was considered at an ACRS Subcommittee meeting in Hampton Beach, New Hampshire, on April 1-2, 1983. Members of the Subcommittee toured the facility on April 1, 1983. In our review, we had the benefit of discussions with representatives of the Applicant, the Yankee Atomic Electric Company, Westinghouse Electric Corporation, United Engineers and Constructors, Inc., the NRC Staff, and with members of the public. We also had the benefit of the documents listed below. The Committee commented on the construction permit application for Seabrook Station, Units 1 and 2 in a report dated December 10, 1974.

The Seabrook Station is located on the western side of Hampton Harbor, in the Township of Seabrook, Rockingham County, New Hampshire, approximately 11 miles south of Portsmouth, New Hampshire and 40 miles north of Boston, Massachusetts.

Each Seabrook unit uses a Westinghouse nuclear steam supply system with a rated core power of 3411 MWt. The containment for each unit consists of a steel lined, reinforced concrete structure which is surrounded by a reinforced concrete containment enclosure. The design pressure of the containment is 52 psig. The annular space between containment and enclosure is maintained at a slight negative pressure.

Seabrook will use Westinghouse Model F steam generators, which incorporate design changes intended to eliminate the problems experienced with earlier models. We wish to be kept informed concerning the performance of these steam generators.

We were favorably impressed by the amount of attention given and resources expended in the area of personnel training. The result appears to be an excellent educational system for operations personnel, including operators and technicians. The resources at the disposal of the Applicant, including those of the Yankee Atomic Electric Company, appear to be appropriate for the operation of this nuclear power station.

The ACRS has on several occasions recommended that evaluations be made of the capability of light water nuclear power plants to be shut down safely in the event of an earthquake of greater severity and lower likelihood than the safe shutdown earthquake. The implications of recent seismic activity, such as the January 1982 earthquakes in central New Brunswick and New Hampshire, are being evaluated. We recommend for the Seabrook Station that specific attention be given to the seismic capability of those components that are important to the accomplishment of safe shutdown including the emergency AC power supplies, the DC power supplies, and small components such as actuators and instrument lines.

The Applicant has undertaken a full-scope probabilistic risk assessment (PRA) which is scheduled for completion about October 1983. The ACRS wishes to be kept informed concerning the results of the NRC Staff's review and evaluation of this PRA.

The Seabrook Station, Units 1 and 2 will be the first commercial nuclear power plant in the state of New Hampshire; the Station is also situated very close to the New Hampshire-Massachusetts border. As a result, the NRC Staff and Applicant must give particular attention to assuring proper coordination with appropriate state and regional agencies in the development of effective emergency plans. There is a large summertime increase in population within a few miles of the site due to the beach areas of Seabrook and Hampton, New Hampshire. The nature of the road network serving the beach requires that special attention be given to the problems associated with evacuation. Because the emergency plan is not yet fully developed, we were unable to review it.

A number of other items have been identified by the NRC Staff as Outstanding Issues. There is also a set of Confirmatory Issues that awaits additional documentation. We found no reason to believe that any of these issues will be especially difficult to resolve. We recommend that they be resolved in a manner satisfactory to the NRC Staff.

Fuel loading for Unit 1 is scheduled for September 1984 and fuel loading for Unit 2 is planned to take place about 2.5 years after fuel loading for Unit 1. Should there be a significant delay in this schedule, we would expect to examine the need for additional review of Unit 2.

We believe that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Seabrook

Station, Units 1 and 2, can be operated at core power levels up to 5 percent of full power without undue risk to the health and safety of the public.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Acting Chairman

References:

1. Public Service Company of New Hampshire, Seabrook Station "Final Safety Analysis Report," Volumes 1-15, with Amendments 45-48
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Seabrook Station, Units 1 and 2," NUREG-0896, dated March 1983.
3. Written Public Comments from J. Doughty, Seacoast Anti-Pollution League (SAPL), Subject: SAPL Comments to the Advisory Committee on Reactor Safeguards Subcommittee Conducting the Independent Technical Review for the Seabrook Nuclear Power Plant, April 1983, received April 1, 1983.
4. Written Public Comments from Rep. Roberta C. Pevear, New Hampshire House of Representatives, Subject: Statement Before Advisory Committee on Reactor Safeguards Meeting on Seabrook Operating License, April 2, 1983, received April 2, 1983.
5. Written Public Comments from Elizabeth Dolly Weinhold, Subject: Seismic Issues, received April 2, 1983.
6. Written Public Comments from Rep. Roberta C. Pevear, New Hampshire House of Representatives, Subject: Response to Kulash Report on evacuation planning, dated April 4, 1983.
7. Written Public Comments from Diana P. Sidebotham, President, New England Coalition on Nuclear Pollution, Inc., Subject: Remarks Prepared for delivery at April 1, 1983 Subcommittee meeting on Seabrook Station, dated April 11, 1983.

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

February 11, 1970

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

Subject: REPORT ON SEQUOYAH NUCLEAR PLANT

Dear Dr. Seaborg:

At its 117th meeting, January 8-10, 1970, and its 118th meeting, February 5-7, 1970, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Tennessee Valley Authority to construct Units 1 and 2 of the Sequoyah Nuclear Plant. A Subcommittee met to review this proposal on December 2, 1969, in Chattanooga, Tennessee and on January 5 and January 31, 1970, in Chicago, Illinois. During its review, the Committee had the benefit of discussions with representatives of the applicant, the Westinghouse Electric Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed below.

The plant will be located on the west shore of Chickamauga Lake on the Tennessee River, approximately 12 miles northeast of Chattanooga, Tennessee (1960 population about 130,000). The minimum exclusion distance will be 1920 ft. and the nearest residence will be approximately 2700 ft. from the plant.

The Sequoyah units will include four-loop pressurized water reactors designed for initial core power levels up to 3411 MWt. The nuclear steam supply systems and the emergency core cooling systems are esentially identical to those provided for the Diablo Canyon units. The proposed power level for the Sequoyah units is approximately five percent higher than the power level of 3250 MWt for which similar units have been approved. This higher power level has been justified by the applicant on the basis of a more detailed calculation of hot channel conditions in the core. The applicant described measurements which have been made or will be made on operating reactors, including some having cores similar to those of the Sequoyah units, to demonstrate the validity of the calculations on which the power level increase is based. If the results of these measurements are not conclusive, similar measurements will be made on the Sequoyah units during start-up. If the designer's expectations should not be adequately confirmed, system modifications or restrictions on operation may be appropriate.

Each containment will utilize the ice-condenser system within a free-standing containment building consisting of a steel dome and walls and a reinforced concrete flat base. A reinforced concrete shield building surrounds the containment. The volume between the two will be provided with a ventilation system employing both particulate and iodine filters. The reinforced concrete divider barrier which separates the upper and lower compartments of the ice-condenser containment system is subjected to pressure loading in the unlikely event of a loss-of-coolant accident. Since this barrier cannot be pressure tested, the Committee believes that it should be designed on a very conservative basis and that an independent check of the design should be made.

The plant will be protected against flooding to an elevation of 705 ft. MSL. In the event that flooding to an elevation of 700 ft. is predicted, the applicant has proposed that the reactors will be brought to a cold shut-down condition. If the flood level should exceed 705 ft., the auxiliary building will be allowed to flood, and decay heat will be removed from the reactors by means of a system which is protected against flooding up to the "probable maximum flood" level of 721 ft. The applicant has described general design bases and design criteria for this system. The Committee believes it important that this system be designed to provide the high standards of performance and reliability required of an engineered safety system. This matter, as well as the development of plans for recovery of the normal decay heat removal systems after flooding, should be resolved in a manner satisfactory to the Regulatory Staff during construction of the plant.

It is expected that the calculated doses to the public in the unlikely event of a design basis accident will be reduced by iodine removal in the ice condenser, by mixing in the volume between the containment and the shield building, and by reduction of leakage from the containment. The applicant should continue his study of these and other means of reducing doses.

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.

Information on a number of items identified in previous reports of the Committee is to be provided by the applicant 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 buildup of hydrogen in the containment following 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 believes that resolution of these items should apply equally to the Sequoyah plant.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during construction and that, if due consideration is given to the foregoing, the Sequoyah Nuclear Plant 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:

1. Sequoyah Nuclear Plant, Units 1 and 2, Preliminary Safety Analysis Report, Volumes 1 - 3

2. Amendments 1 - 9 to Application for Licenses

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 11, 1979

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM LOW POWER OPERATION OF SEQUOYAH NUCLEAR POWER PLANT, UNIT 1

Dear Dr. Ahearne:

During its 236th meeting, December 6-8, 1979, the Committee considered a proposal for interim, low power operation of the Sequoyah Nuclear Power Plant, Unit 1. At its 229th meeting, May 10-12, 1979 and also at its 228th meeting, April 5-7, 1979 the Committee had considered aspects of the application of the Tennessee Valley Authority (hereinafter referred to as the Applicant) for authorization to operate the Sequoyah Nuclear Power Plant, Units 1 and 2. A tour of the facility was made by members of the Subcommittee on January 24, 1976 and the application was considered at Subcommittee meetings on March 12, 1979 and on November 5, 1979. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, the Westinghouse Electric Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for this plant on February 11, 1970.

The Sequoyah Nuclear Power Plant is located on the west bank of the Tennessee River in Hamilton County in southeastern Tennessee approximately 17 miles northeast of the center of Chattanooga, Tennessee. Construction on Unit 1 is essentially complete and construction of Unit 2 is about 90% complete. Each unit will utilize a four-loop pressurized water reactor nuclear steam supply system having a power level of 3411 MWt and an ice condenser system enclosed within a free-standing steel containment vessel which is surrounded by a reinforced concrete shield building. The ice condenser system is similar to that used in the McGuire Nuclear Station and the Donald C. Cook Nuclear Plant. The Applicant has modified the ice condenser system as a result of the operating experience gained in the Donald C. Cook Nuclear Plant. The Applicant and the NRC Staff have made plans to monitor the performance of the ice condenser containments at the Sequoyah Nuclear Power Plant (Generic Item 63 in the ACRS report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 7," dated March 21, 1979). The Committee recommends that such plans be implemented.

The Sequoyah Nuclear Plant will utilize 17x17 fuel assemblies. A surveillance program has been developed by the NRC Staff to follow the behavior of these assemblies, and data are being obtained from several plants now in operation in which such assemblies have been installed for test. Experience to date has been satisfactory. The Committee wishes to be kept informed of the results of the various 17x17 assembly inspections and test programs now under way.

The Sequoyah site is considered by the NRC Staff to be within the Southern Valley and Ridge tectonic province. The maximum historic earthquake within this tectonic province is the 1897 Modified Mercalli Intensity (MMI) VIII earthquake in Giles County, Virginia. During the construction permit review, the NRC Staff concluded that a modified Housner response spectrum anchored at 0.18g was acceptable as the safe shutdown earthquake. Since that time, the NRC Staff has adopted methods which would characterize an MMI VIII earthquake with the more conservative response spectrum specified in Regulatory Guide 1.60 anchored at 0.25g.

The Applicant, in response to NRC Staff recommendations, has evaluated the Sequoyah design using a site-specific safe shutdown response spectrum developed from North American and Italian strong motion records of appropriate magnitude and epicentral distance and has compared the probability of the safe shutdown earthquake being exceeded at Sequoyah to that at other Tennessee Valley Authority plants that meet the Standard Review Plan. It has been concluded that the risk of exceeding the present design spectrum and the risk of exceeding the site-specific spectrum are comparable and that the probability of exceeding the safe shutdown earthquake is not appreciably different from that for other plants in this region. The NRC Staff has reviewed the Applicant's evaluation and has concluded that the Sequoyah plant is adequate to withstand the effects of the safe shutdown earthquake without loss of its capability to perform required safety functions. The NRC Staff, to verify their judgments regarding structural and component design margins, has performed an audit of the design margins in representative critical sections of the reactor and auxiliary building structures and in representative components required for safe shutdown.

The Committee recommends that this program for the quantification of the seismic design margin be continued and expanded to the extent necessary to ensure that all structures and equipment necessary to accomplish safe shutdown do indeed have some margin. Similar recommendations have been made by the Committee for the North Anna Power Station, Units 1 and 2, and the Davis-Besse Unit 1 in its reports dated January 17, 1977 and January 14, 1979. This matter should be resolved on a schedule and in a manner satisfactory to the Staff.

The Emergency Core Cooling Systems (ECCS) for the Sequoyah Nuclear Plant incorporate the Upper Head Injection (UHI) system. The NRC Staff has completed its review of the Westinghouse Electric Corporation ECCS evaluation model for plants equipped with UHI, and the Committee in its April 12, 1978 report on the McGuire Nuclear Station has concurred with the

Staff's conclusions. The NRC Staff has completed its review of the application of this approved evaluation model to the Sequoyah Nuclear Plant and concurs with the Applicant.

The Committee has been reviewing the circumstances relating to the recent accident at the Three Mile Island Nuclear Station Unit 2 and has made recommendations for improvements in plant design and operating procedures which should be considered for all pressurized water reactors. The Committee is continuing its review of the implications of this accident and expects to provide additional recommendations. It is expected that these recommendations will be considered and implemented as appropriate by the NRC Staff. The Committee wishes to be kept informed.

The NRC Staff has identified a number of outstanding issues, confirmatory issues, and licensing conditions, not related to TMI-2 accident considerations, which have not been specifically addressed in this report. These issues should be resolved in a manner satisfactory to the NRC Staff.

Various generic problems are discussed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 7," dated March 21, 1979. Those problems relevant to the Sequoyah Nuclear Plant should be dealt with by the NRC Staff and the Applicant as solutions are found. The relevant items are: 54-60, 63-65, 69, 71, 72, 74, and 76.

The NRC Staff has not completed its review of the Sequoyah Nuclear Power Plant application for a normal operating license at full power, and various implications of the Three Mile Island accident on the Sequoyah Plant remain to be decided. The ACRS has not completed its own review in regard to these matters.

The Applicant has proposed a program of interim low power operation to provide improved operator training and the development of additional experimental information on the behavior of a nuclear unit and its systems under transient conditions. The Applicant has proposed a special test series which includes the following:

1. Natural circulation following a simulated reactor trip.

2. Natural circulation following a simulated loss of offsite power.

3. Natural circulation with loss of pressurizer heaters.

4. Effect of steam generator isolation on natural circulation.

5. Natural circulation at reduced pressure.

6. Cooldown capability of the charging and letdown system.

7. Heat removal following a simulated loss of onsite and offsite AC power.

8. Establishment of natural circulation from stagnant flow conditions.

9. Boron mixing and cooldown.

The NRC Staff plans to review the proposed experimental program in detail to assure itself that all safety-related aspects are being dealt with appropriately. The Committee wishes to be kept informed.

The NRC Staff advised the Committee that it will require that TVA's emergency procedures for Sequoyah be reviewed by Westinghouse. The NRC Staff also stated that an acceptable emergency plan will exist prior to reactor operation.

The Committee believes that there is reasonable assurance that the Sequoyah Nuclear Power Plant, Unit 1 can be operated on an interim basis up to power levels of about five percent of full power without undue risk to the health and safety of the public. Subject to approval of the detailed test program by the NRC Staff, the Committee recommends approval of an interim low power license for the purposes proposed.

Sincerely,

Max W. Carbon
Chairman

References:

1. Tennessee Valley Authority, "Final Safety Analysis Report, Sequoyah Nuclear Power Plant," Volumes 1 to 13, and Amendments 1 to 61.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the operation of Sequoyah Nuclear Plant Units 1 and 2," NUREG-0011, March 1979.
3. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated October 31, 1979, containing revised responses to the Lessons Learned Requirements.
4. Letter, L. M. Mills, TVA, to L. S. Rubinstein, NRC, dated October 30, 1979, containing responses to ACRS questions.
5. Letter from L. M. Mills, TVA, to L. S. Rubinstein, NRC, dated October 23, 1979, containing information on natural circulation in Sequoyah, Unit 1, and Diablo Canyon, Unit 1.
6. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated October 12, 1979, containing responses to ACRS recommendations.

7. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated September 7, 1979, containing responses to the Short-Term Recommendations of the Lessons Learned Task Force.
8. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated July 12, 1979, containing responses to NRC-I&E Bulletin 79-06A and ACRS recommendations.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 15, 1980

The Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: REPORT ON THE SEQUOYAH NUCLEAR POWER PLANT, UNITS 1 & 2

Dear Dr. Ahearne:

During its 243rd meeting, July 10-12, 1980, the Advisory Committee on Reactor Safeguards completed its review of the application of the Tennessee Valley Authority (hereinafter referred to as the Applicant) for authorization to operate the Sequoyah Nuclear Plant, Units 1 & 2 at full power. The Committee had considered aspects of the application during its 242nd meeting, June 5-7, 1980; 236th meeting, December 6-8, 1979; 229th meeting, May 10-12, 1979; and 228th meeting, April 5-7, 1979. A tour of the facility was made by members of the Subcommittee on January 24, 1976 and the application was considered at Subcommittee meetings on July 9, 1980; June 2, 1980; November 5, 1979; and March 12, 1979. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, the Westinghouse Electric Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed. The Committee reported on interim low power operation of Unit 1 on December 11, 1979 and on a construction permit for this plant on February 11, 1970.

In its letter of December 11, 1979 the Committee addressed the proposed special low power test program, to be carried out on Unit 1, the seismic reevaluation of the Sequoyah plant, actions on recommendations resulting from the review of the accident at the Three Mile Island Station, Unit 2, and actions on various generic problems. These generic problems were further discussed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 7," dated March 21, 1979. The Committee's recommendations in its December 11, 1979 letter are also applicable to Unit 2 except that the special low power test program will not be repeated on Unit 2.

The special low power test program has been reviewed by Westinghouse Electric Corporation and by the NRC Staff. The Applicant began these tests on July 11, 1980 and the Applicant, Westinghouse, and the NRC Staff will review the results of these tests. It is expected that the additional operator training and operator experience will prove to be beneficial.

The Committee has reviewed and reported on NUREG-0660, "NRC Action Plans Developed as a Result of the TMI-2 Accident," Draft 3. The status of the Applicant's compliance with the NTOL licensing requirements as well as a number of non-TMI-related items were reviewed during its 243rd meeting. There are a number of both non-TMI and TMI-related requirements not fully resolved. Both the NRC Staff and the Applicant expect that the complete resolution of these outstanding items is essentially a procedural or documentary matter which will be completed within a very few weeks. These items should be resolved to the satisfaction of the NRC Staff. The Committee wishes to be kept informed. The Committee believes that the implementation of the Action Plan as it will be realized at Sequoyah is adequate to assure the safe operation of this plant.

The Committee, in its March 11, 1980 report on the NTOL items, recommended that the licensees develop reliability assessments for their plants and that design studies of possible hydrogen control and filtered vented containment systems be required. The Applicant has conducted studies of a number of means for hydrogen control, and as an interim measure, has proposed installation of a distributed array of ignition sources which it expects to have in place by the fall of 1980. The Applicant has concluded that by this means the containment would be able to cope with the pressure resulting from the combustion of hydrogen released by the reaction with water of up to about 70% of the zirconium in the core. This compares with the 25% which the containment could cope with without any additional control measures and the 30 to 50% estimated to have reacted in the accident at TMI. The NRC Staff plans to review the proposed system in detail to assure itself of its efficacy and that all safety aspects have been taken into account. The Committee wishes to be kept informed of the further conclusions reached by the Staff and the Applicant in their continuing consideration of these matters. The Applicant has conducted reliability assessments of some features of the plant and has considered some aspects of the effects of a possible filtered vented containment. Though the work accomplished to date is limited in scope, these studies are definitely responsive to the Committee's recommendations on these points. The Applicant proposes to continue studies of this nature and to extend the range of their application. While these efforts, as well as those concerned with hydrogen control, should be vigorously pursued, in view of the commitments made by the Applicant, it is the opinion of the Committee that their present incomplete status need not delay the issuance of a full power operating license.

Early this year a differing professional opinion was advanced by a member of the NRC Staff concerning the acceptability of a particular weld repair in the piping to a pressurizer relief valve of Sequoyah Unit No. 1. All other qualified and responsible members of the NRC Staff, as well as professional personnel on the staff of the Applicant, take the position that the weld should be regarded as acceptable since there is no evident reason why it should not be at least as capable as other (more standard) welds which would be considered acceptable. The differing opinion is not that the weld is demonstrably less capable than it need be, but 1) that the evidence available is inconclusive on this point, and 2) that more specifically relevant information could be obtained without serious difficulty. This could be done by constructing a mock-up of the weld in question using material and procedures as similar as possible to those which apply in the actual case and subjecting the mock-up to a through-wall metallographic examination. The results of this examination could then (for example) be compared with those from a full penetration weld in the same material, which has been performed in the standard fashion and deemed acceptable based on satisfactory operational experience with which the majority opinion has compared the present weld. This has not been done. The Committee does not consider it to be particularly likely that this weld repair presents a serious hazard; but it does believe the evidence on this point could be improved. The Committee believes that, in the interest of resolving the question that has been raised to the maximum extent readily possible, steps of the nature outlined should be taken.

The Committee believes, that if due consideration is given to the items mentioned above, the Sequoyah Nuclear Plant, Units 1 and 2 can be operated at levels up to full power without undue risk to the health and safety of the public.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. Tennessee Valley Authority, "Final Safety Analysis Report, Sequoyah Nuclear Power Plant," Volumes 1-13, and Amendments 1-63.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant Units 1 and 2," NUREG-0011, March 1979.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant Units 1 and 2," Supplement No. 1, NUREG-0011, February 1980.
4. U.S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, May 1980.
5. U.S. Nuclear Regulatory Commission, "TMI-Related Requirements for New Operating Licenses," NUREG-0694, June 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 15, 1980

The Honorable Victor Gilinsky
Commissioner
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Gilinsky:

This is in response to your request of July 10, 1980 concerning particular aspects of the Sequoyah Nuclear Plant.

First, as mentioned in the Committee's Sequoyah letter of December 11, 1979, the capability of the ice condenser containment design to cope with the steam resulting from a large LOCA was the subject of detailed discussion over a period of years involving the NRC Staff, the vendor, and the ACRS. As a result of this effort it was concluded that this type of design was fully capable of fulfilling the function mentioned. We have no reason to change that conclusion.

Second, the matter of the control of large amounts of hydrogen is discussed to some extent in the Committee's Sequoyah letter of this date. Although the information available at present is preliminary and will require further detailed confirmation both by the Staff and the Applicant, we expect the present general conclusions to be confirmed. The Applicant has committed to proceed quickly with the installation of a distributed ignition system.

The Committee does not believe that there is any practical need to hold up the issuance of an operating license pending completion of the proposed ignition system.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 8, 1980

Honorable John F. Ahearne
Chairman
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Ahearne:

SUBJECT: SEQUOYAH NUCLEAR POWER PLANT, UNITS 1 AND 2

In connection with the Committee's review of the Sequoyah Nuclear Plant, Commissioner Gilinsky has addressed specific questions to the ACRS regarding ice condenser containments. This is in response to your request for the Committee's comments on the questions raised by Commissioner Gilinsky in his letter of August 7, 1980.

1) "Does the Committee believe additional hydrogen control measures are necessary for ice condenser containments?"

An intensive review of the capability of the Sequoyah containment has recently been completed. Independent estimates have been made by the Applicant, the NRC Staff, various consultants, and the ACRS Subcommittee on Structural Engineering. As a result, it has been concluded that the Sequoyah containment is capable of sustaining a pressure of at least 45 psig without structural failure. On this basis, the containment structure could tolerate burning of all the hydrogen evolved from the oxidation of 20%, or so, of the zirconium in the reactor, assuming the hydrogen was uniformly distributed in the containment atmosphere. Hence, there is a range of accidents involving severe core damage for which additional hydrogen control measures are not necessary. Of course, it would also be necessary to ascertain that all the essential equipment in the containment could withstand such an event. TVA has stated that they are conducting a thorough review of this matter.

For a full scale core meltdown there is no assurance that failure of the containment could be avoided merely by the use of hydrogen control measures. For events involving more than about 30% oxidation of the zirconium, hydrogen control measures may be necessary to avoid containment failure.

A similar situation, though not identical in detail, would be expected to apply to ice condenser plants other than Sequoyah.

The Committee believes that it would be prudent to provide additional hydrogen control measures for ice condenser containments, and that studies to demonstrate the effectiveness, reliability, and absence of significant adverse effects of candidate measures should be pursued actively on a time scale that would permit their application before more than a few additional reactor years of operation of ice condenser containment plants have elapsed. As stated in our Sequoyah Report of July 15, 1980, in the Committee's opinion, there is no need to delay the issuance of a full power operating license for Sequoyah until these studies have been completed.

2) "Is the Committee reasonably persuaded of the effectiveness of distributed igniters in ice condenser containments? Can such igniters be counted on to keep pressure increases caused by hydrogen burns at suitably low values -- which I would define as design pressures -- during accident sequences involving TMI-like quantities of hydrogen?"

On the basis of the preliminary information available, it appears that a distributed ignition system of the type considered for Sequoyah may provide a good capability of controlling the burning of a large amount of hydrogen. It is yet to be established at just what hydrogen concentration a particular style of igniter will provide ignition with high reliability under the conditions anticipated. With the assumption that it can be shown that this concentration is little, if any, higher than the average when the burn occurred at TMI-2, the pressure levels induced by iterated ignition would be well within the 45 psig capability of the Sequoyah containment. There is no present basis for assurance that the pressure increases can be held below the design pressure -- nor would there seem to be any need to do so under the circumstances considered. The hoped for, and expected, performance would be capable of disposing of all the hydrogen that might present itself, up to the point (about 800 kg burned) at which the oxygen level in the containment atmosphere should drop to about 5%, after which no further hydrogen could burn. This, of course, would depend on the continuing operation of the containment heat removal systems.

The action of the igniters will probably reduce the risk, since there will be at least as many ignition events with them in use as if only unintended ignition sources were present. The average amount of hydrogen per burning event should therefore be smaller, and the chance that a large pocket of ignitable or detonable hydrogen could survive without ignition (while waiting for a random source to act) will be reduced.

The results of the present testing program will, of course, be necessary before concluding that the ignition system being studied meets all the necessary objectives.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant, Units 1 and 2," USNRC Report NUREG-0011, Supplement No. 2, August 1980

2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant, Units 1 and 2," USNRC Report NUREG-0011, Supplement No. 3, September 1980

3. Letter from Commissioner V. Gilinsky to M. Plesset, Chairman, Advisory Committee on Reactor Safeguards, dated August 7, 1980

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 8, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Ahearne:

SUBJECT: ADDITIONAL ACRS COMMENTS ON HYDROGEN CONTROL AND IMPROVEMENT OF CONTAINMENT CAPABILITY

We have responded in a letter of this date to your request for comments on the questions raised by Commissioner Gilinsky in his letter of August 7, 1980. In our discussions accompanying the preparation of that response, it became evident that Commissioner Gilinsky's questions need to be considered within a broader context.

In our letter to you dated December 13, 1979, entitled, "Report on TMI-2 Lessons Learned Task Force Final Report," we stated the following concerning "reliability assessments":

> "The ACRS strongly supports the application of reliability assessments to final designs. The Committee supports the Integrated Reliability Evaluation Program (IREP) which is being initiated by the Office of Nuclear Regulatory Research. However, the Committee does not agree that the proposed IREP will fully satisfy the need. The ACRS recommends that the NRC develop a program in which licensees acting individually or jointly develop reliability assessments of their plants, in addition to the NRC IREP, which should be performed concurrently.
>
> "If the reliability assessments were performed in the manner proposed above, it would accelerate obtaining potentially significant safety information and expedite the development of the basis for changes, should they be necessary. It would also provide the operating organizations with better technical insight into the safety of their plants and would provide the benefits to be derived by separate studies of system reliability."

In addition, concerning the topic entitled, "Design Features for Core-Damage and Core-Melt Accidents," we stated the following:

> "The ACRS supports this recommendation. However, the Committee believes that the recommendation should be augmented to require concurrent design studies by each licensee of possible hydrogen control and filtered venting systems which have the potential for mitigation of accidents involving large scale core damage or core melting, including an estimate of the cost, the possible schedule, and the potential for reduction in risk."

The NRC Staff appeared to support this latter recommendation in Task II.B of the Action Plan. However, in the interim rule on degraded core cooling proposed by the NRC Staff in August 1980 and approved for public comment on September 4, 1980, only the study of measures for hydrogen control are requested, leaving other questions of possible improvements in containment design for a rulemaking which appears likely to take some years.

With regard to the reliability assessment of plants in operation or under construction, the NRC Staff appears to be satisfied with an IREP which is moving much more slowly than was being projected in December 1979, when we recommended a major acceleration of such efforts.

If one considers the potential for improving the safety of light water reactors, we believe such consideration will not provide a basis for the rather different priority and emphasis that the NRC is placing on hydrogen control in contrast to the priority and emphasis it is giving to reliability assessment of final design and to a more general approach to improving containment capability.

For many reasons, we believe it is difficult to demonstrate with a high degree of confidence that the frequency of severe core damage or core melt for reactors in operation or under construction is so low that it is not prudent to aggressively pursue measures both to prevent serious accidents and to mitigate them. We believe that the recommendations quoted above from our letter dated December 13, 1979 should be adopted and given priority by the NRC.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References

1. Letter from Commissioner V. Gilinsky to M. Plesset, Chairman, Advisory Committee on Reactor Safeguards, dated August 7, 1980

2. Letter from M. Carbon, Chairman, Advisory Committee on Reactor Safeguards, Subject: Report on TMI-2 Lessons Learned Task Force Final Report, dated December 13, 1979

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 13, 1981

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: REPORT ON THE SEQUOYAH NUCLEAR POWER PLANT, UNITS 1 AND 2

Dear Dr. Ahearne:

During the 249th meeting of the ACRS, January 8-10, 1981, we discussed the NRC Staff's review of the interim hydrogen control system proposed for use in the Sequoyah Nuclear Power Plant, Unit 1. This matter was also discussed at a Subcommittee meeting on January 6, 1981. We have previously commented on this subject in our report dated July 15, 1980 and in two reports dated September 8, 1980. In this previous correspondence we indicated that distributed ignition systems of the type being considered for use in the Sequoyah plant could provide an improved capability for controlling the burning of a large amount of hydrogen and that the use of such a system would probably reduce risk.

We now believe that the results of analyses and tests which we have discussed with the NRC Staff and the Tennessee Valley Authority (TVA) support these conclusions. The NRC Staff and TVA are continuing to work together to resolve the issue of the survivability of the equipment within containment which is important to safety. Although much further study is needed to determine the ability of the many essential components to continue to operate after exposure to the conditions imposed by possible hydrogen burning, the conditions imposed will not be aggravated by the operation of the ignition system, and in all probability will be less severe. We wish to be kept informed of the NRC Staff's and TVA's progress in this work.

We concur with the NRC Staff recommendation to allow the operation of the Interim Distributed Ignition System and believe that this system will provide improved protection against breach of containment in the event that a substantial quantity of hydrogen is generated. We recommend that the NRC Staff and TVA continue their efforts to describe the performance characteristics of the system over a broader range of conditions.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 13, 1982

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2 HYDROGEN CONTROL SYSTEM

During its 272nd meeting, December 9-11, 1982, the Advisory Committee on Reactor Safeguards reviewed the design features of the hydrogen control system which has been proposed by the Tennessee Valley Authority (TVA) for use in the Sequoyah Nuclear Plant Units 1 and 2. This matter was discussed during a Subcommittee meeting held on December 7, 1982. During our review we had the benefit of discussions with representatives of TVA and the NRC Staff. The Committee has previously reported on issues related to hydrogen control for the Sequoyah Nuclear Plant in two letters dated July 15, 1980 and in a letter dated September 8, 1980.

The hydrogen control system reviewed during this meeting has been designated by TVA as the Permanent Hydrogen Mitigation System (PHMS) and replaces the Interim Distributed Ignition System (IDIS). The PHMS utilizes igniters of a different type than those used in the IDIS, and incorporates system changes which are intended to increase the reliability of the igniter system. The TVA proposal for hydrogen control is supported by extensive research and development programs carried out by TVA, the nuclear industry, and the NRC. Some of these programs are currently ongoing and will be continued. We believe that igniter systems represent a viable method for hydrogen control. In addition, we believe that the PHMS is an adequate hydrogen control system and that it will perform its intended function in a manner that provides adequate safety margins.

The NRC Staff has proposed that additional igniters be installed in the upper compartment of the containment. The additional igniters may not be necessary but will do no harm. The NRC Staff has also proposed that the performance of the igniters be tested in a containment spray environment. These proposed tests are intended to ensure the capability of the system to burn small quantities of hydrogen. We are not fully persuaded at this point that the Staff's concern is warranted. We wish to be kept informed on this matter since the questions raised are also relevant to distributed ignition systems in other plants.

The PHMS as presently proposed by TVA uses either offsite power or the emergency diesels as a power source. The PHMS would consequently not control a hydrogen release from a degraded core coincident with a station blackout. We believe that this should be further considered by the NRC Staff and TVA and that, in particular, the use of special emergency procedures should be considered. We wish to be kept informed regarding this matter.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

Reference:

1. U.S. Nuclear Regulatory Commission "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant Units 1 and 2," NUREG-0011, Supplement No. 6, draft dated December, 1982

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

November 4, 1957

Honorable Lewis L. Strauss
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: PRESSURIZED WATER REACTOR

Dear Mr. Strauss:

This letter constitutes the report of the Advisory Committee on Reactor Safeguards with regard to the proposed operation of the Pressurized Water Reactor at the Shippingport Atomic Power Station. The pertinent information is contained in the reports listed on the attached sheet.

On the basis of the information presented, the Committee is convinced that adequate safeguards have been incorporated into the design and construction of the Pressurized Water Reactor and adequate operating procedures have been worked out to insure that it can be operated at designed power with an acceptably low risk to the health and safety of the public.

Inasmuch as this will be the first major nuclear power plant to be operated in this country, the Committee must emphasize that the safety of the installation depends upon competent operation and adequate administrative controls as well as the physical safeguards incorporated into the plant. It is essential that close cooperation exist between the design and operating organizations to assure a safe transition during the startup, from the initial test period through full power operation.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough, Chairman
Advisory Committee on
Reactor Safeguards

Following is a list of reports covering the PRESSURIZED WATER REACTOR:

WAPD-SC-541 dated September 1957

WAPD-SC-542 dated October 1957

WAPD-SC-543 dated May 1957

WAPD-SC-544 dated May 1957

WAPD-SC-545 dated September 1957

WAPD-SC-546 dated September 1957

WAPD-SC-547 dated June 1957

WAPD-SC-548 dated September 1957

WAPD-SC-549 dated June 1957

WAPD-PWR-970 dated June 1957

WAPD-PWR-971 dated July 1957

WAPD-PWR-972 dated July 1957

WAPD-PWR-973 dated May 1957

WAPD-PWR-974 dated May 1957

DL-S-191 dated May 1957

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

December 13, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: SHIPPINGPORT PRESSURIZED WATER REACTOR

Dear Mr. McCone:

On December 7, 1960, as a part of its thirtieth meeting, the Advisory Committee on Reactor Safeguards met at the site of the Shippingport Pressurized Water Reactor. Representatives of the AEC staff, Westinghouse and Duquesne Light Company participated. A resume describing the origin, development and up-to-date reactor and power-plant operation experience, including refueling, was presented. This included a description of the coordination of the responsibilities of the AEC, Westinghouse and Duquesne covering normal operation, test procedures, and personnel training.

Future operation including the proposed design and operational changes required with the installation of PWR Core #2 was described. The ACRS had been furnished previously with a description of the proposed PWR Core #2 and with an AEC staff review. The change-over to accommodate PWR Core #2, which increases the power level to 505 Mw thermal and will increase the fission product inventory, will require major revisions in the mechanical design of the reactor and its components including heat dissipation equipment.

Based upon the preliminary evaluations submitted, the ACRS believes that the proposed modifications including Core #2 will not significantly change the present safety status. However, it is suggested that during the design development period, consideration be continued toward developing methods for routine integral testing of the leakage rate of the containment vessel, methods to reduce the leakage rate, and improvements in the effectiveness of the safety injection system.

Sincerely yours,

/s/
Leslie Silverman
Chairman

cc: A. R. Luedecke, GM
W. F. Finan, AGMRS
H. L. Price, Dir., DL&R

References:

Experience with the PWR Reactor Plant Container, WAPD-PWR-1318 dated July 1960.

Safeguards Aspects of PWR Core 2, WAPD-PWR 2191, July 1960.

Shippingport Operations, from Start-up to First Refueling, Dec. 1957 to Oct. 1959, undated.

The First Refueling of the Shippingport Atomic Power Station, WAPD-233 dated July 1960.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

December 16, 1964

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

Subject: REPORT ON SHIPPINGPORT PRESSURIZED WATER REACTOR

Dear Dr. Seaborg:

At its sixtieth meeting, on December 10-12, 1964, the Advisory Committee on Reactor Safeguards reviewed modifications to the Shippingport Pressurized Water Reactor. These modifications, including installation of a new core, are designed to permit an increase in power level from 231 MW(t) to 505 MW(t). In its letter of December 13, 1960, the Committee commented on the proposed design and associated changes required for the installation of Core No. 2. In its present review, the Committee had the benefit of discussion with representatives of Westinghouse Electric Corporation, the Division of Naval Reactors, and the AEC Regulatory Staff, and of the documents referenced. In addition, a Subcommittee Meeting was held at Shippingport on October 29, 1964.

The Committee believes that methods developed and plans formulated for testing containment leakage rates and penetration integrity will provide reasonable assurance of adequate containment. The installation of a booster pump and provision to inject water above each fuel assembly represents an acceptable improvement to the safety injection system.

The Committee recognizes that the PWR primary pressure vessel has been exposed to a cumulated neutron flux which has increased the temperature at which brittle fracture may occur, to an extent such that careful procedures are required when the primary system is being heated up or cooled down. In the Committee's opinion, pressures should be kept as low as feasible when the vessel temperature is below the Reference Transition Temperature. Accordingly, pump heat-up rather than nuclear heat-up may be preferable.

Release of fission products, iodine in particular, to the environs in the unlikely event of a serious accident is of concern at higher power levels. Although Westinghouse and the Division of Naval Reactors are convinced that doses from accidental releases would not exceed the guideline values given in 10 CFR Part 100, the supporting data are only qualitative. The Committee believes that values for iodine plateout, deposition, and holdup within the concrete cells may be amenable to actual measurement, and recommends that such tests be considered. Should testing not prove practicable, the Committee recommends that an additional engineered safeguard, such as fans arranged to exhaust air from the concrete shielding structures through air cleaning systems, be provided.

With the aforementioned reservations, the Advisory Committee on Reactor Safeguards believes that the modified PWR utilizing Core 2 can be operated at the proposed design power level without undue risk to the health and safety of the public.

Sincerely yours,

/s/
Herbert Kouts
Chairman

References:

1. WAPD-SC-501, "PWR Core 2 Safety Analysis," dated June 1964.

2. "Answers to Questions on PWR Core 2 Safety Analysis," undated, received September 16 and 25, 1964.

3. List of ACRS Questions and Answers, Attachments 1-11 and 4 related drawings, undated, received November 20, 1964 and Attachment 12, undated, received December 9, 1964.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 19, 1976

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON THE SHIPPINGPORT ATOMIC POWER STATION LIGHT WATER BREEDER REACTOR

Dear Mr. Rowden:

At its 196th meeting, August 12-14, 1976, the Advisory Committee on Reactor Safeguards reviewed the Energy Research and Development Administration (ERDA) proposal to install a light water breeder reactor (LWBR) core, and to make numerous modifications, in the Shippingport Atomic Power Station. It is proposed to operate the LWBR core for about three years. A Subcommittee meeting and site visit was held on July 21, 1976. The Committee had the benefit of discussions with representatives of the Westinghouse Electric Corporation (Bettis), the Division of Naval Reactors of ERDA, Duquesne Light Company, the Nuclear Regulatory Commission (NRC) Staff, and of the documents listed. The Committee reported previously on the Shippingport Atomic Power Station at its 2nd meeting, November 1-3, 1957, its 30th meeting, December 7, 1960, and its 60th meeting, December 10-12, 1964.

The proposed LWBR oxide fuel core consists of thorium-232 and uranium-233 and is designed to have a net conversion ratio slightly greater than 1.0, compared to the conventional pressurized water reactor with a conversion ratio less than 1.0. The LWBR core will operate at less than one-half the power density of the preceding core.

The Committee recognizes that the Shippingport Atomic Power Station is the oldest operating commercial reactor. It was designed and built in accordance with the stringent requirements imposed by the Naval Reactors Program at a time prior to the issuance of 10 CFR Part 50. The Committee also recognizes that it is not possible to demonstrate strict compliance with all current safety criteria being applied to new plant construction.

Substantial modifications have been made to the Shippingport Atomic Power Station emergency core cooling system, and safety-related improvements have been made in many other portions of the plant.

The ACRS concurs with the NRC Staff recommendations regarding installation of diverse trip signals for initiation of safety injection, and the installation of a chlorine monitor in the control room.

With regard to lockout of the pressurizer surge line isolation valve, which is being re-examined by the NRC Staff and will be discussed in a supplemental Staff report, the ACRS position regarding valve lockout has been cited previously (e.g., ACRS Report on Trojan Nuclear Station, dated November 20, 1974, and Item IIC-1 of ACRS Report No. 4 on Status of Generic Items Relating to Light-Water Reactors, dated April 16, 1976).

The Committee believes it to be acceptable for the Shippingport Atomic Power Station to operate with the Light Water Breeder Reactor core as proposed.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Additional Remarks by Mr. H. S. Isbin

In the review of this project the Committee was informed that a Division of Naval Reactors' representative monitors the operations of the reactor. The concept of a federal monitor on a watch standing basis with the authority to shut the reactor down appears to me to be a carryover of the initial operating procedures of the Shippingport Atomic Power Station of some twenty years ago. In my opinion, a federal monitor would not enhance safety within the present system of licensing commercial nuclear power reactors. The evolution of the independent federal agency, the Nuclear Regulatory Commission, with its full complement of technical and experienced personnel to set Technical Specifications for operations, to carry out inspections and enforcements, and to require rigid qualifications for the licensee's operators, now constitutes the authority for thorough and effective monitoring of nuclear power operations.

References:

1. Nuclear Regulatory Commission (NRC) Staff's Safety Evaluation Report for the Light Water Breeder Reactor (LWBR), dated July 1976.
2. Shippingport Atomic Power Station Safety Analysis Report for the LWBR, Volumes 1 through 10.
3. Letters, dated January 19, January 27, January 30, February 9, April 27, July 7, and August 9, 1976, Division of Naval Reactors of the Energy Research and Development Administration to the Division of Reactor Licensing of the NRC, forwarding supplementary information related to the LWBR review.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 6, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: EXTENDED OPERATION OF SHIPPINGPORT LIGHT WATER BREEDER REACTOR

Dear Dr. Ahearne:

The Division of Naval Reactors, Department of Energy, in its letter of March 10, 1980, discussed its plan to operate the Light Water Breeder Reactor (LWBR) core at Shippingport Atomic Power Station beyond the 18,000 effective full power hours (EFPH) originally planned, and requested NRC comments by April 30, 1980 regarding the extended operation.

During its 241st meeting, May 1-3, 1980, the Advisory Committee on Reactor Safeguards discussed this proposal with representatives of the Westinghouse Electric Corporation (Bettis), Duquesne Light Company, the Division of Naval Reactors of DOE, and the Nuclear Regulatory Commission Staff. The Committee also had the benefit of the documents listed.

The Committee concurs with the NRC Staff's letter to Admiral Rickover dated April 30, 1980, which recommended that consideration be given to the bulletins, orders, and requests issued to the commercial nuclear power industry as a result of the TMI-2 accident.

Subject to the above, the Committee believes it to be acceptable to operate the Shippingport Atomic Power Station Light Water Breeder Reactor core to 24,000 EFPH as proposed.

Sincerely yours,

Milton S. Plesset

Milton S. Plesset
Chairman

References:
1. Letter from H. R. Denton, NRC, to Adm. H. G. Rickover, DOE Naval Reactors, Subject: Continued Operation of Shippingport LWBR, dated April 30, 1980
2. Letter from H. G. Rickover, DOE Naval Reactors, to H. R. Denton, NRC (NR:D:H.G.Rickover Z#818) Subject: Light Water Breeder Reactor - Plans to Continue Operation of the Present Reactor Core
3. NBI Log No. 0203-80/0081L, "Information Report Concerning Extended Operation of the LWBR Core at Shippingport"

cc: Admiral H. G. Rickover

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

December 18, 1969

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

Subject: REPORT ON SHOREHAM NUCLEAR POWER STATION UNIT 1

Dear Dr. Seaborg:

At its 116th meeting, December 11-13, 1969, the Advisory Committee on Reactor Safeguards completed its review of the application by Long Island Lighting Company for authorization to construct the Shoreham Nuclear Power Station Unit 1. The project was considered at Subcommittee meetings on October 30, 1969, at the plant site, and on December 4, 1969, in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives of Long Island Lighting Company, General Electric Company, Stone and Webster Engineering Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed.

The site for the Shoreham station is located in the Town of Brookhaven, Suffolk County, New York, on the shore of Long Island Sound, north of Brookhaven National Laboratory. The exclusion zone radius for the plant is 1,000 ft. and the reactor will be located 1,500 ft. from the nearest residence. The low population zone radius is five miles. The population within the low population zone was 7,500 in 1960, and is estimated to be 12,300 in 1980. The nearest town with a population over 25,000 is Stratford, Connecticut with a 1960 population of 45,000, 18 miles from the site across Long Island Sound.

The Shoreham Nuclear Power Station Unit 1 will use a single cycle, forced circulation, boiling water reactor similar to the previously reviewed reactors of the Cooper, Hatch, and Brusnwick stations (ACRS reports of March 12, 1968, May 15, 1969, and October 16, 1969). The Shoreham reactor is designed to produce 2436 MWt with a maximum performance rating of 2550 MWt.

The Shoreham primary containment is a steel lined, reinforced concrete structure having a conical drywell section located directly over a cylindrical wetwell section. This arrangement provides a more direct path from the drywell to the wetwell than the "light bulb and torus" design used in other boiling water reactor plants. The applicant reports that this design results in a lower peak pressure in the structure in the event of a loss-of-coolant accident. A reinforced concrete floor separates the drywell and wetwell sections of the primary containment and is connected to the wall of the containment by a flexible seal which includes provisions for periodic leak testing.

The applicant has stated that he will perform analyses to determine the potential dynamic loads on the drywell vent pipes during a loss-of-coolant accident and that he will provide conservatively designed structural restraints at the lower portions of the pipes to resist these forces. In order to provide increased assurance that the design pressures for the vapor-suppression system are conservative, the applicant should perform additional parametric analyses. These analyses should be completed in a manner satisfactory to the Regulatory Staff.

Shear forces in the containment structure caused by earthquakes will be resisted by diagonal steel reinforcing bars. On the basis of recent tests, the applicant has suggested that these shear forces could be resisted by shear friction and aggregate interlock in the concrete alone. The Committee believes that the diagonal steel reinforcing should be provided unless convincing evidence, both analytical and experimental, can be presented to show that it is not needed, both from the standpoint of the integrity of the steel liner and from the standpoint of the possible effects of containment deformations on the reactor vessel support structure.

The reactor building, which serves as the secondary containment, is cylindrical in shape, with reinforced concrete walls up to the level of the main crane rail. Above this point the building is steel framed, with sheet metal siding and a metal deck roof. Leakage from the primary containment, in the event of an accident, is mixed in the reactor building and is filtered through redundant absolute and charcoal filter banks. It is important that no leakage from the primary containment bypass the secondary containment and the associated filtering systems in the event of an accident. The applicant should study the effects of leakage through possible bypass paths, with particular emphasis on the main steam line isolation valves, and should prpopse measures to deal with any such bypass leakage. This matter should be resolved in a manner satisfactory to the Regulatory Staff during construction of the plant. The Committee wishes to be kept informed of progress in this area.

The applicant proposes to carry out a full radiographic examination of all butt welds in the steam system piping from the main steam line isolation valves to the turbine stop valves, and to the first block valve in each branch line 2½ inches or larger in size. He also proposes to examine essentially all of the pressure-containing bodies of the main turbine stop and bypass valves by radiographic means, and to examine branch line valve bodies 2½ inches or larger by radiographic or surface inspection means. The Committee believes that the proposed program is acceptable for this system.

A large number of instrument lines, approximately one inch in diameter, penetrate the primary containment and terminate as closed systems at pressure sensing devices. Some of these lines connect directly to the reactor primary system. The isolation provisions for these instrument lines include a manual shutoff valve and a spring-loaded excess flow check valve for each line, with both valves located outside the primary containment. The Committee believes that such provisions may be satisfactory without inclusion of remotely operable isolation valves. However, since these lines represent a potential source for a primary system and containment leak, it is essential that proper attention be given during design and construction to questions of isolation, control of leak rate, quality assurance, integrity of safety signals, and minimization of the possible mechanical damage while the primary system is pressurized. The applicant should propose design criteria for these lines that are satisfactory to the Regulatory Staff. Also, the applicant should study and propose means to reduce the rate of possible leakage from instrument lines in the event of failure so that such leakage would not damage the secondary containment or bypass the building filters.

Plant grade has been established at 20 feet above mean sea level and protection of essential components against the effects of storm flooding has been planned to an elevation of 25 feet above mean sea level. The applicant is currently recalculating the peak storm surge water level and will increase the elevation to which the plant is to be protected against flooding if the recalculation shows this to be necessary.

The Shoreham station is located about four and three quarter miles from the Peconic River Airport, operated by the Grumman Aerospace Corporation for the U. S. Navy. The applicant and the Regulatory Staff have independently reviewed the probability of an aircraft enroute to or from this airport crashing into the nuclear station. The Regulatory Staff review has included the available statistics on all of the types of aircraft using the airport, the types of flights made, and on both fatal and non-fatal crashes. The Staff concludes that the Shoreham station is far enough from the airport so that the probability of an aircraft striking the Shoreham station is not significantly increased by the presence of the airport, and that special protective measures against an aircraft crash are not required for this station. The Committee concurs in these conclusions.

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 following a loss-of-coolant accident.

(c) Analysis of methods to limit damage to the spent fuel pool and to reduce the release of fission products in the event of a dropped fuel cask.

Other problems related to boiling 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 Shoreham 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 Shoreham 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/

Stephen H. Hanauer
Chairman

References - Shoreham Nuclear Power Station

1) Letter from Long Island Lighting Company dated April 18, 1969; Amendment No. 4 to License Application; Volumes I, II and III of the Preliminary Safety Analysis Report

2) Letter from Long Island Lighting Company dated April 18, 1969; Amendment No. 5 to License Application; Responses to comments contained in DRL's letter of January 19, 1969

3) Letter from Long Island Lighting Company dated June 30, 1969; Amendment No. 6 to License Application; Replacement pages

4) Letter from Long Island Lighting Company dated August 22, 1969; Amendment No. 7 to License Application; Replacement pages

5) Letter from Long Island Lighting Company dated October 24, 1969; Amendment No. 8 to License Application; Replacement pages

6) Letter from Long Island Lighting Company dated November 19, 1969; Amendment No. 9 to License Application; Replacement pages

7) Letter from Long Island Lighting Company dated December 5, 1969; Amendment No. 10 to License Application; Replacement pages

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

February 11, 1970

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

Subject: SUPPLEMENTARY REPORT ON SHOREHAM NUCLEAR POWER STATION UNIT 1

Dear Dr. Seaborg:

At its 118th meeting, February 5-7, 1970, the Advisory Committee on Reactor Safeguards reviewed the plans of the Long Island Lighting Company to accommodate hypothetical piping failures in their nuclear plant containment drywell. The project had been previously reviewed by the Committee at its 116th meeting, December 11-13, 1969, and the Committee's report to you, dated December 18, 1969, discusses the results of that meeting. During its present review, the Committee had the benefit of discussions with representatives of Long Island Lighting Company, General Electric Company, Stone and Webster Engineering Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The applicant stated that it is a design criterion for the Shoreham Station that in the event of pipe rupture within the primary containment, resulting pipe movement will not violate the integrity of the primary containment. It is also a design criterion that, in the event of such rupture, adequate emergency core cooling be assured. The Committee believes that these criteria are acceptable.

The Committee believes that the above item can be resolved during construction and reconfirms its previous conclusion that, if due consideration is given to the items discussed in this report and the Committee's prior report, the nuclear plant proposed for the Shoreham 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/
Joseph M. Hendrie
Chairman

Reference attached.

Reference:

1. Letter from Long Island Lighting Company, dated January 30, 1970; Amendment No. 11 to License Application, additional and replacement pages

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

September 23, 1970

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

Subject: SUPPLEMENTARY REPORT ON SHOREHAM NUCLEAR POWER STATION UNIT 1

Dear Dr. Seaborg:

At its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards considered possible adverse effects of a nearby Nike Hercules battery on the safety of the proposed Shoreham Nuclear Power Station Unit 1 of the Long Island Lighting Company. This matter was also considered at Subcommittee meetings on October 30, 1969 at the site and on September 10, 1970, in Washington, D. C. The Shoreham project was the subject of previous Committee reports to you dated December 18, 1969 and February 11, 1970. During its present review the Committee had the benefit of discussions with representatives and consultants of the Long Island Lighting Company, The General Electric Company, The Stone and Webster Engineering Corporation, the Department of Defense, and the AEC Regulatory Staff, and of the document listed.

The launching area of the Nike Hercules battery is located about 1.4 miles from the Shoreham reactor. The battery control area is about 0.7 miles from the reactor. The missile launchers have a fixed direction, not toward the site, and missiles accidentally launched with guidance control would not pass over the site. Further, the maneuvering limitations of the Nike Hercules are such that a missile under guidance control could not be turned to impact in the site area without breaking up in the air.

The Nike Hercules battery is located far enough from the Shoreham site so that the detonation of all high explosive at the battery would not cause significant damage to the proposed reactor facility.

In the unlikely event of an accidental launching of a missile there would be no guidance control and the missile would break up soon after launching. The resulting fragments, including the warhead, would be distributed over the ground area within several miles of the launching position.

The likelihood of a missile fragment striking the Shoreham plant area and causing significant damage due to impact or explosion is a combination of the probability of a launch which results in breakup of the missile and the probability, given such a breakup, that a large fragment strikes a sensitive portion of the plant. The Committee believes that the overall probability of significant damage to the proposed plant from the Nike Hercules battery is so small that it does not detract from the acceptability of the site for the proposed plant.

The Committee recommends that the Regulatory Staff make arrangements with the Department of Defense to be notified of any contemplated changes in the Nike Hercules installation in the Shoreham area which might change the present evaluation of the likelihood of damage to the reactor facility. These arrangements should be completed before power operation of the reactor.

The Committee believes that if due consideration is given to the items described in the previous reports, the nuclear plant proposed for the Shoreham 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,

Joseph M. Hendrie
Chairman

<u>Reference</u>

1) Department of Defense letter dated September 9, 1970 re: Nike Hercules System at Rocky Point, Long Island

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 19, 1981

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON THE SHOREHAM NUCLEAR POWER STATION UNIT 1

Dear Dr. Palladino:

During its 258th meeting, October 15-17, 1981, the Advisory Committee on Reactor Safeguards completed its review of the application of the Long Island Lighting Company (LILCO) for a license to operate the Shoreham Nuclear Power Station Unit 1. A Subcommittee meeting was held in Washington, D.C. on September 30, 1981 to consider this project. A tour of the facility was made by members of the Subcommittee on April 30, 1981. During its review, the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee reported on the construction permit application for this plant in a letter to AEC Chairman Glenn T. Seaborg dated December 18, 1969.

The Shoreham plant is located on Long Island in the Town of Brookhaven, Suffolk County, New York, about 55 miles east-northeast of downtown New York City. It uses a GE BWR-4 nuclear steam supply system with a rated power level of 2436 MWt and a Mark II pressure suppression containment with a design pressure of 48 psig. The Shoreham plant is one of three included in the Mark II Owners Group lead plant program. The NRC Staff has completed its review of the lead plant program and has issued NUREG-0487 and Supplements I and II, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria." The NRC Staff has concluded that Shoreham satisfies these criteria. In addition, LILCO has committed to evaluate the final generic load definition of NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria," against the load specification used in the interim evaluation (NUREG-0487). Subject to satisfactory completion of this work, the NRC Staff has found the Shoreham containment acceptable. We concur in this finding.

LILCO described the management organization and the technical personnel available for operation of the Shoreham plant. Because of LILCO's lack of BWR operating experience, the NRC Staff is requiring that the control room staff and senior plant management be provided with advisors who have substantial BWR operating experience. We concur with the NRC Staff that supplemental personnel experienced in BWR operation are needed until adequate operating experience is developed by the LILCO staff.

LILCO described three safety review committees which will be a permanent part of the Shoreham organization. We believe that these committees should include some expertise from sources outside LILCO's or its contractors' organizations to provide balanced professional judgment on matters that could affect public health and safety. LILCO should organize the planned safety review committees as soon as practical so they will have time to develop an understanding of plant related safety matters prior to plant operation.

LILCO also described its program and philosophy for training of personnel. The initial training that the operations staff has received using a contractor-run training organization appears adequate. However, LILCO should establish an in-house training program to be maintained on a continuing basis so that operational skills are enhanced.

LILCO has initiated a Shoreham plant assessment based on probabilistic risk assessment techniques. The Applicant's assessment effort in this area will provide a valuable addition to his operational knowledge.

An outstanding issue in the NRC Staff's Safety Evaluation Report dated April 1981 and Supplement 1 to that report dated September 9, 1981 involves the remote shutdown system. The NRC Staff is concerned that a single, random failure in the instruments and controls of systems controlled from the remote panel or in the systems themselves may prevent the remote shutdown panel from performing its function. This item should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

The NRC Staff has identified other outstanding issues in its Safety Evaluation Report. We believe that these outstanding issues can be resolved and recommend that this be done in a manner satisfactory to the NRC Staff before operation at full power.

We believe that if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that Shoreham Nuclear Power Station Unit 1 can be operated at power levels up to 2436 MWt without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. Long Island Lighting Company, "Shoreham Nuclear Power Station Unit 1 Final Safety Analysis Report," Volumes 1-16 and Amendments 1-40.

2. U.S. Nuclear Regulatory Commission "Safety Evaluation Report Related to the Operation of Shoreham Nuclear Power Station Unit 1," NUREG-0420, dated April 1981 and Supplement No. 1 dated September 9, 1981.
3. U.S. Nuclear Regulatory Commission, NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," dated October 1978 with Supplements 1 and 2 dated September 1980 and February 1981.
4. U. S. Nuclear Regulatory Commission, NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria," dated August 1981
5. Letter, SNRC-629, B. R. McCaffrey, Manager, Long Island Lighting Company (LILCO), to Advisory Committee on Reactor Safeguards, regarding response to requests for information made at the ACRS Subcommittee meeting on Shoreham Station Unit 1 of September 30, 1981, dated October 13, 1981.
6. Letters from J. P. Novarro, Project Manager, LILCO to Harold R. Denton, Director, Office of Nuclear Reactor Regulation (NRR), NRC, dated July 17, 1981; June 29, 1981; June 15, 1981; May 29, 1981; May 29, 1981; May 28, 1981; May 27, 1981; May 27, 1981; May 27, 1981; May 21, 1981; May 15, 1981; May 15, 1981; May 15, 1981; May 12, 1981; April 22, 1981; April 15, 1981; March 16, 1981.
7. Letters from B. R. McCaffrey, Manager, LILCO, to Harold R. Denton, Director, NRR, NRC dated August 18, 1981; August 7, 1981; July 31, 1981; July 23, 1981; July 22, 1981; July 21, 1981; July 20, 1981.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 18, 1977

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON THE SKAGIT NUCLEAR POWER PROJECT, UNITS 1 AND 2

Dear Dr. Hendrie:

During its 211th meeting, November 3-5, 1977, the Advisory Committee on Reactor Safeguards completed its review of the application of the Puget Sound Power and Light Company, the Pacific Power and Light Company, the Washington Water Power Company, and the Portland General Electric Company (the Applicants) for a permit to construct the Skagit Nuclear Power Project, Units 1 and 2. The Puget Sound Power and Light Company will be responsible for the design, construction, and operation of the station. The application was reviewed at a Subcommittee meeting in Seattle, Washington on September 30, 1977. A visit was made to the site by Subcommittee members on August 30, 1976. Regional tectonics and matters pertaining to the seismicity of the Skagit site were discussed at Subcommittee meetings on September 1-2, 1977 in San Francisco, California, on October 27-28, 1977 in Portland, Oregon, and during the 209th meeting, September 8-10, 1977. During its review, the Committee had the benefit of discussions with members of the Nuclear Regulatory Commission Staff (NRC Staff), and with representatives of the U.S. Geological Survey (USGS), of the Applicants, the Bechtel Power Corporation and the General Electric Company. The Committee also had the benefit of the documents listed and of participation by members of the public.

The Skagit Nuclear Power Project includes two 3800 MWt boiling water reactors of the BWR-6 type, each housed in a Mark III containment. The design of the Skagit Nuclear Power Project is similar to that of the Grand Gulf Nuclear Station, Units 1 and 2 and the River Bend Station, Units 1 and 2, on which the Committee reported in its letters of May 15, 1974 and January 14, 1975, respectively.

The NSSS for the Skagit plant is similar to but not identical to the GESSAR-251 reference design. Because of differences in the designs and because GESSAR-251 had not received Preliminary Design Approval when the Skagit application was submitted, the NRC Staff made a custom review of the Skagit plant. The Committee reported on GESSAR-251 in its letter of December 17, 1976.

The plant will be located in Skagit County in northwestern Washington, approximately 60 miles north of Seattle, Washington and 37 miles southeast of Bellingham, Washington. The site consists of 1500 acres of land on the north side of the Skagit River Valley in the forested western foothills of the Northern Cascade Mountains. The minimum exclusion area boundary distance is 1,980 feet. The low population zone has a radius of 4 miles and includes a population of 1,563 (1970 Census). The nearest center of population is Mt. Vernon, Washington, which had a population of 8,532 in 1970 and is 9.5 miles from the plant.

In view of statements made by the NRC Staff and the USGS, information presented by the Applicants, and opinions provided by ACRS consultants, the Committee believes that horizontal ground accelerations of 0.35g for the safe shutdown earthquake and 0.175g for the operating basis earthquake, committed to by the Applicants, are acceptable.

The NRC Staff has identified a number of safety items which will require resolution before issuance of a Construction Permit. These matters should be resolved in a manner satisfactory to the NRC Staff. The Committee believes these items can be resolved prior to the issuance of a Construction Permit.

With regard to the generic problems listed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors - Report No. 6," dated November 15, 1977 items considered relevant to the Skagit Nuclear Power Project are: II-4, 5A, 5B, 6, 7, 9, 10; IIA-4; IIB-2, 3, 4; IIC-1, 2, 3A, 3B, 5, 6; IID-1, 2. These problems should be dealt with by the NRC Staff and the Applicants as solutions are found.

The Advisory Committee on Reactor Safeguards believes that if due consideration is given to the foregoing, the Skagit Nuclear Project, 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.

Sincerely yours,

M. Bender

M. Bender
Chairman

References

1. Puget Sound Power and Light Company: "Preliminary Safety Analysis Report for Skagit Nuclear Power Project, Units 1 and 2" with Amendments 1-18 and Supplements 1-18.
2. U.S. Nuclear Regulatory Commission: "Safety Evaluation Report by the Office of Nuclear Reactor Regulation, Related to the Puget Sound Power and Light Company Construction of Skagit Nuclear Power Project, Units 1 and 2, Docket Nos. STN 50-522 and STN 50-523," NUREG-0309, dated September, 1977.
3. E. S. Cheney, "Interim Report on the Seismic and Geologic Hazards to the Proposed Skagit Power Site, Sedro Wooley, Washington," Revision of October, 1977, unpublished manuscript.
4. Letter from Roger M. Leed to Dr. S. H. Bush, concerning the seismic potential for the Skagit site, dated November 1, 1977.
5. Letter from faculty members in the Earth Sciences at the University of Washington, as signed on letter, to the U. S. Geological Survey and the U. S. Nuclear Regulatory Commission, concerning the need for more detailed geological maps of the Skagit site, dated November 2, 1977.
6. Letter from D. B. Vassallo, Assistant Director for Light Water Reactors, Division of Project Management, to Mr. Myer Bender, Chairman, Advisory Committee on Reactor Safeguards, concerning NRC Staff position on Skagit seismic design, dated November 15, 1977.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 16, 1983

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE SKAGIT/HANFORD NUCLEAR PROJECT, UNITS 1 AND 2

During its 274th meeting, February 10-12, 1983, the Advisory Committee on Reactor Safeguards completed its review of the application of the Puget Sound Power and Light Company, the Pacific Power and Light Company, the Washington Water Power Company, and the Portland General Electric Company (the Applicants) for a permit to construct the Skagit/Hanford Nuclear Project, Units 1 and 2. The Puget Sound Power and Light Company will be responsible for the design, construction, and operation of the station.

This project had originally been planned for a site on the Skagit River and was reviewed in that context by the ACRS during its 211th meeting, November 3-5, 1977. The Committee concluded that the Skagit Nuclear Power Project, 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" in its letter to NRC Chairman, Joseph M. Hendrie, dated November 18, 1977. In 1980 the Applicants decided to move the project to a site on the Hanford Reservation, and the project name was changed in 1981 from the Skagit Nuclear Power Project to the Skagit/Hanford Nuclear Project. Taking into account this new site, the request to construct this plant was again reviewed during a Subcommittee meeting in Richland, Washington, on January 24-25, 1983. TMI-related requirements and other matters of interest were also reviewed. A visit to the new site was made by members of the Subcommittee on January 24, 1983.

The Skagit/Hanford Nuclear Project includes two 3800 MW(t) boiling water reactors of the BWR-6 type, each housed in a Mark III containment. The design of the Skagit Nuclear Project is similar to that of the Grand Gulf Nuclear Station, Unit 1 on which the Committee reported in its operating license letter of August 18, 1982.

The NSSS for the Skagit/Hanford plant is similar to, but not identical with, the GESSAR-251 reference design. The Committee reported on the GESSAR-251 design in its letter of December 17, 1976. Because of the differences in design and because GESSAR-251 had not received preliminary design approval when the Skagit application was originally submitted, the

NRC Staff made a custom review of the Skagit plant. Except as required by differences between the original and present sites -- including items such as the water supply, the temperature and humidity ranges of the atmosphere, and a foundation on soil rather than rock -- and changes in regulatory requirements between 1977 and 1982, the present plant design is essentially the same as that considered originally.

The Project will be located on the Hanford Reservation in Benton County, Washington, approximately 5 miles west of the Washington Public Power Supply System Nuclear Project No. 2 (WNP-2) and 4.8 miles northwest of the Fast Flux Test Facility (FFTF). It is 8 miles west of the Columbia River, 7 miles north of the Yakima River at Horns Rapid Dam, and 12 miles northwest of the city of North Richland. The exclusion area boundary is at a radius of one mile. The low population zone has a radius of 4 miles, which includes no residents. The 10 mile radius includes a resident population of 357. In addition, about 5000 persons are employed at the WNP-2 and FFTF sites. The nearest center of population is Richland, Washington with a population of 33,578 (1980 census).

The schedule for the start of construction has not yet been established. In addition to the need for receiving a construction permit, the start of construction will depend on the decision by the regional power planning council to include the Skagit/Hanford Nuclear Project as a power resource in their regional power plan. It is also dependent on the state of the economy.

The NRC Staff has asked the Applicants to perform additional core drilling to determine if capable faults are associated with the May Junction Monocline, which, at its closest point, is about 4 miles north of the site. We agree with this recommendation, and the Applicants have committed to the additional core drilling before any major construction work is initiated. Although it is not expected that such subsurface investigations will resolve small faults with accumulated vertical displacements less than about 20 feet, we believe that such faults would not present an earthquake hazard as large as that already taken into account in the seismic design. The Applicants have designed for a safe shutdown earthquake (SSE) of 0.35g, which is significantly higher than the SSE of 0.25g deemed acceptable for the WPPSS-2 plant located only 5 miles away.

The Applicants have committed to perform a probabilistic risk assessment (PRA) to examine core and containment heat removal reliability. The PRA will include the potential effects of external events such as earthquakes, floods, and other environmental phenomena. The results may be useful in determining whether changes or design improvements are needed.

The ACRS believes that, if due consideration is given to the matters noted, the Skagit/Hanford Nuclear Project, 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. Should there be significant changes in design or regulatory requirements before the actual start of construction, the Committee would expect to review this application again.

Sincerely,

J. J. Ray
Chairman

References:
1. Puget Sound Power and Light Company, "Skagit/Hanford Nuclear Project Preliminary Safety Analysis Report," Volumes 1-12 and Amendments 1-29.
2. Puget Sound Power and Light Company, "Skagit/Hanford Nuclear Project Application for Site Certification/Environmental Report," Volumes 1-4 and Amendments 1-8.
3. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report, Skagit Nuclear Power Project, Units 1 and 2," NUREG-0309, dated September 1977, and Supplement No. 2, dated October 1981, and Supplement No. 3, dated December 1982.

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

March 14, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: SMALL SIZE PRESSURIZED WATER REACTOR, JAMESTOWN, NEW YORK, SITE

Dear Mr. McCone:

At its twenty-fourth meeting, March 10-12, 1960, the Advisory Committee on Reactor Safeguards considered the site proposed for a Small Size Pressurized Water Reactor to be located in the City of Jamestown, New York. The data furnished in the Site Report (referenced below) provided only general information on the reactor which is in the conceptual stage. In addition to the site report, the ACRS had the benefit of comments from the AEC Staff and others as well as a visit to the site by a Subcommittee.

This 60 MW (thermal) pressurized light water moderated reactor is to be built and operated by the Commission on a site furnished by the City of Jamestown, New York, which will also provide the generating plant. The proposed site comprises thirty-five acres of city owned land located in the northwest corner of the city approximately 1.75 miles from the center.

The ACRS believes that such factors as the small size of the site; proximity to the City of Jamestown with its high population density; unfavorable meteorology; lack of control by the City of Jamestown over the area contiguous to the south and west boundaries of the site, which is located within the limits of the town of Celeron; and the long periods of low flow in the Chadakoin River with consequent adverse effects on liquid waste disposal all indicate that this site is not suitable for a power reactor of this size in the present stage of technology.

Sincerely yours,

/s/

Leslie Silverman
Chairman

cc: A.R.Luedecke, GM
W.F.Finan,OGM
H.L.Price, DL&R

References

U. S. Atomic Energy Commission, Oak Ridge, Tennessee, "Small Size Pressurized Water Reactor, Jamestown, New York, Site Report" - February 8, 1960.

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

June 30, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: SMALL SIZE PRESSURIZED WATER REACTOR, JAMESTOWN, NEW YORK SITE

Dear Mr. McCone:

At its twenty-sixth meeting, June 22-24, 1960 the Advisory Committee on Reactor Safeguards again discussed the Jamestown, New York site proposed for a Small Size Pressurized Water Reactor. Members of the Oak Ridge Operations Office and its contractors made a presentation regarding this project at the Special Advisory Committee on Reactor Safeguards meeting in Boston on June 7, 1960. In a letter to you dated March 14, 1960, the Committee presented its conclusions regarding the proposed Jamestown site.

The Committee deplores the tendency on the part of some of those proposing reactor sites to place power reactors containing large quantities of stored energy in or near centers of population at this time to duplicate conditions for conventional power plants for the sake of demonstrating how near a population center such a reactor can be located. We believe that the Jamestown reactor is a case of this kind. We wish to point out that the proximity to a population center will require more rigid specifications of all safety features including containment, leakage rates, power densities, ultimate power, shielding, etc. Thus, it appears that the improved economies of shorter transmission lines may be far outweighed by the increased costs of safety features and more conservative operations. In addition, this reactor will require most stringent surveillance by the AEC during its entire life adding expenses to both the user and the Government.

At the proposed site it appears that any reactor will probably require costly piling foundations, since it is located in a swampy area. Land must be taken in an adjacent township in order to provide the proposed exclusion area. Much safer sites certainly exist in the Jamestown area which are considerably further out.

The Committee believes that any demonstration of economic power is doomed to failure at this site, but might conceivably be improved by movement to some more favorable site.

The Committee can find no serious technical fault with the reactor, the containment, and the safety features proposed, insofar as the partial information supplied to date has presented the case. The Committee emphasizes, however, that power reactors are relatively new and untried, and that there exists a considerable degree of uncertainty in our knowledge of their long-term safe behavior. Accordingly, the Committee doubts that the new and relatively untried technical features for improved safety proposed by the applicant, since our last report, are a satisfactory substitute for the inherent safety implied by a greater distance from population centers.

The Advisory Committee on Reactor Safeguards strongly urges that, as a matter of policy, the Atomic Energy Commission not build this reactor at this site since the reactor cannot safely demonstrate economic nuclear power or anything else here that it could not do more satisfactorily at a better site.

Sincerely yours,

/s/ Leslie Silverman

Leslie Silverman
Chairman

cc: A. R.Luedecke, GM
W.F.Finan, OGM
H.L.Price, DL&R

References:

(1) U.S.Atomic Energy Commission - Oak Ridge, Tennessee; "Small Size Pressurized Water Reactor, Jamestown, New York, Site Report" - dated February 8, 1960 - revised March 11, 1960.
(2) U.S.Atomic Energy Commission - Oak Ridge, Tennessee; "Small Size Pressurized Water Reactor, Jamestown, New York, Site Report" - Supplement No. 2, May 31, 1960.
(3) Small Size Pressurized Water Reactor Meltdown Analysis, ATL-295 Preliminary, May 16, 1960.
(4) Small Pressurized Water Reactor, Jamestown, New York, Site Report - Supplement No. 2- Supplementary Information, June 10, 1960.

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

July 25, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: DAIRYLAND SITE NEAR GENOA, WISCONSIN, FOR THE SMALL PRESSURIZED WATER REACTOR, (SPWR)

Dear Mr. McCone:

At its twenty-seventh meeting in Washington, D. C., on July 20-22, 1960, the Advisory Committee on Reactor Safeguards considered the proposed site for a Small Pressurized Water Reactor to be located in the vicinity of the Dairyland Power Cooperative power plant near Genoa, Wisconsin. The proposed reactor design and hazard analysis have been discussed in our letters dated June 30, 1960, and March 14, 1960, referring to the Small Pressurized Water Reactor, Jamestown, New York site. The only design change suggested by the applicant is the elimination of the recirculating filtration system which was described in the most recent Jamestown report, Reference #1.

It had been added for the purpose of providing additional safety because of the proximity of the site to the City of Jamestown. Because of the greater isolation of the Dairyland site, the designers feel that the filtration system is not necessary there. The Committee agrees that this feature is not essential at this location.

The Committee reviewed the Dairyland site report, and an oral report from its Subcommittee for this site. If the utility is able to obtain control of the proposed exclusion area, it is the opinion of the Committee that this area plus the low population density and generally favorable environmental factors, make this site suitable for a reactor of the type proposed.

Sincerely yours,

/s/Leslie Silverman

Leslie Silverman
Chairman

References:

1. Small Pressurized Water Reactor, Jamestown, New York Site Report, Supplement No. 2, dated May 31, 1960
2. Small Pressurized Water Reactor, Dairyland Power Cooperative, Site Report, dated July 18, 1960
3. Comments on Reactor Siting - Dairyland Power Cooperative, Genoa Site - Oak Ridge Operations Office, dated July 18, 1960

cc: A. R. Luedecke, Gen. Mgr.
W. F. Finan, OGM
H. L. Price, L&R

November 7, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: REPORT ON SMALL PRESSURIZED WATER REACTOR - JAMESTOWN, N. Y.

Dear Mr. McCone:

At its twenty-ninth meeting, November 3-5, 1960, the Advisory Committee on Reactor Safeguards considered two new sites in the vicinity of Jamestown, New York, for the Small Pressurized Water Reactor. Neither of these sites is now under the control of the City of Jamestown, but a preliminary opinion is sought prior to attempts being made to obtain such land. The Committee has previously written two letters dated June 30, 1960 and March 14, 1960, dealing with the problem of locating this reactor at Jamestown, New York. The new proposals are described in the referenced report. This document and discussions with the AEC staff form the basis of this opinion.

Both of the sites are located east of the City of Jamestown. Both will have adequate exclusion radii and low population densities. There are some problems with hydrology which can be satisfactorily resolved by provision of cooling towers and adequate hold-up facilities for liquid wastes. Site #2 appears to have slight but not overriding advantages over Site #1 from the standpoint of hydrology and population density beyond three miles.

The Committee believes that either of these sites is suitable for a reactor of the general type and power level proposed.

Sincerely yours,

Sgd/ LESLIE SILVERMAN

Leslie Silverman
Chairman

Reference:
USAEC - Oak Ridge, Tenn., Small Pressurized Water Reactor, Jamestown, New York, Site Report, Supplement No. 3, dtd October 14, 1960

cc: A. R. Luedecke, GM
W. F. Finan, AGMRS
H. L. Price, Dir., DL&R

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

September 26, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: SODIUM REACTOR EXPERIMENT (SRE)

Dear Mr. McCone:

The Sodium Reactor Experiment has been reviewed by the Advisory Committee on Reactor Safeguards previously and reported in its letters dated April 14, 1955 and September 9, 1955. Because of a required rehabilitation of the facility, it was again considered at the Committee's twenty-eighth meeting on September 22-24, 1960. The ACRS was furnished with the reports listed below. In addition to AEC staff, representatives of Atomics International, the Chicago Operations Office, and the Canoga Park Area Office participated in presentation of data.

The investigation of the loss of fuel elements brought forth recommendations for a complete rehabilitation of the facility, a review of the safety aspects, and complete delineation of required changes in the philosophy of organizational procedures and operational controls. Several major changes have been made; such as, changed fuel element design, substitution of alternate coolants for tetralin, improved pump seals, increased number of in-core and coolant exit thermocouples, improved reactor instrumentation and revised operational organization and control responsibility.

With the changes made and the revised operational controls in effect, the Committee believes the SRE may be operated as proposed without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Leslie Silverman

Leslie Silverman
Chairman

References:

SRE Standard Operating Procedures (NAA-SR-MEMO-5326) dtd June 27, 1960
Organization of the Sodium Reactor Experiment Group (NAA-SR-MEMO-5360) dtd June 3, 1960
Design Modifications to the SRE during FY 1960 (NAA-SR-5348) dtd June 20, 1960
Hazards Summary for the Thorium-Uranium Fuel in the Sodium Reactor Experiment (NAA-SR-3175-Revised) dtd July 1, 1959
Hazards Summary for the Thorium-Uranium Fuel in the Sodium Reactor Experiment (NAA-SR-3175-Rev.- Supplement) dtd April 8, 1960
SRE Fuel Element Damage (NAA-SR-4488) dtd Nov. 30, 1959

cc: A. R. Luedecke, GM
W. F. Finan, AGMRS
H. L. Price, DL&R

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

July 25, 1960

Mr. A. R. Luedecke
General Manager
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: SOUTHERN CALIFORNIA EDISON COMPANY POWER REACTOR SITES

Dear Mr. Luedecke:

In our letter of June 3, 1960, we mentioned arrangements for several members of the Advisory Committee on Reactor Safeguards to visit sites proposed by Southern California Edison Company for location of a nuclear power reactor.

On June 21, 1960, a subcommittee met with representatives of Southern California Edison Company in Los Angeles. Information on three sites was obtained. The Camp Pendleton and Chino sites were inspected.

The report of the subcommittee indicates that the Camp Pendleton site, located on the seacoast midway between the Camp Pendleton Military Reservation boundaries, has the advantage of isolation of approximately four miles from the nearest residence on the reservation and low population density, all subject to military control, for a distance of approximately nine miles from the proposed reactor site. The Chino and Sycamore Canyon sites appear to be somewhat less desirable.

The Advisory Committee on Reactor Safeguards is of the opinion that the Camp Pendleton site is suitable for a properly contained pressurized water reactor of 1150 MWT capacity.

The design of engineering safety features of the proposed reactor should take into account the leak rate of the container versus the unfavorable site meteorology, espectially the episodes of poor atmospheric dilution known to be characteristic of this area. The results of these engineering and environmental studies may lead to the requirement for containment measures in excess of those now used or being designed for large power reactors.

Sincerely yours,

/s/ Leslie Silverman

Leslie Silverman
Chairman

References:
Preliminary Site Examination Report, Southern California Edison Company, Los Angeles, California, undated, received June 14, 1960 by ACRS
Letter from Lt. Gen. W. M. Greene, Chief of Staff, Headquarters, Marine Corps to Chairman, AEC, dated June 30, 1960

cc: W. F. Finan/OGM
H. L. Price/L&R
F. K. Pittman/RD

bc: L. Silverman/Harvard

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

January 16, 1961

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: MULTIPLE REACTOR UNITS (CAMP PENDLETON)

Dear Mr. McCone:

At its thirty-first meeting on January 12-14, 1961, the Advisory Committee on Reactor Safeguards was asked by the Division of Licensing and Regulation to assist in answering the question before the Division regarding multiple nuclear reactors at a given site, and in particular the Camp Pendleton site. The referenced memorandum by Dr. Beck was presented.

The Committee is in agreement with the memorandum that the exclusion area should be substantially more than the 90 acres specifically allocated to the Southern California Edison Company. The Committee is not prepared to advise on any specific values for distances for more than one reactor unit in this general area. Specific proposals would require more data and the opportunity for the Committee to study the data.

In order to give adequate protection to the health and safety of the public against accidental release of radioactive material, the Committee believes the following comments will be helpful:

> It seems reasonable to assume that each of the units can be so located and constructed that an accident to one unit will not precipitate an accident to the others whether initiated by external or internal means.

It is especially important to assure that significant coupling between reactors does not occur. This problem is complex and will require detailed consideration of such items as common facilities, utilities, manpower, maintenance, etc.

Without further knowledge of the exact situation, it is impossible to give more definite answers at this time.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

Reference:
Memorandum, C. K. Beck to A. R. Luedecke, subject, Comments on the Camp Pendleton Site, December 19, 1960

cc: A. R. Luedecke, GM
W. F. Finan, AGMRS
H. L. Price, Dir., DL&R

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 19, 1975

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

Dear Mr. Anders:

Subject: REPORT ON SOUTH TEXAS PROJECT UNITS 1 AND 2

At its 185th meeting, September 11-13, 1975, the Advisory Committee on Reactor Safeguards reviewed the application of Houston Lighting and Power Company, the City Public Service of San Antonio, the Central Power and Light Company and the City of Austin (Applicants) for a permit to construct the South Texas Project Units 1 and 2. The site was visited on August 26, 1975, and the project was considered at a Subcommittee meeting at Bay City, Texas on August 27, 1975. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, Westinghouse Electric Corporation, Brown & Root, Incorporated, and the NRC Staff. The Committee also had the benefit of the documents listed below.

The Plant will be located on the Colorado River in Matagorda County, Texas, approximately 89 miles southwest of Houston and 12 miles south-southwest of Bay City, the designated population center (1970 population, 11,733; projected 2020 population, 24,000). The exclusion area has a minimum boundary distance of 1430 meters. The radius of the low population zone (present population, 55) is three miles. Major land use in the area of the plant site is for the production of rice and cattle.

The South Texas Project will be the first plant to reference the RESAR-41 Westinghouse Standard Design Nuclear Steam Supply System (NSSS). The South Texas Project will be in compliance with the RESAR-41 requirements. The Committee reported on RESAR-41 in its letter of September 18, 1975. Each reactor unit will utilize a four-loop pressurized water nuclear steam supply system having a core power level of 3800 MW(t).

Groundwater at the site area consists of a shallow, low quality aquifer occurring above depths of 90-150 feet and a high quality aquifer commencing at depths in the vicinity of 300 feet. Groundwater usage is almost totally from the deep aquifer. Based upon observations at other areas of similar soil structure, such as the Houston area, continual pumping of ground water from the high quality acquifer is expected to cause subsidence in the vicinity of the plant site. The Applicant has developed design criteria assuming long term settlements, and has committed to the NRC Staff to monitor subsidence at the site over the life of the plant. The Committee believes that the planned actions provide an adequate basis for the safety of the plant structures.

The ultimate heat sink for the plant will be an artificial pond eight feet deep covering over 40 acres. It will be capable of providing the cooling water required for shutdown and maintenance of both reactors in shutdown condition for a minimum of 30 days.

The Committee has reviewed the plans of the Applicant and the NSSS designer to complete the identification and documentation of interface information required by the balance of plant contractor to meet the safety design requirements of the NSSS designer. The Committee believes that this program, when completed in a manner satisfactory to the NRC Staff, will provide an adequate design basis for the balance of plant.

The NRC Staff has identified a number of outstanding issues specific to this application as well as to RESAR-41, some of which will require resolution before the issuance of a construction permit. The Committee recommends that these matters be resolved in a manner satisfactory to the Staff. The Committee wishes to be kept informed on the resolution of the following items:

1. The emergency core cooling system evaluation,

2. Diesel engine building design and location of the storage tanks for the diesel fuel.

The Committee recommends that the NRC Staff and the Applicant review the design features that are intended to prevent the occurrence of damaging fires and to minimize the consequences to safety-related equipment should a fire occur. This matter should be resolved to the satisfaction of the NRC Staff. The Committee wishes to be kept informed.

The NRC Staff is currently reassessing the parameters and mathematical models for calculating releases of radioactive materials in effluents from this plant. Although these calculations include the consideration of additional airborne releases such as carbon-14 and particulates, the Staff does not anticipate that the modifications will result in any substantial increase in the annual population doses previously estimated. The Staff has offered the Applicant the option of including in the South Texas Plant waste management systems meeting the requirements of the earlier proposed Appendix I, 10 CFR 50, or the revised guidance as outlined in the Commission's issuance of April 30, 1975. The revised guidance includes the requirement that cost-benefit analyses be taken into consideration in the determination of waste management needs. The Committee wishes to be kept informed on this matter.

Generic problems relating to large water reactors are discussed in the Committee's report dated March 12, 1975. These problems should be dealt with appropriately by the NRC Staff and the Applicant.

The Committee believes that the above items can be resolved during construction and that if due consideration is given to these items, the South Texas Project 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.

Sincerely yours,

WKerr

W. Kerr
Chairman

References attached.

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

May 20, 1965

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

Subject: REPORT ON SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR (SEFOR)

Dear Dr. Seaborg:

At its sixty-second meeting, March 11-13, 1965, and at its sixty-third meeting, May 13-15, 1965, the Advisory Committee on Reactor Safeguards considered the application of the General Electric Company for a construction permit for the Southwest Experimental Fast Oxide Reactor (SEFOR). The Committee had the benefit of discussions with representatives of the applicant, the AEC Staff and its consultants, and of the documents listed. A preliminary review of the project was held at the Committee's 48th meeting. Members of the Committee visited the site of the proposed reactor on February 19, 1965, and Subcommittee meetings were held on January 14, 1965, and March 4, 1965.

SEFOR will have mixed plutonium oxide - uranium oxide fuel and sodium coolant, and is to be operated at a maximum steady state power level of 20 MW(t). The core volume will be roughly 500 liters; the fuel will contain about 16% plutonium; and approximately 11% by volume of BeO will be added to the core to lower the average neutron energy and provide a Doppler coefficient similar to that of much larger fast reactors.

The principal purpose of operating SEFOR with this core is to study the nuclear characteristics of such a system, particularly the Doppler effect. The planned experimental program includes critical experiments and oscillator measurements at steady power. The intent to perform super-prompt-critical transient tests, after the reactor characteristics are well understood, has been mentioned by the applicant, but a review of transient testing was not requested and has not been made by the Committee.

The proposed reactor site is a 620 acre tract of land owned by the Southwest Atomic Energy Associates and located in Cove Creek Township, Washington County, Arkansas. The site is approximately 16 miles south-southwest of Fayetteville and 29 miles north-northeast of Fort Smith.

As yet no meteorological data exist for the site, but preliminary estimates of atmospheric dispersion have been made, based on the records of surrounding stations. The applicant proposes to make wind velocity and direction measurements on-site in order to obtain meaningful diffusion factors that can be applied to potential release conditions.

The proposed site is in an area of only minor seismic activity, and the SEFOR plant will be designed accordingly. The SEFOR site is located in a region having a relatively high incidence of tornadoes, and the outer containment building and the air blast heat exchangers will be designed to withstand loads resulting from winds up to 300 mph.

The reactor building will provide two containment barriers, an outer cylindrical steel containment vessel, and an inner, reinforced concrete, steel-lined containment. The inner containment will normally contain an oxygen-depleted atmosphere.

A design criterion for the inner containment is that it withstand the energy release and missiles associated with credible reactivity accidents. A few structural details remain to be discussed between the applicant and the Regulatory Staff.

Provisions will be made to prevent re-criticality in the unlikely event of large scale melting and any subsequent motion of the fuel out through the bottom of the reactor vessel.

The primary cooling system of the reactor is comprised of a main and an auxiliary loop. In the event of a major break in a main primary pipe, the sodium level in the reactor vessel would fall rapidly and uncover the main vessel nozzles. The applicant has stated that model experiments will be performed to assure no significant gas entrainment in the sodium of the auxiliary system under these circumstances.

Reactor control will be accomplished by vertical movement of ten reflector segments that are outside the reactor vessel. Design of the control rod system and the fuel element is not complete, and the Committee would like to be informed of the design details when these are available. The General Electric Company has stated that details of the reactor instrumentation will be supplied for review prior to actual fabrication and installation of such équipment.

The shutdown margin is calculated to be 7.4 dollars of reactivity with all control rods inserted. If control rod worth, as measured in ZPR-III critical experiments, is found to be much less than that calculated, General Electric proposes to utilize a considerable reduced shutdown margin, as low as 2 dollars. The Committee does not now agree to the suitability of the smaller shutdown margin and wishes to review this matter, should measured control rod worths prove to be quite low.

The reactor is calculated to have a Doppler coefficient (T dk/dt) of -0.008, and to have a small maximum positive reactivity contribution if centrally preferential sodium voiding occurs. If the reactivity coefficients measured in ZPR-III prove to be considerably different from those calculated, the Committee would like to review the situation.

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

Sincerely yours,

/s/
W. D. Manly
Chairman

References Attached.

References (SEFOR)

1. Preliminary Safeguards Summary Report, Part I, Southwest Experimental Fast Oxide Reactor, undated, received November 5, 1964.
2. Preliminary Survey of the Site of the SEFOR Project, Washington County, Arkansas, prepared by the University of Arkansas, Fayetteville, dated December 16, 1963.
3. Preliminary Safeguards Summary Report, Supplement No. 1, Southwest Experimental Fast Oxide Reactor, undated, received February 24, 1965.
4. Page 3-62, Answer, undated, received February 24, 1965.
5. Supplement No. 2 to License Application for Southwest Experimental Fast Oxide Reactor (SEFOR), dated March 9, 1965.
6. Preliminary Safeguards Summary Report, Supplement No. 3, Southwest Experimental Fast Oxide Reactor, transmitted by Supplement No. 3 to License Application, dated March 29, 1965.
7. Supplement No. 4 to License Application for Southwest Experimental Fast Oxide Reactor, dated April 16, 1965.
8. Report to AEC Regulatory Staff by David B. Hall of the Los Alamos Scientific Laboratory Concerning Evaluation of the Energy Release Due to Large Reactivity Excursions in the Proposed Southwest Experimental Fast Oxide Reactor (SEFOR), dated April 1965.
9. Report to AEC Regulatory Staff, Adequacy of the Structural Criteria for the Proposed Southwest Experimental Fast Oxide Reactor (SEFOR) by N. M. Newmark and W. J. Hall, dated April 1965.
10. Letter dated April 26, 1965 from Coast and Geodetic Survey to AEC Director of Regulation with attached "Report on the Seismicity of the Washington County, Arkansas Area".
11. Geological Survey, "Review of the Hydrology and Geology of the Site for the Southwest Experimental Fast Oxide Reactor (SEFOR) near Fayetteville, Arkansas", dated April 1965.

October 10, 1968

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

Subject: REPORT ON SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR (SEFOR)

Dear Dr. Seaborg:

At its one-hundred-first meeting, September 5-7, 1968, and at its one-hundred-second meeting, October 3-5, 1968, the Advisory Committee on Reactor Safeguards considered the application of the General Electric Company for an operating license for the Southwest Experimental Fast Oxide Reactor (SEFOR). This project was previously discussed by the Committee in its report of May 20, 1965. In its current review, the Committee had the benefit of discussions with representatives of the applicant, the AEC Regulatory Staff, and of the documents listed. A Subcommittee of the ACRS met in June, 1966, January, 1968, and September, 1968, to review plant design and proposed SEFOR operation. The January, 1968, meeting included a visit to the plant site.

SEFOR has been constructed in Washington County, Arkansas on a 620 acre tract of land approximately 16 miles south-southwest of Fayetteville and 29 miles north-northeast of Fort Smith. The reactor employs plutonium oxide-uranium oxide fuel in stainless steel jackets and is sodium-cooled. The reactor building provides two containment barriers, an outer cylindrical steel containment vessel, and an inner, reinforced concrete, steel-lined containment. The inner containment encloses the primary coolant system and will have an oxygen-depleted atmosphere during reactor operation and during certain activities in the refueling cell.

Critical experiments performed in the ZPR-III facility at the National Reactor Testing Station have confirmed control rod worths, and provide an empirical basis for prediction of reactivity coefficients in SEFOR.

The maximum steady state power planned for the first SEFOR core is 20 MWt, although the applicant also plans to perform a limited series of sub-prompt-critical and super-prompt-critical transient tests after the reactor characteristics, based on operation up to 20 MWt, are well understood. Review by the AEC Regulatory Staff and the ACRS is continuing with regard to certain matters related to full power operation and pulsed experiments. The applicant has requested interim approval for reactor operation at powers up to 1 MWt.

The Advisory Committee on Reactor Safeguards believes that the SEFOR reactor can be operated as proposed at power levels up to 1 MWt without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Carroll W. Zabel

Carroll W. Zabel
Chairman

References Attached.

References - Southwest Experimental Fast Oxide Reactor (SEFOR)

1. Letter from General Electric Company, dated July 19, 1965; Amendment No. 3 to License Application for a Construction Permit and Operating License

2. Letter from General Electric Company, dated March 11, 1966; Drawing concerning SEFOR Reflector Drive Control and Position Indication System

3. Letter from General Electric Company, dated March 24, 1966; SEFOR R&D Program

4. Letter from General Electric Company, dated April 25, 1966; Drawing concerning Containment Penetrations for SEFOR

5. Letter from General Electric Company, dated May 4, 1966; Drawing concerning SEFOR Reflector Control Drive Piping

6. Letter from General Electric Company, dated May 9, 1966; Drawing concerning SEFOR Piping and Instrument Symbols

7. Letter from General Electric Company, dated May 9, 1966; Drawing concerning Reactor Safety System

8. Letter from General Electric Company, dated May 25, 1966; Drawings and Specifications concerning SEFOR Safety System and Neutron Monitoring System

9. Letter from General Electric Company, dated May 31, 1966; Drawing concerning Neutron Monitoring System

10. Letter from General Electric Company, dated June 6, 1966; Drawing concerning SEFOR Containment Penetrations

11. Letter from General Electric Company, dated July 21, 1967; Amendment No. 5 to License Application; Including Facility Description and Safety Analysis Report (FDSAR)

12. Amendment No. 4 to License Application, dated July 24, 1967

13. Letter from General Electric Company, dated December 5, 1967; Amendment No. 6 to License Application; Supplements Nos. 1 and 2 to FDSAR

References - SEFOR (cont'd)

14. Letter from General Electric Company, dated December 27, 1967; Amendment No. 7 to License Application; Supplements Nos. 3 and 4 to FDSAR

15. Letter from General Electric Company, dated January 18, 1968; Amendment No. 8 to License Application; Supplement No. 5 to FDSAR

16. Letter from General Electric Company, dated February 14, 1968; Amendment No. 9 to License Application; Supplements Nos. 6 and 7 to FDSAR

17. Letter from General Electric Company, dated February 29, 1968; Amendments Nos. 10 and 11 to License Application; Supplements Nos. 8, 9 and 10 to FDSAR

18. Letter from General Electric Company, dated Marcn 2, 1968; Amendment No. 12 to License Application; Supplement No. 11 to FDSAR

19. Letter from General Electric Company, dated March 11, 1968; Amendment No. 13 to License Application; Supplements Nos. 12, 13, 14 and 15 to FDSAR

20. Letter from General Electric Company, dated April 25, 1968; Amendment No. 14 to License Application; Supplement No. 16 to FDSAR; Errata for FDSAR Volumes I and II, and Supplements

21. Letter from General Electric Company, dated May 17, 1968; Amendment No. 15 to License Application

22. Letter from General Electric Company, dated May 24, 1968; Amendment No. 16 to License Application; Supplement No. 17 to FDSAR

23. Letter from General Electric Company, dated July 18, 1968; Amendment No. 17 to License Application; Errata 2 for FDSAR Volumes I and II, and Supplements

24. Letter from General Electric Company, dated August 6, 1968; Amendment No. 18 to License Application; Supplement No. 18 to FDSAR

25. Letter from General Electric Company, dated September 12, 1968; Final Specifications for the SEFOR Experimental Program

26. Letter from General Electric Company, dated September 25, 1968; Amendment No. 19 to License Application; Supplements Nos. 19 and 20 to FDSAR; Errata 3 for FDSAR Volume I and Supplements

27. Letter from General Electric Company, dated September 26, 1968; Proposed Technical Specifications for SEFOR

28. Letter from General Electric Company, dated September 27, 1968; Amendment No. 20 to License Application

29. Letter from General Electric Company, dated September 30, 1968; Amendment No. 21 to License Application

30. Letter from General Electric Company, dated October 1, 1968; Amendment No. 22 to License Application

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

March 13, 1969

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

Subject: REPORT ON THE SOUTHWEST EXPERIMENTAL FAST OXIDE REACTOR

Dear Dr. Seaborg:

At its 107th meeting, March 6-8, 1969, the Advisory Committee on Reactor Safeguards considered the application of the General Electric Company for a provisional operating license for the Southwest Experimental Fast Oxide Reactor (SEFOR). Operation of this facility at steady-state power levels up to 1 Megawatt was discussed by the Committee in its report of October 10, 1968. A Subcommittee of the ACRS discussed the proposed SEFOR operation on March 5, 1969. In its current review the Committee had the benefit of discussions with representatives of the applicant and the AEC Regulatory Staff, and of the documents listed.

The final design provisions for protection of core integrity against primary system leaks do not meet all the criteria proposed by the applicant during the construction permit review. For example, in the unlikely event of a large primary system leak, siphon breaker action will not be adequate to prevent the core from becoming uncovered temporarily and to prevent interruption of flow in the auxiliary cooling system. Dependence for core cooling in this unlikely event would then be placed upon the ability to replenish the sodium supply in the reactor vessel via the pump-around-loop, and the ability to reprime and initiate cooling via the auxiliary cooling system. Nevertheless, the ACRS believes the provisions for coping with primary system leaks to be acceptable in view of the low steady-state power and power density of the core in the proposed experimental program, and because of the other engineered safety features of the facility. However, for significant changes in power density or level, or for extended operation considerably beyond the experimental program described in the current documentation, the Committee believes that further regulatory review would be appropriate.

Guidelines for determining actions in the event of anomalous behavior have been proposed. Prior to operation at full power (20 MWt), the applicant and the Regulatory Staff should agree on quantitative definitions of suitable limits on unexplained behavior of reactivity, coolant temperatures, and other parameters of significance.

The primary system does not currently include a sampling station that permits routine monitoring of the sodium for fission product or other impurities of significance. However, the applicant stated that he will attempt to provide, at an early date, equipment that would enable the operators to obtain samples much more frequently than is now practical. Because of the potential usefulness of relatively frequent sampling in characterizing impurities in the sodium and in diagnosing and understanding the probable status of fuel element defects which may arise during reactor operation, the Committee recommends that such equipment be made available prior to the start of 20 MWt operation.

Before operation at full power, the applicant and the Regulatory Staff should also agree on the necessary detection steps and on specific criteria, and the bases thereof, for judging the acceptability of continued steady-state operation in the presence of a known loss of fuel clad integrity.

The ACRS believes that, if due attention is given to the foregoing comments and if experience in the stepwise experimental program is favorable, the SEFOR reactor can be operated at steady-state powers up to 20 MWt and in the pulsed mode, as proposed, without undue risk to the health and safety of the public.

Sincerely yours,

/s/
Stephen H. Hanauer
Chairman

References attached.

References - Southwest Experimental Fast Oxide Reactor - (SEFOR)

1. Letter from General Electric Company, dated October 9, 1968; Amendment No. 23 to License Application

2. Letter from General Electric Company, dated November 1, 1968; Amendment No. 24 to License Application

3. Letter from General Electric Company, dated December 10, 1968; Amendment No. 25 to License Application; Supplement No. 21 and Errata 4 to FDSAR

4. Letter from General Electric Company, dated January 14, 1969; Amendment No. 26 to License Application

5. Letter from General Electric Company, dated January 24, 1969; Amendment No. 27 to License Application; Supplement No. 22 and Errata 5 to FDSAR

6. Letter from General Electric Company, dated January 31, 1969; Proposed Technical Specifications for 20 MWt Operation of SEFOR

7. Letter from General Electric Company, dated February 20, 1969; Amendment No. 28 to License Application; Supplement No. 23 and Errata 6 to FDSAR

8. Letter from General Electric Company, dated February 28, 1969

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C.

October 21, 1958

Honorable John A. McCone
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: SPERT-III REACTOR

Dear Mr. McCone:

The operation of the SPERT-III reactor was reviewed by the Advisory Committee on Reactor Safeguards on October 15, 1958, at the request of the Division of Licensing and Regulation. The Committee considered both the report of the Division of Licensing and Regulation and the report from the Phillips Petroleum Company, IDO-16425.

SPERT-III is a light water moderated and cooled experimental reactor designed to operate intermittently at pressures up to 2000 psi and temperatures up to 670°F. It is designed for transient studies of pressurized water reactors operating under full power conditions. The SPERT-III program is an extension of the reactor kinetic studies which have been carried out successfully under the SPERT-I program.

The nature of the SPERT-III experimental program is such that this reactor will be operated closer to failure conditions than normal power reactors. The safe performance of these experiments rests largely on the competence of the operating staff. The Advisory Committee on Reactor Safeguards believes that the SPERT staff by performance has shown that it can handle this type of assignment. The Committee commends, and agrees with, the SPERT proposal to avoid running the more hazardous experiments during times of adverse meteorological conditions.

The Advisory Committee on Reactor Safeguards concurs with the Division of Licensing and Regulation that SPERT-III can be operated without undue hazard to the health and safety of the public.

The Advisory Committee on Reactor Safeguards wishes to comment again that the SPERT program is contributing extremely worthwhile information which is basic to many of the safety problems of the entire atomic reactor industry.

Dr. Richard L. Doan excused himself from participation in the discussion and recommendation in this case.

Sincerely yours,

/s/

C. Rogers McCullough
Chairman

cc: P. F. Foster, GM
H. L. Price, DL&R

Dr. Wolman) not at meeting
Dr. Brooks)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
United States Atomic Energy Commission
Washington 25, 1959

July 25, 1959

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: SPERT-II REACTOR

Dear Mr. McCone:

At its Seventeenth Meeting, July 23-25, 1959, the Advisory Committee on Reactor Safeguards reviewed the design of the SPERT-II Reactor. In addition to the information referenced below, a summary of SPERT-I test results was presented by a representative of the National Reactor Testing Station.

SPERT-II is an experimental heterogeneous, water cooled and moderated reactor designed for operation at a pressure of 375 psig at 400°F. The facility may use either light or heavy water coolant or moderator. SPERT-II will be used as a transient test facility and will involve reactivity insertions above prompt critical.

Due to the nature of the test program SPERT-II will be operated closer to failure conditions than conventional power reactors. Safe operation with this type of program depends largely on the competence of the operating personnel. In the opinion of the Advisory Committee on Reactor Safeguards the SPERT staff has demonstrated its competence in this regard by satisfactory performance of similar experiments on SPERT-I.

The SPERT staff has proposed that experiments of even moderate hazard be suspended under adverse weather conditions so that exposure of persons off site to harmful doses of radiation is extremely unlikely. The Committee concurs in this proposal.

The Advisory Committee on Reactor Safeguards concludes that the SPERT-II reactor can be operated without undue hazard to the health and safety of the public. The Committee notes that information being developed from the SPERT tests is extremely valuable to the safe design of reactors throughout the reactor program.

Dr. Richard L. Doan did not participate in the discussion and recommentation in this case.

Sincerely yours,

C. Rogers McCullough
Chairman

cc: A.R.Luedecke, GM
H.L.Price, DL&R

References:

1) IDO-16491 - SPERT-II Hazards Summary Report, February 17, 1959.

2) Division of Licensing and Regulation Report to the ACRS on the SPERT-II Reactor, April 28, 1959.

3) Comments on Meteorological aspects of SPERT-II Hazards Summary Report prepared by Special Projects Section, Office of Meteorological Research, U. S. Weather Bureau, May 4, 1959.

4) Office Memorandum from G. Victor Beard, Division of Biology and Medicine on SPERT-II, May 12, 1959.

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

August 30, 1962

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

Subject: REPORT ON SPERT I DESTRUCTIVE TESTS

Dear Dr. Seaborg:

At its forty-third meeting on August 23-25, 1962, at Idaho Falls, Idaho, the Advisory Committee on Reactor Safeguards reviewed the proposed destructive tests to be performed with the Spert I reactor facility. The review during this meeting was preceded by a subcommittee review at the site on August 22, 1962. During both the subcommittee meeting and the ACRS meeting referred to, the Committee had the benefit of discussions with the Phillips Petroleum Company personnel, personnel from the Idaho Operations Office of the AEC, who are to conduct environmental monitoring aspects of the tests, and other AEC staff. The hazards aspects of the Spert I destructive tests are covered in the documents referred to below.

The fission product burden of the Spert I reactor during the proposed tests will be very low. The isolation provided by the Spert site is adequate. The group conducting the tests is qualified and experienced. It is therefore the opinion of the Advisory Committee on Reactor Safeguards that the Spert I destructive tests can be performed as proposed without undue hazard to the health and safety of the public. The Committee believes, however, that the number of observers to be posted at the control center and the Spert IV sites during the tests should be held to an absolute minimum.

Sincerely yours,

/s/

F. A. Gifford, Jr.
Chairman

References attached

The SNUPPS will utilize the RESAR-3 Consolidated Version, four-loop pressurized water nuclear reactor with a core power output of 3411 MW(t). This design is similar to that utilized at the Comanche Peak Steam Electric Station, Units 1 and 2, reported on by the Committee in its letter of October 18, 1974. The Committee's continuing review of the SNUPPS was reported on in its Callaway letter of September 17, 1975, and is further reported on in this letter. It is anticipated that the Committee's report on the remainder of its review of SNUPPS will be included in its report on the Tyrone application.

The NRC Staff has identified several items in its review of the Sterling application which are not yet completed. The Committee recommends that any outstanding issues which may develop in the course of completing these reviews be dealt with in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed on the resolution of the following items:

1. The emergency core cooling system evaluation in compliance with the Final Acceptance Criteria.

2. The analyses of the effects of anticipated transients without scram.

3. The evaluation of the plant design to meet the requirements of the new Appendix I of 10 CFR Part 50.

The RESAR-3 Consolidated Version nuclear design utilizes the Westinghouse 17x17 fuel assembly. Westinghouse has identified an integrated test program to confirm the safety margins associated with this design, which it plans to complete late this year. The RESAR-3 reactor core design has been calculated by Westinghouse to be stable against radial xenon oscillations. Westinghouse has agreed to verify this stability in a startup physics test for a 193 fuel assembly core similar to SNUPPS. The Committee will continue to review these matters as appropriate documentation is submitted.

The Committee recommended in its report of September 10, 1973, on acceptance criteria for ECCS, that significantly improved ECCS capability should be provided for reactors for which construction permit requests are filed after January 7, 1972. The SNUPPS design is in this category. These units will use the 17x17 fuel assemblies similar to those to be used in Comanche Peak Steam Electric Station, Units 1 and 2. Although calculated peak clad temperatures in the event of a postulated LOCA are less for 17x17 assemblies than for a 15x15 array, the Committee believes that the applicant should continue studies that are responsive to the Committee's September 10, 1973 report. If studies establish that significant further ECCS improvements can be achieved, consideration should be given to incorporating them into this unit.

The part of the exclusion zone which extends into Lake Ontario, including the points of intake and discharge of emergency service cooling water, will be under control of the United States Coast Guard. The Committee recommends that the NRC Staff and the applicant give particular attention to assure proper coordination between the applicant and the Coast Guard appropriate to protection of the emergency equipment.

The Committee believes that the applicant and the NRC Staff should continue to review the Sterling plant design for features that could reduce the possibility and consequences of sabotage.

The Committee recommends that the NRC Staff and the applicant review the design features that are intended to prevent the occurrence of damaging fires and to minimize the consequences to safety-related equipment should a fire occur. This matter should be resolved to the satisfaction of the NRC Staff. The Committee wishes to be kept informed.

Generic problems relating to large water reactors are discussed in the Committee's report dated March 12, 1975. These problems should be dealt with appropriately by the NRC Staff and the applicant.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above and the items mentioned in its Callaway letter, which are relevant to the Sterling application, can be resolved during construction and that if due consideration is given to the foregoing, the Sterling Power Project Nuclear Unit No. 1 can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

WKerr

W. Kerr
Chairman

References - Spert I

1. IDO-16790 - Spert I Destructive Test Program, Safety Analysis Report, dated June 15, 1962 (Official Use Only).

2. Letter from Argonne National Laboratory to U. S. Atomic Energy Commission, dated August 3, 1962.

3. Memo from F. K. Pittman, Atomic Energy Commission to R. Lowenstein, Atomic Energy Commission, dated August 15, 1962.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 16, 1975

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

Subject: REPORT ON THE STERLING POWER PROJECT NUCLEAR UNIT 1

Dear Mr. Anders:

During its 186th meeting, October 9-11, 1975, the Advisory Committee on Reactor Safeguards reviewed the application of Rochester Gas and Electric Corporation for a permit to construct the Sterling Power Project, Unit No. 1. On September 24, 1975, the site was visited and a Subcommittee meeting was held in Sterling, New York to review site-related matters. The "Standardized Nuclear Unit Power Plant System" (SNUPPS) to be utilized at the Sterling site, and at three other plant sites, was reviewed at Subcommittee meetings held at Washington, D. C. on August 19, 1975, and at Emporia, Kansas on September 26, 1975, and at the 185th and 186th meetings of the Committee. During its reviews, the Committee had the benefit of discussions with the Nuclear Regulatory Commission (NRC) Staff and representatives of the applicant, the Westinghouse Electric Corporation and the Bechtel Corporation. The Committee also had the benefit of the documents listed below.

The Sterling unit will be located on a 2800-acre site of partially wooded rural land located on the southeastern shore of Lake Ontario, approximately 7 miles southwest of Oswego, New York the nearest population center (1970 population: 23,844). The minimum exclusion area boundary distance from the center of the reactor building is 1190 meters. Part of the exclusion area extends into Lake Ontario. In the event the applicant is unable to gain control over those three acres of shore land within the exclusion zone which he does not now own, the minimum exclusion area boundary distance will be reduced to 945 meters. NRC Staff calculations indicate that the applicant can meet the siting dose guidelines at this reduced distance without additional engineered safety features.

REFERENCES

1. SNUPPS Preliminary Safety Analysis Report with Revisions 1 through 10 and the Sterling Site Addendum Report with Revisions 1 through 11.

2. RESAR-3 Consolidated Version, Westinghouse Reference Safety Analysis Report with Amendments 1 through 6.

3. Safety Evaluation Report, NUREG 75/082 related to the Construction of the Sterling Power Project, Nuclear Unit No. 1, Docket No. STN 50-485, September, 1975.

4. Resolution by the Town of Sterling Town Board, dated May 12, 1975.

5. Letter dated September 17, 1975, from Ms. Sue Reinert, Ecology Action of Oswego.

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

November 15, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

Dear Dr. Schlesinger:

At its 151st meeting, November 9-11, 1972, the Advisory Committee on Reactor Safeguards completed its review of the application of the South Carolina Electric and Gas Company for a permit to construct the Virgil C. Summer Nuclear Station, Unit 1. This project had been considered previously at the Committee's 149th meeting, September 14-16, 1972, at the Special Meeting of the Committee, October 26-28, 1972, and at Subcommittee meetings held near the plant site on September 8, 1972, and in Washington, D. C. on October 6, 1972. During its review, the Committee had the benefit of discussions with representatives of the South Carolina Electric and Gas Company, the Westinghouse Electric Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed below.

The plant will be located in a sparsely populated region in Fairfield County, South Carolina, about 26 miles northwest of Columbia (population, 113,000). The minimum distance to the exclusion area boundary is 5347 ft (1630 m) and the low population zone radius has been selected to be three miles.

The plant will employ a 3-loop Westinghouse pressurized water reactor, to be operated at power levels up to 2775 MW(t). The nuclear steam supply system, including the reactor, is similar to other 3-loop Westinghouse systems on which the Committee has reported recently, but with slightly higher reactor power, coolant flow rate, and coolant temperature. The Committee believes that appropriate additional evidence regarding core behavior will be obtained from reactors of similar design prior to operation of the plant.

The plant will be constructed adjacent to a reservoir of approximately 6,800 acres (Lake Monticello) which will be created by constructing a series of earthen dams across Frees Creek. Lake Monticello will store water for a pumped storage facility and provide cooling water for the nuclear plant. A service water pond constructed within the reservoir is impounded by Seismic Category I dams and will function as an ultimate shutdown or post-LOCA heat sink in the event of loss of water from the reservoir. The Committee believes that particular attention should be given to the design of the service water pond dams to assure an adequate ultimate heat sink in the event that an earthquake leads to loss of Lake Monticello.

The applicant plans to design the Virgil C. Summer Station to withstand a bedrock acceleration of 0.15 g for the Safe Shutdown Earthquake (SSE) and an acceleration of 0.10 g for the Operating Basis Earthquake (OBE). The Committee finds these accelerations acceptable for this plant.

In order to satisfy requirements with regard to efficacy of the emergency core cooling systems for this reactor, the applicant proposes to limit the maximum permissible linear power by reducing peaking factors. The applicant described an experimental and analytical program intended to provide improved understanding of phenomena entering into the loss-of-coolant accident, and which can provide the basis for developing improvements in ECCS design. The Committee believes it important that improvements in ECCS effectiveness be included in the Summer plant, and recommends that the final design of the Summer ECCS be reviewed by the Regulatory Staff and the ACRS prior to fabrication and installation of major components.

Although the applicant does not propose to install a fixed in-core flux monitoring system, he stated that it would be possible to install such a system; the Committee believes this capability should be retained.

The turbine-generator for this plant is so arranged that in the unlikely event of a turbine failure, missiles could be generated which might damage the reactor building or other key structures related to safety of the plant. The Committee recommends that a study be made of the probability of unacceptable consequences arising from potential turbine missiles and of the possible need for protective measures if this probability should be unacceptably high. In addition, the Committee believes that analytical and experimental work on the penetration of reinforced concrete by missiles of the type of interest is desirable to provide a suitable basis for establishing the probability of penetration of thick-walled concrete structures and damage to safety-related components.

The Committee believes that protection against pipe whip should be provided by the applicant in accordance with criteria being developed by the AEC Regulatory Staff.

The applicant has proposed criteria for means to mitigate the consequences of a possible main steamline rupture inside the auxiliary building. This matter should be resolved to the satisfaction of the Regulatory Staff; the Committee wishes to be kept informed.

The fuel rod problem involving densification and subsequent movement of the fuel pellets is undergoing intensive investigation. The Regulatory Staff and the ACRS should review the resolution of this matter as it relates to the Summer plant.

Further studies are in progress with regard to the effects of a failure to scram on anticipated transients and of design features which would make tolerable the results of such an event. These studies should be expedited and the matter resolved during construction in a manner satisfactory to the Regulatory Staff and the ACRS.

Other problems related to large water-cooled and moderated 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 Virgil C. Summer Nuclear Station.

The Committee, while noting that the applicant refers to programs for verification of design and equipment capability, believes that a more active participation by the applicant would expedite the resolution of generic items relating to safety of light-water reactors.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items, the Virgil C. Summer Nuclear Station, Unit 1 can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Additional remarks by Dr. David Okrent are presented below.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

References Attached.

Additional Remarks by D. Okrent

Significant uncertainties exist concerning the probable cause of the major 1886 Charleston earthquake, and the ACRS has received conflicting opinions regarding the probability that a major related earthquake might occur closer to the applicant's site. Several geologic and seismic experts recommended that use of a higher Safe Shutdown Earthquake bedrock acceleration of 0.2 g would be prudent unless the applicant can confirm by field studies his theory that the Yamacraw ridge is a fault responsible for the 1886 Charleston earthquake and that there are no structures which might lead to extension of the Charleston earthquake activity toward the site, or confirm the existence of some other source of the Charleston earthquake which permits the same conclusion with regard to the Summer site. I agree with these recommendations.

References

1. South Carolina Electric and Gas Company letter dated June 30, 1971; Application for Licenses; Preliminary Safety Analysis Report (PSAR), Volumes I through VI

2. Amendments 1, 3-14, 16 and 17 to PSAR

3. Safety Evaluation by Directorate of Licensing, dated August 29, 1972

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 18, 1981

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON VIRGIL C. SUMMER NUCLEAR STATION UNIT 1

Dear Dr. Hendrie:

During its 251st meeting, March 12-14, 1981, the ACRS completed its review of the application of the South Carolina Electric and Gas Company for a license to operate the Virgil C. Summer Nuclear Station Unit 1. This project was considered at subcommittee meetings on February 26-27, 1981 in Columbia, South Carolina, and on March 11, 1981 in Washington, D.C. A tour of the facility was made by members of the Subcommittee on February 26, 1981. During its review the Committee had the benefit of discussions with representatives of the Applicant, the NRC Staff, the U.S. Geological Survey, and of the documents listed. The Committee reported on the construction permit application for this plant in a letter to AEC Chairman Schlesinger dated November 15, 1972.

The Summer plant is located in Fairfield County, South Carolina, about 26 miles northwest of Columbia, South Carolina. The nearest community with more than 1000 residents is Winnshore, about 15 miles to the northeast. The plant is adjacent to the Monticello reservoir, which provides cooling water for the main condenser, as well as the ultimate heat sink.

The Summer plant employs a Westinghouse, three-loop, pressurized water, nuclear steam supply system. The containment is a cylindrical, carbon-steel-lined, prestressed concrete structure having a design pressure of 57 psig.

At the construction permit review stage, some of the ACRS consultants were reluctant to accept the position of the Regulatory Staff and its consultants that the 1886 Charleston earthquake could be clearly localized in the Charleston area with regard to recurrence and recommended that a somewhat increased seismic design basis be employed. The ACRS supported the Regulatory Staff position favoring a safe shutdown earthquake (SSE) acceleration of 0.15g. However, in separate reports to the AEC dated May 13, 1971 and May 16, 1973, the ACRS urged initiation of a seismic research program intended to provide a better understanding of the likely causes of earthquakes near Charleston as well as several other areas in the eastern United States. Considerable research has since been undertaken in the Charleston area, and an improved understanding of the possible causes of earthquakes in the eastern United States has been developed. However, there still exists more than one theory with regard to the source of the 1886 Charleston earthquake.

Since the construction permit stage, a new issue has arisen with regard to the choice of seismic design basis; namely, the potential for a moderate earthquake at the site resulting from reservoir-induced seismicity. The Applicant has studied seismic activity in the vicinity of the Monticello reservoir since it was filled in 1977, and combined the results of those studies with information about the local geology and hydrology in arriving at the conclusion that a maximum near-field earthquake magnitude of 4.0 should be considered in evaluating plant safety. The NRC Staff and its consultants have concluded that a near-field magnitude of 4.5 should be used. However, one member of the NRC Staff disagrees with the majority Staff position, suggesting that the available information does not rule out a somewhat larger reservoir-induced earthquake, and that a near-field earthquake having a magnitude of 5.0 to 5.3 should be used for assessing seismic safety.

The ACRS consultants agree that there does not exist a very good basis for choosing a specific near-field event, and generally support the use of a near-field magnitude of about five for evaluation of the plant.

Because it is difficult to judge that the probability of significant exceedence of the original SSE is sufficiently small, the ACRS has requested, and the Applicant has provided, information that indicates there is sufficient margin in the original design to cope safely with accelerations considerably larger than the SSE of 0.15g, including those which might arise from a near-field, magnitude 5 earthquake.

The Applicant's results to date regarding seismic design margin are reassuring. The ACRS recommends that these studies by the Applicant be extended to include all systems and components whose function is important to the assurance of the continuing removal of shutdown heat. Such studies need not be completed prior to operation of the Summer plant.

The discussions relative to the seismic issues at the Summer Nuclear Power Station raise certain questions that we believe should be addressed. These questions, which largely pertain to emergency preparedness, include the ability of certain key systems to function after a major seismic event. Included among such systems are the emergency alarm features to alert the public to an accident in the plant, meteorological and field radiation monitoring networks, communications, and emergency evacuation routes.

As a result of the continuing microseismic activity induced by the reservoir, the Applicant has, at NRC request, agreed to continue seismic monitoring for at least the next two years. We recommend that the NRC Staff assure that the monitoring program is not halted prematurely.

In its review of the Applicant's organization and management, the NRC Staff has identified several areas requiring attention, including the size of the engineering organization and the adequacy of experience with nuclear power reactors within the company, including hands-on operating experience within the operating organization. The Applicant has taken steps to obtain the services of outside groups to provide additional technical capability for the short term while the needed in-house capability is developed. Care should be exercised that, as part of this effort, sufficient technical breadth and independence exists among the members of the Nuclear Safety Review Committee for the plant.

We have previously recommended that probabilistic safety analyses be performed for all plants in operation or under construction. We believe that this recommendation is applicable to this unit, but that such studies need not be performed prior to licensing of the plant.

During construction of the essential service water intake structure and pump house, settlement well beyond that predicted was experienced. While the settlement of the structures appears to have halted, the NRC Staff is still evaluating information addressing the stability of the subsurface materials and foundations of the intake structure and pumphouse. This matter should be resolved in a manner satisfactory to the NRC Staff.

The ACRS believes that, if due consideration is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the Virgil C. Summer Nuclear Station Unit 1 can be operated at power levels up to 2775 MWt without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. South Carolina Electric and Gas Company, "Final Safety Analysis Report, Virgil C. Summer Nuclear Station," Volumes I-XX and Amendments 1-22
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Virgil C. Summer Nuclear Station, Unit No. 1," USNRC Report NUREG-0717, dated February, 1981
3. Letter from J. Devine, USGS, to R. Jackson, NRC, in response to an NRC request for update on USGS information concerning occurrence of earthquakes similar to the 1886 Charleston event, dated December 30, 1980
4. Memorandum from A. Murphy, Site Safety Research Branch, NRC, to R. Jackson, Chief, Geosciences Branch, NRC, Subject: Recommendation of Maximum Reservoir-Induced Earthquake at the V. C. Summer Nuclear Station, dated February 6, 1980
5. "Comments from the Palmetto Alliance, Inc., by Michael Lowe on V. C. Summer Operating License Application Review by the NRC Advisory Committee on Reactor Safeguards," dated February 26, 1981
6. "Testimony Before the Advisory Committee on Reactor Safeguards Related to the Virgil C. Summer Nuclear Station," Ms. Ruth Thomas, received February 26, 1981

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,

WKerr

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 16, 1977

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON PARTIAL REVIEW OF THE SITE FOR THE SUNDESERT NUCLEAR POWER PLANT, UNITS 1 AND 2

Dear Mr. Rowden:

At its 203rd meeting, March 10-12, 1977, the Advisory Committee on Reactor Safeguards completed a partial review of the suitability of a site on which the San Diego Gas and Electric Company (Applicant) proposes to construct the Sundesert Nuclear Power Plant, Units 1 and 2. The site was visited on February 18, 1977, and a Subcommittee meeting was held in Blythe, California, on the same day. During its review of the Sundesert site, the Committee had the benefit of discussions with representatives of the Applicant and its consultants, Fugro, Inc., and Stone and Webster Engineering Corporation, and with the staffs of the Nuclear Regulatory Commission and the United States Geological Survey. The Committee also had the benefit of the documents listed.

The Sundesert site is located in Riverside County, California, about 16 miles southwest of Blythe, California, 2.5 miles west of Palo Verde, California, and about 6 miles from the Colorado River (the California-Arizona boundary). The minimum exclusion distance is 3200 feet; the low population zone (LPZ) radius is 3 miles. The nearest population center is Yuma, Arizona, which is located approximately 50 miles south-southeast of the site and had a 1970 population of 29,007. The 1970 population for the LPZ was reported to be 16; the population actually located within 50 miles was reported as 27,867. Population projections through the year 2020 do not indicate any population centers within 50 miles of the site other than Yuma.

The site is located in an arid region on the mesa adjacent to the Colorado River flood plain. The maximum calculated flood, which is based on the assumption that Hoover Dam fails, is expected to produce water levels no higher than 63 feet below the plant grade. Surface runoff from local intense rain storms will be controlled by diversion of water to dry washes north and south of the plant.

Plant cooling water will be supplied from the Palo Verde Irrigation District at the rate of 17,000 acre-ft. per year per reactor. Contracts to obtain the water have been signed with the Metropolitan Water District of Southern California for Unit No. 1, and water from farm land owned by the Applicant will be diverted for use in Unit No. 2. Blowdown from the plant's cooling towers will go to evaporation ponds.

The Applicant, the NRC Staff, and the USGS have agreed that horizontal ground accelerations of .35g and .175g at the site are appropriate design values for the safe shutdown earthquake (SSE) and the operating basis earthquake, respectively. The vertical accelerations are taken to be 2/3 of the horizontal. The SSE value was based on a postulated 8.5 magnitude (Richter) earthquake on the Sand Hills Fault, a branch of the San Andreas, at a distance of 35 miles. The SSE value also bounds random events, of up to magnitude 5.0, that were postulated at a distance of 5 miles from the site.

The NRC Staff has underway a program of review and reevaluation of several generic matters related to soil-structure interaction and the appropriate response spectrum for use at foundation levels of nuclear power plants. Completion of this reevaluation may result in some change in the development of the appropriate design response. The Committee believes this matter can be resolved prior to completion of the review of a construction permit for use at this site.

Nearby industrial, transportation, and military facilities were evaluated to determine their impacts upon the site. The only potential hazard to the site results from aircraft flights in the area. The Applicant has submitted analyses which conclude that the risk of aircraft impact from present traffic is acceptably low. In addition, by agreement between the NRC and Department of Defense, a directive exists requiring that military training fights be moved further from the site prior to reactor operation.

The NRC Staff has yet to review the ultimate heat sink. In addition, the NRC Staff has identified several items which will require verification during the detailed review of the Preliminary Safety Analysis Report. These items, based upon experience with other plant designs, do not preclude the use of the Sundesert site. The NRC Staff, subject to the Applicant establishing the requirements for the ultimate heat sink, has concluded that the Sundesert site is acceptable under the guidelines of 10 CFR, Part 100 for the construction and operation of a nuclear power plant of the type and size proposed.

The Committee, recognizing that the scope of this review was limited to several site related items, and that complete review of the PSAR will be finished prior to issuance of a construction permit, agrees that the site is acceptable under the guidelines of 10 CFR, Part 100.

Sincerely yours,

M. Bender

M. Bender
Chairman

REFERENCES:

1. San Diego Gas and Electric Company: "Sundesert Nuclear Power Plant, Early Site Review Report" (April 1975) with Amendments 1 through 12.
2. U. S. Nuclear Regulatory Commission: "Early Site Review Report By the Office of Nuclear Reactor Regulation, In the Matter of San Diego Gas and Electric Company Sundesert Site, Project No. 558," NUREG-0171, February 10, 1977.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 6, 1977

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

SUBJECT: ACRS REPORT ON SUNDESERT SITE, DATED MARCH 16, 1977

During the 205th ACRS meeting, the Committee considered the request from B. C. Rusche of April 8, 1977, for clarification of the subject ACRS report. The members discussed the bases for the Committee's report on the Sundesert Site and the comments noted below are reflected in the meeting minutes.

(1) The Committee believes that the horizontal ground acceleration of 0.35g for the SSE and 0.175g for the OBE are acceptable.

(2) The Committee believes that the results of the NRC Staff program of review and reevaluation of the several generic matters related to soil-structure interaction should be considered in the review of the Sundesert Units for a construction permit.

R F Fraley
R. F. Fraley
Executive Director

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

April 29, 1968

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

Subject: REPORT ON SURRY POWER STATION UNITS 1 AND 2

Dear Dr. Seaborg:

At its ninety-sixth meeting, on April 4-6, 1968, the Advisory Committee on Reactor Safeguards completed a review of the application by the Virginia Electric and Power Company for authorization to construct two nuclear units at its Surry Power Station in Surry County, Virginia. This project had previously been considered at Subcommittee meetings at the site on September 5, 1967 and in Washington, D. C. on March 26, 1968. During its review, the Committee had the benefit of discussions with representatives of the Virginia Electric and Power Company and their consultants, the Westinghouse Electric Corporation, the Stone and Webster Company and the AEC Regulatory Staff and their consultants. The Committee also had the benefit of discussions with its own consultants and of the documents listed.

The Surry Station site comprises approximately 840 acres, located on a small peninsula which juts into the James River, 4.7 miles northwest of the nearest corporate limit of Newport News, Virginia. Newport News has a population of approximately 114,000, located from ten to twenty miles southeast of the site. Williamsburg, Virginia, a major tourist attraction, is located seven miles north of the site. The region surrounding the site is rural and agricultural.

Surface deposits at the site consist of layers of sand, silts and clays ranging in thickness from approximately 50 to 80 feet. Below this are Miocene, Eocene, Paleocene and Cretaceous sediments extending to bedrock, about 1300 feet below grade. The reactor buildings are to be founded on ten foot thick, reinforced concrete mats, supported on the Miocene deposits, approximately 70 feet below the surface. The fuel building, between the reactor buildings, is supported on concrete-filled piles driven into the Miocene deposits. The auxiliary building and control room area are supported on four foot thick, reinforced concrete mats, about 36 feet below grade.

The Surry Station units are to be identical, three-loop, pressurized water reactors operated at maximum power levels of 2441 MWt. With respect to core design and other features of the nuclear steam supply system, the reactors are similar to the Diablo Canyon reactor. The units have a power level and average heat flux about 16% higher than the H. B. Robinson reactor with a power density a little less than that of the Diablo Canyon reactor.

Each of the primary system loops is equipped with two valves to enable isolation of the pumps and steam generators for purposes of maintenance. Further consideration of the instrumentation and administrative procedures proposed for protection against potential reactivity transients initiated by the introduction of cold and/or unborated water into the core from a previously isolated loop may be appropriate at the operating license review stage.

In connection with postulated loss-of-coolant accidents, the applicant stated that, using conservative assumptions and allowing appropriately for fuel element distortion from the original core geometry, the emergency core cooling systems will be designed to keep fuel-clad temperatures below the point at which the clad may disintegrate upon subsequent cooling.

Each reactor and its steam generators are enclosed in a steel-lined reinforced concrete containment structure of 45 psig design pressure. A routine operating pressure of 10.0 $\pm$ 0.5 psia is maintained with vacuum pumps. The applicant has stated that either of the two containment spray subsystems, employing chilled, slightly alkaline water, together with two of the four containment recirculation spray subsystems will return the containment to subatmospheric pressure within 40 minutes in the unlikely event of a loss-of-coolant accident.

The applicant has stated that protection will be afforded against the maximum wave runup expected during hurricanes in the vicinity of the station.

The applicant has proposed using signals from certain protection instruments for control purposes. The Committee continues to believe that control and protection instrumentation should be separated to the fullest extent practicable. The Committee believes that the proposed protection system can and should be modified to eliminate or reduce to a minimum the interconnection of control and protection instrumentation. The modified system should be reviewed by the AEC Regulatory Staff.

The Committee continues to call attention to matters that warrant careful consideration with regard to recent reactors of high power density and other matters of significance for all large, water-cooled, power reactors. These matters, stated in our report to you of December 20, 1967 on Diablo Canyon, apply similarly to Surry Station Units 1 and 2.

The Committee believes that, if due consideration is given to the foregoing items, the nuclear units proposed for the Surry Station site 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/

Carroll W. Zabel
Chairman

References:

1. Letter from Hunton, Williams, Gay, Powell & Gibson, dated March 20, 1967; Surry Power Station Units 1 and 2 License Application, Part A; Part B, Preliminary Safety Analysis Report, Vols. I, II, and III.

2. Letter from Virginia Electric and Power Company, dated June 21, 1967; Amendment No. 1 to License Application.

3. Letter from Virginia Electric and Power Company, dated July 5, 1967; Amendment No. 2 to License Application.

4. Letter from Virginia Electric and Power Company, dated August 24, 1967; Amendment No. 3 to License Application.

5. Letter from Virginia Electric and Power Company, dated October 6, 1967; Amendment No. 4 to License Application.

6. Letter from Virginia Electric and Power Company, dated December 7, 1967; Amendment No. 5 to License Application.

7. Letter from Virginia Electric and Power Company, dated December 8, 1967; Amendment No. 6 to License Application.

8. Letter from Virginia Electric and Power Company, dated January 4, 1968; Amendment No. 7 to License Application.

9. Letter from Virginia Electric and Power Company, dated January 19, 1968; Amendment No. 8 to License Application.

10. Letter from Virginia Electric and Power Company, dated February 14, 1968; Amendment No. 9 to License Application.

11. Letter from Virginia Electric and Power Company, dated March 1, 1968; Amendment No. 10 to License Application.

12. Letter from Virginia Electric and Power Company, dated March 18, 1968; Amendment No. 11 to License Application.

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

December 17, 1971

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON SURRY POWER STATION UNITS 1 AND 2

Dear Dr. Schlesinger:

At its 140th meeting, December 9-11, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application of Virginia Electric and Power Company for authorization to operate Surry Generating Units 1 and 2 at power levels up to 2441 MW(t). Unit 2 is expected to be ready for operation about six months after Unit 1. This project was considered at the 139th Committee meeting, November 11-13, 1971, and at Subcommittee meetings at the site July 1, 1971, and in Washington, D. C. on November 5, 1971. During its review, the Committee had the benefit of discussion with representatives of Virginia Electric and Power Company, Westinghouse Electric Corporation, Stone and Webster Engineering Corporation, the Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported on the construction of these units in its letter of April 29, 1968.

Surry Units 1 and 2 are located in Surry County, Virginia on the Gravel Neck Peninsula which extends northwest from the south bank of the James River. The nearest boundary of Newport News, Virginia, population about 140,000, is approximately 4.7 miles southeast of the site. The high population density portion of Newport News is 17 miles from the plant. There is a large transient population in the area of the plant during summer months.

Each nuclear unit employs a pressurized water reactor in a three-loop nuclear steam supply system of essentially the same design as the H. B. Robinson Unit No. 2 and Turkey Point Units 3 and 4, previously reviewed.

The applicant states that he intends to operate Units 1 and 2 in such a manner as to assure that maximum fuel rod linear power density does not exceed 16.4 kw/ft at 102% of the rated power of 2441 MW(t). Performance of the emergency core cooling system (ECCS) during postulated loss-of-coolant accidents has been reevaluated in the light of results from the Commission's FLECHT program, experiments and analyses by the applicant and his contractors, and information developed by the Regulatory Staff in recent studies of ECCS. The Committee believes that the applicant's proposed manner of operation is acceptable.

The applicant proposes to use a catalytic hydrogen recombiner to control the buildup of hydrogen in the containment that could follow in the unlikely event of a loss-of-coolant accident. The recombiner will be operable by the end of 1972. In the meantime, a backup purging system, to be operable before plant startup, will be relied upon for hydrogen concentration control. The Regulatory Staff should assure itself that the criteria for these systems are consistent with those for other engineered safety features.

The applicant has reported that the stress relieving of much of the cold bent type 316 stainless steel piping, some of which is utilized in the emergency core cooling system, resulted in its becoming sensitized to potential stress corrosion cracking under certain conditions. The applicant proposes that an augmented inservice inspection program, together with certain additional, special operating procedures, be implemented to aid in assuring maintenance of integrity of this piping throughout plant life. The details of these programs and procedures should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant should assure himself that instrumentation for determining the course of postulated accidents is on hand at the station and that appropriate calibration methods and calculated bases for interpreting instrument responses are available.

The Committee reiterates its previous comments concerning the need to study further means of preventing common mode failures from negating reactor scram action, and design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. The Committee wishes to be kept informed of the resolution of this matter.

Other problems relating to large water reactors, which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports, should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

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 Surry Nuclear Generating Units 1 and 2 can be operated at power levels up to 2441 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H Bush

Spencer H. Bush
Chairman

References

1. Amendment No. 12, dated January 21, 1970: Final Safety Analysis Report, Surry Power Station Unit Nos. 1 and 2, Volumes 1 through 4 and Supplements Volumes 1 and 2

2. Amendment No. 13, dated February 9, 1970: Amended License Application for Surry Power Station Unit Nos. 1 and 2

3. Amendments Nos. 14 through 29, to License Application for Surry Power Station Unit Nos. 1 and 2

4. "Security Program, Surry Power Station," (Proprietary), dated October 22, 1971, Virginia Electric and Power Company

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

July 16, 1974

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

Subject: REPORT ON SURRY POWER STATION, UNITS 3 AND 4

Dear Dr. Ray:

At its 171st meeting, July 11-13, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application of the Virginia Electric and Power Company for a license to construct the Surry Power Station, Units 3 and 4. This project had been considered previously during a Subcommittee meeting in Williamsburg, Virginia on June 28, 1974, subsequent to a tour of the site. In addition, the ACRS Subcommittee on Babcock and Wilcox Water Reactors discussed topics pertinent to the nuclear steam supply system for this plant at a meeting in Washington, D. C. on July 5, 1974. In the course of its review, the Committee had the benefit of discussions with representatives and consultants of the Virginia Electric and Power Company, the Babcock and Wilcox Company, the Stone and Webster Engineering Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee previously reported to the Commission on the construction and operation of the Surry Power Station, Units 1 and 2, in its letters of April 29, 1968 and December 17, 1971.

The site for the Surry Power Station is an 840-acre tract located in the county of Surry, Virginia. The nearest population center is the city of Newport News, which had a 1970 population of about 138,000 and whose nearest boundary lies 4.5 miles east-southeast of the site. Due to the presence of several places of historical importance, there is a large transient population in the area of the plant during summer months.

Each Nuclear Unit will employ a pressurized water reactor with a two-loop coolant system of essentially the same design as that previously reviewed and approved by the Committee for the North Anna Power Station, Units 3 and 4. Each of the proposed Surry reactors will be designed to operate at a power of 2631 MW(t) with an expected ultimate capability of producing 2763 MW(t).

The applicant proposes to utilize in Surry Units 3 and 4 a new reactor protection system, designated as RPS-II. The system, a hybrid using both analog and digital techniques, represents an evolution from the analog system, RPS-I, currently in use in the Oconee reactors. RPS-II incorporates a single-chip central processor unit as a micro-computer for the more complex trip functions. The applicant has proposed a series of environmental, reliability, and in situ tests for qualification of this system prior to its use in Surry Units 3 and 4. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Committee recommended in its report of January 7, 1972, on Interim Acceptance Criteria for ECCS, that significantly improved ECCS capability should be provided for reactors for which construction permit applications were filed after January 7, 1972. This position was repeated in the Committee's report of September 10, 1973, on Acceptance Criteria for ECCS. The Surry Units 3 and 4 are in this category. The applicant has amended the license application to use the B&W Mark C (17x17) fuel assembly design, instead of the B&W Mark B (15x15) design previously proposed. The new fuel assemblies will be operated at lower linear heat generation rates and are expected to yield greater thermal margins for fuel design limits and improved safety margins in the analyses of the loss-of-coolant accidents. An extensive program has been initiated for determining the mechanical and thermal-hydraulic characteristics of the new fuel assemblies. A program of control rod tests also is proposed, including testing of trip times and control rod wear. Should modifications become necessary as a result of the control rod tests, retesting of the entire control rod drive would be undertaken. While many of the details of the proposed design are available, complete analyses of the performance of the Mark C fuel are not yet available, and the AEC Regulatory Staff has not completed its review. The Committee reserves judgment concerning the final design until the required performance information is presented and has been adequately reviewed. The Committee recommends that the applicant continue studies directed at further improvements in the capability and reliability of the ECCS. The Committee wishes to be kept informed.

The Staff Safety Evaluation Report did not address the matter of turbine missiles. The Committee recommends that the Regulatory Staff review the turbine orientation for Surry Units 3 and 4 to establish that appropriate protection from potential turbine missile damage to safety related equipment will be provided.

The Regulatory Staff has been investigating on a generic basis the problems associated with a potential reactor coolant pump overspeed in the unlikely event of a particular type of rupture at certain locations in a main coolant pipe. Some additional protective measures may be warranted for Surry Units 3 and 4 in this regard. The Committee recommends that resolution of this matter be expedited. The Committee wishes to be kept informed.

The Committee believes the applicant should address more attention to instrumentation for the determination of the course of potentially serious accidents, particularly with regard to upper range limits to fully encompass the spectrum of possible accidents. The instrumentation system should respond on a time scale which would permit necessary emergency action. The applicant should assure himself that appropriate calibration methods and calculated bases for interpreting instrument responses are available.

The applicant has made progress in arrangements for offsite emergency procedures to be followed in case of an accidental release of radioactive materials to the environment. Yet to be confirmed, however, are modifications in the plans of the State agency whose actions would be important in dealing with the population in the unlikely event of a major release. The Committee recommends that the applicant and the AEC Staff continue to collaborate with the State in moving ahead to complete development of an emergency action plan, and that the adequacy of arrangements for implementing such a plan be confirmed as soon as feasible. Also important is the planning for the protection of construction workers at Surry Units 3 and 4 in case of an unexpected release of radioactive materials from operating Units 1 and 2.

The Committee believes it is desirable for the applicant and the Regulatory Staff to continue to review Surry Units 3 and 4 for design features that could reduce the possibility and consequences of sabotage, in accordance with Regulatory Guide 1.17, "Protection of Nuclear Plants Against Industrial Sabotage."

Generic problems relating to large water reactors have been identified by the Regulatory Staff and the ACRS and 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 Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during construction and that, if due consideration is given to the foregoing, Surry Power Station, Units 3 and 4, can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References

1. Virginia Electric and Power Company (VEPCO) Application for a Construction Permit for the Surry Power Station, Units 3 and 4, with Preliminary Safety Analysis Report (PSAR), Vols. 1-10.

2. Amendments 1-13 and 15-17 to the Application.

3. VEPCO letter, dated December 17, 1973, transmitting Report Concerning the Analysis of Postulated High Energy Line Failures Outside Containment.

4. Directorate of Licensing letter, dated May 23, 1974, transmitting Safety Evaluation Report.

5. Directorate of Licensing letter, dated June 14, 1974, regarding Design Safety Factors and Subcompartment Pressure Analyses.

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

April 13, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

Dear Dr. Schlesinger:

At its 144th meeting, April 6-8, 1972, the Advisory Committee on Reactor Safeguards completed its review of the application from the Pennsylvania Power and Light Company for a permit to construct the Susquehanna Steam Electric Station, Units 1 and 2. The project was previously considered at a Subcommittee meeting at the Station site on March 24, 1972. During the review the Committee had the benefit of discussions with representatives and consultants of the applicant, the General Electric Company, the Bechtel Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The Susquehanna Station will be located in Pennsylvania on a 1522 acre site on the west bank of the Susquehanna River approximately 12 miles northwest of Hazleton and 15 miles southwest of Wilkes-Barre, the nearest cities having populations in excess of 25,000. The low population zone radius is 3.0 miles within which the 1970 population was about 2,400 and the projected 2020 population about 4,000. The exclusion zone has a minimum radius of 1,800 feet and is separated from the river on the east by U. S. Route 11 and a single-track line of the Erie-Lackawanna Railroad. The principal facilities are located approximately 3,000 feet from the bank of the river at a grade elevation of about 170 feet above the bank.

The Susquehanna Station will utilize two General Electric boiling water reactors, each to be operated at a power level of 3293 MWt with waste heat rejected to the atmosphere by two natural-draft cooling towers. The primary containment is of the over-under pressure suppression type similar to those previously reviewed for Zimmer, Limerick, and Shoreham. The reactors are of the 1967 General Electric product line and similar to those of other facilities now under construction, particularly Browns Ferry 1, 2, and 3 and Peach Bottom Units 2 and 3.

The applicant does not currently own all portions of the proposed site south of the reactors and within the exclusion radius. Similarly, mineral rights within the exclusion radius are not yet owned by the applicant. Procedures are being initiated to obtain ownership of the needed properties, and the applicant has stated that no construction will begin until this has been accomplished.

The applicant's criteria for protecting low pressure piping from overpressure include interlocks to prevent residual heat removal (RHR) system valves from opening unless the reactor coolant system pressure is below the RHR system design pressure. Although the applicant will design these interlocks to meet the requirements of IEEE 279-1971, the Committee recommends that diverse pressure sensors also be employed to provide greater assurance of performance of this important function.

The Susquehanna Station is the second plant for which the relief valve augmented bypass (REVAB) system is proposed. This system allows a full-load rejection without a reactor scram even though the turbine bypass capacity is only 25% of full-power steam flow. REVAB utilizes rapid-response pressure relief valves discharging into the suppression pool and rapid reactor power reduction to avoid reaching scram setpoints. As this system provides an additional signal causing opening in the primary system coolant boundary, the Committee believes that attention should be given to the possibility of valves remaining open following REVAB action.

The Committee believes that the main steam lines up to and including the turbine stop valves, and all branch lines 2-1/2 inches and larger up to their first valve, should be dynamically analyzed to ensure structural integrity during a design basis earthquake. A sealing system designed to standards applicable to engineered safety features should be provided to minimize leakage through the main steam line isolation valves. These matters should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant has studied design features to make tolerable the consequences of failure to scram during anticipated transients, and has concluded that automatic tripping of the recirculation pumps and injection of boron could provide a suitable backup to the control rod system for this type of event. The Committee believes that this recirculation pump trip represents a substantial improvement and should be provided for the Susquehanna reactors. However, further evaluation of the sufficiency of the approach and the specific means of implementing the proposed pump trip should be made. This matter should be resolved in a manner satisfactory to the Regulatory Staff and the ACRS during construction of the reactors.

The techniques for analysis of the control rod drop accident are being revised by the General Electric Company. The adequacy of the revised model and the acceptability of the results should be established in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed of the resolution of this matter.

Current analysis indicates acceptably low peak clad temperatures following a postulated loss-of-coolant accident. A research program, which was recently begun under the auspices of the General Electric Company and the USAEC, should provide more detailed knowledge of the flow and heat transfer processes during the first stages of such postulated accidents. More detailed analytical studies, particularly as they relate to the time to critical heat flux and the level swell process, should also be performed during construction of the plant. These studies should be reviewed by the Regulatory Staff. 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 Susquehanna Station.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items, the Susquehanna Steam Electric Station, Units 1 and 2, can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

References

List Attached

References

1. Pennsylvania Power and Light Company letter dated 4/1/71 transmitting their Application for Licenses for the Susquehanna Steam Electric Station together with an Environmental Report and Vols. 1 through 6, Preliminary Safety Analysis Report

2. Amendments 1 and 3 through 7 to the Application

3. Pennsylvania Power and Light Company letter dated 4/3/72

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 11, 1981

The Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2

Dear Dr. Palladino:

During its 256th meeting, August 6-8, 1981, the Advisory Committee on Reactor Safeguards completed its review of the application of the Pennsylvania Power and Light Company and Allegheny Electric Cooperative, Inc. (Applicant) for a license to operate the Susquehanna Steam Electric Station Units 1 and 2. The units will be operated by the Pennsylvania Power and Light Company. A Subcommittee meeting was held in Washington, D.C. on July 23, 1981 to consider this project. A tour of the facility was made on July 2, 1981. During its review, the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee commented on the construction permit application for this station in its report dated April 13, 1972.

The Susquehanna station is located in Luzerne County, Pennsylvania about 12 miles northwest of Hazleton and 15 miles southwest of Wilkes-Barre, the nearest cities having populations in excess of 25,000.

Each Susquehanna unit is equipped with a General Electric BWR-4 nuclear steam supply system with a rated power level of 3293 MWt and has a Mark II pressure suppression containment with a design pressure of 53 psig.

In connection with our review of the Susquehanna station, the NRC Staff discussed its generic resolution of the safety issues associated with the Mark II containment design and performance. This resolution is given in the Staff report NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria." This matter has received detailed review by the ACRS Subcommittee on Fluid Dynamics. We believe that the load definitions given in this report are conservative and acceptable. These load definitions are to be applied to BWR Mark II's on a case-by-case basis. We believe that the Susquehanna containment structures will meet these requirements.

The Applicant described the management organization and the technical personnel available for operation of the Susquehanna plant. Although this is the first nuclear power plant to be operated by this Applicant, both management and plant staff are made up of personnel with considerable background and expertise in commercial nuclear power plant operation. We commend the Applicant's efforts to obtain knowledgeable and experienced personnel.

The Applicant described the program and the philosophy for training of personnel. Training has a high priority as it had even prior to the TMI-2 accident. For example, a training simulator was ordered by the Applicant considerably before the accident at TMI-2 and is currently in use. The training program includes consideration of ATWS. The Applicant's training program appears sound and thorough.

The NRC Staff proposes to require the installation of core thermocouples in the Susquehanna station as specified by Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." The Applicant has not yet agreed to this requirement. We supported use of core thermocouples in BWRs in our letter of November 10, 1980 to the NRC Executive Director for Operations but called attention to the need for further study to determine the appropriate vertical location of such thermocouples. Since most of the information of interest from thermocouples may be obtainable from a small number of thermocouples placed in a more accessible location, we recommend that this requirement be reevaluated.

The NRC Staff proposes to require a second meteorological tower at the Susquehanna site for the purpose of collecting additional data for use during an emergency. This issue is still being discussed with the NRC Staff. Additionally, there are several other issues concerning emergency planning which are identified by the NRC Staff in its Safety Evaluation Report and Supplement No. 1 as Outstanding Issues. We believe that these issues should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed.

Another Outstanding Issue involves IE Bulletin 79-27, "Loss of Non-Class-1-E Instrumentation and Control Power System Bus During Operation." The Applicant has stated that this IE Bulletin will be complied with prior to issuance of an operating license. We recommend that this issue be resolved in a manner satisfactory to the NRC Staff.

The Applicant is currently reviewing the issue of station blackout. Analytical work, development of operating procedures, and actual testing of equipment response to simulated blackout conditions are planned by the Applicant. We believe that the Applicant's proposed program is a satisfactory response to this issue.

The NRC Staff has identified other Outstanding Issues in its Safety Evaluation Report dated April 1981 and in Supplement No. 1 to that report dated June 1981 such as turbine missiles, review of the alternate shutdown system, and modification of depressurization logic. We believe the Outstanding Issues can be resolved, and recommend that this be done in a manner satisfactory to the NRC Staff before operation at full power.

The Committee believes that if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that Susquehanna Steam Electric Station Units 1 and 2 can be operated at power levels up to 3293 MWt each without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. Pennsylvania Power and Light Company, "Final Safety Analysis Report, Susquehanna Steam Electric Station, Units 1 and 2," with Amendments 1 through 35.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Susquehanna Steam Electric Station, Units 1 and 2, Docket Nos. 50-387 and 50-388," USNRC Report NUREG-0776, dated April 1981 and Supplement No. 1, dated June 1981.
3. U.S. Nuclear Regulatory Commission IE Bulletin No. 79-27, "Loss of Non-Class-1-E Instrumentation and Control Power System Bus During Operation," dated November 30, 1979.

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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, 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.

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.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C.

April 12, 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-sixth meeting, on April 4-6, 1968, the Advisory Committee on Reactor Safeguards completed its re-examination of the application by Metropolitan Edison Company to construct the Three Mile Island Nuclear Station Unit 1. The Three Mile Island Nuclear Station Unit 1 was the subject of a report to you dated January 17, 1968. The review was re-opened at the ninety-fifth ACRS meeting, on March 7-9, 1968, in consideration of an additional submittal by the applicant evaluating further the effects of the proximity of the Olmsted State Airport. The Committee has had the benefit of discussions with representatives of Metropolitan Edison Company, Babcock and Wilcox Company, Gilbert Associates, and the AEC Regulatory Staff and their consultants. The additional documents submitted are listed in this report.

Although the probability of an airplane hitting the station is very small, the applicant has undertaken to provide principal structures and components of the station with the capability of withstanding aircraft strike loadings over a range of conditions, including effects such as secondary missiles, fire, and pressure and temperature effects. The reactor building, control building, fuel handling building, auxiliary building, and intermediate building will have the necessary modifications to assure the capability of bringing the plant to a safe shutdown condition.

The Committee reaffirms its previous conclusion that, if suitable attention is given to the various items mentioned in the Committee's report of January 17, 1968, 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/ Carroll W. Zabel

Carroll W. Zabel
Chairman

References:

1. Amendment No. 8 to Application; Metropolitan Edison Company letter dated February 23, 1968.

2. Amendment No. 9 to Application; Metropolitan Edison Company letter dated March 6, 1968.

3. Amendment No. 10 to Application; Metropolitan Edison Company letter dated March 28, 1968.

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

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.

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.

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,

/s/ 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.

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

August 14, 1973

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

Subject: REPORT ON THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Dear Dr. Ray:

During its 160th meeting, August 9-11, 1973, the Advisory Committee on Reactor Safeguards completed its review of the application of the Metropolitan Edison Company, Jersey Central Power and Light Company, Pennsylvania Electric Company, and General Public Utilities Corporation for a license to operate Unit 1 of the Three Mile Island Nuclear Station at power levels up to 2535 MW(t). This project was considered during a Subcommittee site visit and meeting conducted on May 27 and 28, 1971. The Subcommittee visited the site again on May 3, 1973, and held a meeting in Washington, D. C. on July 25, 1973. In the course of the review, the Committee had the benefit of discussions with representatives and consultants of the Metropolitan Edison Company, the General Public Utilities Corporation, Gilbert Associates, the Babcock and Wilcox Company, and the AEC Regulatory Staff, and of the documents listed. The Committee reported to the Commission on the construction of this Unit in its letters of January 17 and April 6, 1968, and on the construction of Unit 2 in its letter of July 17, 1969.

Three Mile Island Nuclear Station is located on Three Mile Island in the Susquehanna River, about 10 miles southeast of Harrisburg, Pennsylvania. Harrisburg International Airport is located 2-1/2 miles northwest of Unit 1. The applicant has provided protection of the engineered safety features and safe shutdown equipment in the unlikely event of the impact of an aircraft up to 200,000 pounds, and against fires resulting from crashes of even larger aircraft.

The application for a construction permit proposed initial operation at power levels up to 2452 MW(t), the same as the construction permit power level of Oconee Nuclear Station, Unit 1 which employs a similar reactor. Safety studies and performance analyses have been made for a power level of 2535 MW(t) for Three Mile Island Nuclear Station, Unit 1. The

Committee believes that review of the operation of Oconee Nuclear Station, Unit 1 by the Regulatory Staff should be completed and satisfactory performance of Oconee Nuclear Station, Unit 1 should be demonstrated before Three Mile Island Nuclear Station, Unit 1 is operated at full licensed power.

The hot functional testing of Oconee Nuclear Station, Unit 1 which was conducted in 1972 caused damage of some components, including reactor vessel internals. The design changes which were made for Oconee Nuclear Station, Unit 1 have been applied to Three Mile Island Nuclear Station, Unit 1. The Committee believes that these changes are acceptable.

The applicant has been responsive to the Committee's recommendation that suitable instrumentation be sought to monitor for loose parts and for vibration; such instrumentation has been designed and will be utilized.

The applicant stated that he will propose appropriate additional operating limitations if, at any time during operation, the moderator temperature coefficient of reactivity is positive. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Regulatory Staff has been investigating on a generic basis the problems associated with a potential reactor coolant pump overspeed in the unlikely event of a particular type of rupture at certain locations in a main coolant pipe. Some additional protective measures may be warranted and this matter should be resolved to the satisfaction of the Regulatory Staff. The Committee wishes to be kept informed.

The Committee reiterates its previous comments on the need for further study of means for preventing common mode failures from negating reactor scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. The Committee wishes to be kept informed of the resolution of this matter.

The applicant should assure himself that instrumentation for determining the course of potentially serious accidents, on a time scale that will permit appropriate emergency action, is provided at the station and that appropriate calibration methods and calculated bases for interpreting instrument responses are available.

It was reported that some of the steel bearing plates at the upper ends of the vertical prestressing tendons in the containment wall had depressed into the concrete as much as one-eighth inch during the tensioning operation. The Committee believes that the cause of this behavior should be determined and its possible effects should be evaluated. This matter should be resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The applicant has proposed measures, including alarms and administrative procedures, to prevent operating under conditions which might result in exceeding acceptable fuel limits established from accident studies and other considerations. The current review has been confined to the first fuel cycle and the analyses have been based on the as-built fuel. The ACRS recommends that the Regulatory Staff establish suitable criteria for these measures, and provide suitable bases for evaluating future loadings. The Committee wishes to be kept informed.

The Committee recognizes that re-evaluation of operating limits may be necessary as a result of possible changes in the acceptance criteria for emergency core cooling systems. The Committee wishes to be kept informed.

Other problems relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous reports should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

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 preoperational testing, there is reasonable assurance that Three Mile Island Nuclear Station, Unit 1 can be operated at power levels up to 2535 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/ H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

Attachment:

List of References

<u>References</u>

1. Final Safety Analysis Report, Vols. 1 through 5

2. Amendments 13 through 41 to the Application

3. BAW-1389 (Proprietary), dated June 15, 1973, "Three Mile Island, Unit 1 Fuel Densification Report"

4. DL Technical Report on Densification of B&W Reactor Fuel, dated July 6, 1973

5. DL Safety Evaluation, dated July 11, 1973

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 22, 1976

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON THREE MILE ISLAND NUCLEAR STATION, UNIT 2

Dear Mr. Rowden:

During its 198th meeting, October 14-16, 1976, the Advisory Committee on Reactor Safeguards completed its review of the application of the Metropolitan Edison Company, Jersey Central Power and Light Company, and Pennsylvania Electric Company (Applicants) for a license to operate Three Mile Island Nuclear Station, Unit 2. This project was also considered during a Subcommittee meeting held in Harrisburg, Pennsylvania, on September 23 and 24, 1976. Members of the Committee visited the facility on September 23, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, General Public Utilities Service Corporation, the Babcock and Wilcox Company (B&W), Burns and Rowe, Inc., and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had available the documents listed below. The Committee reported on the application for a construction permit for Unit 1 on January 17 and April 12, 1968, and for an operating license for Unit 1 on August 14, 1973. The Committee reported on the application for a construction permit for Unit 2 on July 17, 1969.

The Three Mile Island Nuclear Station, Units 1 and 2, is located on Three Mile Island near the eastern shore of the Susquehanna River, about 12 miles southeast of Harrisburg, Pennsylvania. About 2380 people live within a two-mile radius of the site (the low population zone). The minimum exclusion distance is 2000 feet. The nearest population center is Harrisburg (1970 population 68,000).

Several changes have been made to bring the Babcock and Wilcox Emergency Core Cooling System (ECCS) evaluation model into conformance with the requirements of 10 CFR 50.46, and Appendix K to Part 50. Analyses of a spectrum of break sizes appropriate to Three Mile Island, Unit 2 have been completed using the approved B&W generic evaluation model. The results of the analyses for the reactor coolant pump discharge break, believed to be the "worst" break, show maximum allowable linear heat generation rates as a function of elevation in the reactor core ranging from 15.5 to 18.0 kilowatts per foot. Corresponding calculated post-accident peak clad temperatures range from 2002°F to 2146°F. The NRC Staff has identified additional information that it will require to complete its review and the Applicants' submittal is expected by the end of 1976. The Applicants propose to use both in-core and ex-core instrumentation to assure accuracy of measurement of core power distributions. The Committee believes that the proposed monitoring methods may be acceptable, but that an augmented startup program should be employed, and that satisfactory experience at 100% steady state power and during transients at less than full power should be obtained. This experience should be reviewed and evaluated by the NRC Staff prior to operating at up to full power in a load following mode. The Committee wishes to be kept informed.

A question has arisen concerning asymmetric loads on the reactor vessel and its internal structures for certain postulated loss-of-coolant accidents in pressurized water reactors. The Staff has required the Applicants to supply further information in order to complete its assessment of this matter. This issue should be resolved in a manner satisfactory to the NRC Staff.

The question of whether Unit 2 requires design modifications in order to comply with WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors", remains an outstanding issue pending the NRC Staff's completion of its review of B&W generic analyses of anticipated transients without scram. The Committee recommends that the NRC Staff, the Applicants and B&W continue to strive for an early resolution of this matter in a manner acceptable to the NRC Staff. The Committee wishes to be kept informed.

Emergency plans have been developed to allow plant shutdown and maintenance of safe shutdown in the event of a maximum probable flood. Such a postulated flood would top the levee surrounding the plant by several feet. Included in the plan is the fastening of water tight steel panels in doorways and other openings of safety related structures. The Committee believes that the details of this plan, particularly relating to re-entry into the station during the post-flood period, need to be more clearly delineated.

The Committee supports the NRC Staff's program for evaluation of fire protection in accordance with Branch Technical Position APCSB 9.5-1, Appendix A, "Guidelines for Fire Protection for Nuclear Power Plants". The Committee recommends that the NRC Staff give high priority to the completion of both owner and Staff evaluations and to recommendations for Three Mile Island Unit 2 and other plants nearing completion of construction in order to maximize the opportunity for improving fire protection while areas are still accessible and changes are more feasible.

The Committee notes that long-term post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptable to ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The Committee believes that appropriate test procedures to confirm continuous long-term seal capability should be developed.

The Committee recommends that further review be made of the battery supplied DC power system to assure that non-essential loads do not interfere with its safety function. The Committee recommends that further review be made to assure no unacceptable effects such as release of hydrogen into the plant can occur from the failure of a hydrogen charging line. The Committee also recommends that studies be made to assure that failure of an instrument line cannot cause plant controllability problems of significance to public safety.

The management organization proposed by the Applicants to delineate the safety related responsibilities of the off-site and on-site personnel of the Three Mile Island Station left open questions as to how these responsibilities are to be discharged during normal working hours and during evening, night, and weekend shifts. This matter should be resolved to the satisfaction of the NRC Staff.

The NRC Staff is still reviewing various issues related to accidents leading to loss of fluid in the steam generator secondary side, such as steam line breaks. The Committee wishes to be kept informed of the resolution of these issues.

The Committee recommends that, prior to commercial power operation of Three Mile Island Unit 2, additional means for evaluating the cause and likely course of various accidents, including those of very low probability, should be in hand in order to provide improved bases for timely decisions concerning possible off-site emergency measures. The Committee wishes to be kept informed.

The Committee believes that the Applicants and the NRC Staff should further review the Three Mile Island Nuclear Station for measures that could significantly reduce the possibility and consequences of sabotage, and that such measures should be implemented where practical.

Other generic problems relating to large water reactors are discussed in the Committee's report entitled "Status of Generic Items Relating to Light Water Reactors: Report No. 4", dated April 16, 1976. Those problems relevant to the Three Mile Island Station should be dealt with appropriately by the NRC Staff and the Applicants as solutions are found. The relevant items are: II - 1, 2, 3, 4, 5, 6, 7, 9, 11; IIA - 1, 4, 5, 6, 7, 8; IIC - 1, 2, 3, 4, 5, 6, 7.

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 Three Mile Island Nuclear Station, Unit 2 can be operated at power levels up to 2772 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

References

1. Three Mile Island Nuclear Station, Unit 2 Final Safety Analysis Report (April, 1974) with Amendments 1 through 44.

2. Safety Evaluation Report (NUREG-0107) related to operation of Three Mile Island Nuclear Station, Unit 2, dated September, 1976.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 7, 1979

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM REPORT ON RECENT ACCIDENT AT THE THREE MILE ISLAND NUCLEAR STATION UNIT 2

Dear Dr. Hendrie

During its 228th meeting, April 5-7, 1979, the Advisory Committee on Reactor Safeguards reviewed the circumstances relating to the recent accident at the Three Mile Island Nuclear Station Unit 2. During this review, the Committee had the benefit of discussions with the NRC Staff.

Our study of the accident at Three Mile Island has shown that it is very difficult for a PWR plant operator to understand and properly control the course of an accident involving a small break in the reactor coolant system accompanied by other abnormal conditions.

The Committee recommends that further analyses be made, as soon as possible, of transients and accidents in PWRs that involve initially, or at some time during their course, a small break in the primary system. The computer codes used for these analyses should be capable of predicting the conditions observed during the accident at Three Mile Island, including thermal-hydraulic effects and clad and fuel temperatures. The range of break sizes considered should include the smallest that could be deemed significant, and should consider a range of break locations.

The Committee believes that the analyses recommended above will demonstrate, as has the accident at Three Mile Island, that additional information regarding the status of the system will be needed in order for the plant operator to follow the course of an accident and thus be able to respond in an appropriate manner. As a minimum, and in the interim, it would be prudent to consider expeditiously the provision of instrumentation that will provide an unambiguous indication of the level of fluid in the reactor vessel. Early consideration should be given also to providing remotely controlled means for venting high points in the reactor system, as practical.

The foregoing recommendations apply to all pressurized water reactors.

The recommendations in IE Bulletin 79-05A, dated April 5, 1979, are believed to be generally suitable for Babcock and Wilcox facilities, on an interim basis. However, the Committee believes that the actions listed in Item 4b. under the heading, "Actions To Be Taken by Licensees," may prove to be unduly prescriptive in view of the uncertainties in predicting the course of anomalous transients or accidents involving small breaks in the primary system.

With regard to Three Mile Island Unit 2, the Committee believes that decisions should be made expeditiously with regard to contingency measures which may be prudent concerning containment and reactor cooldown as a backup to the currently planned cooldown procedure.

The Committee is continuing its review of these and other concerns arising from this accident and will provide further advice as it is developed.

Sincerely,

Max W. Carbon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 18, 1979

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

FROM: R. F. Fraley, Executive Director
Advisory Committee on Reactor Safeguards

Attached for your information and use is a copy of the recommendations of the Advisory Committee on Reactor Safeguards which were orally presented to and discussed with you on April 17, 1979 regarding the recent accident at the Three Mile Island Nuclear Station Unit 2.

R F Fraley
R. F. Fraley
Executive Director

Attachment:
Recommendations of the NRC Advisory Committee
on Reactor Safeguards Re. the 3/28/79 Accident
at The Three Mile Island Nuclear Station Unit 2

April 17, 1979

RECOMMENDATIONS OF THE NUCLEAR REGULATORY COMMISSION ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REGARDING THE MARCH 28, 1979 ACCIDENT AT THE THREE MILE ISLAND NUCLEAR STATION UNIT 2

Presented orally to, and discussed with, the NRC Commissioners during the ACRS-Commissioners Meeting on April 17, 1979 - Washington, D. C.

Natural circulation is an important mode of reactor cooling, both as a planned process and as a process that may be used under abnormal circumstances. The Committee believes that greater understanding of this mode of cooling is required and that detailed analyses should be developed by licensees or their suppliers. The analyses should be supported, as necessary, by experiment. Procedures should be developed for initiating natural circulation in a safe manner and for providing the operator with assurance that circulation has, in fact, been established. This may require installation of instrumentation to measure or indicate flow at low water velocity.

The use of natural circulation for decay heat removal following a loss of offsite power sources requires the maintenance of a suitable overpressure on the reactor coolant system. This overpressure may be assured by placing the pressurizer heaters on a qualified onsite power source with a suitable arrangement of heaters and power distribution to provide redundant capability. Presently operating PWR plants should be surveyed expeditiously to determine whether such arrangements can be provided to assure this aspect of natural circulation capability.

The plant operator should be adequately informed at all times concerning the conditions of reactor coolant system operation which might affect the capability to place the system in the natural circulation mode of operation or to sustain such a mode. Of particular importance is that information which might indicate that the reactor coolant system is approaching the saturation pressure corresponding to the core exit temperature. This impending loss of system overpressure will signal to the operator a possible loss of natural circulation capability. Such a warning may be derived from pressurizer pressure instruments and hot leg temperatures in conjunction with conventional steam tables. A suitable display of this information should be provided to the plant operator at all times. In addition, consideration should be given to the use of the flow exit temperatures from the fuel subassemblies, where available, as an additional indication of natural circulation.

The exit temperature of coolant from the core is currently measured by thermocouples in many PWRs to determine core performance. The Committee recommends that these temperature measurements, as currently available, be used to guide the operator concerning core status. The range of the information displayed and recorded should include the full capability of the thermocouples. It is also recommended that other existing instrumentation be examined for its possible use in assisting operating action during a transient.

The ACRS recommends that operating power reactors be given priority with regard to the definition and implementation of instrumentation which provides additional information to help diagnose and follow the course of a serious accident. This should include improved sampling procedures under accident conditions and techniques to help provide improved guidance to offsite authorities, should this be needed. The Committee recommends that a phased implementation approach be employed so that techniques can be adopted shortly after they are judged to be appropriate.

The ACRS recommends that a high priority be placed on the development and implementation of safety research on the behavior of light water reactors during anomalous transients. The NRC may find it appropriate to develop a capability to simulate a wide range of postulated transient and accident conditions in order to gain increased insight into measures which can be taken to improve reactor safety. The ACRS wishes to reiterate its previous recommendations that a high priority be given to research to improve reactor safety.

Consideration should be given to the desirability of additional equipment status monitoring on various engineered safeguards features and their supporting services to help assure their availability at all times.

The ACRS is continuing its review of the implications of this accident and hope to provide further advice as it is developed.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 20, 1979

Honorable Victor Gilinsky
Acting Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Gilinsky:

This letter is in response to yours of April 18, 1979 which requested that the ACRS notify the Commissioners immediately if we believe any of our oral recommendations of April 17 should be acted upon before our next regularly scheduled meeting at which we could prepare a formal letter. The Committee discussed this topic by conference telephone call on April 19 and offers the following comments.

All of the recommendations made by the ACRS in its meeting with the Commissioners on April 17, 1979, are generic in nature and apply to all PWRs. None were intended to require immediate changes in operating procedures or plant modifications of operating PWRs. Such changes should be made only after study of their effects on overall safety. Such studies should be made by the licensees and their suppliers or consultants and by the NRC Staff. The Committee believes that these studies should be begun in the near future on a time scale that will not divert the NRC Staff or the industry representatives from their tasks relating to the cooldown of Three Mile Island Unit 2. However, the Committee believes that it would be possible and desirable to initiate immediately a survey of operating procedures for achieving natural circulation, including the case when offsite power is lost, and the role of the pressurizer heaters in such procedures.

At its meeting on April 16 and 17, 1979, the Committee discussed with the NRC Staff the matter of natural circulation for the Three Mile Island Unit 2 plant. The Committee believes that this matter is receiving careful attention by the NRC Staff and the licensee.

The Committee's own recommendations to the Commission on April 17 were not intended to apply to Three Mile Island Unit 2.

We plan to write a further report on these matters at our May 10, 1979 meeting.

Sincerely,

Max W. Carbon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 16, 1979

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: INTERIM REPORT NO. 2 ON THREE MILE ISLAND NUCLEAR STATION UNIT 2

Dear Dr. Hendrie:

During its 229th meeting May 10-12, 1979, the Advisory Committee on Reactor Safeguards continued its review of the circumstances relating to the recent accident at Three Mile Island Nuclear Station Unit 2 (TMI-2). The recommendations presented orally to the Commissioners on April 17, 1979 were reviewed by the full Committee and are repeated in somewhat amplified form herein. Amplification of these items is responsive to the request of Acting NRC Chairman Victor Gilinsky dated April 18, 1979.

Natural Circulation - Procedures

It is evident from the experience at TMI-2 that there was failure to establish natural circulation of water in the primary system and failure to recognize in a timely manner that natural circulation had not been achieved. The need for natural circulation under certain circumstances is common to all PWRs.

The Committee recommends that procedures be developed by all operators of PWRs for initiating natural circulation in a safe manner and for providing the operator with assurance that circulation has in fact been established. These procedures should take into account the behavior of the systems under a variety of abnormal conditions.

As a first step, the NRC Staff should initiate immediately a survey of operating procedures for achieving natural circulation, including the case when offsite power is lost. At the same time, the operators of all PWR plants should be requested to develop detailed analyses of the behavior of their plants following anticipated transients and small breaks in the primary system, with appropriate consideration of potential abnormal conditions, operator errors and failures of equipment, power sources, or instrumentation. These analyses are necessary for the development of suitable operating procedures. The review and evaluation of these analyses by the NRC Staff should receive a priority consistent with the priority being given to changes in operating procedures.

Natural Circulation - Pressurizer Heaters

The use of natural circulation for decay heat removal following an accident in a PWR normally requires the maintenance of a suitable over-pressure on the reactor coolant system in order to prevent the generation of steam which can impede circulation. For many transients, maintenance of this overpressure is best accomplished by use of the pressurizer heaters.

Although the pressurizer heaters at TMI-2 continued to receive power from offsite sources during the entire accident, the availability of offsite power cannot be assured for all transients or accidents during which, or following which, natural circulation must be established. The Committee recommends that the NRC Staff initiate immediately a survey of all PWRs licensed for operation to determine whether the pressurizer heaters are now or can be supplied with power from qualified onsite sources with suitable redundancy.

Natural Circulation - Saturation Conditions

The plant operators should be informed adequately at all times of those conditions in the reactor coolant system that might affect their capability to place the system in the natural circulation mode or to sustain it in such a mode. Information indicating that coolant pressure is approaching the saturation pressure corresponding to the core exit temperature would be especially useful, since an impending loss of overpressure would signal to the operator a potential loss of natural circulation. This information can be derived from available pressurizer pressure and hot leg temperature measurements, in conjuction with conventional steam tables.

The Committee recommends that information for detecting an approach to saturation pressure be displayed to the operator in a suitable form at all times. Since there may be several equally acceptable means of providing this information, there is no need for the NRC Staff to assign a high priority to the development of prescriptive requirements for such displays. However, a reasonably early request that licensees and vendors consider and comment on the need for such a display would be appropriate.

Core Exit Thermocouples

The NRC Staff should request licensees and vendors to consider whether the core exit temperature measurements might be utilized, where available, to provide additional indication regarding natural circulation or the status of the core. For the latter purpose, it is recommended that the full temperature range of the core exit thermocouples be utilized. At TMI-2, the temperatures displayed and recorded did not include the full range of the thermocouples.

The Committee believes it would be appropriate for the NRC Staff to request licensees and vendors to consider and comment on this recommendation. This request should be made as soon as convenient and the time allowed for responses should be such as not to degrade responses on higher priority matters. Plant changes that might result eventually from consideration of this recommendation would not at this time seem to require a high priority.

Instrumentation to Follow the Course of an Accident

The ability to follow and predict the course of an accident is essential for its mitigation and for the provision of credible and reliable predictions of potential offsite consequences. Instrumentation to follow the course of an accident in power reactors of all types has long been a concern of the ACRS, is the subject of Regulatory Guide 1.97 (which has not yet been implemented on an operating plant), and is the subject of an NRC Staff Task Action Plan for the resolution of generic issues.

The Committee believes that the positions of Regulatory Guide 1.97 should be reviewed, and redefined as necessary, and that the Task Action Plan should be reexamined, as soon as manpower is available. The lessons learned from TMI-2 should be the bases for these reviews. For example, improved sampling procedures under accident conditions should be considered.

Although review and reexamination of existing criteria may take some time, the studies completed to date, together with the understanding gained from the accident at TMI-2, should provide sufficient basis for planned and appropriately phased actions. The Committee believes that the installation of improved instrumentation on operating reactors of all types should be underway within one year.

Reactor Safety Research

The ACRS recommends that safety research on the behavior of light-water reactors during anomalous transients be initiated as soon as possible and be assigned a high priority. The ACRS would expect to see plans and proposals within about three months, preliminary results within an additional six months, and more comprehensive results within a year.

Of particular interest would be the development of the capability to simulate a wide range of postulated transient or accident conditions, including various abnormal or low probability mechanical failures, electrical failures, or human errors, in order to gain increased insight into measures that can be taken to improve safety.

The new program of research to improve reactor safety has been initiated only recently, and then only on a relatively small scale. The Committee reiterates its previous recommendations that this program be pursued and its expansion sought by the Commission with a greater sense of urgency.

Status Monitoring

Although the closed auxiliary feedwater system valves may not have contributed directly or significantly to the core damage or environmental releases at TMI-2, the potentially much more severe consequences of unavailability of engineered safety features in plants of any type is of concern and deserving of attention. Status monitoring not dependent chiefly on administrative control, and thus possibly less subject to human error, might help assure the availability of essential features.

A request should be made within the next few months that licensees consider additional status monitoring of various engineered safety features and their supporting services. The NRC Staff should begin studies on the advantages and disadvantages of such monitoring on about the same time scale. Responses from licensees should be expected in about one year, at which time the NRC Staff should be in a position to review and evaluate them.

The Committee recognizes that some of the recommended actions in this report have already been taken by the NRC Staff.

Sincerely,

Max W. Carbon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 16, 1979

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: INTERIM REPORT NO. 3 ON THREE MILE ISLAND NUCLEAR STATION UNIT 2

Dear Dr. Hendrie:

During its 229th meeting, May 10-12, 1979, the Advisory Committee on Reactor Safeguards continued its review of the recent accident at Three Mile Island Nuclear Station Unit 2 (TMI-2), including implications drawn from the occurrence of this accident. The Committee has several additional recommendations to make at this time.

Reactor Pressure Vessel Level Indication

The Committee believes that it would be prudent to consider expeditiously the provision of instrumentation that will provide an unambiguous indication of the level of fluid in the reactor vessel. We suggest that licensees of all pressurized water reactors be requested to submit design proposals and schedules for accomplishing this action. This would assure the timely availability of reviewed designs if the Staff ongoing studies should indicate that early implementation is required. The Committee believes that as a minimum, the level indication should range from the bottom of the hot leg piping to the reactor vessel flange area.

Operator Training and Qualification

The NRC Staff should examine operator qualifications, training, and licensing to determine what changes are needed. Consideration should be given to educational background, to training methods, and to content of the training program. Attention should also be given to testing methods, with specific concern for the ability of the testing methods to predict operator capability. Examination of licensing procedures should determine whether they are responsive to new information that is developed about plant or operator performance. Effort should also be made to determine whether results of examinations can be correlated with operator ability. Requalification training and testing should be similarly examined to insure that they take account of information that is developed by operation in the plant, and to determine that relevant information about other plants is made available to operators, and is made part of the training and requalification program. As part of this and of other more extensive studies, continuing attention must be given to the amount of information which an operator can assimilate and use in normal and in emergency situations and to the best method of presenting the information to the operator. The use and limitations of simulators for operator training should receive careful consideration.

Evaluation of Licensee Event Reports

Because of the potentially valuable information contained in Licensee Event Reports (LERs), the Committee recommends that the NRC Staff establish formal procedures for the use of this information in the training of supervisory and maintenance staffs and in the licensing and requalification of operating personnel at commercial nuclear power plants. The information in LERs may also be useful in anticipating safety problems. At the present time some utilities routinely request that they be provided copies of all LERs applicable to plants of the type they operate or to specific systems and components in a given class of plants similar to their plant. Certain reactor vendors have made similar requests and use the LERs to review and evaluate the performance of their plants. In addition, the NRC operator licensing staff has indicated that they use LERs in reviewing operating experience at commercial facilities.

The large number of LERs that attribute the cause to personnel error would tend to indicate that a formalized program of LER review would be useful in the training, licensing and requalification of nuclear power plant personnel. The extent to which such a program could be used to anticipate safety problems should also be considered.

Operating Procedures

Safety aspects of individual reactors during normal operation and under accident conditions are reviewed in detail by the NRC Staff and discussed with the ACRS. Acceptable limits for normal operations are formalized by Technical Specifications, submitted by the licensee and approved by the NRC Staff. Operating procedures for severe transients have received less detailed review by the NRC Staff. It appears that such procedures would benefit from review by an interdisciplinary team which includes personnel expert both in operations and in system behavior. Also, for the longer term, there may be merit in considering the development of more standardized formats for such procedures.

Reliability of Electric Power Supplies

During the past several years there have been several operating experiences involving a loss of AC power to important engineered safeguards. The ACRS believes it important that a comprehensive reexamination be made by the NRC and the reactor licensees of the adequacy of design, testing, and maintenance of offsite and onsite AC and DC power supplies. In particular, failure modes and effects analyses should be made, if not already performed, more systematic testing of power system reliability, including abnormal or anomalous system transients, should be considered, and improved quality assurance and status monitoring of power supply systems should be sought.

Analysis of Transients

The ACRS recommends that each licensee and holder of a construction permit be asked to make a detailed evaluation of his current capability to withstand station blackout (loss of offsite and onsite AC power) including additional complicating factors that might be reasonably considered. The evaluation should include examination of natural circulation capability, the continuing availability of components needed for long-term cooling, and the potential for improvement in capability to survive extended station blackout.

The ACRS also recommends that each licensee and construction permit holder should examine a wide range of anomalous transients and degraded accident conditions which might lead to water hammer. Methods of controlling or preventing such conditions should be evaluated, as should research to provide a better basis for such evaluations. The Committee expects it would be appropriate to have such studies done generically first, for classes of reactor designs and system types.

Emergency Planning

An effort should be undertaken to plan and define the role NRC will play in emergencies and what their contribution and interaction will be with the licensee and other emergency plan participants including other government agencies, industry representatives, and national laboratories. Such planning should consider:

- assurance that formal documentation of plans, procedures and organization are in place for action in an emergency,

- designation of a technical advisory team with names and alternates for the anticipated needs of an emergency situation,

- compilation of an inventory of equipment and materials which may be needed for unusual conditions including its description, location, availability and the organization which controls its release.

The Committee recommends that each licensee be asked to review and revise within about three months:

- his bases for obtaining offsite advice and assistance in emergencies, from within and outside the company,

- current bases for notifying and providing information to authorities offsite in case of emergency.

This review and evaluation should be in terms of accidents having a broad range of consequences. The results of this review should be reported to the NRC.

Decontamination and Recovery

The Committee wishes to call attention to the importance of a program designed to learn directly about the behavior, failure modes, survivability, and other aspects of component and system behavior at TMI-2 as part of the long-term recovery process. This program should also examine the lessons learned at TMI-2 to determine if design changes are necessary to facilitate the decontamination and recovery of major nuclear power plant systems.

Safety Review Procedures

The TMI-2 accident has imposed large new pressures on the availability of manpower resources within the NRC Staff. If progress is to be expedited on the new questions which have arisen and on existing unresolved safety issues, the ACRS believes that new mechanisms should be sought and implemented. For those safety concerns where such a mechanism is appropriate the Committee recommends that the Commission should request licensees to perform suitable studies on a timely basis, including an evaluation of the pros and cons, and proposals for possible implementation of safety improvements. The NRC Staff should concurrently establish its own capability to evaluate such studies by arranging for support by its consultants and contractors. In this fashion, the Committee anticipates that the information on which judgments will be based can be developed much more expeditiously, and an earlier resolution of many safety concerns may be achieved.

Capability of the NRC Staff

The Committee recommends that the capability of the NRC Staff to deal with basic and engineering problems in what may be termed broadly as reactor and fuel cycle chemistry be augmented expeditiously. This should include establishment of expertise within the NRC, with assistance arranged from consultants and contractors, in such important technical areas as the behavior of PWR and BWR coolants and other materials under radiation conditions; generation, handling and disposal of radiolytic or other hydrogen at nuclear facilities; performance of various chemical additives in containment sprays; processing and disposal techniques for low and high level radioactive wastes; chemical operations in other parts of the nuclear fuel cycle; and in the chemical treatment operations involved in recovery, decontamination, or decommissioning of nuclear facilities. The Committee wishes to emphasize the importance of providing this expertise in both the research and licensing management elements of the NRC.

Single Failure Criterion

The NRC should begin a study to determine if use of the single failure criterion establishes an appropriate level of reliability for reactor safety systems. Operating experience suggests that multiple failures and common mode failures are encountered with sufficient frequency that they need more specific consideration. This study should be accompanied by concurrent consideration of how the licensing process can be modified to take account of a new set of criteria as appropriate.

Safety Research

The ACRS believes that, as a result of the TMI-2 accident, various safety research areas will warrant initiation or much greater emphasis, as appropriate. The Committee suggests that consideration be given to an augmentation of the NRC safety research budget for FY 80.

Also, the Committee believes that a larger part of the safety research program should be oriented toward exploratory research as contrasted to confirmatory research, with some degree of freedom from immediate licensing requirements. The ACRS plans to have a Subcommittee meeting on this subject with representatives of the NRC Office of Nuclear Regulatory Research in the near future.

The Committee is continuing to review these matters and will report further as additional recommendations are developed.

Additional comments by Messrs. H. Lewis, D. Moeller, D. Okrent, and J. Ray are presented below.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

Additional Comments by Messrs. H. Lewis, D. Moeller, D. Okrent, and J. Ray

The potential for a reduction in risk to the public in the case of a serious reactor accident by the implementation of a means for controlled, filtered venting of a containment which could retain particulates and the bulk of the iodine has been recognized for more than a decade. The concept was recommended for study more recently in the American Physical Society Report on light-water reactor safety and in the Ford Foundation-Mitre Report, "Nuclear Power - Issues and Choices." It is a high priority item in the NRC plan submitted to Congress for Research to Improve the Safety of Light-Water Nuclear Power Plants (NUREG-0438). The study performed for the State of California on underground siting concluded that filtered, vented containment was a favored option to explore in connection with possible means to mitigate the consequences of serious reactor accidents. However, little progress has been made on the development of sufficiently detailed design information on which to evaluate the efficacy and other factors relevant to a decision on possible implementation of such consequence ameliorating systems.

The TMI-2 accident suggests that the probability of a serious accident in which a filtered vented containment could be useful is larger than many had anticipated.

We recommend that the Commission request each power reactor licensee and construction permit holder to perform design studies of a system which adds the option of filtered venting or purging of containment in the event of a serious accident. The system should be capable of withstanding a steam and hydrogen environment and of removing and retaining for as long a time as necessary radioactive particulates and the great bulk of the iodine for accidents involving degraded situations up to and including core melt. Such studies could be done generically for several reactor-containment types, and should evaluate the practicality, pros and cons, the costs, and the potential for risk reduction. A period of about twelve months for a report to the NRC by licensees and construction permit holders appears to represent a possible schedule.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 13, 1979

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: SHORT-TERM RECOMMENDATIONS OF TMI-2 LESSONS LEARNED TASK FORCE

Dear Dr. Hendrie:

During its 232nd meeting, August 9-11, 1979, the Advisory Committee on Reactor Safeguards completed a review of the short-term recommendations of the TMI-2 Lessons Learned Task Force as reported in NUREG-0578. These recommendations had been reviewed, in part, by an ACRS Subcommittee at a meeting in Washington, D.C., on July 27, 1979. During its review the Committee had the benefit of discussions with members of the Task Force. Comments from representatives of the nuclear industry were also considered.

In its review, the Committee has noted that the recommendations in NUREG-0578 are those deemed by the Task Force to be required in the short term to provide substantial additional protection for the public health and safety.

The Committee has considered both the recommendations themselves and the schedules proposed for their implementation. Regarding the latter, the Committee believes that the orderly and effective implementation and the appropriate level of review and approval by the NRC Staff will require a somewhat more flexible, and in some cases more extended, schedule than is implied by NUREG-0578.

With regard to the requirements themselves, the Committee agrees with the intent and substance of all except those discussed below.

2.1.5 Post-Accident Hydrogen Control Systems

a. The Committee agrees with the recommendations relating to dedicated penetrations for external recombiners or purge systems for operating plants that have such systems.

b. and c. The majority of the Task Force has recommended rule-making to require inerting of BWR Mark I and II reactors. A minority of the Task Force has recommended rule-making to require that all operating light water reactors provide the capability to use a hydrogen recombiner.

The Committee believes that questions relating to hydrogen generation during and following an accident, the rate and amount of generation, the need to control it, and the means of doing so, need to be reexamined. The Task Force has advised the Committee that it is considering this question further in connection with its longer-term recommendations which are scheduled to be completed by September, 1979. The ACRS believes that decisions concerning possible additional measures to deal with hydrogen should be deferred pending early evaluation of the forthcoming longer-term Task Force recommendations.

2.1.8 Instrumentation to Follow the Course of an Accident

With regard to instrumentation to follow the course of an accident, the ACRS believes that containment pressure, containment water level, and on-line monitoring of hydrogen concentration in the containment should also be considered for implementation for all operating reactors on the same schedule as that recommended by the Lessons Learned Task Force.

2.2.1.b Shift Technical Advisor

The Committee agrees completely with the two closely related objectives of this recommendation. One relates to the presence in the control room during off-normal events of an individual having technical and analytical capability and dedicated to concern for safety of the plant. The other relates to the need for an on-site, and perhaps dedicated, engineering staff to review and evaluate safety-related aspects of plant design and operation. The achievement of these objectives will contribute significantly to the safe operation of a plant.

The Committee believes that there may be difficulty in finding a sufficient number of people with the required qualifications and interest in shift work to fill the Technical Advisor positions. The Committee therefore believes the solution proposed by the Staff should not be mandatory but that alternate solutions also should be considered.

2.2.3 Revised Limiting Conditions for Operation

The Committee agrees with the findings of the Task Force that there are too many human or operational errors resulting in the defeat of an entire safety system, that the number of such occurrences should be and can be reduced, and that the ultimate responsibility for doing this must rest with the licensee.

The Committee, however, is not convinced that the Task Force proposal is the best or only way to increase the licensee's awareness of the need to improve operational reliability, and suggests that measures short of shutdown, such as a rule that requires actions similar to those of a show-cause order, may be equally effective.

Sincerely,

Max W. Carbon
Chairman

References:

1. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, July 1979.
2. Letter, D. Knuth, President, KMC, Inc., to Harold Denton, Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, August 8, 1979, Subject: TMI-2 Lessons Learned Task Force Report (NUREG-0578).
3. Letter, Stanley Ragone, President, Virginia Electric and Power Company, to Joseph M. Hendrie, Chairman, U.S. Nuclear Regulatory Commission, August 8, 1979, Subject: Lessons Learned Task Force on TMI-2, NUREG-0578.
4. Letter, Floyd W. Lewis, Chairman, Ad Hoc Nuclear Oversight Committee, to Harold R. Denton, Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, August 1, 1979, Subject: Lessons Learned from TMI-2.
5. Letter, American Nuclear Society, ANS-3 Committee, to Joseph M. Hendrie, Chairman, U.S. Nuclear Regulatory Commission, August 2, 1979, Subject: Lessons Learned Task Force Status Report NUREG-0578.
6. Letter, Robert Szalay, Atomic Industrial Forum, Inc. (AIF), to Harold Denton, Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, August 2, 1979, Subject: "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," (NUREG-0578).
7. Report by the AIF Policy Committee on Follow-up to the Three Mile Island Accident, July 5, 1979.
8. Memorandum, C. G. Long, Lessons Learned Task Force Member, to R. J. Mattson, Director, TMI-2 Lessons Learned Task Force, July 30, 1979, Subject: Review of LERs for Loss of Safety Function Due to Personnel Error and Defective Procedures, (50-320).

REVISED

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 14, 1979

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: NUREG-0600 "INVESTIGATION INTO THE MARCH 28, 1979 THREE MILE ISLAND ACCIDENT BY OFFICE OF INSPECTION AND ENFORCEMENT"

Dear Dr. Hendrie:

During its 235th meeting, November 8-10, 1979, in accordance with the Commission's request, the Advisory Committee on Reactor Safeguards completed its review of NUREG-0600. The report was also discussed at a Subcommittee meeting in Washington, D. C. on October 30, 1979. During its review the Committee had the benefit of discussions with the Nuclear Regulatory Commission (NRC) Inspection and Enforcement (I&E) Staff, and of comments from the licensee.

The stated scope of NUREG-0600 is limited to investigation of the licensee's operational actions prior to and during the course of the accident, and his actions to control release of radioactive materials and to implement his emergency plan during the course of the accident. Consistent with this limitation, emphasis is placed on departure from Technical Specifications prior to the accident and departure from the licensee's procedures during the course of the accident, with little consideration of other factors.

Other investigations and other NRC task force studies have considered not only the actions taken by the licensee, but also other facets of the accident, including peculiarities of the nuclear steam supply system that tended to inhibit recovery or to confuse the operators by leading to pressure and level conditions not anticipated by the written procedures, and deficiencies of the control room and system design that degraded the quality of information available to the operator. Additional details not in NUREG-0600 can be found, for example, in a report entitled "Analysis of Three Mile Island Unit 2 Accident" (NSAC-1, July 1979) prepared by the Electric Power Research Institute, Nuclear Safety Analysis Center.

NUREG-0600 includes a factual chronology with event descriptions, and a finding of operational and administrative shortcomings and errors. It concludes (Appendices IB and IIF) that a total of 36 items of potential operational or administrative noncompliance existed. The Office of Inspection and Enforcement subsequently, by letter of October 25, 1979 to Metropolitan Edison Company, imposed fines for seventeen violations, infractions and deficiencies, many of them multiple occurrences.

Because the limited scope of the report tends to lead to a catalog of violations with only limited recognition of other factors that contributed to errors by the operators, the Committee has some concern that it may be concluded from the charges of failure to follow accident procedures that such failure is automatically a violation.

Accident procedures are prepared by the licensee and are not approved by NRC, but the licensee is required to follow them. The Committee believes that an accident procedure cannot be sufficiently detailed to encompass every possible sequence of events, and that it must be based on the assumption that a particular set of conditions exists; a deviation from this set of conditions may make it necessary to depart from the procedure. As an example, TMI-2 Emergency Procedure 2202-1.3 (Loss of Reactor Coolant/Reactor Coolant System Pressure) which is referred to in NUREG-0600, is believed by the Committee to include confusing symptoms and instructions for the case of a loss of reactor coolant at the top of the pressurizer. Likewise TMI-2 Emergency Procedure 2202-1.5 (Pressurizer System Failure) which calls for pressurizer level control is believed to be unacceptable for the TMI-2 accident or for any other loss of reactor coolant at the top of the pressurizer. The question, therefore, arises whether an operator, using his best judgment, is guilty of a violation if he consciously takes an action that is at variance with procedures which in themselves may contain confusing or incorrect guidance. The Committee believes that, if so, this is the wrong approach to protecting the health and safety of the public during an emergency and that the operator, guided by the written procedures, his training, and available technical advice, should be allowed to use his best judgment to deal with the problem. His judgment will obviously be subject to post-factum appraisal.

The Committee has found this report less than satisfactory, and its title misleading, chiefly because of limitations in its predefined scope. For this reason, the Committee recommends the preparation and issuance of a summary report that consolidates and integrates the findings of the several NRC Task Forces that have investigated and reported on this accident.

Sincerely,

Max W. Carbon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 14, 1979

Honorable Peter A. Bradford
Commissioner
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Bradford:

In your letter of October 9, 1979 to the Advisory Committee on Reactor Safeguards you referred to the Committee's letter to Chairman Hendrie of August 13, 1979 concerning "Short-Term Recommendations of TMI-2 Lessons Learned Task Force" and noted the ACRS statement that "orderly and effective implementation and the appropriate level of review and approval by the NRC Staff will require a somewhat more flexible, and in some cases more extended, schedule than is implied by NUREG-0578." You asked that the ACRS "identify in more detail which of the scheduled items the Committee believes should be extended and the basis for those recommendations."

The ACRS comment was intended as a general observation. The Committee was not favoring any unnecessary delays. However, the Committee anticipated that exceptions to the original schedule might be desirable or even necessary. For example, with regard to the Shift Technical Advisor, the Committee anticipated that not all licensees would be able to obtain within the time specified the services of sufficiently qualified personnel for three-shift, seven-days-a-week duty, including provisions for the ongoing training which is called for and appropriate to the task. In this respect, the Committee believes that, where licensees are not able to comply with the NRC requirements on schedule, they should be required to submit temporary alternative proposals for approval by the Staff.

Other items, such as the establishment of an onsite technical support or operational support center may also be difficult to achieve at all operating reactors by the scheduled time. In addition, some items of equipment or instrumentation may not be available on the time schedule proposed.

Furthermore, some of the changes will require shutdown of the reactor. Some grouping of such changes is likely to be desirable to limit the number of transients associated with shutdowns that are required for this purpose.

The ACRS does not believe public safety will be unduly jeopardized by extending the implementation schedule for some reasonable period.

Sincerely yours,

Max W. Carbon
Chairman

cc: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Ahearne
Samauel Chilk

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 13, 1979

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON TMI-2 LESSONS LEARNED TASK FORCE FINAL REPORT

Dear Dr. Ahearne:

The TMI-2 Lessons Learned Task Force has issued its Final Report, NUREG-0585. The ACRS provides comments herein both on the specific recommendations made by the Task Force and on related subjects. The Committee will first address the recommendations made in NUREG-0585.

1. Personnel Qualifications and Training.

The ACRS gives general support to the recommendations made in this category.

The ACRS believes that, although a broader technical background should be required of Shift Supervisors, it may be neither necessary nor practical to require that all Shift Supervisors have a Bachelor of Science Degree. The Committee recommends that the NRC define its criteria for "equivalent training and experience in engineering or the related physical sciences." The ACRS believes that a training program tailored to the requirements of reactor operation, possibly of less than four years duration, may provide a practical alternative to a formal degree program. The Committee believes that the NRC should define the scope and duration of a training program that may be considered as an acceptable alternative to a degree curriculum. The ACRS also recommends that, if the Technical Advisor system proves satisfactory, consideration should be given to offering licensees the option of retaining that system instead of upgrading the academic education of Shift Supervisors to the specified level.

The ACRS recommends that the adequacy of staffing in the NRC Operator Licensing Branch be reevaluated with respect to the number of personnel and breadth of their background.

The Committee believes that additional emphasis must be given to the determination of what constitutes an adequate degree of in-house technical capability for each licensee and assurance of the continuing development of such capabilities. The ACRS also believes that attention must be given to providing, on a continuing basis, technical backup to review safety-related design changes or to provide assistance under accident conditions by a group having the depth of technical knowledge which exists in the organization of the nuclear steam system supplier and a well-qualified architect-engineer during the period while the plant is being designed.

2. Staffing of Control Room.

   The ACRS supports this recommendation.

3. Working Hours.

   The ACRS supports this recommendation.

4. Emergency Procedures.

   The ACRS, in general, gives strong support to this recommendation. However, the Committee believes that the emergency procedures at licensed power reactors should receive priority. The ACRS recommends that the licensees should give priority to the development of improved emergency procedures with the aid of expert, interdisciplinary review groups and that the NRC Staff should review, in depth, the existing and proposed, emergency procedures for a large sample of licensed reactors on a priority basis.

   The knowledge developed from the concurrent industry and NRC efforts should be used to revise, in a timely fashion, the emergency procedures of all operating plants.

5. Verification of Correct Performance of Operating Activities.

   The ACRS gives general support to this recommendation.

6. Evaluation of Operating Experience.

   The ACRS gives general support to these recommendations.

   Additional Committee comments on this subject are contained in NUREG-0572, "Review of Licensee Event Reports (1976-1978)."

7. Man-Machine Interface.

   The ACRS gives general support to these recommendations.

   In addition to the nine items listed in NUREG-0585, Appendix A, Section 7.1 , the Committee recommends that the licensee should include in his evaluation the data recording requirements and recall capabilities of the minimum set of plant parameters that defines the safety status of a nuclear power plant.

8. Reliability Assessments of Final Designs.

The ACRS strongly supports the application of reliability assessments to final designs. The Committee supports the Integrated Reliability Evaluation Program (IREP) which is being initiated by the Office of Nuclear Regulatory Research. However, the Committee does not agree that the proposed IREP will fully satisfy the need. The ACRS recommends that the NRC develop a program in which licensees acting individually or jointly develop reliability assessments of their plants, in addition to the NRC IREP, which should be performed concurrently.

If the reliability assessments were performed in the manner proposed above, it would accelerate obtaining potentially significant safety information and expedite the development of the basis for changes, should they be necessary. It would also provide the operating organizations with better technical insight into the safety of their plants and would provide the benefits to be derived by separate studies of system reliability.

9. Review of Safety Classifications and Qualifications.

The ACRS supports this recommendation. A particular problem warranting early attention is the qualification of operator information systems. More generally, the Committee believes that more than a year will be needed to accomplish the overall task, partly because of its breadth and depth, and partly because of the very considerable number of knowledgeable personnel which would be needed.

The Committee agrees that completion of the overall task should not be made a condition for the licensing of new plants.

10. Design Features for Core-Damage and Core-Melt Accidents.

The ACRS supports this recommendation. However, the Committee believes that the recommendation should be augmented to require concurrent design studies by each licensee of possible hydrogen control and filtered venting systems which have the potential for mitigation of accidents involving large scale core damage or core melting, including an estimate of the cost, the possible schedule, and the potential for reduction in risk.

The ACRS agrees with the recommendation made by the Lessons Learned Task Force in NUREG-0578 that the Mark I and Mark II BWR containments should be inerted while further studies are made of other possible containment modifications in accordance with the general recommendations in this category. The ACRS also recommends that special attention be given to making a timely decision on possible interim measures for ice-condenser containments.

The Committee also recommends that special attention be given to operating reactors located at densely populated sites.

11. Safety Goal for Reactor Regulation.

    The ACRS supports this recommendation.

12. Staff Review Objectives.

    The ACRS supports this recommendation. However, the ACRS believes that there is a need for review of NRC safety rules, regulations, guides and philosophy on a regular basis in order to ascertain various matters including the following:

    a. Does an appropriate balance exist in the expenditure of NRC financial and manpower resources among the various research areas, on the resolution of safety issues, on the legal requirements of licensing, and on inspection and enforcement?

    b. Is there an appropriate division of effort and responsibility between industry and the NRC?

    c. Has an undesirable inflexibility in the approach to safety developed due to previous decisions, or for other reasons?

    d. Are there any important gaps in the existing safety review process? Is there a mechanism for searching out such gaps?

13. NRR Emergency Response Team.

    The ACRS gives general support to these recommendations. The Committee believes that the timing of implementation should be more flexible. The Committee believes that better definition of the NRC role and responsibilities in an emergency will have an influence on the determination of the makeup, training and abilities of an NRC emergency response team.

The ACRS wishes to make some comments and recommendations on several matters not directly addressed in NUREG-0578 or NUREG-0585.

1. The ACRS believes that the lessons learned from the TMI accident should be viewed in a broader perspective. The Committee agrees that the TMI accident shows a need for considerable improvement

in reactor operations and in knowledge of the behavior of a plant during a wide range of transients. However, the Committee believes that there are other potentially important contributors to the probability of a reactor accident, and they should also receive priority attention.

Reliability assessments and systems interactions studies, as discussed under recommendations 8 and 9 above, should serve this function in part. However, there is a need also to consider, in some more systematic way, methods to uncover significant design errors, to detect system or component degradation, and to test systems under conditions more closely simulating the range of situations which might result from transients and accidents.

2. The Task Force has not addressed the need to reexamine the adequacy of the current design basis for emergency cooling recirculating systems, as recommended by the ACRS in its report of August 14, 1979 on "Studies to Improve Reactor Safety."

   There are several other specific recommendations made by the ACRS in its interim reports Nos. 2 and 3 on Three Mile Island both dated May 16, 1979 and in its report of August 14, 1979 on studies to improve reactor safety. The Committee believes that the NRC Staff should address each such recommendation in formulating its overall action plan.

3. The ACRS recommends that a reevaluation should be made of the potential influence of a serious accident involving significant atmospheric release of radioactive materials from one unit of a multiple unit site on the ability to maintain the other units in a safe shutdown condition.

4. The ACRS recommends that the industry and the NRC Staff undertake studies to ascertain what contingency design measures, beyond those covered in the Task Force recommendations, may ensure improved capabilities for recovering from or mitigating the effects of accidents beyond the design basis. For example, in some cases, it may be possible to provide alternative measures in the event of loss of the safety grade ultimate heat sink for an extended period of time.

5. The ACRS recommends that the NRC Staff give attention to the seismic implications of TMI, for example, the seismic qualifications of auxiliary feedwater supplies, the acceptability of failure of nonseismic Class 1 equipment, and the suitability of emergency procedures for earthquakes.

6. The ACRS recommends that greater consideration be given to the provision of dedicated shutdown heat removal sytems, and to the potential merits of having a shutdown heat removal system capable of operating at normal system pressure.

The ACRS expects to address other considerations of reactor safety and the regulatory process in a separate report.

Sincerely,

Max W. Carbon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 15, 1980

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: DRAFT NUREG-0660, "ACTION PLANS FOR IMPLEMENTING RECOMMENDATIONS OF THE PRESIDENT'S COMMISSION AND OTHER STUDIES OF THE TMI-2 ACCIDENT"

Dear Dr. Ahearne:

During its 237th meeting, January 10-12, 1980, the Advisory Committee on Reactor Safeguards reviewed Draft NUREG-0660, dated December 10, 1979. The draft had previously been discussed at an ACRS Subcommittee meeting in Washington, D.C., on January 7, 1980. During its review, the Committee had the benefit of discussions with the NRC Staff.

The draft is a compilation of recommendations made by the several organizations and commissions that have investigated the TMI-2 accident. The Committee understands that a primary purpose of the document is to establish criteria for termination of the pause in licensing. Other purposes are to provide a complete action plan relating to all the unresolved issues and unimplemented recommendations from the lessons learned from the TMI-2 accident, and to establish priorities and requirements of funds and manpower. The draft gives preliminary target dates and estimates of the necessary resources, but does not yet recommend priorities.

The Committee believes the Plan is comprehensive, but not selective; this comprehensiveness serves to dilute the items important to safety, and therefore important to termination of the licensing pause. In the absence of priorities and identification of the items that the NRC Staff considers important, the ACRS finds it difficult to make objective comments on the Plan. The Committee understands that the Staff is proceeding to develop priorities and identification of items of primary importance, and the Committee will expect to review the important aspects of the Plan when this has been done.

The Committee is also concerned that preoccupation with the Plan may lead to neglect of pre-TMI-2 accident safety concerns, some of which are of long standing and of greater importance than some of the listed items. It is important to establish priorities on an overall consideration of both "old" and "new" items.

The Plan lists a large number of proposed changes in plant equipment, plant staffing, operating procedures, and licensing requirements. The ACRS believes that the scheduled time for establishing a complete plan setting detailed requirements for all items is too short to give reasonable assurance that all changes will be in the direction of greater safety. In illustration of this concern, the Committee points to the controversy that arose over the directive prohibiting tripping of the reactor coolant pumps following high pressure injection initiation.

The Committee believes that a two step process is more appropriate in developing the Action Plan. On an expedited basis, the Staff should develop those recommendations for safety improvement that it believes can and should be adopted as requirements for a termination in the pause in licensing. On a longer but defined time schedule, the Staff should develop a plan for dealing with other issues and implications of the TMI-2 accident.

Additional comments by member H. Lewis are presented below.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Additional Comments by Member H. Lewis

The letter of January 5, 1980 from L. V. Gossick, Executive Director for Operations, to the Commissioners describes the Action Plan as the complete list of all actions necessary as a result of the accident at TMI-2, and states that complete approval of the Plan, in its entirety, by the Commission, should be regarded as a prerequisite for the resumption of licensing. The Staff has further told us that, though they plan to assign priority scores to the items on the list (through a scoring system of dubious relevance), it is expected that all items on the list will be accomplished, in time.

It is my view that such an unselective approach to the lessons of TMI-2 is inappropriate, and that the Plan consists of an uncritical listing of anything anyone has suggested be done in the aftermath of (not necessarily as a result of) the accident at TMI-2. In particular, the Plan provides no guidance, and reflects no analysis, with respect to the safety relevance of the items, or even whether they would enhance safety. I believe adoption of the Plan would make no demonstrated contribution to a reordering of NRC priorities toward those safety weaknesses highlighted in the various reports on TMI-2.

It would be preferable to bite the bullet, and identify those twenty items that need attention, in terms of their impact on safety, as determined by any reasonable analysis. This has not been done, nor is it contemplated.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 11, 1980

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: NUREG-0660 DRAFT 2, "ACTION PLANS FOR IMPLEMENTING RECOMMENDATIONS OF THE PRESIDENT'S COMMISSION AND OTHER STUDIES OF THE TMI-2 ACCIDENT"

Dear Dr. Ahearne:

On February 7, 1980, during its 238th meeting, the ACRS received additional information from Messrs. Denton and Mattson on the status of the Action Plans and the requirements for near term operating licenses (NTOL). The Committee was advised that a large number of NTOL items, including the TMI-2 related NRC Bulletins and Orders, had been approved as a minimal set earlier that day.

The ACRS believes that its input into this process has been largely ignored by the Commission and is concerned that the "rush to judgment" on those important matters may result in, at worst, error, and at best inefficient use of resources important to safety.

During its January 1980 meeting, the ACRS had received a briefing on the Draft Action Plans (following a subcommittee meeting on the same subject) and sent you a letter, noting the lack of priorities within the Plans and the lack of an adequate method to establish such priorities. We further stated that we expected to see and to review the Plans when this had been accomplished.

In view of our letter, the ACRS was surprised to learn that the Staff had requested, and the Commission had approved, a large set of NTOL items without ACRS comment, while an ACRS meeting was in progress. While the Committee recognizes the needs and pressures for action, we believe it is important to be sure that a reasonable rationale exists for the setting of priorities, that there is reasonable assurance that there are no adverse safety effects from new requirements, and that the limitations on total resources have been carefully factored into the decision making.

A principal concern is that a very large number of operational and hardware changes are being mandated with, in most cases, little analysis to establish their safety relevance or impact. Design and operational stability is itself a safety asset and, confident though we are in the engineering judgment of the Staff, we think that there would be merit in ACRS review before, not after adoption.

The ACRS will not be ready to provide its advice on the recommendations of the Bulletins and Orders Task Force until it can hold an additional Subcommittee meeting which will include a discussion of questions that have been raised by reactor vendors and operators.

Messrs. Denton and Mattson also stated on February 7 that they were not sure whether the ACRS would be asked to comment on the final Action Plans before the Commission was asked for its approval. The NRC Staff schedule for the availability of Draft 3 of the Action Plans is not firm. The ACRS is planning to meet with the NRC Staff on the Action Plans at its March meeting if the Committee receives Draft 3 in time. However, there appears to be the element of a timing problem which the Commission must consider in deciding whether, how, and when ACRS input in the decision-making process will be obtained.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 11, 1980

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: ACRS REPORT ON NEAR-TERM OPERATING LICENSE ITEMS FROM DRAFT 3 OF NUREG-0660, NRC ACTION PLANS DEVELOPED AS A RESULT OF THE TMI-2 ACCIDENT

Dear Dr. Ahearne:

In your letter of February 19, 1980 you asked the ACRS to provide its position on whether the NRC Staff Near-Term Operating License (NTOL) list was a necessary and sufficient set of supplementary requirements for authorizing operating licenses. During your meeting with the ACRS on March 6, 1980, there was considerable discussion of the terms "necessary and sufficient," and there was agreement that a definition of these terms in the applicable context is subjective. Reasonable people might conclude that a list half as long would be sufficient, and other reasonable people might require a much longer one. We have, therefore, not sought a collegial definition of the terms, but have instead interpreted your request to be that we look at the list and ask if it is reasonable. We have reviewed the list, item by item, for reasonableness, and the remainder of this letter should be interpreted in that sense.

The ACRS review of the NTOL items, Table A.1 of Draft 3 of NUREG-0660, "NRC Action Plans Developed as a Result of the TMI-2 Accident," was performed during the 239th meeting of the ACRS March 6-8, 1980. A Subcommittee had met with the NRC Staff on March 5, 1980. The Committee had the benefit of discussions with the NRC Staff and with industry representatives who had participated in an intensive Atomic Industrial Forum study of the NTOL proposals as outlined in Draft 2 of NUREG-0660.

The following NTOL items are from Table A.1 of Draft 3 of the Plans.

- Part 1, Requirement (3), Item I.B.1.2, <u>"Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants."</u>

  The Committee is concerned about the specification as an NTOL requirement of an "Interoffice NRC review of licensee management to determine organizational and managerial capabilities, using internal NRC draft criteria pending development of formal criteria." If it is to be assumed that this requirement refers to utility management (rather than plant management), then it appears that assurance of competent management should be obtained as soon as feasible for all utilities that are operating power reactors, independently of NTOL activity. Coupling this determination to an operating license (OL) appears logical only if the reactor is the first to be operated by the applicant.

The Staff has indicated that the criteria for judging management capability are in an early state of development. The ACRS recommends that due regard be given to the need for a learning period in developing and applying the criteria, and that there be a continuing effort to make the criteria as clear as possible to those organizations being evaluated.

- Part 1, Requirement (4), Item I.B.1.2, "Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants."

  The ACRS endorses the objective of improving the engineering capability onsite, but has not studied the criteria that will be used to qualify the group.

- Part 1, Requirement (6), Item I.C.7, "NSSS Vendor Review of Procedures."

  With respect to Emergency Procedures, the ACRS recommends that Architect-Engineers (AE) or the AE component of the operating utility also be required to review and verify the adequacy of such procedures in the context of accuracy and completeness to meet emergency conditions, including the specifications of actions to deal with inadequacies in the single failure criterion.

- Part 1, Requirement (7), Item I.C.8, "Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants."

  To ensure against relaxation of continuous vigilance to meet emergencies, the Committee recommends nonscheduled random checking of operating personnel in respect to verifying their ability to meet unanticipated accident conditions.

- Part 1, Requirement (11), Item II.K.1, "IE Bulletins on Measures to Mitigate Small Break LOCAs and Loss of Feedwater Accidents."

  This list includes some items which are useful, some which are of marginal merit and some which may, upon deeper analysis, turn out to have been wrong. Among those that deserve more careful analysis are: criteria for early RCS pump trip; criteria for HPSI termination; automatic PORV blocking; several requirements that increase scram frequency; subcooling meters (versus void-meters); etc. Each of these is a subject in itself, deserving deliberate study.

- Part 1, Requirement (12), Item II.K.3, "Final Recommendations of B&O Task Force."

  Refer to the ACRS report dated March 11, 1980 on the Bulletins and Orders Task Force report, which documents some of our concerns.

- Part 1 Requirement (13), Item III D.3.4, "Control Room Habitability."

  The ACRS notes that this item merely sets a goal to "confirm compliance with existing Regulatory Guides and Standard Review Plan...." The TMI incident indicates that existing requirements to protect the occupants of the control room against radiation may not be adequate, particularly with respect to leakage control and arrangement of air intakes.

- Part 2, Requirement (4), Item I.C.1, "Short-Term Accident Analysis and Procedure Revision."

  The comments in the first sentence concerning Part 1, Requirement (11) regarding the need for careful analysis apply to a number of unresolved items in this requirement.

- Part 2, Requirement (15), Item II.E.4.1, "Containment Dedicated Penetrations."

  The ACRS recommends that, in design and location of penetrations for the recombiner, the Staff pay particular attention to the possibility of hydrogen accumulation at high points in the containment or containment compartments.

- Part 3, Requirement (4), Item III.A.3.1, "Role of NRC in Emergency Preparedness."

  We believe that the responsibility for handling an emergency should be clear and undiluted, and should rest with the utility. The NRC should be fully informed, prepared to intervene when necessary for the public health and safety, but should not, as a rule, take over responsibility in the event of an accident. This issue must be resolved.

In considering these matters, the ACRS also examined those NTOL requirements that have already been issued in the NRC letters of September 27, 1979 and November 9, 1979 to all pending operating license applicants. Included among this group are several requirements related to improved systems for measuring the concentrations of various contaminants both within containment and in effluent releases. Although the Committee endorses these requirements, it believes that more attention needs to be directed to assuring:

(a) That samples collected are representative with emphasis on the location and nature of the sample collector and the length, diameter, and specific nature of the sampling lines.

(b) The adequacy and reliability of the performance of the associated sampling and monitoring equipment.

The Committee wishes to comment at this time on two items in the Action Plans in order to recommend the initiation of actions which relate to the NTOL plants. In the Committee's opinion, the issuance of an operating license should not be contingent on completion of these matters.

1. In its letter of December 13, 1979 on the TMI-2 Lessons Learned Task Force Report, the ACRS supported the Integrated Reliability Evaluation Program (IREP). However, the ACRS went on to state, "The Committee does not agree that the proposed IREP will fully satisfy the need. The ACRS recommends that the NRC develop a program in which licensees, acting individually or jointly, develop reliability assessments of their plants in addition to the NRC IREP, which would be performed concurrently."

   The ACRS believes that, on an expedited but practical schedule, the NTOL plant owners, as well as current licensees, should be required to perform studies of the type referred to above.

2. In its letter of December 13, 1979, the ACRS supported the recommendation of the Lessons Learned Task Force concerning design features for core-damage and core-melt accidents. The ACRS further recommended that design studies of possible hydrogen control and filtered-venting systems for containment be required from licensees. The ACRS also recommended that special attention be given to making a timely decision on possible interim measures for ice-condenser containments. The ACRS recommends initiation of such studies for NTOL plants.

The ACRS has noted in previous letters that it is important that the improvements in safety proposed as a result of the Three Mile Island accident be considered in a broad perspective and that other matters of importance to safety receive proper priority. The ACRS believes it important that the diversion of resources needed to deal with NTOL related activities not produce neglect of problem areas which should have a high priority. The Committee expects to comment on this in detail when it reports on the NRC Action Plans.

The ACRS believes that, subject to the above comments, the NTOL items identified in the NUREG-0660, Draft 3, provide a satisfactory basis for the resumption of licensing.

Additional comments by ACRS Member H. Lewis are presented below.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Additional Comments by Member H. Lewis

Many items not called out above have still not received sufficient analysis, and silence on these items should not be construed as concurrence in the current Staff position. None of these uncertainties should, in my view, affect the resumption of licensing, but I believe that they should be resolved before the Staff position becomes too frozen.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 11, 1980

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Ahearne:

SUBJECT: RECOMMENDATIONS OF THE NRC TASK FORCE ON BULLETINS AND ORDERS

During its 239th meeting, March 6-8, 1980, the Advisory Committee on Reactor Safeguards completed a review of the recommendations of the NRC Task Force on Bulletins and Orders, hereafter called the Task Force. The ACRS Subcommittee on TMI-2 Accident Bulletins and Orders met with representatives of the NRC Staff and Utility Owners Groups on July 9, 1979, August 2, 1979, January 3-4, 1980, and March 4, 1980. The ACRS previously met with representatives of the Task Force at the Committee's meetings of October 4-6, 1979, January 10-12, 1980 and February 7-9, 1980.

The Task Force, formed in May 1979, was charged with reviewing and directing the TMI-2 related staff activities associated with the NRC I&E Bulletins, Commission Orders, and generic evaluations of loss of feedwater transients and small-break loss-of-coolant accidents for all operating plants to assure their continued safe operation. Specific review areas included systems reliability, vendor analysis methods and operating guidelines, plant procedures, and operator training. The results of the Task Force efforts have been reported in NUREG-0645, Volumes I and II, and a series of vendor specific reports noted below.

In its review, the Committee notes that the recommendations in reports NUREG-0565, 0611, 0623, 0626, and 0635 are those deemed by the Task Force to make the operating light water reactor plants less susceptible to core damage during accidents and transients which are coupled with systems failures and operator errors.

The Task Force has proposed that both the recommendations and the responsibility for their implementation be included in Section II.K.3 of NUREG-0660, "NRC Action Plans Developed As a Result of the TMI-2 Accident". The Committee agrees with this course of action.

With regard to the recommendations the Committee has the following comments:

- Reactor Coolant Pump Trip and High Pressure Injection (HPI) Termination Criteria: The NRC Staff has required prompt trip of the reactor coolant pumps in the event of a small-break LOCA. Recent transients at some operating plants have resulted in RCP trip for non-LOCA events and, in some cases, the use of the NRC approved procedures for HPI termination have resulted in PORV or safety valve actuation due to overfilling of the primary system. The NRC Staff should, in conjunction with the licensees, review the criteria for HPI termination and reactor coolant pump trip to reduce unnecessary challenges to the pressurizer safety valves and prevent unnecessary trips of the reactor coolant pumps which may increase the difficulty in establishing uninterrupted core cooling.

- Feed-and-Bleed Cooling of the Primary System: At the March 4, 1980 Subcommittee meeting, the NRC Staff said that there are presently no requirements for the use of feed-and-bleed cooling for decay heat removal. The Committee believes that the availability of a diverse heat removal path such as feed and bleed is desirable, particularly if all secondary-side cooling is unavailable. The ACRS has established an Ad Hoc Subcommittee to review this matter.

- Reduction of Challenges to the PORVs in B&W Plants: As a result of the TMI-2 accident, the NRC Staff has required that all B&W plants raise the PORV actuation setpoint and lower the high-pressure reactor trip setpoint in order to reduce the number of challenges to the PORV. While recent B&W operating reactor experience indicates that the PORV challenge rate has been reduced, there has been a corresponding increase in the number of reactor scrams. The Committee notes that an increase in the scram rate increases the probability of a deleterious impact on safety, and recommends that the NRC Staff continue to evaluate the overall impact of the above action on plant safety.

- Potential Unreviewed Safety Question with Regard to Automatic Initiation of the Auxiliary Feedwater System: Several utilities have raised the issue of a potential unreviewed safety question with regard to automatic initiation of the AFW system, in the event of a main steamline break inside containment. This issue should be reviewed.

The Task Force has recommended that the vendor methods used for small break LOCA analysis should be revised, documented and submitted for NRC review, and that plant specific calculations using NRC approved methods should be provided thereafter. The NRC Action Plans also include an item which recommends that the NRC develop and issue a position on required conservatisms in small break calculations. The Committee believes that the schedule used for developing a revised NRC approach to small break calculations should, if practical, be made compatible with the schedule required of the NSSS vendors for revising their small break models. This should lead to a more efficient use of available resources and may lead to an earlier development of improved analyses. This implies some increased flexibility in the schedule.

With regard to the schedules proposed for the implementation of these recommendations, the Committee believes that the orderly and effective implementation and the appropriate level of review and approval by the NRC staff will require a somewhat more flexible, and in some cases more extended, schedule than is implied by the Task Force reports.

The Committee is still reviewing the NRC Action Plans which we understand will include the Task Force's recommendations discussed above, as well as many other recommendations.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants", USNRC Report NUREG-0565, January 1980.
2. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants", USNRC Report NUREG-0611, January 1980.
3. U.S. Nuclear Regulatory Commission, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors", USNRC Report NUREG-0623, November 1979.
4. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications", USNRC Report NUREG-0626, January 1980.
5. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants", USNRC Report NUREG-0635, January 1980.
6. U.S. Nuclear Regulatory Commission, "Report of the Bulletins and Orders Task Force", USNRC Report NUREG-0645, Volumes I-II, January 1980.
7. U.S. Nuclear Regulatory Commission, "NRC Action Plans Developed As a Result of the TMI-2 Accident", USNRC Report NUREG-0660, Draft 3, March 5, 1980.

April 17, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: NUREG-0660, "NRC ACTION PLANS DEVELOPED AS A RESULT OF THE TMI-2 ACCIDENT," DRAFT 3

Dear Dr. Ahearne:

The ACRS reported on its review of the Near-Term Operating License Items of NUREG-0660 on March 11, 1980 and completed its review of Draft 3 of the Action Plan during its 240th meeting, April 10-12, 1980. The Committee had the benefit of discussions with the NRC Staff. A Subcommittee met with the NRC Staff to review the Plan on April 1 and 2, 1980, and also met with representatives of the General Electric Company and the Vermont Yankee Nuclear Power Corporation on April 2, 1980.

The Committee believes that the Plan, as represented by the third draft, is a generally well-balanced document that establishes reasonable priorities. The ACRS recognizes that it would be impractical for the NRC Staff to expand the descriptions in NUREG-0660 to convey the detailed scope of each listed item; however, the Committee wants to be sure that sufficient emphasis is being placed on particular aspects of some of the items listed.

The ACRS believes the Plan to be deficient in the following aspects:

- <u>Task II.C.1 "Reliability Engineering and Risk Assessment - Initial Integrated Reliability Evaluation Program (IREP)"</u>

  In its report of March 11, 1980 on NTOL requirements, the ACRS commented favorably on the IREP program as it was then described. However, the Committee also recommended that the NTOL plants as well as current licensees be concurrently required to perform IREP-like studies on an expedited but practical schedule. The ACRS wishes to reiterate that recommendation.

- <u>Task IV.A "Strengthen Enforcement Process"</u>

  The Committee believes that the need to implement and enforce 10 CFR 21 is an important lesson that should be learned from the TMI-2 accident.

  The first paragraph of the Introduction to Chapter I of NUREG-0660 states, "The result of every investigation of the accident at TMI-2 has been the conclusion that, although many factors contributed to the accident, the major contributing factor was the manner in which the plant was operated both before and during the accident." The Committee agrees that this is the tenor of the conclusions of the investigatory reports, and also agrees that appropriate action by the operators would have averted the accident. The Committee believes, however, that greater recognition should be given to the probability that the accident would have been averted if the licensee had been warned that, under the circumstances of the initiating transient, indications could lead operators to take incorrect action. There had been some recognition of this possibility both within and outside the NRC, and the transcript and exhibits of the President's Commission report (but not the reported conclusions) show that this problem had been discussed at a decision-making level by the NSSS vendor as a result of a warning by one of his engineers.

The Committee recognizes that vendors are justified, in some cases, in assuming the responsibility for deciding whether a safety issue exists. However, when an issue of this significance is raised by competent and responsible engineers, including those at a supervisory level, the Committee believes that NRC should be made a party to the decision. In this case, it is reasonable to suppose that notification to NRC of a serious concern expressed by vendor personnel would have prompted NRC participation, including an expedited review of a similar warning by an NRC engineer, and would have led to an order to the TMI-2 licensee that should have averted the accident. The Committee believes that the industry has, in general, acted in a responsible manner in notifying NRC of potential safety issues as they arise, but it believes that real NRC control of reporting procedures is necessary. The Committee believes this should be specifically listed as a Priority Group 1 item in Section III of NUREG-0660.

The ACRS understands that this matter is to be addressed as a sub-item of Task IV.A but is concerned that preoccupation with the operators' role in the TMI-2 accident tends to de-emphasize the urgency of enforcement with respect to vendors and architect engineers.

The following items may be covered by the Plan or by non-TMI-2 generic items, but are listed as items that the ACRS believes should receive early attention:

- The ACRS supports the recommendation of the Office of Standards Development that the Action Plan should include a task which considers the possible establishment of classes of equipment between those most important to safety and those least important to safety.

- The Action Plan includes several tasks which bear on means of shutdown heat removal such as the auxiliary feedwater system and the feed and bleed method. However, the Action Plan appears to lack a coordinated effort to evaluate shutdown heat removal requirements in a comprehensive manner, thereby permitting a judgment of adequacy in terms of overall system requirements. The Committee recommends the development of such a function.

The ACRS has noted in previous letters that it is important that the improvements in safety proposed as a result of the TMI-2 accident be considered in a broad perspective and that other matters of importance to safety receive proper priority. The ACRS wishes to make several comments in this regard.

- In its report of December 13, 1979 on the TMI-2 Lessons Learned Task Force Final Report and its report of December 17, 1979 on A Review of NRC Regulatory Processes and Functions (NUREG-0642), the ACRS recommended the development of more effective methods of uncovering design errors. The Committee believes that resources should be allocated to initiate the formulation of an appropriate approach.

- The ACRS has previously noted the need to reconsider the present regulatory approach to control systems as they relate to safety. The Rancho Seco transient of March 20, 1978 had provided an important illustration of how control systems can both cause and aggravate transients. The more recent transients at Oconee on November 10, 1979 and Crystal River on February 26, 1980 add further emphasis. The NRC Staff has initiated efforts to correct the specific issues raised by these transients. However, the ACRS wishes to reiterate its belief that there is also need for a broad study which reevaluates in a systematic way the regulatory approach to what have been previously considered non-safety systems, controls, and instrumentation. The ACRS recommends that an appropriate resource level be allocated to this important task.

- The ACRS recommends that the Regulatory Staff review its current priorities on unresolved safety issues and generic items to see whether the priorities established prior to the TMI-2 accident are still valid. Although the NRC Staff had earlier expected that the demands of the Task Action Plan would delay significant work on the DC power issue, the Staff advised the ACRS at its April 1980 meeting that this issue would now be elevated in priority and receive early attention. The ACRS strongly supports a high priority for resolution of this issue.

- The ACRS believes that, in preparation of the Action Plan, insufficient attention was given to both general and specific policy questions which require consideration in connection with near-term construction permits. The Committee recommends that the appropriate resources be devoted to this matter in a timely fashion. In a similar vein, the ACRS recommends that the NRC initiate appropriate efforts on the development of safety criteria for LWRs for which construction permits have not yet been requested, including consideration of the potential augmentation in safety that might accrue from the development of a limited number of standard plant designs.

Several items of the Action Plan include sub-items relating to research needs and programs. These have not been reviewed in detail but will be reviewed and commented on, as appropriate, as part of the Committee's annual review of the NRC Research Program.

Subject to the foregoing comments, and those in its March 11, 1980 report on NTOL requirements, the ACRS finds that Draft 3 of NUREG-0660 with modifications that the ACRS understands will be incorporated into Draft 4, is a satisfactory plan for dealing with issues identified as a result of the TMI-2 accident. As the Plan develops, the ACRS will continue its interest in relative priorities among pre-TMI-2 and post-TMI-2 items.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. U. S. Nuclear Regulatory Commission, "NRC Action Plans Developed as a Result of the TMI-2 Accident" USNRC Report NUREG-0660, Draft 3, March 5, 1980.

2. Atomic Industrial Forum, Inc. letter dated February 22, 1980 forwarding "Report to the AIF Policy Committee on Follow-Up to the Three Mile Island Accident by the Working Group on Action Plan Priorities and Resources."

3. U. S. Nuclear Regulatory Commission memorandum dated April 1, 1980 for Chairman Ahearne from W. J. Dircks, Subject: "ACRS Report on Near-Term Operating License Requirements."

4. U. S. Nuclear Regulatory Commission memorandum dated March 26, 1980 for W. J. Dircks from R. J. Budnitz, Subject: "Management Review of Draft 3 of TMI Action Plan."

5. U.S. Nuclear Regulatory Commission memorandum undated for W. J. Dircks from Victor Stello, Jr., Subject: "Management Review of Draft 3 of TMI-Action Plan."

6. U.S. Nuclear Regulatory Commission memorandum dated April 1, 1980 for W. J. Dircks from Harold R. Denton, Subject: "NRR Management Review of Draft 3 of TMI Action Plan."

7. U. S. Nuclear Regulatory Commission memorandum dated March 27, 1980 for W. J. Dircks and R. J. Mattson from R. B. Minogue, Subject: "SD Comments and Resource Information for Draft 3a of TMI Action Plan."

8. General Electric Company letter dated March 7, 1980 to John F. Ahearne, Chairman NRC, Subject: "BWR Mark I & II Containment Inerting."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 6, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: NEAR-TERM CONSTRUCTION PERMIT APPLICATIONS

Dear Dr. Ahearne:

During its 241st meeting, May 1-3, 1980, the ACRS reviewed the status of applications for near-term construction permits (NTCPs). In its review the Committee had the benefit of discussions with the NRC Staff and with representatives of the applicants for the NTCPs. A subcommittee meeting on this subject was held on April 9, 1980.

The six NTCP applicants and the reactor types involved are as follows:

Black Fox Station, Units 1 and 2, Public Service Company of Oklahoma, General Electric BWR/6, Mark III pressure suppression containment

Skagit Nuclear Power Project, Units 1 and 2, Puget Sound Power & Light Company, General Electric BWR/6, Mark III pressure suppression containment

Pilgrim Station, Unit 2, Boston Edison Company, Combustion Engineering custom NSSS, large dry containment

Perkins Nuclear Station, Units 1, 2 and 3, Duke Power Company, Combustion Engineering CESSAR System 80 NSSS, large dry containment

Allens Creek Nuclear Generating Station, Houston Lighting & Power Company, General Electric BWR/6, Mark III pressure suppression containment

Pebble Springs Nuclear Plant, Units 1 and 2, Portland General Electric Company, Babcock and Wilcox custom NSSS, large dry containment

The NRC Staff has approached this matter primarily by examining the Action Plan and judging the applicability and scheduling of each item to an NTCP. This procedure has resulted in placing many important items in a category wherein the NRC has yet to develop criteria applicable to construction permit applicants. Action Plan item II.A on siting introduces questions whose resolution must be achieved prior to issuance of a construction permit.

Item II.B on degraded or melted cores bears directly on containment design, as well as other safety features. Item II.C on reliability engineering and risk assessment could bear significantly on the design requirements for many important plant systems. There are many other items in the Action Plan and in the ACRS report of April 17, 1980 which also might impact directly on important design aspects of these plants.

Mr. Harold Denton advised the Committee that he envisaged permitting construction to proceed if there are no obvious site-related questions in terms of the Report of the Siting Policy Task Force (NUREG-0625) and if the containment design pressure were such as to withstand hydrogen combustion, on the assumption that other design aspects could be changed later if so required.

The utility representatives advised the ACRS that, in their opinion, there was a need for the resolution of several policy questions which relate to how and whether construction permit applications will be processed in the near term. The utilities identified the following six policy issues as being in most urgent need of resolution:

1. Siting
2. Emergency planning
3. Degraded core conditions
4. Control room design
5. Management for design and construction
6. Reliability and risk assessment

The utility representatives recommended that a concerted effort be undertaken to develop an acceptable interim approach to resolution by the Commission of such issues in the next few months. The ACRS supports this recommendation and urges that appropriate Staff resources be made available for this purpose. An ACRS Subcommittee plans to work actively with the Staff on the topic with the anticipation that the full Committee would review the NTCP matter within a few months.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. Memorandum from D. F. Ross, NRC, to R. F. Fraley, ACRS, Subject: Transmittal of NTCP Requirements List, dated April 22, 1980.
2. Memorandum from William F. Kane, NRC, to Addressees, Subject: Request for Review of Proposed TMI-2-Related Requirements for NTCP Applicants, dated April 4, 1980.
3. U. S. Nuclear Regulatory Commission, "NRC Action Plans Developed as a Result of the TMI-2 Accident," USNRC Report NUREG-0660 Draft 3, dated March 5, 1980.
4. U. S. Nuclear Regulatory Commission, "Report of the Siting Policy Task Force," USNRC Report NUREG-0625, dated August, 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 10, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: ADDITIONAL INFORMATION CONCERNING NTOL ITEMS FROM DRAFT 3 OF THE NRC ACTION PLAN

Dear Dr. Ahearne:

In your letter of April 1, 1980 to the Advisory Committee on Reactor Safeguards, you requested additional information on some of the Committee's comments in its letter to you of March 11, 1980 concerning near-term operating license (NTOL) items from Draft 3 of the NRC Action Plan. The following is in response to your inquiry:

<u>Question</u>

"1. Which of the items from the list in Part 1, Requirement (11), Item II.K.1 does the Committee consider to be useful, which to be of marginal merit, and which to be wrong?" and

"3. Which of the items from Part 2, Requirement (4), Item I.C.1 does the Committee consider to be useful, which to be of marginal merit, and which to be wrong?"

The Committee did not wish to imply that the referenced conclusions were wrong, but only that they "... may, upon deeper analysis, turn out to have been wrong." The Committee believes that only after the recommended deliberate study will it be possible to appraise the merit of these proposals.

<u>Question</u>

"2. In commenting on Part 1, Requirement (13), Item III.D.3.4 the Committee noted, 'The TMI incident indicates that existing requirements to protect the occupants of the control room against radiation may not be adequate, particularly with respect to leakage control and arrangement of air intakes.' Does the Committee have any specific suggestions as to how these requirements should be upgraded?"

Item III.D.3.4 deals with, "Control Room Habitability" and Table A.1 gives the following charge to NTOL applicants: "Confirm compliance with existing Regulatory Guides and Standard Review Plan or establish schedule for necessary modifications to achieve compliance." The general sense of the Committee's comment was to encourage a look beyond existing criteria to protect the occupants of the control room against radiation. Consideration of the effects of greater than currently assumed containment leakage and/or a larger source term due to a degraded core was envisaged. In the area of control room ventilation, the Committee would encourage the placing of air intakes in such a way that they could be used selectively during an accident to take advantage of prevailing wind direction and radiation source locations. Recall that during the TMI-2 accident, control room operators actually used respirators for a time. The Committee takes note of Item 7 in William Dircks' April 1, 1980 memorandum to you in response to the ACRS NTOL letter in which concerns like those mentioned above are scheduled for review in the longer term.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 16, 1980

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: ADDITIONAL ACRS COMMENTS ON THE RCP TRIP AND HPI TERMINATION CRITERIA

Dear Dr. Ahearne:

In your letter of April 1, 1980, you requested that we clarify our concerns with the present reactor coolant pump (RCP) trip and the high pressure injection (HPI) termination criterion. You also indicated in a memorandum to R. Fraley on February 22, 1980 that you would welcome our comments on NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors."

The present requirements for RCP trip and HPI termination have developed from the lessons learned from the Three Mile Island accident and from the extensive number of small break LOCA calculations subsequently carried out. There are two distinct requirements in the I&E Bulletins issued, as referenced below, which can be considered separately. The first concerns the directive which requires prompt shutdown of all reactor coolant pumps in PWRs following a depressurization transient which initiates safety injection. The second is the requirement that the safety injection system continue to be operated until a specified degree of subcooling is attained in the primary system.

The prompt reactor coolant pump trip mandated by the Bulletins followed analyses by the vendors of nuclear steam supply systems which seemed to show that there was a "window" of break sizes and pump trip delay times which would lead to calculated peak cladding temperatures in excess of the 2200°F licensing limit. These same methods of analysis indicated that with prompt pump trip the peak cladding temperatures would remain below 2200°F. The NRC Staff prepared a useful critique in NUREG-0623 of these vendor calculations and, while this report clearly presented the deficiencies in the analytical methods used, the report agreed with the vendors' conclusions. The short-term action by the Staff therefore was the requirement of prompt trip of the reactor coolant pumps; as a long-term action the Staff recommended that licensees propose and submit design changes that will assure automatic trip of all reactor coolant pumps.

We do not, at this time, disagree entirely with the Staff's requirement of prompt coolant pump trip, but in view of the analytical limitations upon which prompt trip is based we believe that the emphasis on immediacy of the trip and on eventual automatic trip may not be desirable. Recent experimental data has put doubt on the existence of the "window" which is the basis

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for requiring prompt pump trip. Additional experimental data will become available before the end of the year. The prompt trip has been carried out in four transients since the Bulletins have appeared. In none of these was there a LOCA in the primary system; all of these transients arose from disturbances on the secondary side. No significant plant damage ensued in these transients and there was no harm to plant personnel or to the public. There has been complaint, however, that without reactor coolant pump flow the operator loses reactor pressurizer control since, in many PWRs, pressurizer spray flow depends on coolant pump flow. Further, natural circulation must also be established to remove decay heat. It must be said that the Staff's hope to develop a clear distinction between depressurization from a small break on the primary side and depressurization from a secondary side transient seems quite optimistic.

We believe that reactor coolant pump trip upon primary depressurization is an acceptable procedure, but we see no urgency at this time for installation of automatic pump trip. With regard to primary pressure control, we believe that it is desirable to provide pressurizer spray flow which is independent of main coolant pump flow.

The present set of requirements for HPI termination criteria is based upon achieving a specified degree of subcooling in the primary coolant system along with, in some cases, a specified water level in the pressurizer and steam generators. These requirements are intended to prevent a recurrence of the TMI-2 situation in which HPI flow was terminated while still necessary; these requirements, however, do not address the conditions in which HPI should be terminated when not required. We are concerned that relatively frequent system transients which activate HPI might progress to liquid discharge through safety valves or PORVs, valve failure under liquid flow, and a resultant small break LOCA. It should also be pointed out that Westinghouse has recently reported a significant deficiency under 10 CFR 50.55(e) for a number of reactors with high head centrifugal charging/safety injection pumps. Failure to stop these pumps promptly when high pressures are reached could result in pump failure from low flow - a common mode failure of the redundant HPI pumps. Changes in operational procedures may also affect the design limits of other components. These interactions need to be carefully reviewed.

We note that a number of plant transients that have occurred in the past year have been affected by the NRC approved HPI termination and RCP trip criteria. These include events, as referenced below, at North Anna, Unit 1, September 26, 1979; Prairie Island, Unit 1, October 2, 1979; and ANO, Unit 2, January 29, 1980. Some changes have been made in criteria in response to these events. We believe that continued Staff attention in this area is required.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "I&E Bulletin 79-05A," April 5, 1979.
2. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "I&E Bulletin 79-06A," April 14, 1979.
3. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "I&E Bulletin 79-06B," April 14, 1979.
4. U. S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "I&E Bulletin 79-05C and 79-06C," July 26, 1979.
5. NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," November 1979.
6. Letter, C. M. Stallings, VEPCO, to J. P. O'Reilly, NRC, Submitting Licensee Event Report for September 25, 1979 North Anna Number 1 Cooldown Incident (October 9, 1979).
7. Letter, L. O. Mayer, NSP, to J. G. Keppler, NRC, Submitting Licensee Event Report for October 2, 1979 Steam Generator Tube Rupture Incident (October 16, 1979).
8. U.S. Nuclear Regulatory Commission Preliminary Notification of Event or Unusual Occurrence, PNO-IV-80-05, January 30, 1980.
9. Letter, D. C. Trimble, AP&L, to R. W. Reid, NRC, Submitting Startup Report, Supplement 2 for ANO-Unit 2, March 6, 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 11, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: STATUS REPORT ON RESTART OF THE THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Dear Dr. Ahearne:

During its 248th meeting, December 4-6, 1980, the Advisory Committee on Reactor Safeguards continued its review of the status of the proposed restart of the Three Mile Island Nuclear Station, Unit 1 (TMI-1) with representatives of the Metropolitan Edison Company (Licensee), General Public Utilities Nuclear Group, the Babcock and Wilcox Company (B&W), and members of the NRC Staff. This matter was also the subject of Subcommittee meetings in Middletown, PA, on January 31 - February 1, 1980, and in Washington, DC, on November 28 and 29, 1980.

One of the primary results of these reviews is an indication of the need for a statement of policy by the NRC on how and when the various components of the Action Plan, the NTOL list, and items in the NRC order of August 9, 1979, are to be applied in the evaluation of the TMI-1 restart.

There is also a need for the NRC Staff to prepare a concise summary of the issues that remain open on the TMI-1 review, a statement as to the status of each, the degree to which each is considered significant from the standpoint of health and safety, and an indication as to which items must be resolved prior to restart. For those items whose resolution can be delayed until after restart, there is a need for the specification of a date when their associated review and implementation must be completed. Because of the importance the Committee attaches to this subject, we requested at our meeting on December 4, 1980, that the NRC Staff complete and submit such a summary to the Committee.

In terms of the response of the Licensee, the ACRS was encouraged by their actions in several areas. These include: (a) the qualifications of management personnel who have been brought into the organization; (b) the thorough, in-depth training program they have established for their operators and plant support personnel; (c) the program they have developed for keeping up to date on operating experiences elsewhere within the nuclear power industry; (d) the degree to which human factors considerations have been used in modifying and upgrading the TMI-1 control room; and (e) the commitment of the Licensee to a restart testing program, which includes confirmation of natural circulation.

On the basis of its review, the Committee offers the following comments:

1. In accordance with our previous recommendations, we believe that the Licensee should conduct reliability assessments of the plant as modified. Such assessments should accelerate the acquisition of potentially significant safety information and would expedite the development of the basis for further changes, should they be necessary. They would also provide the Licensee with additional technical insight into the safety of the plant. In addition, we believe the Licensee should examine the plant from the standpoint of systems interactions that may degrade safety. Although both of these studies should be conducted on a timely basis, their completion should not be a condition for restart.

2. The Committee has previously recommended that a means be considered which would provide an unambiguous indication of water level in the reactor pressure vessel. Although we do not believe that installation of such a system should be a requirement for restart, we believe the Licensee should give additional consideration to this matter on a timely basis.

3. The Committee believes there is a need for instrumentation to monitor the position (i.e., opened or closed) of the pressurizer PORV and safety valves in an unambiguous manner. The sensitivity of the currently proposed method to monitor valve position remains an open issue between the Staff and the Licensee. This matter should be resolved in a manner acceptable to the Staff prior to restart.

4. The Licensee reported on the thermal/mechanical effect of high pressure injection on reactor pressure vessel integrity for a small break LOCA with no emergency feedwater flow. This concern, raised by the Bulletins and Orders Task Force, showed a possible conflict between the need for keeping the fuel cool during bleed-and-feed cooling versus keeping the vessel within 10 CFR 50, Appendix G limits. Although B&W personnel have performed calculations relative to this matter, their calculations were limited to the small break LOCA bleed-and-feed procedure. There may be certain accident combinations which result in much more severe chilling of the pressure vessel coincident with vessel repressurization. The Committee believes that the Licensee should review a broader spectrum of accident scenarios to assure better bounding of the range of possibilities. Although these studies should be completed on a timely basis, they need not be a condition for restart.

5. The Licensee has discussed the consequences of DC power failure at TMI-1 and has evaluated them in a manner similar to that outlined in NUREG-0305, "Technical Report On D.C. Power Supplies In Nuclear Power Plants." The Licensee is performing additional studies to identify possible events which might lead to the loss of both battery trains. We encourage completion of these studies on a timely basis.

We will schedule follow-up Subcommittee meetings as soon as practicable and will arrange for the Licensee and NRC Staff to meet with the full Committee when progress warrants.

Additional comments by Messrs. D. Moeller and D. Okrent are presented below.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Additional Comments by Messrs. D. Moeller and D. Okrent

In its letter dated December 13, 1979 entitled, "Report on TMI-2 Lessons Learned Task Force Final Report," concerning the topic entitled "Design Features for Core-Damage and Core-Melt Accidents," the ACRS said, "The ACRS supports this recommendation. However, the Committee believes that the recommendation should be augmented to require concurrent design studies by each licensee of possible hydrogen control and filtered venting systems which have the potential for mitigation of accidents involving large scale core damage or core melting, including an estimate of the cost, the possible schedule and the potential for reduction in risk."

In its letter dated September 8, 1980 entitled "Additional ACRS Comments on Hydrogen Control and Improvement Of Containment Capability," the ACRS reiterated this recommendation, stating its belief that it, "should be adopted and given priority by the NRC."

We believe that this recommendation is especially applicable to a higher population density site such as TMI, and that the prior history of an accident at this site reinforces the desirability of examining design measures which have the potential for reducing significantly the quantity of radioactive material released for a range of postulated serious accidents leading to severe core damage or a molten core. We recommend that the restart of Three Mile Island Nuclear Station, Unit 1 be made contingent on a commitment by the Licensee to perform, within a reasonable period following restart, a study such as that recommended in the ACRS letter of December 13, 1979 referred to above.

References:

1. Metropolitan Edison Company, "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1," Volumes 1-3, and Amendments 1-22.
2. U.S. Nuclear Regulatory Commission, "TMI-1 Restart, Evaluation of Licensee's Compliance with the Short- and Long-Term Items of Section II of the NRC Order Dated August 9, 1979, Metropolitan Edison Company, et al., Three Mile Island Nuclear Station Unit 1, Docket 50-289," NUREG-0680, June 1980.

References Cont'd:

3. U.S. Nuclear Regulatory Commission, "TMI-Related Requirements for New Operating Licenses," NUREG-0694, June 1980.
4. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.
5. U.S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident" NUREG-0660, Volumes 1 and 2, May 1980 (Revised: August 1980).
6. Letter from Marvin Lewis, member of the public, to Richard Major, ACRS Staff, regarding the restart of the Three Mile Island Nuclear Station Unit 1, dated November 16, 1980.
7. Letter from B. Lehmann, GPU Service Corporation, to Richard Major, ACRS Staff, transmitting Testimony outlines - TMI-1 Restart Proceeding, dated October 29, 1980.
8. Letter from H. Dieckamp, President, General Public Utilities Corporation, to J. Ahearne, Chairman, U.S. Nuclear Regulatory Commission, regarding request that the Commission reconsider and modify its Orders of July 2, 1979 and August 9, 1979 dealing with the restart of Three Mile Island Unit No. 1, dated December 1, 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 12, 1981

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: REQUIREMENTS FOR NEAR-TERM CONSTRUCTION PERMIT APPLICATIONS

Dear Dr. Ahearne:

During its 249th meeting, January 8-10, 1981, the ACRS again reviewed the status of the requirements for near-term construction permits (NTCPs). The Committee reported to you previously on this subject in a letter dated May 6, 1980. In the present review we had the benefit of a Subcommittee meeting on January 6, 1981 and of discussions with members of the NRC Staff and with representatives of applicants for NTCPs and Offshore Power Systems, the applicant for a manufacturing license (ML).

In our letter of May 6, 1980 we noted that the utility representatives had advised the Committee that there was a need for resolution of several policy issues which related to how and whether construction permit applications would be processed in the near term. The principal policy issues identified dealt with siting, degraded core conditions, reliability and risk assessment, and emergency planning. In May 1980, the utilities expressed a desire to have the chance to propose an acceptable interim approach to resolution of these issues. However, the utilities did not present any common proposal for dealing with this matter during the next several months.

The NRC Staff did develop a proposed policy and on October 2, 1980 the NRC published for comment in the Federal Register "Proposed Licensing Requirements for Pending Construction Permit and Manufacturing License Applications." The Federal Register notice identified the following three options as having been considered by the NRC Staff.

1. Resume licensing using the pre-TMI CP requirements augmented by the applicable requirements identified in the TMI Action Plan, NUREG-0660. In effect, this treats the pending CP and ML applications as if they were the last of the present generation of nuclear power plants.

2. Take no further action on the pending CP and ML applications until the rulemaking actions described in the Action Plan have been completed. This would, in effect, treat the pending applications as the first of a new generation of nuclear power plants.

3. Resume licensing using the pre-TMI CP and ML requirements augmented by the applicable requirements identified in the TMI Action Plan, NUREG-0660, and require certain additional measures or commitments in related areas, e.g., those that would be the subject of rulemaking.

The NRC Staff favored Option 3 as a suitable compromise and identified their current positions for NTCP and ML plants with regard to siting, degraded core rulemaking, reliability engineering and emergency preparedness.

The comments from representatives of the nuclear industry on the proposed licensing requirements generally opposed the Staff's preference for Option 3, and favored Option 1. In addition to opposing additional requirements for NTCP plants, the industry representatives argued that the Staff's position concerning degraded core rulemaking was open-ended and would lead to protracted delays and case-by-case adjudication of the matter at ASLB hearings. Industry representatives provided a varied set of comments concerning reliability engineering and argued against adoption of the NRC Staff's position on siting. Offshore Power Systems favored Option 1 but stated that they believed they could live with Option 3.

During the 249th ACRS meeting, the NRC Staff advised the Committee that it now favored adoption of a revised Option 3. The new NRC Staff position was described as follows:

Emergency Preparedness

The Commission has adopted a rule which addresses this subject. The NTCP Applicants will be required to comply with this rule.

Siting

In view of the demographic and hydrological characteristics of the proposed sites, no additional measures with regard to siting would be required in connection with these construction permit applications.

Reliability Engineering

Each applicant would be required to submit a site/plant probabilistic risk assessment as part of the application for an operating license.

Degraded Core Rulemaking

In order to minimize foreclosure of plant modifications in the structural design area, at least those applicants whose designs incorporate a relatively low-design-pressure reactor containment would have to strengthen the containment structure against internal pressure. In addition, all applicants would be required to commit to making provisions for an approximately three foot diameter, or equivalent, containment penetration which could be used in conjunction with a filtered venting design feature, should the latter be judged to be needed.

We agree with the NRC Staff's currently proposed approach on siting. We also agree with the current NRC Staff position on reliability engineering. During the discussion with us, the NRC Staff indicated that, although they did not propose making a formal requirement to that effect, one intent of the proposed position on reliability engineering was to strongly encourage each applicant to perform the relevant portions of the probabilistic assessment early enough that the results could be factored into a safety-related reliability optimization of the design. We strongly support this point of view and recommend that each applicant give high priority to such efforts.

The NRC Staff's position on the degree of containment strengthening that should be required had not yet been definitively formulated by the time the 249th ACRS meeting was held. Since the NRC Staff's position was new, industry representatives did not have time to review the position and provide comments.

Furthermore, we were advised by representatives of the Houston Lighting and Power Company, the Applicant for the Allens Creek Nuclear Generating Station, that they had authorized a study of possible accident prevention and mitigation features for their plant in order to ascertain the advantages, disadvantages, and practicality of these features. The results of this study are to be presented to Houston Lighting and Power in mid-January and representatives of the company requested an opportunity to meet with the ACRS in early February to discuss these results.

We agree with the general approach outlined by Harold Denton at the 249th ACRS meeting concerning provisions for degraded core rulemaking on NTCP plants. However, we believe that the NRC Staff needs to define its proposal more precisely. We believe that both the NRC Staff and the ACRS should have the benefit of further discussions with the NTCP and ML applicants. Hence, we recommend that the Nuclear Regulatory Commission defer any final action on the overall matter at least until after the 250th ACRS meeting on February 5-7, 1981 during which this matter is scheduled for discussion.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 10, 1981

The Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: ACRS REPORT ON REQUIREMENTS FOR NEAR-TERM CONSTRUCTION PERMITS AND MANUFACTURING LICENSES

Dear Dr. Ahearne:

During its 250th meeting, February 5-7, 1981, the ACRS again reviewed the status of requirements for near-term construction permits (NTCPs) and manufacturing licenses (MLs). The Committee reported to you previously on this subject in letters dated May 6, 1980 and January 12, 1981. In the present review we had the benefit of a Subcommittee meeting on February 4, 1981 and of discussions with members of the NRC Staff and representatives of the Houston Lighting and Power Company, Offshore Power Systems, Boston Edison Company, and the General Electric Company.

In our letter dated January 12, 1981, we agreed with the general position outlined by Harold Denton to the ACRS but recommended that a decision be deferred while the NRC Staff better defined its proposal and the Houston Lighting and Power Company was provided an opportunity to present the results of their study of the merits of possible preventive and mitigative design features for the proposed Allens Creek boiling water reactor.

During the 250th ACRS meeting, the NRC Staff presented the attached proposed position regarding requirements for NTCP and ML applicants. We have the following comments on these proposed requirements:

<u>Item 1 - Site/plant specific probabilistic risk analysis</u>

The current NRC Staff position is similar to the Staff position of January 9, 1981 which the ACRS supported. The new position on reliability engineering is more specific in that it would require the applicant to submit the risk assessment within two years after issuance of the construction permit and call for an NRC review at that time to determine possible requirements for preventive and mitigative actions. The criteria which would be used in this selection process have not been specified nor are they easily specified at this time. The Committee suggests that the Commission consider stating as an aim the seeking of such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant, with the intent of encouraging each applicant to take those steps which are in harmony with such an aim.

Item 2 - Dedicated penetration for possible installation of systems to prevent containment failure

This is identical to the Staff position discussed in January and has the support of the ACRS.

Items 3 and 4 - Hydrogen control measures and containment strengthening requirements

These represent a modified statement of the position proposed by Harold Denton in January to strengthen relatively low-design pressure containments against internal pressure as practical, within the existing design concept and without excessive impact. Items 3 and 4 require hydrogen control measures and pose some specific requirements with regard to minimum internal pressure capability. The ACRS believes that the NRC Staff approach in this regard is acceptable. However, while the ACRS wishes to encourage applicants to provide containment strengthening of the type proposed in Item 4 a., we believe that, if proposed by any of the applicants, modest deviations from the specific requirements should be considered on their merits.

In a letter to you dated September 8, 1980 providing additional comments on hydrogen control and improvement of containment capability, the ACRS stated its belief that each licensee should be required to perform design studies of possible hydrogen control and filtered venting systems which have the potential for mitigation of accidents involving large scale core damage or core melting, including an estimate of the cost, the possible schedule, and the potential for reduction in risk. The Committee believes that such studies should also be made by NTCP and ML plants during construction and that the final choice of hydrogen control system for each plant should be made with the benefit of such broader studies.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

Attachment:
Staff Position With Regard to NTCP Requirements With Respect to Degraded Core Rulemaking, dated 2/6/81

STAFF POSITION WITH REGARD TO NEAR-TERM CONSTRUCTION PERMIT REQUIREMENTS WITH RESPECT TO DEGRADED CORE RULEMAKING - FEBRUARY 6, 1981

1. Applicants shall commit to performing a site/plant-specific probabilistic risk assessment and incorporating the results of the assessment into the design of the facility. The commitment must include a program plan, acceptable to the Staff, that demonstrates how the risk assessment program will be scheduled so as to influence system designs as they are being developed. The assessment shall be completed and submitted to NRC within two years of issuance of the construction permit. The outcome of this study and the NRC review of it will be a determination of specific preventive and mitigative actions to be implemented to reduce these risks. A prevention feature that must be considered is an additional decay heat removal system whose functional requirements and criteria would be derived from the probabilistic risk assessment study.

2. In order not to preclude the installation of systems to prevent containment failure, such as a filtered vented containment system, the containment design shall include provisions for one or more dedicated penetrations, equivalent in size to a single three foot diameter opening.

3. Hydrogen control measures shall be provided.

4. Applicants shall provide preliminary design information at a level consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

   a. Containment integrity will be maintained (i.e., for steel containments, ASME Service Level C based on ASME code specified minimum yield values and considering pressure and dead load alone. For concrete containments, an equivalent approach based on ASME Div. 2) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon-dioxide is the inerting agent depending upon which option is chosen for control of hydrogen. As a minimum, for steel containments ASME Service Level C (based on ASME Code specified minimum yield values and considering pressure and dead load alone) will not be exceeded at an internal pressure of 45 psig. For reinforced concrete containment structures, an equivalent standard based on ASME Division 2 is satisfied at the same internal pressure. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

b. The containment and associated systems will provide reasonable assurance that uniformly distributed hydrogen concentrations do not exceed 10% associated with an accident that releases hydrogen generated from 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

c. The facility design will provide reasonable assurance that, based on a 100% fuel clad metal-water reaction, combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

d. If the option chosen for hydrogen control is post-accident inerting:

   (1) Containment structure loadings produced by an inadvertent full inerting (assuming carbon dioxide) but not including seismic or design basis accident loadings, will not produce stresses in excess of the acceptable maximum for Service Level A specified in ASME Code Section III, Subsection NE (ASME Div. 2 for concrete containments).

   (2) A pressure test of the containment at 1.15 times the pressure calculated to result from carbon dioxide inerting can be safely conducted.

   (3) Inadvertent full inerting of the containment can be safely accommodated during plant operation and demonstrated by test.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 13, 1981

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

Subject: APPLICATION OF TMI-2 ACTION PLAN TO NEAR-TERM CONSTRUCTION PERMITS AND MANUFACTURING LICENSES

During its 255th meeting, July 9-11, 1981, the ACRS heard presentations from the NRC Staff and the Applicant regarding application of the NRC TMI-2 Action Plan items to the Pilgrim Nuclear Power Station Unit 2.

The NRC Staff has established a review team especially for post-TMI-2 issues concerning near-term construction permit (NTCP) and manufacturing license applications. The Committee believes that this approach is providing effective reviews. The Committee concluded that it has no objection to NRC Staff approval of a construction permit for Pilgrim Unit 2, subject to the conditions in its letter of October 12, 1977.

In addition, the Committee concluded that it is not necessary for the ACRS to review application of TMI-2 Action Plan items to the Allens Creek Nuclear Generating Station Units 1 and 2, although the members would like a briefing regarding the resolution of questions regarding hydrogen generation and control for the Allens Creek Mark III containment. This briefing has tentatively been scheduled for the 257th meeting of the ACRS on September 10-12, 1981.

The Committee also desires an opportunity, with respect to the five additional NTCPs and the manufacturing license, to determine on a case-by-case basis if ACRS review of changes resulting from application of the TMI-2 Action Plan is appropriate.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

July 14, 1981

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON RESTART OF THE THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Dear Dr. Palladino:

During its 255th meeting, July 9-11, 1981, the Advisory Committee on Reactor Safeguards completed its review of the proposed restart of the Three Mile Island Nuclear Station, Unit 1 (TMI-1). The Committee had the benefit of discussions with representatives of the Metropolitan Edison Company (Licensee), General Public Utilities Nuclear Group, the Babcock and Wilcox Company, and members of the NRC Staff. The Committee also had the benefit of the documents listed. This matter was the subject of Subcommittee meetings in Middletown, PA on January 31 and February 1, 1980, and in Washington, D.C. on November 28-29, 1980, and June 25-26, 1981. This matter was also discussed during the 248th ACRS meeting and a Status Report was issued on December 11, 1980.

The question of restarting TMI-1 cannot be separated from the important issues revealed by the accident at TMI-2. Among these issues were significant deficiencies in the Licensee's management. Since that time the management structure which will be responsible for operation of TMI-1 has undergone substantial change. These changes have included establishment of a full-time organization dedicated solely to nuclear operations and considerable expansion of the full-time, in-house staff. The Committee believes that the management of a nuclear power plant deserves the highest degree of attention. Although we believe the current structure is sufficient for restart, we urge continued diligence on the part of the Licensee and the NRC Staff to assure the maintenance of adequate managerial capabilities for TMI-1.

In response to the requirements in NUREG-0737, "Clarification of TMI Action Plan Requirements," the Licensee is considering a variety of instruments that might be installed for detecting inadequate core cooling but has not established a plan for selecting an appropriate device. The Licensee and the NRC Staff should act promptly to establish a basis for selecting a suitable monitoring system at TMI-1 taking into account the questions raised by the ACRS in its letter to Mr. Dircks of June 9, 1981, concerning the conditions under which such a system would be of use.

Currently the Licensee has a number of studies underway in response to suggestions of the Committee and to requirements of the Action Plan. These studies include a reliability assessment of the plant as modified, an identification of possible events which might lead to the loss of both battery trains, and evaluations of outages of emergency core cooling systems to indicate areas where improvements in availability can be made. The Licensee has made some limited progress in each of these areas. We believe these studies should be concluded in a timely manner, but need not be conditions for restart, with the exception of those studies required by the NRC Staff prior to restart.

Recently the security system for TMI-1 was reviewed by a team of specialists from the Los Alamos National Laboratory and a formal report detailing necessary improvements was issued. The Licensee indicated that the recommended changes are being addressed and will be implemented over the next few months. Because of the decontamination operations underway within TMI-2, and the large number of people present, we recommend that plant security continue to be given special attention.

One of the primary deficiencies noted in reviews of the TMI-2 accident was the inadequacy of the exchange of information among the Licensee, state, and local organizations and federal agencies, such as the NRC. Although information reported by the Licensee to state and federal authorities during the course of the TMI-2 accident was incomplete and misleading with regard to the severity of the accident, the deficiencies which contributed to this problem have been addressed in the revised organization and in the improved procedures for collecting, evaluating, and reporting information during the course of an accident. Other improvements have included an upgrading of emergency procedures and emergency response capabilities at all levels, including the periodic conduct of drills and exercises, and the development of better mechanisms for keeping those agencies informed who have responsibility for alerting the public in case of an accident. These steps represent significant progress and are considered by the Committee as sufficient for the restart of TMI-1.

The Licensee has proposed a start-up test program for TMI-1 similar to that being conducted at the near-term operating license plants. The Committee agrees that such a program is desirable, particularly in view of the length of time that TMI-1 has been out of service and the number of modifications made. Such a program should also provide useful additional operator training and experience. The review of this program by the NRC Staff is not yet complete. Those issues remaining should be resolved to the NRC Staff's satisfaction.

The Committee believes it is acceptable to allow TMI-1 to complete the remainder of the required Action Plan items as outlined by the NRC Staff on a schedule consistent with other operating reactors.

The Advisory Committee on Reactor Safeguards believes the Licensee has demonstrated reasonable progress toward completion of those requirements necessary to restart this facility. Subject to the satisfactory completion of the NRC Staff review and the comments noted above, the Committee believes that Three Mile Island Nuclear Station, Unit 1 can be restarted and operated without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. Metropolitan Edison Company, "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1," Volumes 1-3, and Amendments 1-24.
2. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "TMI-1 Restart, Evaluation of Licensee's Compliance with the Short- and Long-Term Items of Section II of the NRC Order Dated August 9, 1979, Metropolitan Edison Company, et al., Three Mile Island Nuclear Station Unit 1, Docket 50-289," NUREG-0680, June 1980 and Supplements 1, 2, and 3 dated November 1980, March 1981, and April 1981, respectively.
3. U. S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "Emergency Preparedness Evaluation for TMI-1, Metropolitan Edison Company et al., Three Mile Island Nuclear Station Unit 1, Docket 50-289," NUREG-0746, December 1980 and Supplement 1 dated May 29, 1981.
4. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Control Room Design Review Report for TMI-1, Metropolitan Edison Company, et al., Three Mile Island Nuclear Station Unit 1, Docket 50-289," NUREG-0752, December 1980 and Supplement 1 dated April 1981.
5. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.
6. U. S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," Vols. 1 & 2, May 1980, revised August 1980.
7. Two letters from J. Stolz, NRC, to H. D. Hukill, Metropolitan Edison Company, transmitting "Safety Evaluation Reports for Items Contained in NUREG-0694 and Enclosure 1 to NUREG-0737 Outside of the Content of the Commission's Orders of August 9, 1979 and March 6, 1980 Required by Restart (October 1981)," April 22, 1981.
8. "Report of the Presidential Commission on The Accident at Three Mile Island - The Need for Change: The Legacy of TMI," October, 1979.
9. "Three Mile Island -- A Report to the Commissioners and to the Public" Vols. I & II, Nuclear Regulatory Commission Special Inquiry Group, January 1980.

References Cont'd:

10. Memorandum to Chairman Ahearne, NRC, from Mitchell Rogovin and George T. Frampton, Jr., Special Inquiry Group, Subject: Questions Submitted by Congressman Udall, March 4, 1980.
11. Staff Studies "Nuclear Accident and Recovery at Three Mile Island, A Special Investigation, Subcommittee on Nuclear Regulation for the Senate Committee on Environment and Public Works," July 1980.
12. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "Investigation Into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, August 1979.
13. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "Investigation Into Information Flow During the Accident at Three Mile Island," NUREG-0760, January 1981.
14. Memorandum for William J. Dircks, Executive Director for Operations, from Carlyle Michelson, Director, Office of Analysis and Evaluation of Operational Data, Subject: Investigations Into Information Flow Concerning the TMI Accident, February 20, 1981.
15. "Reporting of Information Concerning the Accident at Three Mile Island," A Report Prepared by the Majority Staff of the Committee on Interior and Insular Affairs of the U.S. House of Representatives, Ninety-Seventh Congress First Session, March 1981.
16. Letter from Congressman Morris K. Udall, Chairman, Committee on Interior and Insular Affairs, U.S. House of Representatives, to Dr. D. W. Moeller, Chairman, Advisory Committee on Reactor Safeguards Subcommittee on Three Mile Island Nuclear Station, Unit 1, enclosing information for the record of the June 25, 1981 meeting of the TMI-1 Subcommittee, July 6, 1981.
17. Letter from Marvin Lewis, member of the public, to Richard Major, ACRS Staff, regarding the restart of the Three Mile Island Nuclear Station Unit 1, undated, received April 1981.
18. Letter from Marvin Lewis, member of the public, to M. Plesset, Chairman, ACRS, regarding the restart of the Three Mile Island Nuclear Station Unit 1, undated, received December 31, 1980.
19. Letter from Marvin Lewis, member of the public, to M. Plesset, Chairman, ACRS, regarding the restart of the Three Mile Island Nuclear Station Unit 1, December 31, 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 21, 1984

Mr. William J. Dircks
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

SUBJECT: REVIEW OF GPU NUCLEAR CORPORATION'S CLEANUP PLAN FOR TMI-2 AND THE NRC STAFF'S DRAFT SUPPLEMENT TO THE CLEANUP PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT

During its 287th meeting, March 15-17, 1984, the ACRS considered the recommendations of its Subcommittee on Reactor Radiological Effects regarding the TMI-2 cleanup. The Subcommittee had the benefit of the presentations by the NRC's TMI Program Office and by GPU Nuclear Corporation personnel during meetings on January 24 and February 24, 1984, respectively.

The ACRS approved forwarding the Subcommittee comments to you for your consideration.

Sincerely,

Jesse C. Ebersole
Chairman

Enclosure:
Feb. 24, 1984 Subcommittee Comments on TMI-2
Cleanup and Related Issues

Reference:
Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979 Accident, Three Mile Island Nuclear Station, Unit 2 (Draft Supplement Dealing with Occupational Radiation Dose) NUREG-0683, Supp. No. 1, Draft Report, 12/83

cc: B. Snyder, TMIPO
L. Barrett, TMIPO
H. Denton, NRR
R. Minogue, RES

COMMENTS ON
GPU NUCLEAR CORPORATION'S CLEANUP PLAN FOR TMI-2 AND
ON THE NRC STAFF'S DRAFT SUPPLEMENT TO THE CLEANUP
PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT (PEIS)

ACRS SUBCOMMITTEE ON REACTOR RADIOLOGICAL EFFECTS
FEBRUARY 24, 1984

During a meeting on January 24, 1984, the Subcommittee heard presentations by representatives of the NRC's TMI Program Office on the Staff's draft supplement to the Programmatic Environmental Impact Statement (PEIS) Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979 Accident, Three Mile Island Nuclear Station, Unit 2. This supplement was issued for comment in December, 1983 and deals with occupational radiation doses associated with the cleanup effort. On February 24, 1984, the Subcommittee met again and was briefed by GPU Nuclear Corporation on its detailed cleanup plan for TMI-2. Based on the above, we offer the following comments:

1. The TMI-2 GPU Recovery Staff appeared to be professional in their approach, and they were thorough in their presentations. However, they do not appear to have on their staff (or serving as consultants to them) an adequate number of people who have had previous direct experience in nuclear facility cleanup operations. The Subcommittee believes that the provision of such expertise would be helpful.

2. The discussions of the cleanup at TMI-2 clearly indicated that Cs-137 accounts for a major part of the external exposures that are occurring, and those that are projected in terms of the collective occupational doses for the total cleanup operation.

   Accordingly, the Subcommitee urges that GPU obtain the services of professional personnel expert in the chemical behavior of cesium so that they can effectively address the problems represented by this radionuclide. They apparently do not now have such expertise.

3. There appear to be several aspects of the recovery operations wherein a better understanding of the radiation protection problems and a better knowledge of more effective control measures would be helpful. These aspects include:

   a. Nature of Airborne Radionuclides

      In connection with potential internal exposures of workers within TMI-2 containment, there is a need to specify the radionuclide composition of the various airborne particulates according to particle size. This has not apparently been done, yet it is essential to the assessment of the accompanying potential health hazard. The Subcommittee believes that studies should be undertaken to more clearly delineate the nature of the airborne radionuclides.

b. Internal Versus External Exposures

Workers entering containment for decontamination and recovery operations are currently required to wear full-scale protective equipment, including respirators. Closer examination of the increased external exposures, because of the impediments caused by the utilization of protective equipment, might show that it would be better to alter this approach (such as working faster without protective equipment). This needs further evaluation.

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

July 17, 1970

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

Subject: REPORT ON TROJAN NUCLEAR PLANT UNIT 1

Dear Dr. Seaborg:

At its 123rd meeting, July 9-11, 1970, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Portland General Electric Company to construct Unit 1 of the Trojan Nuclear Plant. A Subcommittee met to review this proposal on April 28, 1970, in Portland, Oregon, and on June 26, 1970, in Denver, Colorado. During its review, the Committee had the benefit of discussions with representatives of the applicant, the Westinghouse Electric Corporation, the Bechtel Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed.

The plant will be located on the west bank of the Columbia River, approximately 30 miles northwest of Portland, Oregon. The nearest population center is the Kelso-Longview community, six miles from the site, with a population of 39,000. The minimum exclusion distance is about 0.4 miles and the low population zone distance is 2.5 miles. Approximately 6,400 people live within five miles of the site.

The applicant has examined the region for geologic faults by trenching and aerial surveys with special attention to the fault exposed in a highway cut 4.7 miles north of the site. Neither the trenched exposures nor the aerial surveys revealed evidence of surface faulting at or in the vicinity of the site.

The applicant has chosen 0.15g operating basis and 0.25g design basis earthquakes for the site. The Committee believes these values are adequate and should be used in conjunction with conservatively derived response spectra. These spectra should be consistent with a magnification of about 3.2 at 2-percent damping.

The Trojan unit will include a four-loop pressurized water reactor designed for an initial core power level of 3411 MWt. The unit is very similar to the Sequoyah units which represent a five-percent higher power level than the 3250 MWt for similar units, recently reviewed. The Committee believes that, as is the case with the Sequoyah units, if the designer's expectations for this plant should not be adequately confirmed, system modifications or restrictions on operation may be appropriate.

The applicant proposes to increase the reactor coolant inlet and outlet temperatures for the Trojan unit by about 7.5°F over the Sequoyah temperatures, in order to improve the thermal efficiency. The Regulatory Staff should satisfy itself during construction that this increase is justified and that suitable thermal-hydraulic margins remain. The Committee wishes to be kept informed.

The applicant proposes to provide redundant containment spray systems, utilizing the Residual Heat Removal heat exchangers as heat sinks. The Committee notes that the total heat removal capability of this system is less than other systems approved, which use both spray and fan coolers. The Committee believes that additional containment cooling capacity should be included and notes that fan coolers, designed to engineered safety feature standards, may be of benefit in accomplishing more complete mixing of hydrogen in the atmosphere in the unlikely event of a loss-of-coolant accident.

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 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 prestressed reinforced concrete containment building differs from those for previously reviewed plants in that a hemispherical dome is used and the vertical prestressing tendons in the walls are extended and made continuous over the dome to provide the desired prestress. These changes have permitted elimination of the ring girder at the junction of the wall and dome and of the anchorages for the wall and dome tendons in this region. The haunch at the bottom of the wall has also been eliminated in this design. The Committee believes that this design is acceptable. However, because of the significant reduction in the number of individual prestressing tendons, it is especially important that the construction procedures be such as to insure that no tendon ducts will be made unusable because of collapse or blockage by concrete or grout.

The applicant is conducting an on-site meteorological monitoring program to provide the meteorological data which will be needed to establish the operating release limits and the diffusion rates following postulated accidents. If the results of this program should show the need for additional fission product removal equipment, the applicant has agreed to install the equipment.

The applicant has not designed the plant specifically to resist tornado loadings because of the very low probability of tornadoes in the Northwestern United States. The Committee recognizes the low probability of a tornado striking the plant. However, it believes that the vulnerability to tornado loads of critical safety components, such as emergency power and emergency cooling systems, should be reviewed to determine if additional protection may be desirable.

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 loss-of-coolant accident. The applicant proposes to make use of a venting technique 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 venting. The capability for venting 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 to prevent 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 Trojan Nuclear Plant.

The Committee believes that the above items can be resolved during construction and that, if due consideration is given to these items, the Trojan Nuclear Plant Unit 1 can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Joseph M. Hendrie
Chairman

References:

1. License Application, dated June 24, 1969; Volumes 1, 2 and 3 of the Preliminary Safety Analysis Report

2. Amendments Nos. 1 through 8 to the License Application

3. Supplement to Amendment No. 6, dated April 30, 1970

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

November 20, 1974

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

Subject: REPORT ON TROJAN NUCLEAR PLANT

Dear Dr. Ray:

At its 175th meeting, November 14-16, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application of the Portland General Electric Company, the City of Eugene, Oregon, acting by and through the Eugene Water and Electric Board, and the Pacific Power and Light Company for authorization to operate the Trojan Nuclear Plant. The project has been previously considered at a Subcommittee meeting on October 17, 1974, and a tour of the facility was made by Subcommittee members on October 18, 1974. The new Westinghouse 17x17 fuel rod assembly to be employed in the Trojan reactor was also reviewed by the Committee at the 175th meeting; it was previously considered at a Subcommittee meeting on July 30, 1974. In the review of the Trojan plant and of the 17x17 fuel assembly, the Committee had the benefit of discussions with representatives and consultants of the Applicant, the Bechtel Corporation, the Westinghouse Electric Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The plant will be located on a 635-acre site on the west bank of the Columbia River approximately 40 miles north of Portland, Oregon. The nearest population center is the Kelso and Longview, Washington area, six miles from the site, with a 1970 population of about 39,000.

At the time of the construction permit review, design for tornado loadings was not required for plants located west of the Rocky Mountains because of the very low probability of tornadoes in this region. Subsequent to that time, the design of those portions of the plant critical to safety was reviewed and changes were made as needed to provide resistance to tornado loadings corresponding to wind velocities of at least 200 miles per hour. The Committee believes that the plant as constructed is in reasonable conformance with the tornado criteria now incorporated in Regulatory Guide 1.76, and that a satisfactory level of safety has been achieved.

The Trojan Nuclear Plant will include a four-loop Westinghouse nuclear steam supply system with a design core power level of 3411 MWt. The reactor will be one of the first to operate with fuel assemblies having a 17x17 rod array.

At the construction permit stage, the Trojan plant design included redundant spray systems for the removal of heat from the containment following a postulated loss-of-coolant accident. As suggested by the Committee in its letter of July 17, 1970, the design has been changed to include both spray and fan-cooler systems and thus provides diversity as well as redundancy.

The Regulatory Staff has proposed that the power be disconnected from certain motor-operated valves in the ECCS in order to prevent a single failure in the electrical system from disabling a part of an essential safety system. The Applicant has argued that a spurious signal is highly improbable and that locking out power to these valves will not necessarily lead to greater safety. The Committee believes that a complete systems analysis of this generic problem has not been made which takes into account all possible failures, both electrical and mechanical, for these valves in both the locked-out and normal configurations, and recommends that additional studies be made of possible alternatives. This matter should be resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The Committee believes that the Trojan plant should be provided with a system capable of prompt detection of gross failure of a fuel element. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

Several changes are to be made in the Westinghouse ECCS evaluation model to bring it into conformance with the Commission Criteria as given in 10 CFR 50.46. The performance of the emergency core cooling systems will be re-evaluated with the approved evaluation model, and appropriate operating limits and procedures for ensuring monitoring of the power distribution are to be incorporated in the Technical Specifications. The Committee wishes to be kept informed.

The evaluation of Anticipated Transients Without Scram (ATWS) has been made generically for Westinghouse plants, and the Applicant has made comparisons indicating that the results obtained are applicable to the Trojan plant. Regulatory review should be completed and this matter resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The Westinghouse 17x17 fuel rod array is identical to that to be used in Catawba Units 1 and 2 and in several other nuclear power stations which have recently been reviewed for construction by the Committee. The Trojan plant is scheduled to be one of the first to go into operation using a full core of 17x17 fuel. While many of the various required verification programs have been completed and reviewed by the Regulatory Staff, other tests and analyses are still to be completed and documented. These include: DNB tests with non-uniform heat flux, single-rod burst tests, fuel assembly flow tests, guide tube tests, and the effect of bowing on DNB. The results of such tests and analyses should be evaluated fully by the Regulatory Staff and resolved to their satisfaction prior to the full core use of 17x17 fuel to produce power. Four prototype 17x17 fuel rod assemblies are to be loaded into other operating pressurized water reactors in the near future; the results of these irradiations should be followed closely. The Committee wishes to be kept informed concerning the results of the various ongoing 17x17 test and analytical programs and any design changes which may be proposed in the future.

The Applicant has proposed a fuel surveillance program to follow the behavior of the fuel as its irradiation progresses. Following each cycle of operation, 17x17 fuel assemblies will be examined for fuel rod integrity, fuel rod and assembly dimension and alignment, and surface deposits. In addition, one fuel assembly will contain fuel rods which can be removed to facilitate interim and end-of-life fuel rod evaluation as a function of exposure. In view of the fact that the 17x17 fuel array is a new design and that no prototype irradiations are planned for 17x17 fuel containing eight spacer-grids (which will be employed only in full - core operation), the results of this surveillance program should be followed closely. The Committee wishes to be kept informed.

The recently proposed method of constant axial offset control will be used for in-core power distribution monitoring and control. The Regulatory Staff should review carefully the effectiveness of this method of control in protecting against adverse consequences of postulated reactor transients and accidents. The Committee wishes to be kept informed.

The Trojan reactor may be the first reactor of its type to operate with a rated power as high as 3411 MWt. Because there is limited operating experience with very large, high power density reactors, the ACRS believes that a more cautious than normal approach to full power is prudent, with longer periods of operation at power levels in the range of 70 to 90% of full power, and with additional monitoring of core and systems performance throughout the life of the first core. The Committee recommends that the Regulatory Staff evaluate the overall operating experience prior to sustained operation at full power.

Generic problems relating to large water reactors have been identified by the Regulatory Staff and the ACRS and discussed in the Committee's report dated February 13, 1974. These problems should be dealt with appropriately by the Regulatory Staff and the Applicant as suitable approaches are developed.

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 preoperational testing, there is reasonable assurance that the Trojan Nuclear Plant can be operated at power levels up to 3411 MWt without undue risk to the health and safety of the public.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References

1. Final Safety Analysis Report for the Trojan Nuclear Plant, Volumes 1-9 (including Amendments Nos. 1-17)
2. "Safety Evaluation Report, Trojan Nuclear Plant," U.S.A.E.C. Directorate of Licensing, dated October 7, 1974
3. "Trojan Nuclear Plant Safety-Related Schematic Diagrams," PGE-1001, Volumes 1-2, (including Amendments Nos. 1-4)
4. "Trojan Nuclear Plant Safety-Related Schematic Diagrams," PGE-1002, dated March, 1973 (PROPRIETARY)
5. "Trojan Nuclear Site - Results of Laboratory Rock Testing," Bechtel Engineering Corporation (undated)
6. "Geophysical Survey of Trojan Nuclear Power Plant Site, Longview, Washington" PC Exploration (undated)
7. "Trojan Nuclear Plant Analysis of Pipe System Breaks Outside Containment," PGE-1004 (including Revision #1, dated January, 1974)
8. Letter, Portland General Electric Company (PGE) to U. S. AEC Directorate of Licensing (DL), dated June 17m 1974, concerning preoperational testing of emergency core cooling systems
9. "Radiological Emergency Response Plan Trojan Nuclear Plant," Revision A PGE-1008, dated April, 1974

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

January 18, 1967

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

Subject: REPORT ON TURKEY POINT NUCLEAR GENERATING UNITS NO. 3 AND NO. 4

Dear Dr. Seaborg:

At its eighty-first meeting, January 12-14, 1967, the Advisory Committee on Reactor Safeguards completed its review of the application of Florida Power and Light Company for authorization to construct Turkey Point Nuclear Generating Units No. 3 and No. 4. This project had previously been considered at the seventy-ninth meeting of the Committee, November 10-12, 1966, and at Subcommittee meetings on September 7, November 9, and December 7, 1966, and January 7, 1967. Representatives of the Committee visited the site on December 16, 1966. During its review, the Committee had the benefit of discussions with representatives of Florida Power and Light Company, Westinghouse Electric Corporation, Bechtel Corporation, and the AEC Regulatory Staff and its consultants. The Committee also had the benefit of the documents listed.

The Turkey Point Units are to be located in Dade County, Florida, on the west shore of Biscayne Bay approximately 25 miles south of Miami. Each unit includes a pressurized water reactor to be operated at an initial maximum power level of 2097 MWt but designed to operate ultimately at a maximum power level of 2300 MWt.

The containment structure for each unit consists of a steel-lined concrete shell with shallow spherical dome and flat slab base. The shell and dome are fully prestressed, with steel tendon systems carrying the principal loads. Provisions are made for in-service inspectability, replaceability, and corrosion control of the tendons over the lifetime of the structure.

The complex of emergency core cooling systems provided for each unit includes a high head safety injection system and a low head residual heat removal system with an accumulator subsystem. The accumulators are capable of very rapid addition of borated water to the reactor in the unlikely event of a large scale loss-of-coolant accident, and increase the time margin available for initiation of emergency cooling flow by pumping. The high head safety injection system pumps (three) are shared by Units 3 and 4. These systems appear to be adequate for the Turkey Point reactors. The AEC Regulatory Staff should review carefully the final design of the emergency core cooling systems, including the analyses of system characteristics and the effects of blowdown on reactor internals.

The reactor is calculated to have a positive moderator coefficient during a portion of core life. The applicant will give careful attention to the influence of positive coefficients on reactor transients, including the loss-of-coolant accident, rapid control rod motion, and xenon oscillations. If necessary, the moderator coefficient will be modified by the addition of solid burnable poison to the core. The Committee feels that the Regulatory Staff should follow closely the status of this aspect of design. The ACRS would like to be kept informed with respect to both the emergency core cooling and the moderator coefficient studies.

The frequency and intensity of hurricanes at the Turkey Point site present problems of potential flooding and wind damage. The applicant has made preliminary estimates of wind forces, water levels, and wave heights associated with the maximum probable hurricane against which vital components of the plant are to be protected. Remaining questions on the appropriate degree of protection will be resolved between the applicant and the AEC Regulatory Staff.

The applicant desires to continue uninterrupted operation of the reactor in the event one of two or more redundant active components in an engineered safeguard system becomes temporarily inoperable. The associated operable components would be maintained in continuous operation, however, until the inoperable component is again ready for service. The Committee feels that this matter may require review at the time of application for an operating license.

The Advisory Committee on Reactor Safeguards believes that the various items mentioned can be resolved during construction and that the proposed reactors can be built at the Turkey Point site with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Mr. Harold Etherington did not participate in review of the Turkey Point Nuclear Generating Units No. 3 and No. 4.

Sincerely yours,

/s/ by N. J. Palladino

N. J. Palladino
Chairman

References Attached.

References - Turkey Point

1. Florida Power & Light Company, Turkey Point Plant Units No. 3 and No. 4 Application for Licenses, dated March 22, 1966.
2. Preliminary Safety Analysis Report, Volumes 1, 2, and 3.
3. Supplement No. 1 to Application for Licenses, dated May 4, 1966.
4. Florida Power and Light Company Letter, dated May 9, 1966, submitting corrected pages for Supplement No. 1.
5. Supplement No. 2 to Application for Licenses, dated August 11, 1966.
6. Supplement No. 3 to Application for Licenses, dated September 1, 1966.
7. Florida Power and Light Company Letter, dated September 8, 1966, regarding correction to Supplement No. 3.
8. Supplement No. 4 to Application for Licenses, dated September 6, 1966.
9. Supplement No. 5 to Application for Licenses, dated October 7, 1966.
10. Florida Power and Light Company Letter, dated November 17, 1966, submitting additional information regarding Question No. 6 in Supplement No. 5.
11. Supplement No. 6 to Application for Licenses, dated September 30, 1966.
12. Supplement No. 7 to Application for Licenses, dated October 12, 1966.
13. The Effect of Xenon Spatial Variations and the Moderator Coefficient on Core Stability, WCAP-2983, dated August 1966.
14. Supplement No. 8 to Application for Licenses, dated November 4, 1966.
15. Supplement No. 9 to Application for Licenses, dated November 29, 1966.
16. Supplement No. 10 to Application for Licenses, dated December 9, 1966.
17. Supplement No. 11 to Application for Licenses, dated January 4, 1967.

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

May 15, 1968

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

Subject: TURKEY POINT NUCLEAR GENERATING UNITS NO. 3 and NO. 4

Dear Dr. Seaborg:

At its ninety-seventh meeting, May 9-11, 1968, the Advisory Committee on Reactor Safeguards reviewed the proposal of Florida Power and Light Company in Supplements 14, 15, and 16 to their application for authorization to construct Turkey Point Nuclear Generating Units No. 3 and No. 4. The Committee had previously considered the project at its seventy-ninth meeting, November 10-12, 1966, and its eighty-first meeting, January 12-14, 1967, as reported to you by letter on January 18, 1967. During its review, the Committee had the benefit of discussions with representatives of Florida Power and Light Company and its consultants, Westinghouse Electric Corporation, Bechtel Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

In the original application to construct Turkey Point Nuclear Generating Units No. 3 and No. 4, the applicant had included in the exclusion zone land not owned by him. At the Public Hearing held on February 28-March 3, 1967, the Atomic Safety and Licensing Board questioned the ability of the applicant to control access to this land. In Supplements 14-16, the applicant proposes to revise the exclusion area and to install a filter system in the containment structure. The applicant now proposes to reduce the exclusion zone to land he owns and to the waters of Biscayne Bay. In order to reduce post-accident iodine concentrations at the nearest exclusion zone boundary to below the 10 CFR Part 100 guideline values, he proposes to install an iodine removal system in each containment building.

The iodine removal system consists of three filter units, each containing a demister filter bank, a high efficiency particulate air filter bank, and a charcoal filter bank installed with an electric-motor-driven blower and air diffuser in a common enclosure. The units are designed to operate under the adverse conditions expected in the containment in the case of a loss-of-coolant accident. Provisions have also been made to insure operation in case of loss of outside power. Arrangements for maintenance and testing appear adequate. Calculations, using a highly conservative model, indicate that the radiation dose to the thyroid at and beyond the nearest site boundary will be well within the guidelines of 10 CFR Part 100.

The Advisory Committee on Reactor Safeguards believes the proposed iodine removal system provides assurance that the proposed reactors can be built at the Turkey Point site with the modified exclusion zone so that, with suitable attention to the items discussed in our letter of January 18, 1967, they can be operated without undue risk to the health and safety of the public.

Mr. Harold Etherington did not participate in review of the Turkey Point Nuclear Generating Units No. 3 and No. 4.

Sincerely yours,

/s/ Carroll W. Zabel

Carroll W. Zabel
Chairman

References:

1. Supplement 14, Preliminary Safety Analysis Report, Turkey Point Nuclear Generating Units No. 3 and No. 4.

2. Supplement 15, Preliminary Safety Analysis Report, Turkey Point Nuclear Generating Units No. 3 and No. 4.

3. Supplement 16, Preliminary Safety Analysis Report, Turkey Point Nuclear Generating Units No. 3 and No. 4.

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

June 18, 1971

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

Subject: REPORT ON TURKEY POINT NUCLEAR GENERATING PLANT UNITS 3 AND 4

Dear Dr. Seaborg:

At its 134th meeting, June 10-12, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application of Florida Power and Light Company for authorization to operate Turkey Point Nuclear Generating Units 3 and 4 at power levels up to 2200 MW(t). This project had been considered previously at the 127th, 131st, and 132nd Committee meetings of November 12-14, 1970, March 4-6, 1971, and April 1-3, 1971, respectively, and at Subcommittee meetings at the site on November 7, 1970 and March 19, 1971. During its review, the Committee had the benefit of discussions with representatives of Florida Power and Light Company, Westinghouse Electric Corporation, Bechtel Corporation, and the Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported to you on the construction of these units in its letters of January 18, 1967 and May 15, 1968.

Turkey Point Units 3 and 4 are located in Dade County, Florida on the west shore of Biscayne Bay approximately 25 miles south of Miami. They share the site with two oil and gas fired units. Each nuclear unit employs a pressurized water reactor in a three-loop nuclear steam supply system of essentially the same design as the H. B. Robinson Unit No. 2, previously reviewed.

The containment structure for each unit consists of a steel-lined concrete cylinder with a flat base and a shallow domed roof. The wall is prestressed with vertical and horizontal tendons; the dome is prestressed with a three-way tendon system.

During construction, the concrete in a portion of the Unit 3 containment building dome was found to contain extensive cracks parallel to and at depths as much as 15 inches below the outer surface. The applicant, together with his contractor and consultants, developed and implemented procedures for removing the damaged concrete; repairing or replacing tendon sheaths, tendon wires, and reinforcing bars damaged during the concrete removal; replacing the concrete; and, retensioning the tendons. Although the reasons for the cracking have not been established conclusively, several possible mechanisms have been identified and measures have been taken to prevent their recurrence. The Committee believes that the repairs made, together with the much more frequent and more extensive surveillance program which will be carried out, provide reasonable assurance that the containment will be able to perform its design function in the unlikely event of a loss-of-coolant accident.

The applicant states that he intends to operate Units 3 and 4 in such a manner as to assure that maximum fuel rod linear power does not exceed 15.8 KW/ft at full reactor power of 2200 MW(t). Performance of the emergency core cooling system (ECCS) during postulated loss-of-coolant accidents has been reevaluated in the light of results from the Commission's FLECHT program, experiments and analyses by the applicant and his contractors, and information developed by the Regulatory Staff in recent studies of ECCS. The Committee believes that the indicated performance is satisfactory.

Conservative pressure-temperature relationships should be established to cover reactor start-up and shut-down. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Committee reiterates its previous comments concerning the need to study further means of preventing common mode failures from negating reactor scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. The Committee wishes to be kept informed of the resolution of this matter.

The applicant proposes to use a purging technique to control the buildup of hydrogen in the containment that could follow in the unlikely event of a loss-of-coolant accident. Installation of the purge system should be completed prior to start of routine operation. The Regulatory Staff should assure itself that the design criteria for the system are consistent with those for other engineered safety features.

An extensive integrated program for measuring vibration of reactor vessel internals and primary system components is being carried out on several previously licensed pressurized water reactors. The Committee believes that some confirmatory vibration measurements are desirable for the Turkey Point Units, as for all reactors. The Regulatory Staff should review the results of vibration measurements on other plants with regard to their applicability to Turkey Point and should determine the confirmatory measurements to be made.

Other problems relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Turkey Point Nuclear Generating Units 3 and 4 can be operated at power levels up to 2200 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H Bush

Spencer H. Bush
Chairman

References
1) Supplement Nos. 17 through 28 and 30 through 36 to the application and Final Safety Analysis Report
2) Florida Power & Light Company letter dated December 23, 1970 transmitting a report describing the distress observed in Turkey Point Unit 3 containment dome
3) Florida Power & Light Company letter dated January 25, 1971 transmitting a report describing the concrete replacement program for the Turkey Point Unit 3 containment dome
4) Florida Power & Light Company letter dated April 22, 1971 transmitting Security Plan for Turkey Point Units 3 and 4

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

December 11, 1975

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

SUBJECT: REPORT ON THE TYRONE ENERGY PARK, UNIT NO. 1

Dear Mr. Anders:

During its 188th meeting, December 4-6, 1975, the Advisory Committee on Reactor Safeguards reviewed the application of the Northern States Power Company of Wisconsin and the Northern States Power Company of Minnesota (Applicants) for a permit to construct the Tyrone Energy Park, Unit No. 1. A visit to the site of the proposed plant was made on November 20, 1975, and a Subcommittee meeting was held at Eau Claire, Wisconsin on November 21, 1975. The "Standardized Nuclear Unit Power Plant System" (SNUPPS), to be utilized at the Tyrone Energy Park site and at three other plant sites, was reviewed at Subcommittee meetings held at Washington, DC, on August 19, 1975, and at Emporia, Kansas on September 26, 1975, and at the 185th meeting of the Committee, September 11-13, 1975, and the 186th meeting of the Committee, October 9-11, 1975. During its review of the Tyrone Energy Park Unit 1, the Committee had the benefit of discussions with the Nuclear Regulatory Commission (NRC) Staff and representatives and consultants of the Applicants, the Westinghouse Electric Corporation, the Bechtel Power Corporation, and Commonwealth Associates, Inc. The Committee also had the benefit of the documents listed.

The Tyrone unit will be located on a 4597-acre site of partially wooded rural land about 19 miles southwest of Eau Claire, Wisconsin, the nearest population center (1970 population: about 45,000). The exclusion area has a minimum boundary distance of 1470 meters from the reactor centerline. The radius of the low population zone is 2.5 miles.

The SNUPPS will utilize the RESAR-3 Consolidated Version, four-loop, pressurized water reactor with a core power output of 3411 MW(t). This design is similar to that utilized at the Comanche Peak Steam Electric Station, Units 1 and 2, reported on by the Committee in its report of October 18, 1974. The Committee's review of the SNUPPS was reported on in its Callaway letter of September 17, 1975, and is further reported on in this letter.

The NRC Staff has identified several items in the Tyrone application the reviews for which are not yet completed. The Committee recommends that any outstanding issues which may develop in the course of completing these reviews be dealt with in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed on the resolution of the following items:

1. The analyses of the effects of anticipated transients without scram.

2. The evaluation of the plant design to meet the requirements of Appendix I of 10 CFR Part 50.

The RESAR-3 Consolidated Version nuclear design utilizes assemblies with the 17x17 fuel rod array. Westinghouse has completed an integrated test program to confirm safety margins associated with this design. The last of a series of reports on this program is expected soon. The RESAR-3 reactor core design has been analyzed by Westinghouse with respect to stability against radial xenon oscillations. Westinghouse has agreed to verify this stability in startup physics tests for a 193 fuel assembly core similar to SNUPPS. The Committee will continue to review these matters as appropriate documentation is submitted.

The Committee recommended in its report of September 10, 1973, on acceptance criteria for ECCS, that significantly improved ECCS capability should be provided for reactors for which construction permit requests are filed after January 7, 1972. The SNUPPS design is in this category. These units will use assemblies with a 17x17 fuel rod array similar to those to be used in Comanche Peak Steam Electric Station, Units 1 and 2. Although calculated peak clad temperatures in the event of a postulated LOCA are less for assemblies with a 17x17 than with a 15x15 fuel rod array, the Committee believes that the Applicants and the reactor vendors should actively pursue studies that are responsive to the Committee's September 10, 1973 report. If studies establish that significant further ECCS improvements can be achieved, consideration should be given to their incorporation into the Tyrone unit.

Although the NRC Staff has concluded that the Applicants will comply with the Final Acceptance Criteria for the Emergency Core Cooling Systems, the Committee wishes to be kept informed on the resolution including possible effects from rod bowing.

The Committee believes that the Applicants and the NRC Staff should continue to review the Tyrone plant design for features that could reduce the possibility and consequences of sabotage.

The Committee recommends that the NRC Staff and the Applicants review the design features that are intended to prevent the occurrence of damaging fires and to minimize the consequences to safety-related equipment should a fire occur. The Committee wishes to be kept informed.

Generic problems relating to large water reactors are discussed in the Committee's report dated March 12, 1975. These problems should be dealt with appropriately by the NRC Staff and the Applicants.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above and the items mentioned in its Callaway report, which are relevant to the Tyrone application, can be resolved during construction and that if due consideration is given to the foregoing, the Tyrone Nuclear Energy Park, Unit No. 1 can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,
WKerr

W. Kerr
Chairman

References

1. SNUPPS Preliminary Safety Analysis Report with Revision 1 through 13 and the Tyrone Site Addendum Report with Revision 1 through 6.

2. RESAR-3 Consolidated Version, Westinghouse Safety Analysis Report with Amendments 1 through 6.

3. Safety Evaluation Report, NUREG-75/102 related to the Construction of Tyrone Energy Park, Unit No. 1, Docket No. STN 50-484, October 1975.

4. Letter dated November 21, 1975, from Mrs. Galen C. Radle to Advisory Committee on Reactor Safeguards, Nuclear Regulatory Commission.

5. Supplement No. 1 to the Safety Evaluation Report, NUREG-75/076 related to Construction of Callaway Plant Units 1 and 2, Docket Nos. STN 50-483 and STN 50-486, November 1975.

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

March 8, 1958

Honorable Lewis L. Strauss
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: GENERAL ELECTRIC BOILING WATER REACTOR AT VALLECITOS

Dear Mr. Strauss:

The problems of the Vallecitos Boiling Water Reactor were considered by the Advisory Committee on Reactor Safeguards with representatives of the Licensee, General Electric Company, and the Hazards Evaluation Branch. The pertinent documents are listed below.

General Electric has asked for three revisions to their license, Construction Permit CPPR-3 and License Nos. CX-2 and DPR-1 as amended, for this reactor: Amendment No. 14 would increase the number of rod type elements in a loading, from one to 14 for thermal powers up to 20 megawatts. Amendment No. 15 would increase allowable thermal power with mixed flat plate and fourteen rod type elements, from 20 to 30 megawatts. Amendment No. 16 would allow heating of the reactor from cold to operating temperature by nuclear heat at a rate not exceeding 1 Mw instead of by heat supplied to the coolant externally.

In view of the reported performance so far obtained in operations of this reactor, the Committee sees no significant increase in hazard in following these proposals provided that rod type elements be so dispersed in the core as to be as uniformly surrounded by plate type elements as possible.

There seems to be some possibility of a "cold-water" type accident in operation of the system and its controls as now constituted, i.e., sudden introduction of cold coolant into the reactor causing excessive increase in reactivity and power. This possibility should receive further study and if modifications of control systems or operating procedures are necessary to ensure against such an accident they should be made before resuming operation.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman
Advisory Committee on
Reactor Safeguards

cc: K. E. Fields, GM
H. L. Price, DL&R

References:

1. GE-BWR Report SG-VAL 1
2. GE-BWR Report SG-VAL 2
3. Amendments No. 13 thru No. 16
4. HEB Staff Report of Feb. 25, 1958

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C.

August 5, 1958

Honorable John A. McCone
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: GENERAL ELECTRIC VALLECITOS BOILING WATER REACTOR (GE-VBWR)

Dear Mr. McCone:

The Advisory Committee on Reactor Safeguards at its Ninth Meeting on August 4, 1958, considered Amendment No. 24 to the License Application of the General Electric Company for the operation of the Vallecitos Boiling Water Reactor. This amendment is designed to provide the operator with greater latitude in the choice of fuel elements and operating limits. In reviewing the amendment application the Committee considered the supporting material submitted in the reports referenced below and held a meeting with representatives of the applicant and members of the Hazards Evaluation Branch.

The Committee believes that the technical specifications set out in Section 1 of Amendment No. 24 define a scope of operations within which it is possible to operate without undue hazard to the public. This belief is based on the assumption that the specifications outlined do not affect the magnitude of the postulated maximum credible accident, which the applicant has shown does not result in the release of dangerous amounts of radioactivity beyond the site boundary provided the container maintains its specified leak tightness. While, as stated, it is the Committee's belief that the proposed technical limitations have no influence on the magnitude of the maximum credible accident, this has not in fact been clearly demonstrated by the applicant, and we believe this point should be documented more definitely.

On the other hand, the Committee would like to emphasize that the technical specifications alone do not guarantee the safe operation of the reactor, especially from the standpoint of hazards within the boundaries of the site.

The Committee is concerned with the mounting number of amendments to the licensee's application on the VBWR operation which apparently stems from attempts to cover a multiplicity of specific situations pertinent to the safe operation of the reactor. Rather than improving the safety it seems possible that the real issues of safe operation may get beclouded by changing one set of circumstances for another.

The Advisory Committee on Reactor Safeguards suggests that the applicant be permitted within the scope of Amendment No. 24 to assume technical responsibility concerning hazards in connection with his experimental program.

The latitude contemplated in the proposed amendment imposes a special responsibility on the applicant to review each proposed change in operating conditions in the light of its effect on the probability of an accident and on the possible severity of the accident if it occurs.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman

cc: Paul F. Foster, GM
H. L. Price, DL&R

References:

1) Amendment No. 24 to License Application for Vallecitor Boiling Water Reactor, May 14, 1958.

2) SG-VAL-2, Second Edition, Final Hazards Summary Report, May 8, 1958.

3) Key to Second Edition of SG-VAL-2, May 8, 1958.

4) Report to ACRS by Division of Licensing and Regulation on GE-VBWR, August 1, 1958.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

February 8, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: GENERAL ELECTRIC VALLECITOS BOILING WATER REACTOR (GEVBWR)

Dear Mr. McCone:

At its twenty-third meeting on January 28-30, 1960, the Advisory Committee on Reactor Safeguards reviewed the General Electric Vallecitos Boiling Water Reactor Final Hazards Summary Report, SG-Val-2 up to and including the Third Edition; the Amendments to License Application for Vallecitos Boiling Water Reactor up to and including Amendment #43; the Division of Licensing and Regulation Reports to the ACRS on Reactor Safeguards on GEVBWR up to and including the one dated January 12, 1960; and the material presented at this meeting by representatives of the General Electric Company.

The reactor control rods have two features which might cause difficulties: 1) a pneumatic system for the rod motion and 2) sliding seals on the rod shafts. The Committee anticipates satisfactory operation of the control rods because of the applicant's past experience with this type of mechanism and because of his stated intent to:

a) Use a separate pneumatic system for each set of two rods;

b) Test initially, at operating temperature and pressure, at least <u>one</u> pair of the rods sufficient times to obtain reliable statistics on drop time, and the remaining rods sufficient times to show that all rods behave in a similar way;

c) Eliminate conventional lubricating oil from the pneumatic scram system and to minimize the oil explosion hazard from this system.

The multiple purpose nature of the plant inherently results in a somewhat unusual number of interlocks and administrative procedures to insure that each component fulfills at any given time its required function. This places a heavy responsibility on the operational managment.

The Committee believes that the worth of the control rods for each substantially different core should be measured and evaluated during core loading in terms of the excess reactivity possibly available in that core loading, and in terms of any possible malfunction of the control rods to insure an adequate shutdown margin under forseeable conditions. This may limit or preclude certain types of operation.

The exclusion area is being used for agricultural purposes. The problem of the use of a reactor exclusion area for purposes unrelated to reactor operation is obviously applicable to many other sites. The Committee believes that such use is compatible with the health and safety of the public in many cases, including the case of the Vallecitos Boiling Water Reactor, and recommends that the required monitoring rules for people, crops, livestock, and equipment be established.

The Committee believes the plant can be modified as proposed, and operated in accord with the above without undue hazard to the health and safety of the public.

Sincerely yours,

/s/ Leslie Silverman

Leslie Silverman
Chairman

cc: A. R. Luedecke, GM
H. L. Price, DL&R
W. F. Finan, OGM

References:

1) SG-VAL-2, Third Edition - Final Hazards Summary Report, November 30, 1959.

2) Amendment No. 43 to License Application for Vallecitos Boiling Water Reactor, January 22, 1960.

3) Report to ACRS by Division of Licensing and Regulation dated January 12, 1960.

4) General Electric Company letter to H. L. Price dated November 7, 1959.

5) U. S. Weather Bureau Comments on SG-VAL-2, Third Edition, dated January 8, 1960.

6) Memorandum from J. E. Turner to E. R. Price, Subject - Vallecitos Boiling Water Reactor, Amendment #41 dated January 29, 1960.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON 25, D.C.

May 20, 1961

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

Subject: REPORT ON VALLECITOS EXPERIMENTAL SUPERHEAT REACTOR (VESR)

Dear Dr. Seaborg:

At its thirty-fourth meeting on May 18-20, 1961, in Cambridge, Massachusetts, the Advisory Committee on Reactor Safeguards considered the Vallecitos Experimental Superheat Reactor (VESR), on the basis of the documents referenced below and discussion with representatives of the General Electric Company and the staff of the Atomic Energy Commission. Prior to this meeting, the VESR had been considered by the Advisory Committee on Reactor Safeguards at its thirty-third meeting on April 6-8, 1961; at the ACRS subcommittee meeting on March 14, 1961 at San Jose, California and at the Vallecitos Site; and at the ACRS subcommittee meeting on April 25, 1961 in Washington, D. C.

The Committee notes three points of interest to reactor safety:

1. The VESR is an experimental reactor. Tests will include operation with purposely defected fuel elements and, generally, operation outside previously established experience.

2. Because the VESR is an all-superheater reactor, the positive reactivity effects connected with unintended flooding or unflooding of steam passages are relatively great.

3. The main control system of the VESR inserts nuclear poison rods against the forces of gravity and of reactor pressure, and the main control system fulfills the functions of shim, regulation, and scram. It is consequently somewhat complex.

The applicant has stated that he intends to provide safety rods which move in the direction of gravity and not against reactor pressure while inserting poison. These rods will be designed solely for the scram function. The reactivity worth of these rods -- approximately $2\%\Delta k$ -- if added to the prompt (negative) reactivity change available prior to destructive fuel melting, should override the maximum positive reactivity effect of flooding or unflooding of the steam passages.

The following items are among those which are not intimately connected with the construction of the main part of the plant and which the Committee would like to consider at a later time:

1. Operation at a power level above 12.5 MW(t).

2. Operation with steam supplied by the Vallecitos Boiling Water Reactor and operation in connection with the VBWR turbine.

3. The shutdown margin.

4. Containment of the steam line to the condenser, and of the condenser itself.

5. Specifications and measurement of the containment leakage rate.

6. Control of routine radioactivity release to the atmosphere, on-site environmental radioactivity monitoring, and interaction in these respects of the VESR with other plants on the Vallecitos site.

The Advisory Committee on Reactor Safeguards believes that, with the addition of the safety rods discussed above, a reactor of the type proposed can be constructed at the Vallecitos site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/

T. J. Thompson
Chairman

References:

1. GEAP-3643, Preliminary Hazards Summary Report for the Vallecitos Superheat Reactor, dated February 1, 1961.
2. Amendment #1 to License Application, dated March 14, 1961.
3. Amendment #2 to License Application, dated March 24, 1961.
4. Amendment #3 to License Application, dated April 14, 1961.
5. Amendment #4 to License Application, dated May 1, 1961.

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

August 30, 1962

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

SUBJECT: REPORT ON VALLECITOS BOILING WATER REACTOR (VBWR)

Dear Dr. Seaborg:

At the request of the Division of Licensing and Regulation, the Advisory Committee on Reactor Safeguards at its forty-third meeting, August 23-25, 1962, at Idaho Falls, Idaho, considered the documents referenced below regarding the shutdown margin of the VBWR. At present, it is possible by the physical removal of a single VBWR control rod to cause the reactor to become supercritical when cold and xenon-free.

With the mechanical and electrical safeguards, and the administrative procedures as described in the referenced documents made effective, and because the diminished shutdown margin will exist for a definitely limited time during which the consequences of excessive control rod withdrawal will be continuously recognized by operating personnel, the ACRS believes that the VBWR may be operated as proposed without undue risk to the health and safety of the public.

Sincerely yours,

/s/ F. A. Gifford, Jr.

F. A. Gifford, Jr.
Chairman

References:
1. Letter from General Electric to AEC, dtd 4/19/62 - Change No. 22.
2. Letter from General Electric to AEC, dtd 8/10/62.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

April 18, 1963

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

Subject: REPORT ON ESADA VALLECITOS EXPERIMENTAL SUPERHEAT REACTOR (EVESR)

Dear Dr. Seaborg:

At its forty-seventh meeting, April 11-13, 1963, the Advisory Committee on Reactor Safeguards considered the ESADA Vallecitos Experimental Superheat Reactor (EVESR) on the basis of the documents listed and discussion with representatives of the General Electric Company and the Staff of the Atomic Energy Commission. The EVESR was the subject of an ACRS report dated May 20, 1961 and was discussed at a subcommittee meeting on November 29, 1962.

Criteria and limits restricting the replacement of Mark II fuel bundles with experimental fuel (including bundles incorporating thoria and the use of small amounts of plutonia) should be made final by discussions between the applicant and the AEC Regulatory Staff.

The points and items noted in the ACRS report of May 20, 1961 appear now resolved to the extent that the Committee concludes that there is reasonable assurance that the EVESR can be operated at power levels up to 12.5 MW(t) with steam from the gas-fired boiler without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

D. B. Hall
Chairman

References attached.

EVESR References

1. Amendment No. 5 to License Application for ESADA Vallecitos Experimental Superheat Reactor (EVESR), dated October 5, 1962.
2. APED-3958, Final Hazards Summary Report for the ESADA Vallecitos Experimental Superheat Reactor, dated October 1, 1962.
3. Amendment No. 6 to License Application for ESADA Vallecitos Experimental Superheat Reactor (EVESR), dated December 21, 1962, with attachments.
4. Letter from L. C. Koke, General Electric, to Saul Levine, AEC, dated January 14, 1963, with attachments. Subject: ESADA Vallecitos Experimental Superheat Reactor, Docket No. 50-183.
5. Amendment No. 7 to License Application for ESADA Vallecitos Experimemtal Superheat Reactor (EVESR), dated March 5, 1963, with attachments.
6. Amendment No. 8,to License Application for ESADA Vallecitos Experimental Superheat Reactor (EVESR), dated March 27, 1963.
7. Letter from L. C. Koke, General Electric, to Saul Levine, AEC-DL&R, dated April 3, 1963. Subject: ESADA Reactor, Docket No. 50-183.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

October 15, 1964

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

Subject: REPORT ON ESADA-VALLECITOS EXPERIMENTAL SUPERHEAT REACTOR (EVESR)

Dear Dr. Seaborg:

At its 58th meeting, October 7-10, 1964, the Advisory Committee on Reactor Safeguards considered Proposed Change No. 9 for the ESADA-Vallecitos Experimental Superheat Reactor (EVESR) on the basis of the documents listed below and discussion with representatives of the General Electric Company and the AEC Regulatory Staff. The ACRS reported on this reactor in its letters of May 20, 1961 and April 18, 1963.

The proposed changes would permit an increase in power level to 17 MW(t), a 100° F increase in maximum allowable fuel clad temperature, and a 44% increase in specific power. The applicant has also analyzed operation of the reactor with four bundles of Mark III type fuel in the superheat region.

It was reported that operation to date at powers up to 12.5 MW(t) has not raised any unexpected safety problems.

The Committee concludes that there is reasonable assurance that the EVESR can be operated at power levels up to 17 MW(t) in accordance with Proposed Change No. 9 without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References Attached.

References:

1. Letter from B. D. Wilson, General Electric Company to R. L. Doan, AEC, dated August 14, 1964 with attached Proposed Change No. 9, ESADA-Vallecitos Experimental Superheat Reactor (EVESR).

2. Letter from B. D. Wilson, General Electric Company to Saul Levine, AEC, dated August 21, 1964 with enclosures.

3. TWX from E. W. O'Rorke, General Electric Company to Saul Levine, AEC, dated September 21, 1964, subject: Modification PC No. 9 EVESR.

4. Letter from E. W. O'Rorke, General Electric Company to R. L. Doan, AEC, dated September 25, 1964, with enclosures.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

October 15, 1965

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

Subject: REPORT ON ESADA VALLECITOS EXPERIMENTAL SUPERHEAT REACTOR (EVESR)

Dear Dr. Seaborg:

At its sixty-seventh meeting, October 7-9, 1965, the Advisory Committee on Reactor Safeguards considered the request by the General Electric Company that the provisional license to operate the ESADA Vallecitos Experimental Superheat Reactor (EVESR) be converted to a ten year license. The Committee had the benefit of discussion with representatives of the General Electric Company and the AEC Regulatory Staff, and of the documents listed below. The ACRS has reported on this reactor previously in its letters of May 20, 1961, April 18, 1963, and October 15, 1964. The last of these letters discussed operation under Change No. 9 at power levels up to 17 MW(t), including a 100° F increase in maximum allowable fuel clad temperature and a 44% increase in specific power, and with up to four bundles of Mark III type fuel in the superheat region.

The EVESR operation is reported to have been generally satisfactory. Some difficulties have occurred with the main steam line isolation valves, with the air lock doors into the containment building, and with the activated carbon filters. Modifications have been made to remedy deficiencies in the design and in operating procedures.

In view of the successful operation of the facility at power levels up to 17 MW(t), the ACRS believes that the EVESR reactor can continue to be operated under a ten year license without undue risk to the health and safety of the public.

Sincerely yours,

/s/
W. D. Manly
Chairman

References Attached.

References (EVESR)

1. Letter dated April 7, 1965 from E. W. O'Rorke, General Electric Company, to Dr. R. L. Doan, AEC.

2. Report of Operating Experience During Second Six Months After Achieving Full Power, ESADA - Vallecitos Experimental Superheat Reactor (EVESR), dated July 2, 1965.

3. Report of Tests and Operations During Ascent to Full Power, ESADA - Vallecitos Experimental Superheat Reactor, dated July 10, 1964.

4. Report of Operating Experience During First Six Months After Achieving Full Power, ESADA - Vallecitos Experimental Superheat Reactor (EVESR), dated January 12, 1965.

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

August 16, 1966

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

Subject: REPORT ON ESADA VALLECITOS EXPERIMENTAL SUPERHEAT REACTOR (EVESR)

Dear Dr. Seaborg:

At its seventy-sixth meeting, on August 11-13, 1966, the Advisory Committee on Reactor Safeguards reviewed Proposed Change No. 17 for the ESADA Vallecitos Experimental Superheat Reactor (EVESR). The Committee had the benefit of discussion with representatives of the General Electric Company and the AEC Regulatory Staff, of a Subcommittee meeting on August 3, 1966, and of the documents listed.

The principal proposed changes in Technical Specifications include:

1. Increase in maximum allowable specific power from 33 to 50 watts/gram UO_2.

2. Increase in allowable non-standard fuel, changed criteria regarding the amount permitted, and use of BWR-type fuel.

3. Increase in maximum allowable clad temperatures.

4. Changes in control-rod materials, worth, and withdrawal pattern.

Item 4 above and proposed modifications in piping supports will decrease the already low probability and the potential consequences of a severe reactivity accident, and thus result in enhanced public safety.

Items 1-3 will permit the use of fuel elements of more advanced design than those previously used. These changes may increase the probability of fuel failures during normal operation, but experience with similar fuel at EVESR and elsewhere indicates that the failure rate should not be excessive. Moreover, EVESR has been operated successfully with intentiaonlly as well as unintentionally defected fuel. Some problems have been encountered with the efficiency of the off-gas charcoal filter under service conditions. Accordingly the licensee has pursued a vigorous program of filter system improvement and performance surveillance. The Committee recommends that the AEC Regulatory Staff continue to follow this program closely.

The allowable number of "experimental" and/or "new" fuel bundles, or combinations, and the technical basis for the limits, need to be clarified. The Committee recommends that this be resolved by the Regulatory Staff and the licensee.

The Committee believes that the EVESR facility can be operated as proposed without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

David Okrent
Chairman

References:

1. Proposed Change No. 17, with attachments, dated June 3, 1966.

2. Letter dated June 30, 1966 from B. D. Wilson, General Electric Company, to Dr. R. L. Doan, AEC, with enclosed "Supplement to Proposed Change No. 17".

3. Letter dated July 21, 1966 from B. D. Wilson, General Electric Company, to Dr. R. L. Doan, AEC, with attachment.

4. Letter dated August 4, 1966 from L. H. McEwen, General Electric Company, to Dr. R. L. Doan, AEC, with attachments.

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

June 15, 1967

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

Subject: REPORT ON VERMONT YANKEE NUCLEAR POWER STATION

Dear Dr. Seaborg:

At its eighty-sixth meeting, on June 8-10, 1967, the Advisory Committee on Reactor Safeguards completed its review of the application by Vermont Yankee Nuclear Power Corporation for authorization to construct the Vermont Yankee Nuclear Power Station. This project was previously considered at ACRS Subcommittee meetings held in Washington, D. C. on May 10, 1967, and in Vermont on June 7, 1967. On the latter date, the Subcommittee also visited the reactor site. During its review, the Committee had the benefit of discussions with representatives of Vermont Yankee Nuclear Power Corporation, General Electric Company, EBASCO Services Incorporated, Chicago Bridge and Iron Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The Vermont Yankee Nuclear Power Station is to be located in southern Vermont, on the west bank of the Connecticut River in the town of Vernon. The Vermont Yankee reactor will be a single cycle, forced circulation boiling water unit with a design power level of 1593 MW(t). The average core power density of the Vermont Yankee unit is essentially the same as that of the previously reviewed Browns Ferry reactors, and the complex of emergency core cooling systems is similar to that proposed for the Browns Ferry reactors. The Committee believes that several of the comments made in the March 14, 1967 report on Browns Ferry apply to the Vermont Yankee application:

1. Analysis indicates that a large fraction of the reactor fuel elements may be expected to fail in certain loss-of-coolant accidents. The applicant states that the principal mode of failure is expected to be by localized perforation of the clad, and that damage within the fuel assembly of such nature or extent as to interfere with heat removal sufficiently to cause clad melting would not occur. The Committee believes that additional evidence, both analytical and experimental, is needed and should be obtained to demonstrate that this model is adequately conservative for the power density and fuel burnup proposed.*

2. The applicant considers the possibility of melting and subsequent disintegration of a portion of a fuel assembly, by inlet coolant orifice blockage or by other means, to be remote. However, the resulting effects in terms of fission product release, local high pressure production, and possible initiation of failure in adjacent fuel elements are not well known. Information should be developed to show that such an incident will not lead to unacceptable conditions.*

3. A linear heat generation rate of 28 KW/ft is used by the applicant as a fuel element damage limit. Experimental verification of this criterion is incomplete, and the applicant plans to conduct additional tests. The Committee recommends that such tests include heat generation rates in excess of those calculated for the worst anticipated transient and fuel burnups comparable to the maximum expected in the reactor.*

4. In a loss-of-coolant accident, the core spray and flooding systems are required to function effectively under circumstances in which some areas of fuel clad may have attained temperatures higher than those at which such cooling mechanisms have been tested to date. The applicant is conducting tests of these devices at increased temperatures and has reported preliminary results which are promising. The Committee again urges that these tests be extended to temperatures as high as practicable. The use of stainless steel in these tests for simulation of the Zircaloy clad appears suitable, but some corroborating tests employing Zircaloy should be included.

The reactor vessel for Vermont Yankee will be a field-fabricated vessel quite similar to that proposed for the previously reviewed Monticello Nuclear Generating Plant. The Committee recommends that great care and diligence be exercised in the quality control program for this vessel to ensure the soundness of this important plant component.

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

The integrity of Vernon Dam, just downstream of the plant site, is essential to maintain the normal cooling water supply to the plant. The applicant has examined the design of the dam and states that it should withstand, without gross failure, the maximum hypothetical earthquake selected for the site.

He has proposed, however, to provide an alternate means of removing shutdown heat from the plant in the event that the river level should fall below the normal cooling water inlet. The Committee believes that shutdown heat removal can be accomplished by one of the several methods being considered by the applicant.

The Committee recommends that the applicant give special attention to the design of critical elements of the plant piping, including the drywell-torus connections, to ensure that these elements are not overstressed under maximum earthquake forces.

The applicant proposes to use sensing devices in the recirculation loops of the reactor to detect the location of a pipe break. Signals from these devices would be used automatically to select various valve actions that are essential to the proper operation of the emergency core cooling systems. In view of the importance of the proper valve actions in the unlikely event of a major pipe break, the Committee recommends that the sensing instrumentation and valve control system be designed to full reactor protection syste standards, and that consideration be given to providing more than one type of sensing device in the system.

Fuel clad temperatures following a steam line break should be further evaluated during detailed design, with due attention to using conservative assumptions and methods in calculating these temperatures. Steam line isolation valve closure times as short as three seconds may be required to maintain acceptably low fuel clad temperatures in this accident. The applicant has stated that isolation valves with closure times adjustable from 3 to 10 second will be obtained for the plant.

The rod block monitor system for the Vermont Yankee reactor is a two-channel system, with one channel required for rod blocking action. The applicant has proposed that, if one channel is bypassed for maintenance, an appropriately short interval between tests will be used for the operating channel. The Committee believes that, if one channel of the rod block monitor system is to be out of service for a long period of time, other measures, in addition to frequent testing of the operative channel, should be taken to ensure that improper rod withdrawal is not allowed to occur.

In view of the high design power density of the core, an especially careful and extensive start-up program will be required for this plant. If the start-up program or the additional information on fuel behavior referred to above should fail to confirm adequately the designer's expectations, plant modifications or restrictions on operation may be appropriate.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during construction of the reactor and should be followed by the Regulatory Staff. On the basis of the foregoing comments, and in view of the favorable characteristics of the site, the Committee believes that the proposed reactor can be constructed at the Vernon site 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 of significance for all large water-cooled power reactors, and warrant careful attention.

References:

1. Letter from Vermont Yankee Nuclear Power Corporation, dated November 30, 1966, including License Application.
2. Vermont Yankee Nuclear Power Station Plant Design and Analysis Report, Volumes I, II, and III.
3. Letter from Vermont Yankee Nuclear Power Corporation, dated January 10, 1967, with Amendment No. 1 to License Application.
4. Letter from Vermont Yankee Nuclear Power Corporation, dated January 23, 1967, with Amendment No. 2 to License Application.
5. Letter from Vermont Yankee Nuclear Power Corporation, dated April 12, 1967, with Amendment No. 3 to License Application.
6. Letter from Vermont Yankee Nuclear Power Corporation, dated April 28, 1967, with Amendment No. 4 to License Application.
7. Letter from Vermont Yankee Nuclear Power Corporation, dated May 19, 1967, with Amendment No. 5 to License Application.
8. Letter from Vermont Yankee Nuclear Power Corporation, dated May 24, 1967, with Amendment No. 6 to License Application.
9. Amendment No. 7 to License Application, dated June 2, 1967.

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

October 12, 1967

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

Subject: REPORT ON VERMONT YANKEE NUCLEAR POWER STATION

Dear Dr. Seaborg:

At its ninetieth meeting, on October 5-7, 1967, the Advisory Committee on Reactor Safeguards extended its review of the application by Vermont Yankee Nuclear Power Corporation to include the proposed use of cooling towers in the Vermont Yankee Nuclear Power Station. This project was the subject of a previous report to you dated June 15, 1967. In extending its review, the Committee has had the benefit of discussions with representatives of the Vermont Yankee Nuclear Power Corporation, Ebasco Services, Inc., the General Electric Company, and the AEC Regulatory Staff, and of the documents listed below.

The Vermont Yankee Station will be located on the west bank of the Connecticut River, in the town of Vernon, Vermont. The reactor will be a single cycle, forced circulation boiling water unit with a design power level of 1593 MW(t). At the time of the previous review by the Committee, the applicant planned to use the Connecticut River as a heat sink by drawing cooling water for the main condenser from the river, heating it in the condenser, and returning the heated water to the river. Since that time, limitations by state agencies on the allowable temperature rise and maximum temperature of water returned to the river have led the applicant to propose the use of cooling towers to reject a portion of the waste heat from the plant to the atmosphere.

The Committee believes that the use of cooling towers, as proposed, is acceptable and reaffirms its previous conclusion that the proposed reactor can be constructed at the Vernon site 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

References:

1. Letter from Vermont Yankee Nuclear Power Corp., dated Sept. 8, 1967, to Valentine B. Deale, Esq., Chairman Atomic Safety and Licensing Board
2. Letter from Vermont Yankee Nuclear Power Corp., dated Sept. 15, 1967, to Valentine B. Deale, Esq., Chairman Atomic Safety and Licensing Board, with revised section, "Possible Radiological Effects"
3. Letter from Vermont Yankee Nuclear Power Corp., dated Oct. 6, 1967, to Valentine B. Deale, Esq., Chairman, Atomic Safety and Licensing Board

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

March 9, 1971

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

Subject: REPORT ON VERMONT YANKEE NUCLEAR POWER STATION

Dear Dr. Seaborg:

At its one hundred thirty-first meeting, on March 4-6, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application by Vermont Yankee Nuclear Power Corporation for authorization to operate the Vermont Yankee Nuclear Power Station. This project was previously considered at Subcommittee meetings held at the plant site on October 22, 1970 and in Washington, D. C. on December 22, 1970, February 23, 1971, and March 3, 1971. During its review, the Committee had the benefit of discussions with representatives and consultants of Vermont Yankee Nuclear Power Corporation, General Electric Company, EBASCO Services Incorporated, and the AEC Regulatory Staff, and of the documents listed below. The results of the Committee's review of this project at the construction permit stage were given in reports to you on June 15, 1967 and October 12, 1967.

The Vermont Yankee Nuclear Power Station is located about five miles south of Brattleboro, Vermont, on the west bank of the Connecticut River in the town of Vernon. The Vermont Yankee reactor is a single cycle, forced circulation boiling water unit with a design power level of 1593 MW(t). The reactor is generally similar to that of the previously reviewed Monticello Unit 1, but with a core average power density about 25 percent greater. The Vermont Yankee reactor has the highest power density and linear heat generation rate of any boiling water reactor reviewed for operation.

Forced draft cooling towers have been provided and the condenser cooling system arranged so that waste heat may be rejected to the atmosphere or the river, as required to meet temperature limitations on cooling water returned to the river. The applicant plans to process all of the radioactive liquid wastes from plant operation and to recycle the low conductivity fraction to the reactor coolant supply system. The Committee recommends that maximum use be made of the installed liquid waste treatment systems so that releases to the river are limited to very low levels with regard to both the concentration and the total amount of radioactivity. The applicant proposes to supplement the installed conventional boiling water reactor gaseous waste treatment system with a holdup system of advanced design to reduce offsite doses by an additional factor of at least 50. This system should be installed by the end of the first refueling outage.

The containment is penetrated by a number of small diameter instrument lines. The applicant proposes to install flow-limiting orifices in these lines inside the containment so that the reactor building, which serves as a secondary containment, would not be damaged and the building filters would not be bypassed in the event of instrument line failure.

The applicant is continuing to study further means of preventing common mode failures from negating reactor scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. The Committee wishes to be kept informed of the resolution of this matter.

The inservice inspection program proposed for the reactor primary system for the first five years of operation complies with the recently-adopted inspection code to the extent permitted by the existing design. The applicant has prepared for inservice inspection by such measures as grinding of hand-deposited welds and by making a reference inspection of the as-built vessel and primary piping to provide a basis for comparison for future inspection results. The Committee believes the proposed program is adequate for the first five years of operation, but recommends that the applicant study additional means of improving access to various pressure vessel welds and to other methods of assuring vessel integrity.

The applicant proposes to supplement the installed primary system leak detection methods, which are based on measurements of sump accumulation rate and drywell temperature and pressure, with an air monitoring system. The new system will sample the containment atmosphere for analysis for radioactive gases and particulate matter.

The Committee has commented in previous reports on the development of systems to control the buildup of hydrogen in the containment that might follow in the unlikely event of a loss-of-coolant accident. The applicant proposes to use a purging technique after a suitable time delay subsequent to the accident. The Committee believes that purging capability should be retained, but that the primary protection in this regard should utilize a method of hydrogen control other than purging. The applicant should submit, on a reasonable time scale, a proposed design for hydrogen control for review by the Regulatory Staff. The Committee wishes to be kept informed of the resolution of this matter.

The Committee believes the containment should be inerted during operation of the reactor. The Committee recognizes that inerting makes inspection and repair of the primary system more difficult, and believes it acceptable to de-inert during operation just prior to a shutdown and to re-inert during startup and operation following a shutdown. It is recommended that the need for inerting be reviewed periodically as operating experience and further knowledge from current development work are obtained, and as other means of coping with the hazards from accident-generated hydrogen are found.

Performance of the emergency core cooling system has been evaluated for the higher power density of the Vermont Yankee reactor core. The effects of possible variations in heat transfer coefficients and other parameters have been analyzed with regard to fuel clad temperatures. Additional studies are underway by the applicant and his contractors to provide further assurance that postulated loss-of-coolant accidents, as analyzed with conservative assumptions, will not lead to peak clad temperatures which exceed limits acceptable to the Regulatory Staff. The Committee believes that these studies should be expedited and the matter resolved in a manner satisfactory to the Regulatory Staff prior to routine operation at full power. The Committee wishes to be kept informed.

The applicant proposes to install protective structures to further assure the integrity of the containment in the unlikely event of failure of large pipes within the containment.

The applicant has proposed several changes in the plant instrument and electrical systems to improve the ability to test instruments during operation, to increase the separation of redundant protection system elements, and to increase the reliability of instrument and electrical power systems. The details of these changes should be reviewed and approved by the Regulatory Staff.

Further studies should be made of the possible effects of a dropped fuel cask on the integrity of the spent fuel pool. Means of reducing potential damage should be examined and measures taken, if necessary, to provide the needed degree ot integrity. This matter should be resolved on a reasonable time scale in a manner satisfactory to the Regulatory Staff.

An extensive integrated program for measuring vibration of reactor vessel internals is being carried out on several previously-licensed boiling water reactors. The Committee believes that some confirmatory vibration measurements are desirable for Vermont Yankee, as for all reactors. The Regulatory Staff should review the results of vibration measurements on other plants with regard to their applicability to Vermont Yankee and should determine the confirmatory measurements to be made.

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 Vermont Yankee Nuclear Power Station can be operated at power levels up to 1593 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/ S. H. Bush

Spencer H. Bush
Chairman

References
1) Amendment Nos. 10 - 22 to the application for the Vermont Yankee Nuclear Power Station

2) Environmental Report dated September 1, 1970, Vermont Yankee Nuclear Power Station

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

June 12, 1974

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

Subject: REPORT ON VERMONT YANKEE NUCLEAR POWER STATION

Dear Dr. Ray:

At its 169th and 170th meetings, May 9-11 and June 6-8, 1974, the Advisory Committee on Reactor Safeguards reviewed the question of whether the Vermont Yankee Nuclear Power Station operating license should be amended to permit operation without inerting of the containment. During its review, the Committee had the benefit of discussions with representatives and consultants of Vermont Yankee Nuclear Power Corporation, General Electric Company, and the AEC Regulatory Staff, and of the documents listed. The results of the Committee's review of this project at the operating license stage were given in a report dated March 9, 1971.

The Vermont Yankee reactor has been operated at power for some time, but has not yet completed its startup program or completed its "warranty run" at full power. The Technical Specifications now in effect require inerting of containment for power operation subsequent to completion of the startup program and warranty run.

The present review of the need for inerting has been conducted at the request of the Director of Licensing, and stems from the Memorandum and Order, April 16, 1974 (ALAB-194) of the Atomic Safety and Licensing Appeal Board which directs that the requirement for inerting be deleted from the Technical Specifications for Vermont Yankee. The Regulatory Staff, however, believes that operation with an inerted containment is necessary to provide reasonable assurance that Vermont Yankee can be operated without undue risk to the health and safety of the public.

In its March 9, 1971 report on this plant, the Committee indicated that:

> "The Committee believes the containment should be inerted during operation of the reactor. The Committee recognizes that inerting makes inspection and repair of the primary system more difficult, and believes it acceptable to de-inert during operation just prior to a shutdown and to re-inert during startup and operation following a shutdown. It is recommended that the need for inerting be reviewed periodically as operating experience and further knowledge from current development work are obtained, and as other means of coping with the hazards from accident-generated hydrogen are found."

At this time, the Committee believes that there is insufficient new information to warrant relaxation of its earlier recommendation that inerting be employed. With the current core loading and mode of operation, high core thermal performance is demanded and peak clad temperature during a postulated LOCA calculated in accordance with the Interim Acceptance Criteria provisions would be close to 2300°F. For such calculated clad temperature, the Committee considers it prudent to assume a very conservative margin on possible magnitude of hydrogen generation by fuel clad-water reaction and, therefore, on the necessity of inerting. Accordingly, the Committee recommends that the Vermont Yankee operating license not be amended at this time to permit operation without inerting of containment.

It should be noted that revisions to Regulatory Guide 1.7 are being considered and that these, together with a major change in core loading involving use of 8x8 fuel being proposed by the applicant, may provide a basis for reconsidering the need for inerting of the Vermont Yankee containment. In this connection, the Committee has provided comments on the proposed Guide revisions, and a copy is attached.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References Attached.

References

1. AEC (DL), letter dated April 30, 1974 to ACRS regarding Operation of the Vermont Yankee Nuclear Power Station Without Inerting of the Primary Containment

2. Memorandum and Order, April 16, 1974, (ALAB-194) Atomic Safety and Licensing Appeal Board Initial Decision

3. DL memo dated May 1, 1974 to R. F. Fraley regarding Responses of the Regulatory Staff to the Board's Memorandum and Order, April 16, 1974, (ALAB-194) concerning the inerting issue on the Vermont Yankee Nuclear Power Station

4. AEC (DL) letter dated May 2, 1974 to Vermont Yankee Nuclear Power Corporation regarding Operation of the Vermont Yankee Nuclear Power Station without inerting of the primary containment

5. Letter from Berlin, Roisman and Kessler dated May 6, 1974 regarding the Inerting Issue submitted by the New England Coalition of Nuclear Pollution

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 12, 1976

Honorable William A. Anders
Chairman
Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON CONTAINMENT SYSTEM MODIFICATIONS, VERMONT YANKEE NUCLEAR POWER STATION

Dear Mr. Anders:

During its 191st meeting, March 4-6, 1976, the Advisory Committee on Reactor Safeguards met with representatives of the Vermont Yankee Electric Company, the Mark-I Owners Group, the General Electric Company, the Bechtel Power Corporation, Teledyne Materials Research Corporation, the Nutech Corporation, and the Nuclear Regulatory Commission Staff to discuss the modifications in the containment system for the Vermont Yankee Nuclear Power Station. A Subcommittee of the ACRS and its consultants also considered these modifications at a meeting on March 3, 1976. During its review, the Committee had the benefit of the documents listed. The Committee also heard oral comments by Mr. David M. Scott, Department of Health, State of Vermont.

The modifications consist first, of an increase in the drywell pressure of 1.25 psi and a reduction in the wetwell, or torus, pressure of 0.45 psi, and second, of the installation of a tie-down system for the torus. The Committee was informed that the tie-down system will be completed in the near future, and also that the water volume in the torus, the torus pressure, and the drywell pressure will be closely monitored.

The Committee concurs in the view that these modifications represent an improvement in the safety margin for the Vermont Yankee Nuclear Power Station containment performance capability. The modifications were based on the Short-Term Program which consisted of small scale tests and analytical studies. The Mark-I Owners Group also informed the Committee of its Long-Term Program to perform large scale tests which will also include three-dimensional effects. The Committee believes that the Long-Term Program will lead to significant findings and recommends that this program proceed in an expeditious manner. The Committee wishes to be kept informed.

The Advisory Committee on Reactor Safeguards wishes to point out that there are several generic items of importance for boiling water reactors, and the resolution of these items will, of necessity, be relevant to the Vermont Yankee Nuclear Power Station.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

REFERENCES:

1. Mark I Containment Evaluation Short-Term Program - Final Report; Volumes I-V, dated September 1975.
2. Vermont Yankee Nuclear Power Corporation Letter, dated February 6, 1976, transmitting an evaluation of torus behavior.
3. Order for Modification of License for Vermont Yankee Power Corporation, dated February 13, 1976.
4. Safety Evaluation Report, dated February 13, 1976.
5. Testimony of Mr. David M. Scott before the Advisory Committee on Reactor Safeguards, Nuclear Regulatory Commission, March 5, 1976.

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

April 16, 1974

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

Subject: REPORT ON ALVIN W. VOGTLE NUCLEAR PLANT, UNITS 1, 2, 3, AND 4

Dear Dr. Ray:

At its 168th meeting, April 11-13, 1974, the Advisory Committee on Reactor Safeguards reviewed the application by the Georgia Power Company for a permit to construct the four-unit Alvin W. Vogtle Nuclear Plant. The Subcommittee made a visit to the plant site on March 28, 1974, and the project was considered at a Subcommittee meeting at Bush Field, Augusta, Georgia, on March 29, 1974. During its review the Committee had the benefit of discussions with representatives and consultants of the Applicant, Southern Services, Incorporated, Westinghouse Electric Corporation, Bechtel Power Corporation and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The plant will be located on the Savannah River in Burke County, Georgia, approximately 26 miles south-southeast of Augusta, Georgia, the nearest population center (reported 1970 population of 59,864). A minimum exclusion area radius of 3600 feet has been specified. The radius of the low population zone (1977 estimated population of 15) has been selected to be two miles. Major land use in the area of the plant site is devoted to timber, with agriculture using about 30 percent of the land within a radius of five miles.

Each unit of the plant will utilize a Westinghouse four-loop pressurized water nuclear steam supply system having a design power level of 3411 MWt and a design essentially the same as that provided for the Catawba Station which was previously reviewed and reported by the Committee in its letter of November 13, 1973.

The seismic design bases for the plant are 0.2g horizontal ground acceleration for the safe shutdown earthquake and 0.12g horizontal ground acceleration for the operating basis earthquake. These values have been derived from experience with the Charleston, South Carolina, earthquake of 1886 as it affected the Vogtle site surroundings, and the Committee believes that they are appropriate to this site location.

The foundation structures will be supported on a marl deposit that has been investigated by the Applicant and found to be suitable for the purpose. Tests of the marl, whose minimum thickness is approximately 70 feet, have shown that it effectively separates the reactor site from the lower Tuscaloosa aquifer, a major regional water distribution channel. The Applicant has indicated that he will carefully evaluate the foundation excavation to verify the properties of the marl and to identify any conditions relevant to the seismic design of the plant.

The ultimate heat sink for the plant is provided by two seismic Category I mechanical-draft cooling towers for each unit. The Applicant has determined that, based on present design requirements, the cooling tower basins will have ample storage capacity for a 30-day emergency cooling demand. He also plans to install two seismic Category I wells for each unit, which would supply makeup water to the ultimate heat sink if future design shows a need for further emergency cooling water capacity. The seismic Category I requirements for these wells are still being evaluated. If the wells are needed for emergency cooling water purposes, these requirements should be met in a manner satisfactory to the Regulatory Staff.

The Committee recommended in its report of September 10, 1973, on acceptance criteria for ECCS, that significantly improved ECCS capability should be provided for reactors filing for construction permits after January 7, 1973. The Vogtle Plant is in this category. This plant will use 17x17 fuel assemblies similar to those to be used in Catawba Units 1 and 2, recently reviewed by the Committee. While details of the proposed design are available, complete analyses of the performance of this fuel arrangement are not yet available from the Applicant, and the AEC Regulatory Staff has not completed their review. The Committee has been informed that performance analyses and reviews will be conducted during the coming year in connection with operating license applications for other nuclear units. The Committee believes that the Applicant should continue studies that are responsive to the Committee's examples of design improvements. If studies establish that significant further improvements can be achieved, consideration should be given to including such additions to this plant.

The proposed emergency diesel-generators are larger than any previously qualified for nuclear service. The Applicant has proposed reliability tests as required for qualification. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The proximity of the AEC's Savannah River Plant and the Barnwell Nuclear Fuel Plant makes it important to have effective emergency arrangements to deal with unusual circumstances that may be of interrelated safety significance to the three plants. The Applicant has indicated that he will establish an emergency plan in cooperation with these other nuclear installations to ensure effective emergency response if demanded by events in the immediate area. Consideration should be given by the AEC to periodic evaluations of the combined routine liquid and airborne radionuclide releases from these two plants and the Vogtle Plant as they may affect the health and safety of the public.

Generic problems relating to large water reactors have been identified by the Regulatory Staff and the ACRS and discussed in the Committee's report dated February 13, 1974. These problems should be dealt with appropriately by the Regulatory Staff and the Applicant, taking into account the nine-year construction period for the four-unit plant.

The ACRS believes that the above items can be resolved during construction and that, if due consideration is given to these items, the Alvin W. Vogtle Nuclear Plant, Units 1, 2, 3, and 4 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References:

1) Preliminary Safety Analysis Report (PSAR), Volumes I-IX, Alvin W. Vogtle Nuclear Plant, dated February 8, 1973

2) Amendments 1 through 17 to the PSAR

3) Safety Evaluation Report by the Directorate of Licensing, USAEC, Alvin W. Vogtle Plant, Units 1, 2, 3, and 4, dated March 8, 1974

4) Letter, Ernest L. Dodson, Department of the Army, Office of the Chief of Engineers, to T. Cardone, Directorate of Licensing, USAEC, dated January 31, 1974 (with enclosure dated January 30, 1974, regarding Amendment 13 to the PSAR)

5) Letter, Elmer H. Baltz, U. S. Department of the Interior, Geological Survey, to William P. Gammill, Directorate of Licensing, USAEC, dated February 21, 1974 (with enclosure dated February 8, 1974, regarding geologic aspects of the Alvin W. Vogtle Nuclear Plant)

References (cont'd)

6) Written Statement by Solomon K. Brown, dated March 19, 1974

7) Written Statement by Solomon K. Brown, dated March 22, 1974

8) Written Statement by Neill Herring, Georgia Power Project, submitted March 29, 1974

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

December 15, 1958

Honorable John A. McCone
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: THE WAHLUKE SLOPE

Dear Mr. McCone:

At the request of the Atomic Energy Commission the Advisory Committee on Reactor Safeguards summarizes its views with respect to the proposal to remove restrictions which have heretofore limited the development of the so-called secondary zone of the Wahluke Slope as an agricultural area.

The program, which the Commission has underway, increasing the degree of confinement of fission products in case of accidents to the Hanford reactors will substantially decrease the hazard to the occupants of the Wahluke Slope. After these changes have been completed, the risk from the reactor plant to the health and safety of occupants of this area should be low enough to allow normal use of the secondary zone. Since it is expected that the population growth in this area will be slow in the first few years and the changes will be completed in this time the secondary zone may be released now.

Pertinent to this question are the following facts:

1) The Hanford reactors have been in successful operation for many years without experiencing any incident that created a significant hazard on the Slope.

2) There have been over the years continuing improvements in the design and operation of these reactors which have substantially reduced the probability of serious accidents.

3) Despite these favorable developments, the Hanford reactors continue to pose potential risks to the public that are greater than those of many other reactors, including the large power reactors now under construction at other locations. The reasons for this are associated partly with the early basic design of the Hanford reactors and partly with their role in national security.

4) Recent studies have indicated the possiblity of effecting a significant additional reduction in potential hazard to the public by improvements in the airtightness of the present reactor buildings and by the provision of suitable filters that will permit better confinement of any radioactive products that may be accidentally released from the reactors.

After careful consideration of all known factors affecting the overall safety of the Hanford operation, and to the things that have been done and can still be accomplished to reduce the hazard to the public, the Committee has reasoned as follows:

A) While distance from the reactors offers no certain protection against the radioactivity that may be released in a reactor accident, it does provide an important factor of safety which should always be preserved at Hanford by the permanent retention of the exclusion area known as the primary control zone.

B) The settlement of the Wahluke Slope, to the extent that it attracts settlers from distant locations, will expose increasing numbers of people to the possible consequences of a reactor accident.

C) The risks of living on the Slope, while not negligible, are significantly less than they have been in the past and with the proposed changes in confinement will not be much greater than those existing at more distant locations.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman

cc: Alvine R. Luedecke, GM
Harold L. Price, DLR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 11, 1975

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

Subject: REPORT ON WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR POWER STATIONS WNP 1 and 4

Dear Mr. Anders:

At its 182nd meeting, June 5-7, 1975, the Advisory Committee on Reactor Safeguards completed its review of the application of the Washington Public Power Supply System for permission to construct the Washington Nuclear Power (WNP) Stations 1 and 4. These plants were previously considered at a Subcommittee meeting on May 16, at Richland, Washington, and the site was visited on May 15, 1975. During its review the Committee had the benefit of discussions with representatives of Washington Public Power Supply System and consultants, the Babcock and Wilcox Company (B&W), and the NRC Staff. The Committee also had the benefit of the documents listed.

The WNP Station site is located on the Energy Research and Development Administration's Hanford Reservation in Benton County, Washington, eight miles north of Richland, Washington, the nearest population center (1970 population 26,290). The exclusion radius is 6400 feet. The low population zone is four miles in radius. In 1970 there were 38 residents within the low population zone. The Fast Flux Test Facility and WPPSS Hanford-2 (WNP-2) Reactor are the only installations within the low population zone.

The safe shutdown earthquake is 0.25g horizontal acceleration at the foundations. The operating basis earthquake is 0.125g.

For shutdown heat removal the plant has two sources of water, the operating water supply from a river intake on the Columbia River, which is not Seismic Category I, and Seismic Category I spray ponds designed to provide a 30 day emergency water supply for each unit.

The nuclear steam supply system supplied by B&W is identical in design to that of Bellefonte Nuclear Plant,Units 1 and 2, previously reported on in the ACRS letter of July 16,1974. The design operating power is 3600 MW(t). The reactor core will use 205 B&W Mark C (17x17) fuel assemblies. The Committee recommended in its report of January 7, 1972, on Interim Acceptance Criteria for ECCS, that significantly improved ECCS capability should be provided for reactors for which construction permit applications were filed after January 7, 1972. This position was repeated in its report of September 10, 1973, on Acceptance Criteria for ECCS. The Mark C fuel assemblies are responsive to this recommendation. The new fuel assemblies will be operated at lower linear heat generation rates and are expected to yield greater thermal margins for fuel design limits and improved safety margins in the analyses of the loss of coolant accidents. An extensive program has been initiated for determining the mechanical and thermal/hydraulic characteristics of the new fuel assemblies. A program of control rod tests also is proposed, including testing of trip times and control rod wear. Should modifications become necessary as a result of the control rod tests, retesting of the entire control rod drive would be undertaken. While many of the details of the proposed design are available, complete analyses of the performance of the Mark C fuel are not yet available, and the NRC Staff has not completed its review. The Committee reserves judgment concerning the final design until the required performance information is presented and has been reviewed. The Committee recommends that the applicant continue studies directed at further improvement in the capability and reliability of the ECCS. The Committee wishes to be kept informed.

The NRC Staff has determined that the ECCS performance evaluation for WNP Stations 1 and 4 meets the Interim Acceptance Criteria of June 1971. In addition the Applicant's ECCS performance evaluation, using an approved B&W model, to show compliance with the Final Acceptance Criteria of 10CFR50.46 and Appendix K, must be reviewed and approved by the NRC Staff. The Committee wishes to be kept informed.

The applicant proposes to utilize a new reactor protection system designated as RPS-II. The system, a hybrid using both analog and digital techniques, represents an evolution from the analog system, RPS-I, currently in use in the Oconee reactors. The applicant has proposed a series of environmental, reliability, and <u>in situ</u> tests for qualification of this system prior to its use in Bellefonte Units 1 and 2, the lead plant. This matter should be resolved in a manner satisfactory to the NRC Staff.

A problem considered to be generic by the ACRS is the environmental and seismic qualifications of Class I instrumentation and electrical equipment. An important aspect is that of defining what represents an acceptable aging procedure for multi-component systems. This issue should be resolved by the applicant and the NRC Staff. The Committee wishes to be kept informed.

A question has arisen concerning loads on the vessel support structure for certain postulated loss-of-coolant accidents in pressurized water reactors. This matter should be resolved for the WNP 1&4 Plants, in a manner satisfactory to the NRC Staff.

Generic problems relating to large water reactors have been identified by the NRC Staff and the ACRS and discussed in the Committee's report dated March 12, 1975. These problems should be dealt with appropriately by the NRC Staff and the applicant.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during construction and that, if due consideration is given to the foregoing, the Washington Public Power Supply System Plants WNP 1 and 4, can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

WKerr

William Kerr
Chairman

References Attached.

References

1. Preliminary Safety Analysis Report, Washington Nuclear Projects 1 and 4. (Including Amendments 1 thru 17).

2. "Safety Evaluation of the Washington Nuclear Projects 1 and 4", NUREG - 75/036, Docket Nos. 50-460, 50-513, May, 1975, ONRR, U. S. Nuclear Regulatory Commission, Washington, D. C.

3. WPPSS Letter dated May 14, 1975, J. J. Stein to Angelo Giambusso, DRL, ONRR, USNRC, Subject: WPPSS Nuclear Projects Nos. 1 and 4, On-site Meteorological data.

4. Supplement 1 to the Safety Evaluation Report, Letter from Voss A. Moore, Asst. Dir. for Light Water Reactors, Group 2 Division of Reactor Licensing, USNRC to Dr. William Kerr, Chairman ACRS dated June 2, 1975.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 16, 1976

Honorable Marcus A. Rowden
Acting Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECTS NO. 3 AND NO. 5

Dear Mr. Rowden:

During its 192nd meeting, April 8-10, 1976, the Advisory Committee on Reactor Safeguards completed a review of the application of the Washington Public Power Supply System (WPPSS) for permission to construct the WPPSS Nuclear Project No. 3 and WPPSS Nuclear Project No. 5 (WNP-3 and WNP-5). The site was visited on August 4, 1975, and Subcommittee meetings were held that same day in Elma, Washington, and on February 24, 1976, in Richland, Washington. The project was also considered during the 191st meeting of the Committee in Washington, D. C., March 4-6, 1976. During its review, the Committee had the benefit of discussions with representatives of WPPSS and its consultants, Combustion Engineering, Inc., Ebasco Services, Inc., and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

The WNP-3 and WNP-5 site is located in Grays Harbor County, Washington, approximately thirteen miles east of Aberdeen-Hoquiam-Cosmopolis, Washington, the nearest population center (1970 population 28,549). The minimum exclusion distance is 1310 meters and the low population zone (LPZ) radius is three miles. The total 1970 resident population within the LPZ was 260.

The WNP-3 and WNP-5 application is submitted in accordance with the Commission's standardization policy as described in Appendix O to Part 50, "Licensing of Production and Utilization Facilities," and Section 2.110 of Part 2, "Rules of Practice," of Title 10 of the Code of Federal Regulations. For this application the reference system is the Combustion Engineering Standardized Nuclear Steam Supply System known as its Standard Reference System-80. This design has been reviewed by the ACRS and discussed in its report of September 17, 1975, "Combustion Engineering Standard Safety Analysis Report - CESSAR-80."

The ultimate heat sink for each reactor will consist of a system of dry cooling towers and components that reject excess heat to the atmosphere. Because of its design the ultimate heat sink does not require a makeup water supply.

The Applicant described his investigations of the geologic and seismic characteristics of the site and the surrounding region. While the geology of the surrounding area is complex, and there is definite tectonic activity, there are no known geologic or seismic problems that cannot be solved by design. The proposed safe shutdown earthquake is 0.32g horizontal acceleration at the foundations. The operating basis earthquake is 0.16g.

Each WNP reactor will employ a containment system including a free standing steel vessel surrounded by a reinforced concrete shield building. The inner steel vessel is designed for an internal pressure of 44 psig. The annulus, between the two structures, is maintained at subatmospheric pressure to permit the collection of leakage from the steel vessel, in the unlikely event of a LOCA, and permit its processing before release to the environment.

The Committee recommended in its report of September 10, 1973, on acceptance criteria for ECCS, that significantly improved ECCS capability should be provided for reactors for which construction permit requests were filed after January 7, 1972. The WNP-3 and WNP-5 design is in this category. These projects will use the 16 X 16 fuel assemblies similar to those to be used in Arkansas Nuclear One Unit 2 and St. Lucie Plant Unit 2. Although calculated peak clad temperatures, in the event of a postulated LOCA, may be less for 16 X 16 than for the 14 X 14 array, the Committee believes that the Applicant should continue studies that are responsive to the Committee's September 10, 1973, report. If studies, conducted with the best available techniques, establish that significant further ECCS improvements can be achieved, consideration should be given to incorporating them into WNP-3 and WNP-5.

A generic question has arisen concerning loads on the vessel support structure for certain postulated loss-of-coolant accidents in pressurized water reactors. This matter should be resolved for WNP-3 and WNP-5 in a manner satisfactory to the NRC Staff.

The Committee believes that the Applicant and the NRC Staff should continue to review the WNP-3 and WNP-5 design for features that could reduce the possibility and consequences of sabotage.

Following the Browns Ferry fire the NRC Executive Director for Operations set up a special review group to determine what could be learned from this incident. This group has made recommendations that apply to future reactors, to reactors that are already operating, and to the NRC regulatory process. The review group points out that its recommendations are not specific to any single plant and that its recommendations are based on knowledge at the time of this investigation. The ACRS wishes to be kept informed of the specific application of the review group's recommendations, as they apply to WNP-3 and WNP-5, for the development of additional information on fire prevention, fire fighting, quality assurance, and the improvement of NRC policies, procedures, and criteria.

Other generic problems relating to large water reactors are discussed in the Committee's report dated April 16, 1976. These problems should be dealt with appropriately by the NRC Staff and the Applicant.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during construction and that, if due consideration is given to the foregoing and to items mentioned in its CESSAR-80 report of September 17, 1975, the Washington Public Power Supply System Nuclear Projects No. 3 and No. 5 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Additional comments by Members Max W. Carbon, David Okrent, Milton S. Plesset, Stephen Lawroski, and Myer Bender are presented below.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Additional Comments by Members Max W. Carbon, David Okrent, Milton S. Plesset, and Stephen Lawroski

The site for WPPSS Nuclear Projects No. 3 and No. 5 lies in a seismically active region that has been subject to large earthquakes in historic time and includes active major faults. While we do not disagree with the proposed seismic design basis, we believe it would be desirable to have the geologic and siesmic aspects of such sites, and perhaps most sites, also reviewed by the U. S. Geological Survey to provide the benefit of an additional independent evaluation.

Additional Comments by Members David Okrent and Milton S. Plesset

The recurrence interval of an earthquake of the order of the safe shutdown earthquake may be about 1,000 years for this site. For such a recurrence interval the probability of not achieving safe shutdown, given the SSE, must be very small if the NRC Staff goal of less than 10^{-7} per year, of a serious accident from any single cause, is to be achieved. Since seismic design adequacy is not subject to direct experimental confirmation, we believe that other measures, including independent design review, low-amplitude shaking measurements of the completed structure, as-built construction validation, and detection of possible inservice degradation, should be evaluated and the necessary steps taken to provide the high degree of detailed specific assurance required with regard to seismic capability of all safety-related features.

Additional Comments by Member Myer Bender

With increasing frequency, questions have arisen concerning the appropriate degree of conservatism to be included in the seismic design criteria for nuclear power plants. The needs of public safety would be best served if the design practices currently in vogue were altered to permit inelastic response so as to enhance the energy absorption characteristics of nuclear structures under severe seismic loadings. For the more severe seismic conditions inelastic design principles should be applied to foundations, concrete containments, floors, and support structures in order to assure a high degree of damping and thus minimize the forces transmitted to critical safety features and to the primary coolant circuitry. This would eliminate the need for many of the complex supplemental structural features of questionable reliability which are now used to meet extreme seismic design conditions. This design approach would allow nuclear structures to satisfy even the most pessimistic loading requirements of the most extreme seismic prophet. If it is not used there is doubtful value, and possibly some loss in public safety margin, from the use of ultraconservative seismic design requirements because the reliability of the structural restraints cannot be assessed from relevant structural experience or post-construction vibrational testing.

References:

1. Washington Public Power Supply Systems (WPPSS) Nuclear Projects No. 3 and No. 5 Preliminary Safety Analysis Report (PSAR) Volumes 1-18
2. Amendments 1-30 to the PSAR

References Continued

3. Division of Reactor Licensing (DRL) Safety Evaluation Report, dated February 1976
4. Letter, dated December 31, 1976, WPPSS to DRL, concerning reactor pressure vessel support design, shutdown cooling system, and containment purging
5. Letter, dated January 12, 1976, WPPSS to DRL, concerning atmospheric dump valve sizing

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 13, 1982

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

Subject: ACRS REPORT ON WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2

During its 270th meeting, October 7-8, 1982, the ACRS completed its review of the application of the Washington Public Power Supply System (WPPSS) (Applicant) for a license to operate the WPPSS Nuclear Project No. 2 (WNP-2). A Subcommittee meeting was held in Richland, Washington on September 2-3, 1982 to consider this project. A tour of the facility was made by members of the Subcommittee on September 2, 1982. During its review, the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee reported on the construction permit application for this plant (then designated as Hanford No. 2 Nuclear Power Plant) in a report to AEC Chairman James R. Schlesinger, dated October 19, 1972.

WNP-2 is located in the southeastern area of the U.S. Department of Energy's Hanford site in Benton County, Washington. WNP-2 uses a General Electric BWR/5 nuclear steam supply system with a rated power level of 3323 MWt and has a Mark II pressure suppression containment system with a design pressure of 45 psig. Fuel loading is scheduled to begin during September 1983.

WPPSS is a municipal corporation and a joint operating agency of the state of Washington. WNP-2 is the first WPPSS unit (of an originally planned five units) to be considered for an operating license. A work stoppage was initiated in June 1980, in order to assess QA/QC problems. WPPSS has since brought in more experienced management that has resulted in substantial improvement in the Licensee's project management teams and management controls, and in the attitude of the project personnel toward quality. Presently, at WNP-2 there is an ongoing reverification program involving sampling, inspection, and documentation review. The NRC will continue to monitor progress of this program. The Committee wishes to be kept informed.

The NRC Staff and the Applicant are in the process of determining the proper criteria and verification procedures for electrical isolation and for identification of Class 1E and associated circuits. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

The WNP-2 design uses multiplexing in the service water control system. This is the first plant we have reviewed that uses this type of service water control. Because of the importance of this system to safety, we recommend that the NRC Staff confirm that service water needed for shutdown decay heat removal can be made available even if the multiplexing system malfunctions. The Committee wishes to be kept informed regarding this matter.

The NRC Staff has identified a number of Unresolved Safety Issues as being applicable to WNP-2. There are also a number of Outstanding Issues, Confirmatory Issues, and License Conditions. We believe that these matters can be resolved in an acceptable manner.

The ACRS believes that if due consideration is given to the matters noted and to our recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that WNP-2 can be operated at power levels up to 3323 MWt without undue risk to the health and safety of the public.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. Washington Public Power Supply System, "WPPSS Nuclear Project No. 2 Final Safety Analysis Report," Volumes 1-24 and Amendments 1-25.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of WPPSS Nuclear Project No. 2," NUREG-0892, dated March 1982.
3. U.S. Nuclear Regulatory Commission, Supplement No. 1 to NUREG-0892, "Safety Evaluation Report Related to the Operation of WPPSS Nuclear Project No. 2," dated August 1982.

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

January 17, 1973

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON WATERFORD STEAM ELECTRIC STATION UNIT NO. 3

Dear Dr. Schlesinger:

At its 153rd meeting, January 11-13, 1973, the Advisory Committee on Reactor Safeguards completed its review of the application of the Louisiana Power and Light Company to construct Waterford Unit No. 3. This project was considered at Subcommittee meetings on November 2, 1972, at the site, and on January 9, 1973, in Washington, D. C. During its review the Committee had the benefit of discussions with representatives and consultants of the Louisiana Power and Light Company, Ebasco Services Incorporated, Combustion Engineering Incorporated, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Waterford site is in an industrial area on the west bank of the Mississippi River at a point about 21 miles upstream from the closest boundary of New Orleans. The site has about 7500 feet of river frontage, and contains more than 3600 acres of flatland. The plant is about 900 feet from the Mississippi River landward of the levee. It is about 500 feet from Louisiana State Highway No. 18, which is adjacent to the levee. The Texas and Pacific Railroad crosses the property about 2500 feet south of the reactor and a highway is under construction, crossing the property some 3000 feet south of the railroad. Two fossil fired units (Waterford No. 1 and No. 2) are under construction 2000 feet upstream from Waterford No. 3. The closest residence is 4000 feet from the reactor site. The closest industrial property is about 3000 feet downstream.

Waterford Unit No. 3 is founded upon some 30,000 feet of alluvial deposits. The upper 50 feet of these deposits is soft, recently deposited material. The soils below the upper material are much older, firm clays and sands. All Class 1 structures will be placed on a mat resting on the lower material. The Committee finds this satisfactory.

The nuclear steam supply system will be provided by Combustion Engineering and will include a 3390 MWt pressurized water reactor essentially identical to those to be provided for San Onofre Units 2 and 3 and Forked River Unit 1, previously reviewed. The Committee reiterates its previous statements with respect to similar reactors that adequate confirmation of the predicted core performance must be obtained to justify the higher power density of this reactor.

The Waterford containment will be a steel structure separated by an annulus from a surrounding concrete structure. The annulus will be maintained at a negative pressure under normal and accident conditions. The Committee understands that the Regulatory Staff is reviewing the adequacy of the proposed design pressure for the reactor containment building. The Committee wishes to be kept informed.

Explosions of material transported on the river, State Highway 18, or the Texas and Pacific Railroad were reviewed for possible danger to Waterford Unit No. 3. The applicant's studies indicate that the potential magnitude of such explosions, or the infrequency of their occurrence, eliminates need for additional protective measures at the plant. The Regulatory Staff should evaluate the adequacy of the analysis.

The applicant has committed himself to inclusion of two trains of wet and dry cooling towers to serve normal and emergency component cooling. When the design is completed it should be reviewed for adequacy by the Regulatory Staff.

The applicant described an experimental and analytical program intended to provide improved understanding of phenomena entering into the loss-of-coolant accident, which can provide the basis for developing improvements in ECCS design. He also described flexibility in design which can be used to improve ECCS effectiveness. The Committee believes it important that improvements in ECCS effectiveness be included in Waterford Unit No. 3, and recommends that the final design of the ECCS be reviewed by the Regulatory Staff and the ACRS prior to fabrication and installation of major components.

The Committee recommends that a study be made of the probability of unacceptable consequences arising from potential missiles in the unlikely event of turbine failure, and of the possible need for protective measures if this probability should be unacceptably high.

In addition, the Committee believes that analytical and experimental work on the penetration of reinforced concrete by missiles of the type of interest is desirable to provide a suitable basis for establishing the probability of penetration of thick-walled concrete structures and damage to safety-related components.

The applicant intends to use pre-pressurized fuel and is considering other modifications of the fuel assemblies. The fuel rod problem involving densification and associated movement of the fuel pellets is undergoing intensive investigation. The Regulatory Staff and the ACRS should review the resolution of this matter.

The Committee recommends that the applicant give careful attention to the use and improvement of instrumentation capable of providing continuing quantitative information of the local performance characteristics of high power density cores.

The Committee believes that protection against pipe whip should be provided by the applicant in accordance with criteria being developed by the AEC Regulatory Staff.

The Committee believes it desirable for the applicant and the Regulatory Staff to review further Waterford Unit No. 3 for design features, in accordance with Safety Guide No. 17, that should reduce the possibility of sabotage.

The Committee reiterates its previous comments concerning the need to study further means of preventing common mode failures from negating reactor scram action, and the design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it is desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. This matter should be resolved during construction in a manner satisfactory to the Regulatory Staff and the ACRS.

Other problems relating to large water reactors, which have been identified by the Regulatory Staff and the ACRS and cited in previous reports, should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during construction and that, if due consideration is given to the foregoing, the Waterford Steam Electric Station, Unit No. 3 can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References

1. Louisiana Power and Light Company Application to Construct and Operate Waterford Steam Electric Station, Unit No. 3, with Preliminary Safety Analysis Report, Volumes 1 through 4

2. Amendments 1 through 28 to the Application

3. Louisiana Power and Light Company Letter, dated January 5, 1973, "Effects of Fuel Densification"

FOR SEPTEMBER 11, 1973 LTR TO DIXY LEE RAY, TRANSMITTING MANGELSDORF MEMO TO MUNTZING RE FORKED RIVER, SAN ONOFRE 2&3, AND WATERFORD 3 ECCS DESIGNS, SEE PAGES 613-615 UNDER "FORKED RIVER".

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 11, 1981

The Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: INTERIM REPORT ON THE WATERFORD STEAM ELECTRIC STATION UNIT 3

Dear Dr. Palladino:

During its 256th meeting, August 6-8, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of Louisiana Power & Light Company (Applicant) for a license to operate the Waterford Steam Electric Station Unit 3 (Waterford-3). This project has been considered at Subcommittee meetings on June 18-19, 1981 in St. Charles Parish, Louisiana, and on August 5, 1981 in Washington, D.C. A tour of the facility was made by Subcommittee members on June 18, 1981. During its review, the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee commented on the construction permit application for this unit in its report dated January 17, 1973.

Waterford-3 is located on the bank of the Mississippi River near Taft, Louisiana in St. Charles Parish. The city of New Orleans is approximately 25 miles east-southeast from the plant site and Baton Rouge is approximately 50 miles north-northwest. The largest town within 10 miles of the site is Reserve, Louisiana, which had a population of approximately 7000 in 1977.

Waterford-3 uses a Combustion Engineering nuclear steam supply system with a rated power level of 3410 MWt. The architect-engineer is Ebasco Services, Inc. The containment is a free standing steel pressure vessel enclosed within a reinforced concrete shield building. The containment building, auxiliary building, fuel handling building, and ultimate heat sink are located on a common base mat, forming a self-contained nuclear island.

Louisiana Power & Light (LP&L) is a part of Middle South Utilities (MSU). Although Waterford-3 is the first nuclear plant to be operated by the Applicant, the MSU system has two operating nuclear plants, Arkansas Nuclear One Units 1 and 2, which are being operated by Arkansas Power and Light Company. Two additional plants in the MSU system, Grand Gulf Nuclear Station Units 1 and 2, are under construction by Mississippi Power and Light. MSU provides some technical services to support the nuclear units in its system.

The Applicant described the management, the operating organization, and the status of staffing. The NRC Staff has not completed its review of these matters, but reported its conclusion that the management and staffing at Waterford-3 is less well established than at other nuclear plants at a similar time during their construction and startup schedule. The LP&L management has not yet been successful in putting together the team of experienced and qualified personnel which we believe will be necessary to successfully operate the plant. Of particular concern is the lack of nuclear experience throughout the organization and the apparent lack of appreciation by high-level management of the magnitude of the project it is undertaking. We believe that an extraordinary effort will be required to prepare the LP&L management and staff for operation of the Waterford-3 plant. We also believe that a more concerted effort is needed to build an integrated organization of LP&L and contractor personnel for startup and operation of Waterford-3. We recommend that the adequacy of management and staffing be established prior to fuel loading. We will continue to review this matter with the Applicant and the NRC Staff.

The Applicant described the three safety review committees which will be a permanent part of the Waterford-3 organization. We believe that better use could be made of experts from sources other than the Applicant's organization and its contractors to provide professional experience in areas such as training, human factors engineering, and reactor safety. We recommend that the Applicant make a greater effort to include recognized experts, especially on its Safety Review Committee.

Although a sincere effort has been made to establish a comprehensive training program at Waterford-3, it has suffered from a lack of professional direction. We believe the Applicant should move as soon as possible to employ a highly qualified professional for the key position of training director and provide him with the resources needed to build an effective program.

Waterford-3 is located in a highly industrialized area with an unusually large concentration of sources of hazardous substances from nearby industries and transportation routes. We believe the Applicant has done a commendable job in analyzing these hazards and providing for protection of the plant by both equipment design and administrative procedures. The NRC Staff has not completed its review of this matter, but we believe it can be resolved satisfactorily.

The Waterford-3 control room makes extensive use of a computer system for monitoring and control of the plant, and for evaluating plant performance. We commend the initiative the Applicant has shown in this area and the continuing effort to integrate the control room equipment with operating procedures and human factors considerations.

Waterford-3 has a unique ultimate heat sink design. It is contained within the nuclear island and is protected from extreme environmental effects. It consists of two trains of wet and dry cooling towers. Sufficient water is stored on the nuclear island to meet the needs for shutdown decay heat removal. We believe the design is acceptable.

The Applicant has performed an analysis of total loss of AC power. The DC power supply is capable of supplying essential loads for at least two hours and the condensate supply is sufficient for a longer period. We recommend that the Applicant expand this analysis to consider the effect of loss of space cooling on essential electrical equipment and to also consider the effect of coolant leakage from the primary system. Evaluation of these matters is a generic issue. Studies for this plant need not be completed prior to startup.

We note that a number of items have been identified as Outstanding Issues in the NRC Staff Safety Evaluation Report dated July 1981. These include some TMI-2 Action Plan requirements. We believe these issues can be resolved in a manner satisfactory to the NRC Staff, subject to the concerns on instrumentation for detection of inadequate core cooling expressed in the ACRS letter to the Executive Director for Operations dated June 9, 1981.

The Committee believes that, contingent on the Applicant's attainment of an adequate level of management and staffing, if due consideration is given to the recommendations above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that Waterford Steam Electric Station Unit 3 can be operated at power levels up to 3410 MWt without undue risk to the health and safety of the public.

We expect to report further on the adequacy of the staffing and management as progress is made toward improvement.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. Louisiana Power & Light Company, "Waterford Steam Electric Station, Unit 3 Final Safety Analysis Report," with Amendments 1 through 20.

2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit 3," Docket No. 50-382, USNRC Report NUREG-0787, July 1981.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 9, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON THE WATERFORD STEAM ELECTRIC STATION UNIT 3

Dear Dr. Palladino:

During its 263rd meeting, March 4-6, 1982, the Advisory Committee on Reactor Safeguards continued its review of the application of Louisiana Power and Light Company (Applicant) for a license to operate the Waterford Steam Electric Station Unit 3 (Waterford-3). This project was considered at a Subcommittee meeting on March 3, 1982 in Washington, D.C. and at a previous full Committee meeting on August 6-8, 1981. During the August meeting, the Committee prepared an interim report to you dated August 11, 1981. In its review the Committee had the benefit of discussions with the Applicant and the NRC Staff. The Committee also had the benefit of the documents listed.

In its interim report the Committee expressed concern about the organizational readiness of the Applicant to operate the plant and about the adequacy of the Applicant's training program. The report made several specific suggestions, and we indicated that we would report to you further on the adequacy of staffing and management.

During the meetings on March 3 and 4, 1982, the NRC Staff reported its conclusion that the Applicant's organization, staff, and management will be adequate to operate Waterford-3 in a safe manner by the time of fuel loading, currently scheduled for January 1983. The Applicant described efforts over the past six months to strengthen the Waterford-3 organization and training program. These efforts include important changes in the corporate structure to provide increased dedication of management to the task of completing and operating Waterford-3, changes in the operating organization to permit improved focus on direct operational and technical support functions, substantial progress toward completion of staffing, the formation of a comprehensive training program, and establishment of a strong Safety Review Committee. In addition, the Applicant described the integration of the Waterford-3 and contract personnel into an effective startup organization.

The Committee believes that the Applicant has effectively responded to the concerns regarding organization and management expressed in our August 11, 1981 report. We believe that with continued dedication of Louisiana Power and Light Company management, satisfactory completion of staffing and the planned program for training, and due consideration to other matters noted in our August 11, 1981 report, there is reasonable assurance the Waterford Steam Electric Station Unit 3 can be operated without undue risk to the health and safety of the public.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References

1. Louisiana Power and Light Company, "Waterford Steam Electric Station Unit No. 3, Final Safety Analysis Report," with Amendments 1-25.

2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit No. 3," NUREG-0787, dated July 1981 with Supplement 1, dated October 1981 and Supplement 2, dated January 1982.

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

September 21, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2

Dear Dr. Schlesinger:

At its 149th meeting, September 14-16, 1972, the Advisory Committee on Reactor Safeguards reviewed the application of the Tennessee Valley Authority to construct Units 1 and 2 of the Watts Bar Nuclear Plant. The project was considered at Subcommittee meetings at the plant site on July 7, 1972, and in Washington, D. C., on September 13, 1972. During its review, the Committee had the benefit of discussions with the representatives of the applicant, the Westinghouse Electric Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed below.

The plant will be located on a 1770-acre site on the west shore of Chickamauga Lake on the Tennessee River, about 50 miles northeast of Chattanooga, Tennessee. The site is 1.9 miles downstream from the Watts Bar Dam and Hydroelectric Plant, and 0.65 miles from the Watts Bar Steam Plant. It is 31 miles upstream from the Sequoyah Nuclear Plant which is also on Chickamauga Lake.

The minimum exclusion distance is 3940 feet. The low population zone has a three mile radius. The 1970 census indicated that 570 people lived within this zone. This site is in a rural area. The nearest population center with a 1970 population greater than 25,000 people is Oak Ridge, Tennessee (1970 population 28,140), which is 40 miles from the site.

The Watts Bar units will include four-loop pressurized water reactors designed for initial core power levels up to 3411 MW(t). These reactors are substantially the same as those previously reviewed for the Sequoyah, Trojan, and McGuire plants.

The plant will employ two natural draft cooling towers. Makeup water for the towers and cooling water during emergencies will be taken from a canal about 900 feet long supplied from the Tennessee River. The stability of this canal under seismic conditions, and the adequacy of the water supply under emergency conditions, should be established to the satisfaction of the AEC Regulatory Staff.

Plant grade is 728.0 feet MSL. The probable maximum flood has a still water level of 737.5 feet. The applicant has agreed to protect safety-related structures and equipment against wave effects to elevation 743.5 feet with the understanding that further study may require this elevation to be increased. This matter should be resolved to the satisfaction of the Regulatory Staff.

In order to satisfy the AEC "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors", the applicant proposes as one possibility a reduction in the maximum permissible linear power to 14.9 kw per foot at full power. However, the applicant is conducting an experimental and analytical program intended to provide improved understanding of phenomena entering into the loss-of-coolant accident, and is studying various possible improvements in ECCS design, including the addition of emergency core cooling water to the vessel upper head cavity. The Committee believes it important that improvements in ECCS design be included in the Watts Bar plant, and recommends that the final design of the Watts Bar ECCS be reviewed by the Regulatory Staff and the ACRS prior to fabrication and installation of major components.

The applicant stated that the fuel rod problem involving densification and subsequent movement of the fuel pellets is undergoing intensive investigation. The Regulatory Staff and the ACRS should review the resolution of this matter.

The applicant will submit the results of recent additional analytical studies of local and overall pressures in the ice-condenser containment for various postulated loss-of-coolant accidents. The Committee recommends that the Regulatory Staff obtain independent confirmation of containment accident pressures and assure itself that adequate margin exists to cover uncertainties.

The Committee believes that protection against pipe whip should be provided by the applicant in accordance with criteria being developed by the AEC Regulatory Staff.

Further studies are in progress with regard to the effects of a failure to scram on anticipated transients and of design features which would make tolerable the results of such an event. These studies should be expedited and the matter resolved during construction in a manner satisfactory to the Regulatory Staff and the ACRS.

Other problems related to large water-cooled and moderated 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 Watts Bar Plant.

The ACRS believes that the above items can be resolved during construction and that, if due consideration is given to these items, the Watts Bar Nuclear Plant 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.

Additional remarks by Dr. H. S. Isbin are presented below.

Sincerely yours,

/s/ C. P. Siess

C. P. Siess
Chairman

Additional Remarks by Dr. H. S. Isbin

I believe that it is inappropriate to reduce the design peaking factor by 21% just in order to meet the AEC "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors." Instead, increased efforts should be devoted to the experimental and analytical programs, together with possible improvements in the ECCS design. These matters were noted in the Committee's October 9, 1971 Report on McGuire Nuclear Station Units 1 and 2.

References

1. Tennessee Valley Authority letter dated May 14, 1971; License Application; Preliminary Safety Analysis Report (PSAR), Volumes 1, 2, 3 and 4

2. Amendments 1-5, 7, 9-11, and 13 to PSAR

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

May 17, 1974

J. F. O'Leary, Director
Directorate of Licensing

ACRS REVIEW OF WATTS BAR ECCS

During its 169th Meeting, May 9-11, 1974, the ACRS discussed the possibility of conducting a review of the final design of the Watts Bar Emergency Core Cooling System in connection with its review of Sequoyah Units 1 and 2, as proposed by the applicant. The Committee had originally recommended, in its report of September 21, 1972, that a review of the Watts Bar ECCS be carried out by the Regulatory Staff and the ACRS prior to fabrication and installation of major components for the Watts Bar Station.

Since TVA has stated in its letter to you of April 29, 1974, that final design, fabrication, and system construction for Watts Bar are now underway, the Committee concluded that review of Sequoyah is too far in the future (May 1975) to be compatible with its original recommendation.

It appears that this review should be scheduled on a more expedited basis either as a generic review for this class of reactor or as part of a specific project review.

R. F. Fraley
R. F. Fraley
Executive Secretary

cc: ACRS Members
S. Varga, DL
M. W. Libarkin, ACRS
J. H. Conran, ACRS
A. Giambusso, DL

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 16, 1982

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2

During its 268th meeting, August 12-14, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of the Tennessee Valley Authority (TVA) for authorization to operate the Watts Bar Nuclear Plant, Units 1 and 2. The project was considered at ACRS Subcommittee meetings in Knoxville, Tennessee on April 30, 1982, and in Washington, D.C. on August 10, 1982. Members of the Subcommittee toured the facility on April 30, 1982. In its review, the Committee had the benefit of discussions with representatives of TVA, Westinghouse Electric Corporation, and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee commented on the construction permit application for the Watts Bar Nuclear Plant in a report dated September 21, 1972.

The Watts Bar Nuclear Plant is located in Rhea County in southeastern Tennessee, about 45 miles north-northeast of Chattanooga, Tennessee. Each of the two identical units uses a Westinghouse nuclear steam supply system with a rated core power of 3411 MWt and has an ice-condenser containment with a design pressure of 15 psig. TVA estimates that Watts Bar Nuclear Plant, Units 1 and 2 will be ready for fuel loading by August 1983 and August 1984, respectively.

A number of items have been identified by the NRC Staff as Outstanding Issues, Confirmatory Issues, and License Conditions. These matters should be resolved in a manner satisfactory to the NRC Staff.

Late in the construction program a serious quality assurance breakdown was identified - principally in the construction area, but also in the design area. The effects of the breakdown persist, and corrective work on the plant will continue at least throughout 1982. TVA invoked major quality assurance programmatic changes, including plans to have an independent contractor review the design and construction of a typical "vertical section" of the plant, to confirm the adequacy and safety of the as-completed plant. This issue should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed.

Both Watts Bar Nuclear Plant units have Westinghouse Model D-3 steam generators. Steam generators of this design have experienced tube failures, apparently related to flow-induced vibrations in the preheater region. TVA has stated that this problem is being worked on by Westinghouse and that a resolution involving internal modifications is expected before the projected fuel load date for Unit 1. We wish to be kept informed.

TVA is using a cement mortar lining in the essential raw cooling water system piping to reduce the pressure drop from corrosion-induced roughness. We believe that periodic inspections and tests of this lined piping should be carried out so that, if the bonding or quality of the coating should unduly deteriorate, the system will not be subject to sudden entrainment of debris.

TVA is developing a hydrogen ignition system using controlled distributed ignition sources. The system to be used at the Watts Bar Plant will be of the same design as the permanent system to be installed at the Sequoyah Nuclear Plant. We expect to review that system in the near future. We recommend that specific attention be given by the NRC Staff to assuring the reliability of the hydrogen monitors used in conjunction with this system. Acceptability of this system has been designated as a License Condition by the NRC Staff.

The ACRS believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Watts Bar Nuclear Plant, Units 1 and 2 can be operated at core power levels up to 3411 MWt without undue risk to the health and safety of the public.

Additional comments by ACRS member D. Okrent are presented below.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

<u>Additional Comments by ACRS Member D. Okrent</u>

With regard to the seismic design, I recommend that TVA and the NRC Staff conduct studies to evaluate the margins available to accomplish safe shutdown, including long-term heat removal, following an earthquake of somewhat greater severity and lower likelihood than the safe shutdown earthquake. I believe it is important that there be considerable assurance that the combination of seismic design basis and margins in the seismic design is such that this accident source represents an acceptably low contribution to the overall risk from this plant.

References:

1. Tennessee Valley Authority, "Watts Bar Nuclear Plant Final Safety Analysis Report," with Amendments 1-46.
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," NUREG-0847, dated June 1982.

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

November 12, 1958

Honorable John A. McCone
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: WESTINGHOUSE TESTING REACTOR (WTR)

Dear Mr. McCone:

During its Eleventh Meeting, November 7, 1958, the Advisory Committee on Reactor Safeguards reviewed the Westinghouse Testing Reactor. The WTR is a water moderated and cooled heterogeneous reactor located at Waltz Mills, Pennsylvania, and nearing completion under a construction permit issued by the Commission. The Westinghouse Company is now requesting a license to operate this reactor at a power of 20 megawatts. For its review, the ACRS was furnished Westinghouse report, WCAP-369 (Rev.), and discussed the reactor with the Division of Licensing and Regulation and with Westinghouse personnel.

In many respects the WTR is similar to the Materials Testing Reactor for which eight years of operating experience is available. Thus both the characteristics of this type of reactor and the operating problems associated with its testing function are well known. Like the MTR, the WTR will also be operated in conjunction with a critical facility with which the reactivity of new experiments can be determined with fair precision. In addition, the WTR is housed in a large steel vessel designed to contain, with nominal leakage, the fission products which might be released in a severe reactor accident.

The Advisory Committee on Reactor Safeguards concludes that the Westinghouse Testing Reactor can be operated without undue hazards to the health and safety of the public.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman

cc: P. F. Foster, GM
H. L. Price, DLR

References:
WCAP-369 (Rev.)
Amendment No. 8 to License Application, 9/29/58
Amendment No. 9 to License Application, 10/30/58
HEB Staff Analysis, 10/7/58

November 14, 1959

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D.C.

Subject: WESTINGHOUSE TESTING REACTOR (WTR)

Dear Mr. McCone:

On November 13, 1959, at its twenty-first meeting the Advisory Committee on Reactor Safeguards reviewed the application of the Westinghouse Testing Reactor (WTR) to increase the power of the reactor from 20 to 60 megawatts thermal power.

The Westinghouse Testing Reactor was originally designed for an ultimate operating power of 60 megawatts (thermal). Initial tests and operation at 20 megawatts indicate that the reactor is capable of this ultimate level. Studies by Westinghouse, concurred in by the Hazards Evaluation Branch, indicate that the hot channel factors, film rise, thermal and hydraulic characteristics and other pertinent features compare favorably with similar ones at the MTR and ETR.

The WTR has demonstrated the effect of bubble formation on the reactor power level fluctuations. They have stated their willingness to retain apparatus and a special detection channel in the reactor during their step-wise rise to 60 Mw (thermal). In this way they can demonstrate the existence of boiling, should it occur. Calculations indicate that oscillations in power may occur when boiling results in 1.8 per cent void by volume.

The Committee recommends that this bubble formation apparatus and detector remain within the reactor until the 60 megawatt level has been reached. The Committee further recommends that the heat flux not be allowed to go above half of that required for burnout and that no more than one percent of the core volume be voided by boiling.

With these reservations the Advisory Committee on Reactor Safeguards believes that this reactor can be operated as proposed at 60 megawatts (thermal) without undue hazard to the health and safety of the general public.

Sincerely yours,
/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman

cc: A. R. Luedecke, GM
H. L. Price, DL&R
ACRS Members & Dr. Duffey - 11/19/59

bc: M. Axelrad, OGC

<u>References</u>

1) WCAP-369 (Rev.) - Final Safety Report for the Westinghouse Testing Reactor, August 7, 1958.

2) Amendment No. 8 to License Application for the Westinghouse Testing Reactor, September 1958.

3) Amendment No. 9 to License Application for the Westinghouse Testing Reactor, October 1958.

4) Amendment No. 11 - Description of a Revised Core Structure for the Westinghouse Testing Reactor, January 1959.

5) Amendment No. 12 to Class 104 License Application for the Westinghouse Testing Reactor, February 5, 1959.

6) Amendment No. 14 to Class 104 License Application for the Westinghouse Testing Reactor, May 1959.

7) Supplementary information to Amendment No. 14 (WTR-22, "Report on Early Operation of the Westinghouse Testing Reactor", September 3, 1959; WTR-23, "Preliminary Report Hydraulic Evaluation Tests"; WTR-21, Appendix II, "Method of Calculating Thermal Performance of WTR at 60 Megawatts," August 7, 1959.

8) WTR-25 - Thermal and Hydraulic Investigation of Testing Reactors with Appendix I (WTR-SS-TA-258), October 1959.

9) U.S. Weather Bureau comments on "Amendment No. 14", June 22, 1959.

10) Division of Licensing and Regulation Report to the ACRS on the Westinghouse Testing Reactor, October 7, 1958.

11) Division of Licensing and Regulation Report to the ACRS on the Westinghouse Testing Reactor, September 23, 1959.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

December 10, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

SUBJECT: REACTOR ACCIDENTS

Dear Mr. McCone:

On April 3, 1960, the Westinghouse Testing Reactor experienced an accident of such severity that the radiation contamination and other consequences caused the reactor to be shut down for a period of five months.

Subsequently engineering modifications to the reactor system and changes in operating procedures were proposed and carried out by the applicant. These changes were reviewed by the Commission staff and on September 7, 1960, the Commission granted authority for re-start and operation of the reactor.

Copies of the documents describing the system changes were furnished to the Advisory Committee on Reactor Safeguards by the Commission, but no advice was requested.

At the request of the Committee, a representative of Westinghouse Electric Corporation, in an information session on September 22, 1960, briefed the ACRS on the accident and the corrective measures taken.

The ACRS believes that when any reactor because of a serious accident requires substantial repairs, change of design, or operational procedures, the Committee should be requested to give advice as to whether the changes proposed insure adequate protection for the health and safety of the public.

Sincerely yours,

Sgd/LESLIE SILVERMAN

Leslie Silverman
Chairman

cc: A. R. Luedecke, GM
W. F. Finan, AGMRS
H. L. Price, Dir., DL&R

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

March 4, 1961

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

Subject: WESTINGHOUSE TESTING REACTOR (WTR)

Dear Dr. Seaborg:

The Committee is in receipt of the letter from A. R. Luedecke, General Manager, on this subject, dated March 1, 1961.

This reactor is a testing reactor which experienced rather serious difficulties subsequent to the last review by the Advisory Committee on Reactor Safeguards. The Committee believes that it should review the modified design and method of operation of this reactor and the experimental program which it will carry out. The desirability of this review is further emphasized since it is understood that there have been changes in design and method of operation of this reactor since its last review by the Committee.

The Committee would like to schedule a review of the design and method of operation of this reactor, together with the experiments currently being performed and those proposed. In particular, the Committee desires to obtain information on the design and testing of the control system and the verification of the margin of safety with regard to burnout of fuel elements or their melting due to failure of the coolant system. A proposed time for the review would be during the May 1961 meeting. The Committee would appreciate receiving this information prior to its meeting. The AEC staff and the owners and operators of the reactor should be present to enable the Committee to obtain all pertinent facts.

The Committee is not aware of any reason for suspending the operation of this reactor pending its study and review.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

Reference:
Letter - A. R. Luedecke to T. J. Thompson, dated March 1, 1961

cc: A. R. Luedecke, GM
W. F. Finan, AGMRS
H. L. Price, Dir., DL&R

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 16, 1975

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

Subject: REPORT ON THE WOLF CREEK GENERATING STATION, UNIT 1

Dear Mr. Anders:

During its 186th meeting, October 9-11, 1975, the Advisory Committee on Reactor Safeguards reviewed the application of Kansas Gas and Electric Company and Kansas City Power and Light Company for a permit to construct Wolf Creek Generating Station Unit 1. The site was visited on September 25, 1975, and a Subcommittee meeting was held in Emporia, Kansas on September 26, 1975. The "Standardized Nuclear Unit Power Plant System" (SNUPPS), to be utilized at the Wolf Creek site and three other plant sites, was also reviewed at a Subcommittee meeting held in Washington, D. C. on August 19, 1975, and at the 185th and the 186th meetings of the Committee. During its reviews, the Committee had the benefit of discussions with the Nuclear Regulatory Commission (NRC) Staff, and representatives of the applicants, the Westinghouse Electric Corporation and the Bechtel Corporation. The Committee also had the benefit of the documents listed below.

The Wolf Creek plant will be located on a 10,000-acre site in the Neosho River Basin in Coffey County, Kansas, about 28 miles east-southeast of Emporia, the nearest population center (1970 population: about 23,000). The exclusion area extends radially from the center of the reactor building a distance of 1200 meters. Except for several existing public roads which the applicants have assured can and will be abandoned prior to the start of construction, the applicants own all the area within the exclusion zone.

The SNUPPS will utilize the RESAR-3 Consolidated Version, four-loop pressurized water nuclear reactor having a core power output of 3411 MW(t). This design is similar to that utilized at the Comanche Peak Steam Electric Station Units 1 and 2, reported on by the Committee in its letter of October 18, 1974. The Committee's continuing review of the SNUPPS was reported on in its Callaway letter dated September 17, 1975. It is anticipated that the Committee's report on the remainder of its review of SNUPPS will be included in its report on the Tyrone application.

The NRC Staff has identified several items in its review of the Wolf Creek application which are not yet completed. The Committee recommends that any outstanding issues which may develop in the course of completing these reviews be dealt with in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed on the resolution of the following items:

1. The emergency core cooling system evaluation in compliance with the Final Acceptance Criteria.

2. The analyses of the effects of anticipated transients without scram.

3. The evaluation of the plant design to meet the requirements of the new Appendix I of 10CFR Part 50.

The RESAR-3 Consolidated Version nuclear design utilizes the Westinghouse 17x17 fuel assembly. Westinghouse has identified an integrated test program to confirm the safety margins associated with this design, which it plans to complete late this year. The RESAR-3 reactor core design has been calculated by Westinghouse to be stable against radial xenon oscillations. Westinghouse has agreed to verify this stability in a startup physics test for a 193 fuel assembly core similar to SNUPPS. The Committee will continue to review these matters as appropriate documentation is submitted.

The Committee recommended in its report of September 10, 1973, on acceptance criteria for ECCS, that significantly improved ECCS capability should be provided for reactors for which construction permit requests are filed after January 7, 1972. The SNUPPS design is in this category. These units will use the 17x17 fuel assemblies similar to those to be used in Comanche Peak Steam Electric Station, Units 1 and 2. Although calculated peak clad temperatures in the event of a postulated LOCA are less for 17x17 assemblies than for a 15x15 array, the Committee believes that the applicants should continue studies that are responsive to the Committee's September 10, 1973 report. If studies establish that significant further ECCS improvements can be achieved, consideration should be given to incorporating them into this unit.

The Wolf Creek Plant, Unit 1 will be the first commercial nuclear power plant in the State of Kansas. For this reason, the Committee recommends that the applicant and the NRC Staff give particular attention to assuring proper coordination with appropriate state agencies in the development of effective emergency plans for this facility.

The Committee believes that the applicants and the NRC Staff should continue to review the Wolf Creek plant design for features that could reduce the possibility and consequences of sabotage.

The Committee recommends that the NRC Staff and the applicants review the design features that are intended to prevent the occurrence of damaging fires and to minimize the consequences to safety-related equipment should a fire occur. This matter should be resolved to the satisfaction of the NRC Staff. The Committee wishes to be kept informed.

Generic problems relating to large water reactors are discussed in the Committee's report dated March 12, 1975. These problems should be dealt with appropriately by the NRC Staff and the applicants.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above and the items mentioned in its Callaway letter, which are relevant to the Wolf Creek application, can be resolved during construction, and that if due consideration is given to the foregoing,the Wolf Creek Generating Station Unit 1 can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

WKerr

W. Kerr
Chairman

References

1. SNUPPS Preliminary Safety Analysis Report with Revisions 1 through 10 and the Wolf Creek Site Addendum Report with Revisions 1 through 7.

2. RESAR-3 Consolidated Version, Westinghouse Reference Safety Analysis Report with Amendments 1 through 6.

3. Safety Evaluation Report NUREG 75/080 related to the construction of the Wolf Creek Generating Station, Unit No. 1, Docket No. STN 50-482, September, 1975.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 11, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE WOLF CREEK GENERATING STATION, UNIT NO. 1

During its 265th meeting, May 6-8, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of Kansas Gas and Electric Company (KG&E), Kansas City Power and Light Co. and Kansas Electric Power Cooperative, Inc. (Applicants) for a license to operate the Wolf Creek Generating Station, Unit No. 1. The Station is to be operated by KG&E. A Subcommittee meeting was held in Emporia, Kansas, on April 21-22, 1982, to consider this project. A tour of the facility was made by members of the Subcommittee on April 21, 1982. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, Westinghouse Electric Corporation, Bechtel Power Corporation, the Nuclear Regulatory Commission (NRC) Staff, and with members of the public. The Committee also had the benefit of the documents listed below. The Committee commented on the construction permit application for this plant in its report dated October 16, 1975.

The Wolf Creek Generating Station is located in Hampdon Township, Coffey County, Kansas. The site is in eastern Kansas approximately 53 miles south of Topeka, and 100 miles east-northeast of Wichita. The nearest population center is Emporia, Kansas, 28 miles west-northwest of the site (estimated 1980 population of 25,019).

The Wolf Creek Generating Station will be the first commercial nuclear power plant in the state of Kansas. It should be assured that state and local agencies are qualified to respond to possible emergency situations associated with the operation of the Wolf Creek Generating Station.

The Station will use a Westinghouse, four-loop, pressurized water, nuclear steam supply system having a rated power level of 3425 MWt. Unit 1 employs a cylindrical, steel-lined, reinforced, post-tensioned concrete containment structure with a free volume of 2.5 million cubic feet. The Wolf Creek Generating Station uses the Standardized Nuclear Unit Power Plant System (SNUPPS) design. It is one of two plants built to this design. The Committee reported on the operating license application of the other plant (Callaway Plant Unit No. 1) in its November 17, 1981 report to you.

The Wolf Creek Generating Station is the first nuclear power plant to be operated by KG&E. The Committee reviewed KG&E's management organization, experience, and training programs. We were favorably impressed by the general competence and attitude of KG&E's personnel. Nevertheless, we wish to emphasize the importance of KG&E's building a strong in-house capability for analyzing and understanding the nuclear-thermal-hydraulic behavior and systems performance of this plant.

To strengthen the shift structure during the initial period of operation, KG&E plans to augment each shift with a consultant who is an experienced, previously licensed PWR operator. These consultants will serve for a period of one year after startup. In addition, KG&E has retained the services of a consultant with considerable commercial nuclear experience to act as a technical assistant to the Plant Superintendent through the initial loading of fuel. We believe the technical assistant to the Plant Superintendent and the "experienced operator consultants" should be retained until the operating organization has developed an experience base involving those operational duties of importance to public safety. This experience base should be defined by the NRC Staff in consultation with operational experts and incorporated into the regulatory requirements instead of using arbitrary operating time periods as a basis for measuring skill. We encourage the practice of assigning the Senior Reactor Operator (SRO) candidates to extended tours of service at operating nuclear power plants, and recommend that others in the operations staff participate in such a program to the extent practical.

KG&E has proposed, as an alternative to a Shift Technical Advisor (STA), that at least one SRO on each shift have the training and background required for an STA. This approach appears to us to meet the need which originally led to the requirement of an STA. However, it is not clear that the level of training given to the SROs will correspond to that intended for STAs, and we recommend that the Staff review this matter carefully.

The site-specific portions of the plant, including vital aspects of the ultimate heat sink and associated systems, were designed for a 0.12 g earthquake, and are being reanalyzed for an earthquake represented by site-specific response spectra that are encompassed by Regulatory Guide 1.60 spectra anchored at a zero-period acceleration of 0.15 g. The standard portion of the plant, on the other hand, was designed for a 0.20 g earthquake with the usual margins of safety and thus would be expected to withstand a considerably larger earthquake without failing in such a manner as to cause a severe accident.

We do not have confidence that all vital aspects of the ultimate heat sink and associated systems have margins sufficient to provide an appropriate level of resistance to a lower probability, more severe earthquake. We recommend therefore that the seismic margins inherent in the components of the ultimate heat sink and associated systems be investigated further and that any needed modifications be made before the plant resumes operation after the second refueling.

Other issues have been identified as Outstanding Issues, License Conditions, and Confirmatory Issues in the Staff's Safety Evaluation Report dated April 1982; these include some TMI Action Plan requirements. Except as noted above, we believe these issues can be resolved in a manner satisfactory to the NRC Staff and recommend that this be done.

We believe that, if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, training, and preoperational testing, there is reasonable assurance that the Wolf Creek Generating Station, Unit No. 1 can be operated at power levels up to 3425 MWt without undue risk to the health and safety of the public.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. "Final Safety Analysis Report for Standardized Nuclear Unit Power Plant System," with Revisions 1-8.
2. "Final Safety Analysis Report, Wolf Creek Generating Station Unit No. 1," with Revisions 1-8.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Wolf Creek Generating Station, Unit No. 1," NUREG-0881, dated April 1982.
4. Written statement by John M. Simpson, Attorney for Intervenors, Re: Emergency Planning Procedures and Plans - Wolf Creek Plant, dated April 22, 1982.

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

September 16, 1957

Honorable Lewis L. Strauss
Chairman, Atomic Energy Commission

Dear Mr. Strauss:

Pursuant to Section 182 of the Atomic Energy Act of 1954, as amended, this letter constitutes the report of the Advisory Committee on Reactor Safeguards with respect to the application for a construction permit by the Yankee Atomic Electric Company, Docket No. 50-29.

The proposed reactor is a pressurized light water reactor, which is designed to produce 492 megawatts of heat and 134 megawatts of electrical power, to be located near Rowe, Massachusetts.

There are three novel features of the reactor that bear on the safety of the system chosen. These are:

1. The addition of neutron absorbers.

2. Intentional design into the reactor of nucleate boiling.

3. Large plutonium build-up.

Experimental programs have been proposed by the applicant to establish the effect of these three novel features. These programs are to determine whether any undesirable instabilities could result from these modifications of pressurized light water systems, with which satisfactory operating experience is available. We regard these experimental programs, together with one additional one discussed below, as being the most important from the standpoint of ensuring the safety of the design finally adopted. They are, respectively:

1. Experimental studies of the conditions under which solid phases form when aqueous solutions containing a suitable corrosion inhibitor and nuclear poison are exposed to reactor radiation at the temperature and pressures of the proposed reactor. These studies should be carried to a point to establish that no significant amounts of poison-containing deposits will form in the reactor under the operating conditions finally adopted.

2. The part-core critical experiments. These studies should be completed and the results correlated. This information should be used in determining the final design of the reactor core.

3. Studies on the effect of the plutonium isotope build-up on the reactivity, flux distribution, temperature coefficient and void coefficients. The applicant's proposal to investigate data from Hanford and Savannah River on these effects of plutonium and to prepare synthetic fuel elements containing long-exposure plutonium and to measure their effect on reactivity in the part-core critical assembly are especially important. These studies should be completed and the information from these studies should be used in determining the final design of the reactor.

The other important experimental program is as follows: In arriving at the final design parameters, it is recommended that the design criteria be so chosen as to prevent the attainment of the burnout heat flux under abnormal, but credible, transient conditions. The relevant criteria are those concerned with temperature and void coefficients and flux each as a function of position with the reactor. It is likely that all of the pertinent design factors may not be confirmed until critical experiments are actually carried out in the power reactor itself. It is important that the applicant conduct the critical experiments proposed and make use of the results of these critical experiments in arriving at the final design of the control instrumentation and in establishing the operating conditions for the reactor.

The Committee is convinced that a reactor of the general type proposed in the application and amendments can be operated at the proposed location with an acceptably low risk of any injury to the health and safety of the public. By this, we mean that the possibilities of any incident which could cause such injury are remote and the consequences of such an incident in terms of endangering the health and safety of the public would be in our judgment low.

The Committee in reaching its generally favorable opinion regarding the safety of the proposed reactor has been influenced by the following considerations: the design parameters of pressurized light water reactors are largely known; the reactor is provided with a containment sphere which the applicant states will be sealed tightly during operation, and we are certain that it can be; the site appears to be adequate with respect to meteorology, hydrology and isolation.

Further, we are confident that the safety aspects of the novel features of the proposed reactor can be satisfactorily resolved by incorporating suitable design features which ought to result from the general type of experimental program proposed by the applicant. Since these programs have not been presented to us in detail, we cannot be entirely assured of their adequacy. Therefore, we have suggested above, as to those programs that are of major importance, the kind of conclusion to which they should be carried.

The Committee's conclusions are based on a reactor of the general design features specified in the application and amendments. However, it must be recognized that development work on the reactor is still in progress and that further experimental and theoretical studies are propsoed to be accomplished before the detailed design of this reactor is finalized. Therefore, before the Committee could recommend approval of the operation of the reactor, it would have to review the detailed design, the results of the experimental programs, and other information which subsequently might be developed and have a bearing on this particular reactor.

I have been authorized by the Advisory Committee on Reactor Safeguards to submit this report to you.

/s/ Reuel C. Stratton

Reuel C. Stratton
Vice Chairman
Advisory Committee on Reactor Safeguards

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

October 21, 1958

Mr. Harold L. Price, Director
Division of Licensing and Regulation
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: YANKEE ATOMIC ELECTRIC COMPANY

Dear Mr. Price:

The Committee finds that there are some matters relating to the Yankee reactor on which it should have more information before it can give its final opinion on the overall safety of this reactor. These are:

1) The means that will be used to estimate the distribution of neutron flux and heat flux in the reactor so that the margin of operation below burnout conditions can be determined.

2) The results of experiments on the extent of precipitation of solids from water containing the amounts of boric acid and lithium hydroxide expected in the operating reactor under the conditions of temperature, pressure and radiation intensity which will be experienced in the reactor.

3) The principles and procedures to be used in operating this reactor.

Would you please arrange to have this information developed for us?

Sincerely yours,

/s/

C. Rogers McCullough
Chairman

cc: Dr. Brooks)
Dr. Wolman) not at meeting

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

October 21, 1958

Honorable John A. McCone
Chairman, U.S. Atomic Energy Commission
Washington 25, D. C.

Subject: YANKEE ATOMIC ELECTRIC COMPANY

Dear Mr. McCone:

At its tenth meeting on October 15, 1958, the Advisory Committee on Reactor Safeguards reviewed Amendments No. 7 and No. 8 to the application of the Yankee Atomic Electric Company for a license to operate the nuclear power plant the company is constructing at Rowe, Massachusetts. The Committee had available to it the material referenced on the last page.

Amendment No. 7 proposed an alternative to the measures recommended by the Committee in its letter of September 16, 1957, for determining the effects of plutonium buildup on the nuclear characteristics and stability of this reactor. Amendment No. 8 described the waste disposal facilities planned for the Rowe plant.

Amendment No. 7

In Amendment No. 7, Yankee states that determination of the effect of plutonium buildup by measurements on synthetic fuel elements made up of uranium and plutonium in a part-core critical facility, as suggested in the Committee's earlier letter, would be difficult to interpret because of the impossibility of duplicating the temperatures and neutron spectrum of the actual Yankee reactor in this facility. As an alternative and more dependable procedure for determining the effect of plutonium, Yankee proposes an experimental program to measure temperature coefficients, prompt and overall, in the actual power reactor at startup, after 2000 hours of operation, and at intervals while plutonium is growing into the core.

To establish that the reactor can be operated safely while plutonium is building up in the core during the interval between experimental measurements, Yankee described calculations made by Westinghouse and by Nuclear Development Corporation of America which show that the buildup of plutonium during even the entire anticipated fuel lifetime of 10,000 hours will have only a minor effect on the overall temperature coefficient of reactivity.

The Committee concurs with Yankee's proposal to determine these temperature coefficients in the actual reactor rather than in the part-core critical facility and agrees with Yankee's judgment that the effect of plutonium buildup on these coefficients will be small enough to permit these measurements to be made with safety in the actual reactor. In this, the Committee concurs with the determination of the Hazards Evaluation Branch.

The Committee recommends that Yankee be asked to provide a description of the specific experiments which will be made to determine the effects of plutonium buildup on prompt and overall temperature coefficients. It suggests that controlled transient experiements, with known sinusoidal or stop changes in reactivity, may be a convenient means of measuring these coefficients.

Amendment No. 8

The Committee heard a detailed description of the facilities planned by Yankee for disposal of gaseous, liquid and combustible solid wastes. It believes that the design of these facilities is conservative and concurs with the conclusion of the Hazard Evaluation Branch that the proposed facilities will permit the disposal of wastes without undue hazard to on-site or off-site personnel.

Sincerely yours,

/s/

C. Rogers McCullough
Chairman

cc: P. F. Foster, GM
H. L. Price, DL&R

References:

1) Amendment No. 3 to Preliminary Hazards Summary Report by Yankee Atomic Electric Company, dated April 1957.

2) Amendment No. 4 to Preliminary Hazards Summary Report by Yankee Atomic Electric Company, dated July 15, 1957.

3) Amendment No. 6 to Preliminary Hazards Summary Report by Yankee Atomic Electric Company, dated March 5, 1958.

4) Amendment No. 7 to Preliminary Hazards Summary Report by Yankee Atomic Electric Company, dated July 21, 1958.

5) Amendment No. 8 to Preliminary Hazards Summary Report by Yankee Atomic Electric Company, dated July 28, 1958.

6) Report to ACRS by Division of Licensing and Regulation, dated September 29, 1958.

7) Memorandum from R. C. Dalzell to H. L. Price, subject: Yankee Atomic Electric Company License Application Amendments No. 7 and No. 8, dated September 2, 1958.

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

February 1, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: YANKEE NUCLEAR POWER STATION - YANKEE ATOMIC ELECTRIC COMPANY*

Dear Mr. McCone:

At its twenty-third meeting, January 28-29-30, 1960, the Advisory Committee on Reactor Safeguards considered various safety aspects of the Yankee Atomic Electric Company 485 MW (thermal) pressurized water power plant. In addition to the reports referenced below, discussions were held with the Hazards Evaluation Branch, Stone & Webster Engineering Corporation, Yankee Atomic Electric Company, and Westinghouse Electric Corporation.

In our letter of September 16, 1957, relative to the Yankee construction permit, the ACRS pointed out that the design of this reactor included three novel features: addition of soluble neutron absorbers, intentional design into the reactor of nucleate boiling, and large plutonium buildup, all of which would require extensive investigation.

The problem of nucleate boiling and the use of boric acid as a soluble poison to supplement the control rods during cold shut down have been thoroughly investigated and solutions to these problems satisfactory to the HEB and the Committee have been reported in the Final Hazards Summary Report.

Amendment No. 7 proposes determination of effect of plutonium buildup in an experimental program to measure temperature coefficients, prompt and overall, in the actual power reactor at startup, after 2000 hours of operation, and at intervals while plutonium is growing into the core.

In a letter dated October 21, 1958, the ACRS agreed that effect of plutonium buildup on these coefficients would be small enough to permit these measurements to be made with safety in the actual reactor.

Amendment No. 8 covered a plant for disposal of gaseous liquid and combustible solid wastes. In the letter of October 21, 1958, the ACRS agreed with HEB that proposed facilities will permit disposal of wastes without undue hazard to on-site or off-site personnel.

Amendment No. 18 is a proposal stating intent to modify the reactor design to permit continued operation of the plant even with leakage from the primary to the secondary system. The changes are discussed in general, but no design details were supplied. The Committee concurs in principle that the general plan can permit this leakage without undue hazard to the public, but cannot comment on design detail.

The general design of the reactor and the proposed startup procedures and schedules are considered acceptable. The applicant's proposal to review its operation at the 392 MW (thermal) power level before proceeding to higher powers is endorsed.

The Committee believes that the broad problems indicated at the time of the issuance of the construction permit have been resolved. It is the Committee's opinion that this reactor can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/S/

Leslie Silverman
Chairman

cc: A.R. Luedecke, GM
W.F. Finan, OGM
H.L. Price, DL&R
ACRS Members & Dr. Duffey (except Dr. Thompson)

bc: L.K. Olson, GC

References:

1) Final Hazards Summary Report, Volumes I and II, (undated) received September 1959.
2) Amendment No. 15 to License Application dated July 29, 1956, October 2, 1959.
3) Amendment No. 16 to License Application, December 4, 1959.
4) Amendment No. 17 to License Application, January 11, 1960.
5) Amendment No. 18 to License Application, January 13, 1960.
6) Office of Health and Safety Comments, October 23, 1959.
7) Division of Licensing Report to ACRS, October 28, 1959.
8) Division of Licensing and Regulation Report to ACRS, January 12, 1960.

*Theos J. Thompson did not participate in these reviews or discussions.

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

May 9, 1960

Honorable John A. McCone
Chairman
U S. Atomic Energy Commission
Washington 25, D. C.

Subject: YANKEE NUCLEAR POWER STATION - YANKEE ATOMIC ELECTRIC COMPANY

Dear Mr. McCone:

At its twenty-fifth meeting, May 5-7, 1960, the Advisory Committee on Reactor Safeguards considered the safety aspects of Amendments Nos. 19 and 20 to the Yankee license application, and reviewed the proposal for evaluating the effect of plutonium buildup through an experimental program which was discussed in the ACRS letters dated October 21, 1958, and February 1, 1960. These matters were also discussed with the Hazards Evaluation Branch.

In a letter dated February 1, 1960, the Committee indicated that it concurred in principle with the general plan for permitting operation of the plant, even with some leakage from the primary to the secondary system as proposed in Amendment No. 18. The Committee was unable to comment on the design details for accomplishing this because they were not available when that letter was written. The Committee agrees that the proposals set forth in Amendment No. 19 for disposal of any radioactivity from such steam generator leakage can provide satisfactory control.

The remainder of Amendment No. 19 and Amendment No. 20 cover several design and procedural changes which the Committee also considers to be satisfactory.

In order to clarify the Committee's position relative to the proposal to test the effects of plutonium buildup at 2000-hour intervals, we wish to point out that we have already indicated in our letters of October 21, 1958, and February 1, 1960, our belief that such tests can be made in the reactor without undue hazard. The exact method of making these measurements has not yet been specified, but it is our understanding that details of this program will be submitted for approval before the tests are made. We are in agreement with this schedule and will review this test program when it becomes available. The absence of precise information at this time does not affect our judgment as to the safety of the reactor.

In view of the material discussed above, the Committee reaffirms the opinion stated in its letter of February 1, 1960, that this reactor can be operated without undue risk to the health and safety of the public.

Dr. Theos J. Thompson did not participate in the Committee's consideration of this reactor.

Sincerely yours,

Leslie Silverman
Chairman

cc: A. R. Luedecke, GM
W. F. Finan, OGM
H. L. Price, DL&R

References

1) Amendment No. 19 to License Application, February 24, 1960

2) Amendment No. 20 to License Application, April 6, 1960

3) AEC Staff's Proposed Technical Specifications, April 20, 1960

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

June 27, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: YANKEE NUCLEAR POWER STATION -- YANKEE ATOMIC ELECTRIC COMPANY

Dear Mr. McCone:

At its 26th meeting held at Lawrence Radiation Laboratory, Livermore, California, June 22, 1960, Amendments 22 and 23 to the Yankee Atomic Electric Company license application were reviewed and discussed with the AEC staff. These amendments do not modify the conclusions expressed in our previous letters, (dated February 1, 1960 and May 9, 1960) that this reactor may be operated without undue risk to the health and safety of the public.

Dr. Theos J. Thompson did not participate in the Committee's consideration of this reactor.

Sincerely yours,

Original Signed By
Leslie Silverman
Leslie Silverman
Chairman

References:
(1) Amendment #22 to License Application, May 16, 1960
(2) Amendment #23 to License Application, May 23, 1960

cc: A. R. Luedecke, GM
W. F. Finan, OGM
H. L. Price, DL&R

May 22, 1961

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

Subject: REPORT ON YANKEE ATOMIC ELECTRIC COMPANY, ROWE, MASSACHUSETTS

Dear Dr. Seaborg:

As part of its thirty-fourth meeting at Quincy, Massachusetts, on May 18, 1961, the Advisory Committee on Reactor Safeguards considered safety aspects of the Yankee Atomic Electric Company Power Plant, including Proposed Changes 1-8, Amendments 24-28, and Operating Reports 1-4, referenced below. The Committee had the benefit of discussions with representatives of Yankee, Westinghouse Electric Corporation, and the AEC staff as well as an oral report from its subcommittee covering its meeting in Boston, Massachusetts, on April 28, 1961.

In previous letters dated February 1, 1960, May 9, 1960, and June 27, 1960, the Committee dealt with safety matters including those covered in the Final Hazards Summary Report, Technical Specifications, and all pertinent amendments through No. 23. The one major point which was unresolved related to testing the reactor for effects of plutonium build-up at about 2000-hour intervals. In letters dated October 21, 1958, February 1, 1960, and May 9, 1960, the Committee indicated that such testing could be done in the reactor without undue hazard, but that the program and its results should be reviewed by the Committee. The program and its results to date have been reported by the applicant. The Committee finds the procedures to be acceptable and notes that there have been no detectable effects of plutonium build-up during the first 2000-hour period. The Committee believes that continued use of in-core monitoring of at least the first core is essential to an understanding of how the core is changing with time. The Committee wishes to be kept informed of any significant data that may be developed in this program.

Amendments 24, 26, 27, and 28, and Proposed Changes 1-8 deal with minor modifications to the plant and changes in the Technical Specifications. These should be worked out by Yankee Atomic Electric Company and the AEC staff. Amendment 25 is a request to amend License No. DPR-3 so as to authorize operation of the reactor at steady state power levels to 485 MW(t) and to extend the expiration date of the license to a date forty years after the expiration date of the construction permit.

It is the opinion of the ACRS that with continued surveillance of the plant by the applicant, as proposed, the plant can be operated at steady state power levels of approximately 485 MW(t), with the changes requested, without undue hazard to the health and safety of the public.

Dr. T. J. Thompson did not participate in the reviews or discussions of this project.

Sincerely yours,

/s/

C. Rogers McCullough
Acting Chairman

References:

1. Amendment #24 to License Application, dated 2/10/61.
2. Amendment #25 to License Application, dated 3/31/61.
3. Amendment #26 to License Application, dated 4/3/61.
4. Amendment #27 to License Application, dated 4/12/61.
5. Amendment #28 to License Application, dated 5/8/61.
6. Proposed Change #1, dated 2/10/61.
7. Proposed Change #2, dated 2/14/61.
8. Proposed Change #3, dated 2/21/61.
9. Proposed Change #4, dated 5/1/61.
10. Proposed Change #5, dated 5/1/61.
11. Proposed Change #6, dated 5/8/61.
12. Proposed Change #7, dated 5/8/61.
13. Proposed Change #8, dated 5/8/61.
14. Operation Report #1, dated 2/13/61.
15. Operation Report #2, dated 3/8/61.
16. Operation Report #3, dated 4/7/61.
17. Operation Report #4, dated 5/10/61.
18. Letter R. J. Coe to USAEC, dated 2/3/61.
19. Letter R. J. Coe to USAEC, dated 4/10/61.
20. Letter R. J. Coe to USAEC, dated 5/5/61.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

February 10, 1962

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

Subject: REPORT ON YANKEE ATOMIC ELECTRIC COMPANY - CORE III

Dear Dr. Seaborg:

At its thirty-ninth meeting on February 8-10, 1962, the Advisory Committee on Reactor Safeguards reviewed the request of Yankee Atomic Electric Company for Commission advice concerning the use of boric acid during operation at power with Core III. The purpose of the request is to provide a basis for specifying increased enrichment (4.1%) and thus increased reactivity core life for the fuel for Core III. In connection with this enrichment it will be advantageous to use boric acid dissolved in the primary coolant, at power, to provide adequate control. The Committee had the benefit of the referenced reports and discussion with representatives of Yankee Atomic Electric Company, Westinghouse, and the AEC Staff.

In September 1961, Yankee Atomic Electric Company conducted tests on Core I to study the effects of using boric acid in the primary coolant at full power. The results of these tests were inconclusive, particularly with respect to evidence as to the possibility of depositing boron within the system and then suddenly releasing it to the coolant.

There are several other questions which must be answered before a definite approval can be given for the operation of this reactor with Core III. These relate, for example, to shut down margin, integrity of the boron injection system, and control rod worth. The applicant will submit full information as to the design of the new core and a complete safety analysis of its operation prior to consideration by the staff and by the ACRS.

In the interim, in the light of available information, and recognizing that there are possible alternate steps which can be taken if serious difficulties arise, the Committee can see no reason at this time to advise against specifying the higher enrichment with a view to using boric acid in the primary coolant at power with Core III.

Dr. Theos J. Thompson did not participate in the Committee's consideration of this project.

Sincerely yours,

/s/ F. A. Gifford, Jr.

F. A. Gifford, Jr.
Chairman

References:

1. Letter from Yankee Atomic Electric Co., to USAEC, dated Jan. 3, 1962, regarding design and fabrication of Core III.

2. Yankee Nuclear Power Station -"Operation Report No. 10 for October 1961", dated Nov. 20, 1961.

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

August 25, 1962

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

SUBJECT: REPORT ON YANKEE ATOMIC ELECTRIC COMPANY

Dear Dr. Seaborg:

At its forty-third meeting on August 23-25, 1962, at Idaho Falls, Idaho, the Advisory Committee on Reactor Safeguards considered the proposal of this licensee to increase the present reactor power level of 485 Mw(t) to 540 Mw(t) and to eliminate the present requirement for temperature coefficient measurements at 2000 operating hour intervals. Discussions were presented by representatives of the Yankee Atomic Electric Company and the Westinghouse Electric Company concerning operational and plant changes to be introduced to permit the increase in power. In addition, the Committee had the benefit of information contributed by the AEC Staff. Data covering the proposed changes are recorded in the documents listed.

This power plant has operated successfully throughout the life of Core I. The experience obtained during this period has indicated the possibility of operation at a higher power level with Core II. The major operational and plant changes proposed include: (a) an additional safety injection pump, (b) automatic low pressure activation of the third charging pump, (c) changed scram requirements, (d) elimination of automatic rod withdrawal, and (e) a limitation of amount of rod withdrawal at power during any eight-hour period.

Power coefficients and moderator temperature coefficients have been determined at 2000 equivalent full power hour intervals during Core I life to determine the possible effect of plutonium build-up. These tests were carried on to substantiate previous theoretical calculations which had indicated that the effect of plutonium build-up would not be significant. Data developed during Core I life have shown that there has been no significant effect. The licensee proposes now to make these determinations during initial start-up after fuel changes, and during scheduled generator load changes and plant shut-down only.

The Advisory Committee on Reactor Safeguards believes that the change in frequency of the 2000-hour tests may be allowed and that this reactor may be operated at a power level of 540 Mw thermal without undue hazard to the health and safety of the public.

Dr. Theos J. Thompson did not participate in the discussion of this project.

Sincerely yours,

/s/ F. A. Gifford, Jr.

F. A. Gifford, Jr.
Chairman

References:

1. WCAP-1997, "New DNB (Burn-out) Correlations", dated June 1, 1962.
2. Amendment No. 41, dated June 4, 1962.
3. Proposed Change No. 26, dated July 20, 1962.
4. Letter from Yankee Atomic Electric Company, dated August 7, 1962.

July 18, 1963

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

Subject: REPORT ON YANKEE ATOMIC ELECTRIC COMPANY, CORE III

Dear Dr. Seaborg:

At its forty-eighth meeting on July 11-13, 1963, at Los Alamos, New Mexico, the Advisory Committee on Reactor Safeguards considered Proposed Change No. 36 which refers to refueling the Yankee reactor with Core III. The proposal involves a change in the loading pattern and fuel enrichment as well as the use of boric acid at power. The Committee had the benefit of the referenced documents and discussions with representatives of Yankee Atomic Electric Company, Westinghouse Electric Corporation, and the AEC Regulatory staff.

On February 10, 1962, the Committee, in response to a request from the Commission, reported on increased enrichment for the proposed Core III and the use of boric acid in the primary coolant at power. At that time, information on the design of the new core and a complete safety analysis were not available and several unresolved areas were indicated.

Since then, complete design details have been provided and questions relative to such items as shut down margin, integrity of the boron injection system, and control rod worth have been resolved. On the basis of the information presented, the Advisory Committee on Reactor Safeguards believes that the proposed Core III loading and operation up to full power with boric acid in the primary coolant may be carried out without undue hazard to the health and safety of the public.

Dr. T. J. Thompson did not participate in this review.

Sincerely yours,

/s/ D. B. Hall

D. B. Hall
Chairman

References:
1. Proposed Change No. 36, dated June 7, 1963.
2. Supplement to Proposed Change No. 36, dated July 1, 1963.

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

September 12, 1963

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

Subject: REPORT ON YANKEE ATOMIC ELECTRIC COMPANY

Dear Dr. Seaborg:

At its forty-ninth meeting on September 5 and 6, 1963, the Advisory Committee on Reactor Safeguards considered Amendment No. 45 in which Yankee Atomic Electric Company requests an increase in maximum power level of Core III from the present limit of 540 to 600 Mw(t) with the main coolant average temperature increased from 514 to 527^{o}F. The Committee had the benefit of referenced documents and discussion with representatives of Yankee Atomic Electric Company, Westinghouse Electric Corporation, and the AEC Regulatory Staff.

Review of the design of the turbine and secondary plant, recalculation of the DNB ratios for 600 Mw(t), and re-analysis of all accidents which could be affected by the proposed power increase were reported. Frequent measurement of flux distribution to be made during Core III life when DNB ratios are expected to be smallest, limitation of exit temperature of the hottest channel, and installation of some new control rods with hafnium absorber sections are proposed. The Committee believes that the proposed operation does not present an undue hazard to the health and safety of the public.

Dr. T. J. Thompson did not participate in the review of this project.

Sincerely yours,

/s/

David B. Hall
Chairman

Reference:
Letter from Yankee Atomic Electric Co. dated July 17, 1963, Amendment No. 45

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 19, 1983

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE SYSTEMATIC EVALUATION PROGRAM REVIEW OF THE YANKEE NUCLEAR POWER STATION

During its 276th meeting, April 14-16, 1983, the Advisory Committee on Reactor Safeguards reviewed the results of the Systematic Evaluation Program (SEP), Phase II, as it has been applied to the Yankee Nuclear Power Station. These matters were also discussed during a subcommittee meeting in Washington, D. C. on February 23, 1983. During our review, we had the benefit of discussions with representatives of the Yankee Atomic Electric Company (Licensee) and the NRC Staff. We also had the benefit of the documents listed.

The Committee has reported previously on its reviews of the SEP evaluations of the five plants in Group 1: Palisades, Ginna, Oyster Creek, Dresden Unit 2, and Millstone Unit 1. The Yankee plant is the first in Group 2 to be reviewed, and differs from the plants in Group 1 in several respects. Whereas none of the plants in Group 1 have yet received a full-term operating license (FTOL), the Yankee plant received an FTOL in June 1961. The plants in Group 1 were all designed and constructed during the period 1963-1971 as compared to a corresponding period of 1955-1961 for the Yankee plant. Yankee is the oldest nuclear power plant still in operation; it has been in commercial operation since 1961. And finally, the Yankee plant with authorized power ratings of 600 MWt (185 MWe) is much smaller than any of the plants in Group 1, the smallest of which, Ginna, is rated at 490 MWe. All of these differences are pertinent to the NRC Staff's evaluation and our review of the SEP in relation to the Yankee plant. Some, but not all, of these differences exist also between the other plants in Group 2 and those in Group 1.

In our report dated May 11, 1982 on the SEP evaluation of the Palisades plant, we commented on the objectives of the SEP and the extent to which they had been achieved. Our review of the SEP in relation to the Yankee plant has led to no changes in our previous findings regarding the extent to which the objectives of the SEP have been achieved and the manner in which the NRC Staff has conducted its review and assessment.

Of the 137 topics to be addressed in Phase II of the SEP, 24 were not applicable to the Yankee plant and another 24 were deleted because they were being reviewed generically under either the Unresolved Safety Issues program or the Three Mile Island Action Plan. Of the 89 topics addressed in the NRC Staff's review, 51 were found to meet current NRC criteria or to be acceptable on another defined basis. We have reviewed the assessments and conclusions of the NRC Staff relating to these topics and have found them appropriate.

The 38 remaining topics involved 80 issues relating to areas in which the Yankee plant did not meet current criteria. These issues were addressed by the Integrated Plant Safety Assessment and various resolutions have been proposed. It is of interest to note that the number of topics and issues in this category is not notably greater for the Yankee plant than for the plants in Group 1. However, there are significant differences, relating chiefly to criteria for protection against external events and to the size of the plant, as discussed further below.

For 36 of the 80 issues included in the Integrated Assessment, the NRC Staff concluded that no backfit is required. We concur.

For 10 of the remaining issues, changes to the Technical Specifications or procedures were recommended by the NRC Staff and agreed to by the Licensee.

For the 9 remaining issues for which the assessment has been completed, the NRC Staff has proposed hardware backfits. The Licensee has agreed to all but one of these. The NRC Staff believes that an ammeter should be installed to indicate charge and discharge of the DC battery current in order to ensure the availability of DC power. We believe that this matter should be resolved in a manner satisfactory to the NRC Staff.

As has been the case for the other plants in the SEP, the Integrated Assessment has not been completed for 25 of the issues, for which the Licensee has agreed to provide the results of studies, analyses and evaluations needed by the NRC Staff for its assessments and decisions. All of these issues are of such a nature that hardware backfits may be required for their resolution. The resolution of these issues will be addressed by the NRC Staff in a supplemental report.

Several of the issues requiring further evaluation result from the fact that the Yankee plant was not designed to resist earthquakes, floods or tornadoes at anywhere near the level required by current criteria.

Failure of Harriman Dam would inundate the site. Determination of whether the dam will fail depends on the value assigned to the Probable Maximum Precipitation in the Deerfield River Basin and on the capacity of the spillway for the dam. Both of these questions are in dispute and the NRC Staff has elected to leave their resolution to the National Oceanic and

Atmospheric Administration for the Probable Maximum Precipitation, and to the Federal Energy Regulatory Commission for resolution of the Dam's integrity for both hydrologic and seismic events. We find this approach acceptable.

Although the evaluations have not been completed, it seems almost certain that extensive modifications would have to be made to structures and systems in order to provide the ability to shut the plant down safely following an earthquake or tornado of the magnitude required by current criteria. The Licensee has proposed the design and installation of a dedicated Hot Shutdown System (HSS) that would be able to remove decay heat and maintain primary inventory following an earthquake or tornado that disables all other means of providing these functions. Both the HSS and those systems in the existing plant that must maintain their integrity will be qualified to survive earthquakes and tornadoes with site-specific intensities prescribed by the NRC Staff. The Staff has agreed with the concept of a dedicated HSS and will evaluate its design and report its findings in a supplemental report. We find this approach acceptable.

Although a plant-specific Probabilistic Risk Assessment (PRA) (excluding external events) has been performed by Energy Incorporated for the Licensee, it was not complete and had not been reviewed fully by the NRC Staff at the time the Integrated Assessment was carried out. The NRC Staff's PRA for the Yankee plant included qualitative consideration of the fault trees from the Yankee PRA aided by results and insights from other PRAs. Eighteen of the SEP topics considered in the Integrated Assessment were evaluated for their significance to risk and the results utilized by the NRC Staff in their evaluations. As in previous reviews, we believe that this use of PRA was appropriate and that suitable use was made of the results.

Our conclusions regarding the Yankee SEP review are as follows:

1. The SEP has been carried out in such a manner that the stated objectives have been achieved for the most part for the Yankee plant.

2. The actions taken thus far by the NRC Staff in its SEP assessment of the Yankee plant are acceptable.

3. Several Outstanding Issues, notably those relating to protection against external events, remain to be resolved. We have been informed of the bases for the resolution of these issues but have not yet reviewed them in detail. At this time, we are satisfied with the SEP evaluation of the Yankee plant; we expect to review further the design

bases for protection against external events, and we wish to review the resolution of the remaining issues when the supplemental report is available.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Acting Chairman

References:

1. U. S. Nuclear Regulatory Commission, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Yankee Nuclear Power Station," NUREG-0825, dated February 1983
2. U. S. Nuclear Regulatory Commission, Safety Evaluation Reports, Yankee Nuclear Power Station, Volumes 1-4, received February 4, 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 13, 1978

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON YELLOW CREEK NUCLEAR PLANT, UNITS 1 AND 2

Dear Dr. Hendrie:

During its 213th meeting, January 5-7, 1978, the Advisory Committee on Reactor Safeguards completed its review of the application of the Tennessee Valley Authority (the Applicant) for a permit to construct the Yellow Creek Nuclear Plant, Units 1 and 2. The application was reviewed at a Subcommittee meeting in Corinth, Mississippi on December 16, 1977. A tour of the site was made by Subcommittee members on December 15, 1977. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, Combustion Engineering Incorporated, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

The Yellow Creek Plant site is located in Tishomingo County, approximately 15 miles east of Corinth, Mississippi. The minimum exclusion area distance is 695 meters; the low population zone radius is three miles. The nearest population center is the Florence-Muscle Shoals-Sheffield-Tuscumbia, Alabama complex (1970 population of about 62,900) which is located approximately 35 miles east of the site.

The application for the Yellow Creek Plant was submitted in accordance with the Commission's standardization policy as described in Appendix 0 to Part 50, "Licensing of Production and Utilization Facilities," and Section 2.110 of Part 2, "Rules of Practice," of Title 10 of the Code of Federal Regulations. For this application, the reference system is the Combustion Engineering Standardized Nuclear Steam Supply System known as Standard Reference System-80. This design has been reviewed by the ACRS and was discussed in its report of September 17, 1975, "Combustion Engineering Standard Safety Analysis Report - CESSAR-80."

The reactor containment scheme for the Yellow Creek plant consists of a spherical steel vessel and a reinforced concrete shield building, generally similar to those for the Cherokee and Perkins plants.

An acceleration of 0.25g for rock-supported structures and 0.30g for soil-supported structures has been specified for the safe shutdown earthquake selected for the Yellow Creek Plant. The applicant has used a probabilistic treatment to choose an operating basis earthquake with which are associated ground level accelerations of 0.08g for rock supported structures and 0.10g for those supported on soil. The Committee considers these values acceptable.

The NRC Staff has identified a number of safety items which will require resolution before issuance of a construction permit. These matters should be resolved in a manner satisfactory to the NRC Staff.

With regard to generic problems cited in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977, items considered relevant to the Yellow Creek Plant are: II-1, 2, 3, 4, 5B, 6, 7, 9, 10; IIA-2, 3, 4; IIB-1, 2; IIC-1, 2, 3A, 3B, 4, 5, 6; IID-1, 2; IIE-1. These problems should be dealt with by the Staff and Applicant as solutions are found.

The Advisory Committee on Reactor Safeguards believes that if due consideration is given to the foregoing, the Yellow Creek Nuclear Plant, Units 1 and 2 can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

References

1. Yellow Creek Nuclear Plant, Units 1 and 2 Preliminary Safety Analysis Report, Volumes 1 through 6 and Amendments 1 through 10.
2. Safety Evaluation Report related to construction of Yellow Creek Nuclear Plant, Units 1 and 2, NUREG-0347, December 1, 1977, Docket Nos. STN 50-566 and STN 50-567.
3. Letter from J. E. Gilleland, TVA to O. D. Parr, NRR, on safety grade instrumentation to detect the loss of cooling water to the reactor coolant pumps, dated September 14, 1977.
4. Letter from J. E. Gilleland, TVA to D. B. Vassallo, NRR, on revised operating basis earthquake, dated October 14, 1977.
5. Letter from J. E. Gilleland, TVA, to O. D. Parr, NRR, on plant monitoring system, dated November 7, 1977.

6. Letter from J. E. Gilleland, TVA, to D. B. Vassallo, NRR on outstanding issues, dated November 14, 1977.
7. Letter from J. E. Gilleland, TVA, to O. D. Parr, NRR, on outstanding issues, dated November 23, 1977.
8. Letter from G. R. Lanning, Jr., to the Nuclear Regulatory Commission Advisory Committee [sic], on geology and emergency plans, dated December 16, 1977.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

May 12, 1966

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

Subject: REPORT ON ZERO POWER PLUTONIUM REACTOR

Dear Dr. Seaborg:

At its seventy-third meeting on May 5-7, 1966, the Advisory Committee on Reactor Safeguards considered the proposal by Argonne National Laboratory (ANL) to construct the Zero Power Plutonium Reactor (ZPPR) at the National Reactor Testing Station in Idaho. The Committee had the benefit of Subcommittee meetings on July 7, 1965 and April 23, 1966; of discussion with representatives of ANL and the AEC Staff; and of the documents listed below.

The proposed ZPPR utilizes a split-table critical machine for constructing large fast critical assemblies. It is capable of handling critical assemblies of about 6000 liters containing about 3000 kg of plutonium. The design and proposed method of operation of ZPPR are similar to the ZPR-III and ZPR-VI critical facilities which have been designed and operated by ANL. The distinguishing features of ZPPR are the larger size, the addition of forced-air cooling of the assembly, provisions for safe handling of large quantities of plutonium, and the uniqueness of the containment.

The reactor is housed in a concrete-walled cell covered with a gravel-sand roof 15 to 20 feet thick to serve as a plutonium filter in the unlikely event of an accident. Further protection from possible venting is afforded by an absolute filter housing over the gravel-sand roof.

ANL has agreed to develop a calculational procedure to estimate the cell overpressure resulting from a design-basis nuclear excursion and to operate only assemblies for which the predicted cell overpressure is less than 12.5 psi, the levitation pressure of the gravel-sand filter. The details of the calculational procedures should be resolved with the Regulatory Staff before operation of the facility is begun.

The ACRS believes that the ZPPR facility can be constructed at the site proposed with reasonable assurance that it can be operated without undue hazard to the health and safety of the public.

Dr. David Okrent and Dr. Harry O. Monson did not participate in the review of this project.

Sincerely yours,

/s/

Nunzio J. Palladino
Acting Chairman

References:

1. ZPPR, Argonne National Laboratory, Zero Power Plutonium Reactor, Preliminary Safety Analysis Report, approval copy, revised January, 1965.

2. "Information for Safety Evaluation of ZPPR", Argonne National Laboratory, dated June 4, 1965.

3. ZPPR, Argonne National Laboratory, Zero Power Plutonium Reactor, Supplement to Preliminary Safety Analysis Report, February 1966.

4. "Logging, Coring & Analysis of the Gravel in the Gravel Gertie Test Cell", Eberline & Associates, Inc., December 1965.

5. "Answers to Questions in the Letter from K. A. Dunbar to A. V. Crewe, March 21, 1966", undated, received April 7, 1966.

6. "More Detailed Discussion of the Excursion Calculations", undated, received April 21, 1966.

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

September 17, 1971

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON WILLIAM H. ZIMMER NUCLEAR POWER STATION, UNIT 1

Dear Dr. Schlesinger:

At its 137th meeting, September 9-11, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application from the Cincinnati Gas and Electric Company, the Columbus and Southern Ohio Electric Company, and the Dayton Power and Light Company for a permit to construct the William H. Zimmer Nuclear Power Station, Unit 1. The Cincinnati Gas and Electric Company is responsible for the design, construction, and operation of the plant and is authorized to act as sole agent during construction and for licensing negotiations. The project was considered at Subcommittee meetings on August 27, 1971, at the plant site, and on September 1 and September 8, 1971, in Washington, D. C. During its review the Committee had the benefit of discussions with representatives and consultants of the applicants, Sargent and Lundy, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The Zimmer Station will be located in Ohio on a 635-acre site on the Ohio River approximately 24 miles southeast of Cincinnati and one-half mile north of Moscow, Ohio. The population of Moscow is estimated by the applicants to be 348. The nearest population center is Covington, Kentucky which is located 20 miles northwest of the site and has a population of 60,000. The low population zone radius is 3.0 miles within which the 1960 population was less than 1,900 and the projected 1985 population less than 2,800. The projected 1985 population within 10 miles of the site is 30,100. The exclusion zone has a minimum radius of 1,250 feet, is bounded on the north by U. S. Route 52, and includes a small manufacturing plant located on the periphery. Provisions have been made to evacuate the employees of this plant in the unlikely event of an accident.

The Zimmer Station will utilize a General Electric boiling water reactor to be operated at a power level of 2436 MWt. It is the first reactor of the GE 1969 product line reviewed by the Committee. Waste heat is rejected to the atmosphere by a natural-draft cooling tower.

The primary containment is of the over-under pressure suppression type similar to those of the previously reviewed Limerick and Shoreham units. The drywell is a steel-lined prestressed concrete truncated cone; the pressure suppression chamber is a cylinder of similar construction. The drywell and pressure suppression chamber are separated by a reinforced concrete deck penetrated by 88 vent pipes. The reactor building is constructed of reinforced concrete up to the refueling floor and of structural steel and paneling at higher levels. The design is intended to limit inleakage to 100% of the building volume per day at a pressure of 1/4 inch of water during operation of the standby gas treatment system. This system, which includes provisions for circulating air throughout the reactor building, exhausts through redundant sets of double high efficiency particulate air filters and deep-bed activated carbon sorbers.

The emergency core cooling system of the GE 1969 product line incorporates several changes. The high pressure injection system has been modified to inject water through a sparger directly over the top of the core, rather than into the downcomer region via the feedwater line. Also, an electric motor drive instead of a steam turbine drive is used for the pump. This system now also serves as one of the two core spray systems. The low pressure coolant injection system has been modified to inject water from the suppression pool directly into the core region through three separate lines, each of which is supplied water by a separate pump. The maximum diameter of the reactor recirculation piping has been reduced from 28 to 20 inches.

The applicants have proposed to design the main steam lines and turbine stop and bypass valves to requirements which are substantially similar to AEC quality assurance Classification Group B. The Committee believes that the main steam lines should be designed to retain their integrity during a design basis earthquake. The applicants propose to install a sealing system, designed as an engineered safety feature, in connection with the main steam line isolation valves to minimize leakage. These matters should be resolved in a manner satisfactory to the Regulatory Staff prior to completion of construction of the station.

The applicants have studied design features to make tolerable the consequences of failure to scram during anticipated transients, and have concluded that automatic tripping of the recirculation pumps and injection of boron could provide a suitable backup to the control rod system for this type of event. The Committee believes that this recirculation pump trip represents a substantial improvement and should be provided for the Zimmer reactor. However, further evaluation of the sufficiency of this approach and the specific means of implementing the proposed pump trip should be made. This matter should be resolved in a manner satisfactory to the Regulatory Staff and the ACRS during construction of the reactor.

The radioactive waste disposal systems process high and low conductivity liquid wastes by demineralizers or evaporators and the decontaminated effluent is recycled to the condensate storage tank for reuse. Chemical and detergent wastes normally are to be processed through evaporators and, if necessary, further processed by demineralizers before discharge. The gaseous waste treatment system includes a high temperature catalytic recombiner followed by a 30-minute holdup system. The applicants will provide an additional holdup system which results in the substantial reduction of all isotopes except long-lived krypton. The applicants have stated that both the liquid and gaseous waste handling systems will be used to the fullest extent and will limit releases of radioactivity or exposures to man to values less than those specified in the proposed 10 CFR 50, Appendix I. An environmental monitoring program has been established, and the applicants have stated that it will permit the calculation of radiation exposures to man from records of radioactivity released from the plant.

The applicants have stated a system will be provided to control the concentration of hydrogen in the primary containment that might follow in the unlikely event of a loss-of-coolant accident. The Committee believes that the containment should be inerted and that the hydrogen control system should be designed to maintain the hydrogen concentration within acceptable limits using the assumptions listed in the AEC Safety Guide 7, Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident.

The applicants' pipe whip criteria consider both longitudinal and circumferential pipe breaks and provide for the installation of piping restraints as required to prevent damage to essential reactor coolant systems and equipment or to the containment.

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 Zimmer Station.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items, the William H. Zimmer Nuclear Power Station, Unit 1 can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H Bush

Spencer H. Bush
Chairman

References

1. Cincinnati Gas and Electric Company, Columbus and Southern Ohio Electric Company, and The Dayton Power and Light Company, License Application and Preliminary Safety Analysis Report (Volumes 1 through 5) for the William H. Zimmer Nuclear Power Station

2. Amendments 1 through 7 and 9 through 19 to the License Application for the William H. Zimmer Nuclear Power Station

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 13, 1979

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON WILLIAM H. ZIMMER NUCLEAR POWER STATION, UNIT 1

Dear Dr. Hendrie:

During its 227th meeting, March 8-10, 1979, the Advisory Committee on Reactor Safeguards completed its review of the application of the Cincinnati Gas and Electric Company (CG&E), the Columbus and Southern Ohio Electric Company, and the Dayton Power and Light Company (hereinafter referred to collectively as the Applicants) for authorization to operate the William H. Zimmer Nuclear Power Station, Unit 1. CG&E will be responsible for operating the plant. A tour of the facility was made by members of the Subcommittee on November 16, 1978 and the application was considered at Subcommittee meetings on November 17, 1978 and February 27, 1979. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, the General Electric Company, Sargent and Lundy Company, Kaiser Engineers Incorporated and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for this plant on September 17, 1971.

The Zimmer Nuclear Power Station is located in Ohio on the Ohio River approximately 24 miles southeast of Cincinnati and one-half mile north of Moscow, Ohio. The plant will utilize a 2436 MWt BWR/5 boiling water reactor which is similar to the BWR/4 used in the Edwin I. Hatch Nuclear Plant, Unit No. 2. A principal difference is the use of recirculation flow control valves to regulate power rather than pump speed control which has been used on plants of the BWR/4 type.

The Zimmer Nuclear Power Station has a Mark II pressure suppression containment and is designated as one of the lead plants for this type containment. The NRC Staff has reviewed the generic aspects of the Mark II containment system and has reported its findings in NUREG-0487. The generic aspects of Mark II load evaluation and acceptance criteria were considered at Subcommittee meetings on July 7-8, 1977, November 30, 1977, May 23, 1978, and November 28-30, 1978. The Committee believes that the acceptance criteria are suitable for the lead Mark II plants.

The Applicants have taken exception to some of the acceptance criteria developed by the NRC Staff. The Staff and the Applicants are continuing to work together to resolve this matter. The Committee wishes to be kept informed.

The Mark II Owners Group and the NRC Staff are continuing to develop information relating to the method of combining loads on the containment structure. However, the Applicants have indicated that they will accept the NRC Staff's current, perhaps overly conservative, methodology, to expedite the licensing action. The Committee considers this acceptable.

The NRC Staff has determined that the present Emergency Core Cooling System analysis contains adequate margins for assessing the performance of the Zimmer Plant. It should be noted that recent tests in the Two Loop Test Apparatus (TLTA) have produced new data on the rate of vaporization of emergency core cooling water. The NRC Staff believes that further analysis of the TLTA test results may require changes in the General Electric model for calculation of this vaporization rate in order to reflect more accurately the observed physical phenomena. The Committee wishes to be kept informed.

In view of the important role of the Operational Review Committee in providing continuing reviews, and in updating and implementing safety measures, the ACRS recommends that the Operational Review Committee include additional experienced personnel from outside the corporate structure as voting members for the first few years of operation.

With regard to the generic items cited in the Committee's report, "Status of Generic Items Relating to Light Water Reactors: Report No. 6," dated November 15, 1977, those items considered relevant to Zimmer are: II-4, 5b, 6, 7, 8, 10; IIA-4; IIB-4; IIC-1, 3A, 3B, 5; IID-2. These items should be dealt with by the NRC Staff and the Applicants as solutions are found.

The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, the William H. Zimmer Nuclear Power Station, Unit 1 can be operated without undue risk to the health and safety of the public.

Sincerely,

Max W. Carbon
Chairman

References:

1. Cincinnati Gas and Electric Company, "Final Safety Analysis Report, William H. Zimmer Nuclear Power Station, Unit 1," with Amendments 23 through 82.

2. U. S. Nuclear Regulatory Commission (USNRC), "Safety Evaluation Report Related to the Operation of William H. Zimmer Nuclear Power Station, Unit 1, Docket No. 50-358," USNRC Report NUREG-0528, dated January 31, 1979.

3. U. S. Nuclear Regulatory Commission, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," USNRC Report NUREG-0487, dated October, 1978.

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

July 24, 1968

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

Subject: REPORT ON ZION STATION UNITS 1 AND 2

Dear Dr. Seaborg:

At its ninety-ninth meeting, July 11-13 and 21, 1968, the Advisory Committee on Reactor Safeguards completed its review of the application by the Commonwealth Edison Company for authorization to construct nuclear generating Units 1 and 2 at its Zion Station in Zion, Illinois. This application was considered also at the ninety-sixth, ninety-seventh, and ninety-eighth meetings, on April 4-6, 1968, May 9-11, 1968, and June 5-8, 1968, respectively. Members of the ACRS visited the site on June 6, 1967, and Subcommittee meetings were held at the Argonne National Laboratory on March 21, 1968, and in Washington, D. C., on April 17 and May 29, 1968. During its review, the Committee had the benefit of discussions with representatives of the Commonwealth Edison Company and their consultants, with the Westinghouse Electric Corporation, and with the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents referenced in this report.

The Zion Station is located on the west shore of Lake Michigan in Zion, Illinois. Zion has a population of 14,000, and Waukegan, Illinois, with a population of 65,000, has its nearest boundary 3.6 miles from the site. The site comprises 250 acres.

Each of the two 3250 MWt pressurized water reactors is similar in design to the Diablo Canyon reactor. The containment for each reactor is a prestressed concrete vessel similar to previously approved designs (e.g., Turkey Point, Palisades, and Point Beach). The reactors to be built at the Zion Station are the largest reactors reviewed to date for construction in a region of relatively high population density.

The applicant has considered the possibility of reactor vessel failure as a result of thermal shock caused by emergency core cooling system action in the unlikely event of a loss-of-coolant accident during the later portions of vessel life. He has conducted engineering studies which have established the feasibility of a cavity flooding system that could flood to a level above the top of the core and thereby provide additional protection in the event of such failure. He stated that this system would be installed at a future time if studies now under way indicated that vessel failure as a result of thermal shock could occur. The present design provides for reactor cavity flooding to about two feet above the bottom of the core. Additionally, the reactor cavity has been designed, as at Indian Point 2, to limit vessel movement in the highly unlikely event of failure of the reactor vessel by longitudinal splitting during operation. The Committee continues to favor such protection for large reactors in regions of relatively high population density.

The applicant has proposed using signals from the protection system for control and override purposes. The Committee reiterates its belief that control and protection instrumentation should be as nearly independent of common failure modes as possible, so that the protection will not be impaired by the same fault that initiates a transient requiring protection. The applicant and the AEC Regulatory Staff should review the proposed design 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. In cases where hypothesized control or override failure could lead to the need for action by interconnected protection instrumentation, separate protection instrumentation channels should be provided or some other design approach be used to provide equivalent safety.

The applicant described programs for development and utilization of instrumentation for prompt detection of gross fuel failure and for detection of primary coolant leakage.

The Committee continues to emphasize the need for quality in the manufacture, storage, and installation of the reactor and primary system components. The applicant described the quality assurance program that he and his contractors intend to carry out for this purpose. In this connection, the applicant described the testing program for engineered safety features, including a full flow test of the emergency core cooling system delivering water to the reactor vessel. The Committee recommends that the applicant give further consideration to testing the containment spray systems with full flow to the spray nozzles at least once at an appropriate time during construction.

The applicant described his emergency plans for the Zion Station, which are based partly on experience acquired in developing plans for the Dresden Nuclear Station.

The Committee continues to call attention to matters that warrant careful consideration with regard to reactors of high power density and other matters of significance for all large, water-cooled power reactors. In addition, attention is called to safety-related questions specifically identified for the Diablo Canyon reactor class. The applicant reviewed his research and development program designed to resolve safety-related problems and stated that he expects resolution of these problems before operation of the reactors. System modifications or restrictions on operation may be appropriate if the startup program, additional operating experience, or the research and development should fail to confirm adequately the proposed safety margins.

The Committee believes that the items mentioned can be resolved during construction and that, if due consideration is given to the foregoing, the nuclear Units 1 and 2 proposed for the Zion Station can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Additional remarks by Dr. David Okrent are appended. The matters discussed by him were considered by the Committee during its meetings. The Committee believes that the status of these matters, as they pertain to the Zion units, is satisfactory.

Sincerely yours,

/s/
Carroll W. Zabel
Chairman

Attachments:
1. References
2. Additional Remarks of
   Member David Okrent

References - Zion Station

1. Letter from Commonwealth Edison Company, dated July 12, 1967; Application for Construction Permit and Operating License; Volumes I and II of Preliminary Safety Analysis Report, Zion Station

2. Letter from Commonwealth Edison Company, dated August 15, 1967; Amendment No. 1 to Application

3. Letter from Commonwealth Edison Company, dated November 28, 1967; Amendment No. 2 to Application; Volumes III and IV of PSAR

4. Letter from Commonwealth Edison Company, dated December 20, 1967; Amendment No. 3 to Application.

5. Letter from Commonwealth Edison Company, dated January 29, 1968; Amendment No. 4 to Application; Volume V of PSAR

6. Letter from Commonwealth Edison Company, dated March 1, 1968; Amendment No. 5 to Application

7. Letter from Commonwealth Edison Company, dated April 4, 1968; Amendment No. 6 to Application

8. Letter from Commonwealth Edison Company, dated April 17, 1968; Amendment No. 7 to Application

9. Letter from Commonwealth Edison Company, dated May 3, 1968; Amendment No. 8 to Application

10. Letter from Commonwealth Edison Company, dated June 6, 1968; Amendment No. 9 to Application

11. Letter from Commonwealth Edison Company, dated June 27, 1968; Amendment No. 10 to Application

July 24, 1968

Additional Remarks of Member David Okrent

While I am not objecting to a construction permit for the Zion reactors, I am suggesting that in connection with its issuance there are certain matters that warrant consideration and resolution before construction is completed.

In its report of November 24, 1965, on reactor pressure vessels, the ACRS recommended that further attention be given "to methods and details of stress analysis, to the development and implementation of improved methods of inspection during fabrication and vessel service life, and to the improvement of means for evaluating the factors that may affect the nil ductility transition temperature and the propagation of flaws during vessel life". The ACRS also recommended that "means be developed to ameliorate the consequences of a major pressure vessel rupture" and suggested as a possible approach the provision of "adequate core cooling or flooding which will function reliably in spite of vessel movement and rupture". The ACRS went on to state that "the orderly growth of the industry, with concomitant increase in number, size, power level and proximity of nuclear power reactors to large population centers will in the future make desirable, even prudent, incorporating in many reactors the design approaches whose development is recommended above".

Since November, 1965, considerable additional emphasis has been placed by the nuclear industry and the AEC on providing still greater quality in pressure vessel fabrication. An important research program is under way by the AEC to provide a better understanding of the behavior of thick-walled, steel pressure vessels. Our reactor vessel operating experience, although limited, has been good.

On the other hand, some questions have arisen in connection with specific design and fabrication aspects of pressure vessels. Resolution is required concerning the potentially adverse effect on vessel integrity of thermal shock arising from operation of the emergency core cooling system in the unlikely event of a sizable primary system leak, and questions exist with regard to the behavior of highly irradiated, thick section, pressure vessel walls in the presence of flaws and at significant vessel pressure.

Increasing attention has been given to the development of in-service inspection techniques and to the provision during reactor design of the necessary accessibility for thorough in-service inspection. Both industry and AEC regulatory groups are currently working on access and periodic inspection requirements for water reactor primary systems, including the pressure vessel. Means of remote, volumetric inspection of pressure vessels in service are under development by the nuclear industry, as are other flaw detection devices.

I believe that, with regard to water reactors of current design to be sited in less populated areas, the efforts under way to provide improved vessel quality and adequate, thorough, in-service inspection, in conjunction with satisfactory resolution of the thermal shock matter, with acceptable results from safety research programs on irradiation effects, subcritical flaw growth, etc., in thick-walled vessels, and with deliberate conservatism and thoroughness in pressure vessel design and fabrication practice, should provide an acceptable basis for dealing with safety questions arising from pressure vessel integrity.

The Zion site has a relatively large surrounding population density. For large water reactors proposed for such a site, I believe that, in addition to the above steps, careful consideration should be given in the initial engineering design to provision of the capability to cope with a loss in primary system integrity arising from a leak or split in the pressure vessel wall. Such provisions should include necessary steps to maintain the containment integrity. It appears likely that means to maintain the general core geometry and to provide the necessary emergency cooling water would be required. It is important that such provisions, if they are to be implemented, provide a significant degree of additional protection, albeit not perfect or complete, and that they should not, of themselves, provide a means of detracting from the integrity of the pressure vessel. It is to be expected that the development of means to deal with a loss of primary system integrity arising with the pressure vessel will be a process of evolution. Careful and thorough study should lead to a definition of those potential areas of degradation in pressure vessel integrity for which protective meausres are practical and appropriate. In view of the very low probability of a pressure vessel rupture, the design of these protective features could be based on fairly realistic rather than highly conservative analyses. A reliance on off-site power sources in connection with these protective features may be acceptable, if the capability of the external power system to withstand sudden, unexpected shutdown of the reactor can be clearly demonstrated and periodically verified.

For the Zion reactors, where the engineering design is now well along and could not be readily modified without major delays and significant additional costs, I believe that the applicant should study what provisions could be made, within the limitations of the existing design, to provide further protection against a loss in primary system integrity arising from a limited size leak or split in the pressure vessel wall, particularly in the region that receives the highest neutron irradiation dose during reactor lifetime.

I also believe that, at this time, additional conservatism in design, construction and operation is desirable for the Zion reactors, as compared to similar reactors at less populated sites. To be most effective, this additional conservatism should be part of the applicant's basic philosophic approach. The following aspects might be included:

1. Both for the primary coolant system and for other features of vital importance to the protection of the health and safety of the public, additional conservatism in design and further steps to assure quality of construction and continued integrity and reliability during operation should be used, where practical.

2. Safety issues remaining to be resolved between the start of construction and the initiation of operation at power should be minimized; well-defined research and development programs, adequate to clearly resolve the issues in timely fashion, should be committed. Where questions remain to be resolved, and where complete resolution may not be accomplished by the time of reactor operation, the reactor design should proceed on the basis of incorporating the appropriate safety provisions.

3. Since it is highly unlikely that a clear demonstration of the efficacy of the several engineered safety systems and other protective features under representative accident conditions will occur as a consequence of actual accident experience in the reasonably near future, it is desirable that extra margins be provided in the design of the usual engineered safety systems, particularly those for which some degree of uncertainty or some problem requiring resolution remains.

4. Additional, detailed examination of potential accidents leading to moderate releases of radioactivity to the environment (small accidents) should be made, and steps be taken to reduce still further the probability of occurrence of such accidents.

In my opinion, additional steps such as these, which are taken to protect the health and safety of the public with regard to reactors to be sited close to population centers, need not necessarily be applied to reactors in less populated sites.

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

August 17, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON ZION STATION UNITS 1 AND 2

Dear Dr. Schlesinger:

At its 148th Meeting, August 10-12, 1972, the Advisory Committee on Reactor Safeguards completed its review of the application of Commonwealth Edison Company for authorization to operate Zion Station Units 1 and 2 at power levels up to 3250 MW(t). This project had been considered previously at the Committee's 147th Meeting, July 13-15, 1972, and at Subcommittee meetings at the site on June 1, 1972, and in Washington, D.C. on July 6, July 12, and August 9, 1972. Unit 2 is expected to be ready for operation in slightly less than one year after Unit 1. During its review, the Committee had the benefit of discussions with representatives of Commonwealth Edison Company, Westinghouse Electric Corporation, Sargent and Lundy, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported to the Commission on the construction of these units in its report of July 24, 1968.

The Waukegan Memorial Airport is about 3½ miles from the plant, and activity has increased since the Construction Permit was issued. There are plans for enlarging the airport for greater usage and larger aircraft. The applicant should, on a continuing basis, appraise the potential effect on his plant of the changing airport operations, including the probabilities of crashes by the various categories of aircraft, the vulnerability of the plant structures, and measures that might be taken to minimize the effect of impact on critical structures. The Regulatory Staff is currently discussing with the applicant measures that can be taken to minimize the effect of fires arising from spillage of aircraft fuel in the event of an airplane crash. The Committee believes that the applicant should take measures to limit the consequences of such fuel spillage and believes that this matter can be resolved between the applicant and the Regulatory Staff prior to commencement of operation. In the event of any major change in the character of the airport usage that may affect the safety of the plant, the Committee recommends that the Regulatory Staff review the situation.

The Committee's report of July 24, 1968, called attention to specific matters of ACRS concern, including the need for adequate reliability of the protection system and adequate independence of protection and control systems; the need for prompt detection of gross fuel failure and primary coolant leakage; the importance of quality assurance and of testing engineered safety features; and other matters identified as being significant for all large water reactors. Most of these items are generic, not unique to Zion. During the four years that have elapsed since the Zion construction permit review, much progress has been made in resolving such problems. AEC Regulations and Safety Guides and industry codes and standards have formalized positions on many items of immediate concern, and additional work is in progress on these problems. The Committee recommends that as the results of additional research, analyses, and design studies become available, they should be used by the applicant for evaluation and possible improvement of the existing Emergency Core Cooling System. The Committee wishes to be kept informed.

The applicant should assure himself that instrumentation for determining the course of postulated accidents is on hand at the station and that appropriate calibration methods and calculated bases for interpreting instrument responses are available.

The Committee recommends that the Regulatory Staff confirm the adequacy of the applicant's analysis of peak overall accident pressures during postulated loss-of-coolant accidents, as well as the response of compartment walls within the containment to dynamic forces during such events.

In its report of July 24, 1968, the Committee called attention to the possibility of reactor vessel failure, during the later part of the reactor life, as a result of thermal shock caused by emergency core cooling system action in the unlikely event of a loss-of-coolant accident. This possibility could materialize only after many years of vessel irradiation, and the Heavy Section Steel Technology Program should yield data that will show whether the possibility is real. The applicant has made provision, as suggested in the Committee letter of July 24, 1968, for installing a reactor cavity flooding system if this should prove desirable. The Committee believes it is satisfactory to defer a decision on installation of this system.

In the unlikely event of a loss-of-coolant accident, hydrogen buildup in the containment would be controlled on an interim basis by purging through a filter system. The applicant is committed to add a hydrogen recombining system, as recommended by Safety Guide No. 7, within one year after initial criticality. The Committee finds this satisfactory.

The Committee reiterates its previous comments concerning the need to study further means of preventing common mode failures from negating reactor scram action, and design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee believes it desirable to expedite these studies and to implement in timely fashion such design modifications as are found to improve significantly the safety of the plant in this regard. The Committee wishes to be kept informed of the resolution of this matter.

Defects have developed in unpressurized fuel in some plants. The Zion fuel is pre-pressurized and there is reason to expect improved performance with such fuel. However, the phenomena are not fully understood, and some effects on fuel performance are anticipated. The applicant will submit further information with regard to this matter and will propose acceptable upper limits for linear power and procedures for adequate surveillance of core power distribution and fuel condition. The Regulatory Staff and the ACRS should review these proposals prior to operation at appreciable power.

Because of limited experience with very large high power density reactors such as Zion, and residual uncertainty relating to other matters mentioned, the Committee believes it would be prudent to restrict initial operation to somewhat below full power. The Committee recommends operation at power levels not exceeding 2760 MW(t) (85 percent of full power) until the first refueling of Zion Unit 1, at which time operating experience will have been gained and the condition of the fuel can be observed visually. The Regulatory Staff and the ACRS should review the matter prior to operation at higher power.

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 preoperational testing, there is reasonable assurance that the Zion Station Units 1 and 2 can be operated initially at power levels up to 2760 MW(t) without undue risk to the health and safety of the public. Subsequent to the first refueling of Unit 1 and satisfactory operation up to that time, and subject to review by the Regulatory Staff and the ACRS, the Committee believes there will be reasonable assurance that the units can be operated at power levels up to 3250 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

References attached

References - Zion

1. Commonwealth Edison letter dated 12/1/70 (Amendment 12) transmitting Final Safety Analysis Report (FSAR) for the Zion Station Units 1 and 2 and the Technical Specifications

2. Amendments 13-21 to the Application for Construction Permits and Operating Licenses

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

MAY 17 1973

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

Subject: REPORT ON ZION STATION UNITS 1 AND 2

Dear Dr. Ray:

In our August 17, 1972 report on Zion Station Units 1 and 2, the Committee recommended that the Regulatory Staff and the ACRS review, prior to operation of the plant at appreciable power, the applicant's proposals for operating power limits as related to the possible densification of fuel in the Zion reactors. In addition, the Committee recommended that operation not exceed 2760 Mw(t) (85 per cent of full power) until the first refueling of Unit 1 and that the Regulatory Staff and the ACRS should review the matter prior to operation at higher power. The applicant has addressed the densification effects and has proposed an extended startup program. These matters were considered at a Subcommittee meeting held in Washington, D. C., April 21, 1973, and at the 157th ACRS meeting, May 10-12, 1973. During its review, the Committee had the benefit of discussions with representatives of Commonwealth Edison Company, Westinghouse Electric Corporation, Sargent and Lundy, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The extended startup program to acquire operating data and experience will be initiated following satisfactory completion of preoperational testing. The ACRS concurs with the Regulatory Staff and the applicant on the general features of the power ascension program and the specific attention being given to in-core monitoring of the fuel rods. However, the Committee reaffirms its recommendation that neither Unit 1 nor Unit 2 be operated at power levels greater than 85 per cent of full power until after the first refueling of Unit 1. In addition, for use in establishing suitable limits on power for subsequent operation, acquisition of data relating to fuel behavior in both units should be continued during the entire period of operation prior to first refueling of Unit 1 and the condition of the fuel in Unit 1 should be determined at the time of the refueling outage. The Regulatory Staff and the ACRS should review any proposals for changes in maximum power levels during subsequent operation, taking into account also the progress made in the analytical and experimental developments for emergency core cooling systems and the implementation of the resolutions for generic items mentioned in previous ACRS reports.

Dr. Edward A. Mason did not participate in the Committee's review of this project.

Additional comments by Dr. W. R. Stratton and Dr. W. Kerr are presented below.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

Additional Comment by Dr. W. R. Stratton and Dr. W. Kerr

"We disagree with the position of the Committee that power levels above 85% should not be considered until after the first refueling of Unit 1. We believe that it would be appropriate for the Regulatory Staff and the ACRS to consider the possibility of operation at higher power levels after a review of the operating experience and data acquired during the augmented and extended startup program."

References

1. Safety Evaluation of the Zion Nuclear Power Station Units 1 & 2, USAEC/DL, October 6, 1972.

2. Amendments 22-27 to the Zion Station FSAR.

3. Directorate of Licensing letter to ACRS Chairman, dated April 10, 1973 and attachments.

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

December 9, 1974

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

Subject: REPORT ON ZION STATION, UNITS 1 AND 2

Dear Dr. Ray:

At its 176th meeting, December 5-7, 1974, the Advisory Committee on Reactor Safeguards completed its review of the request of Commonwealth Edison Company for authorization to increase the power level of Zion Station Units 1 and 2 from the current maximum power of 2760 MW(t) to the full power level of 3250 MW(t) in the immediate future. The matter had been previously considered at a Subcommittee meeting on December 4, 1974. In its review, the Committee had the benefit of discussion with representatives of Commonwealth Edison and its consultants, the AEC Regulatory Staff, and the documents listed. The Committee reported previously on the operation of Zion Station Units 1 and 2 on August 17, 1972, and May 17, 1973.

In its previous reports, the Committee recommended that for the Zion Station, neither Unit 1 nor Unit 2 be operated at greater than 2760 MW(t) (85% of full power) until after the first refueling of Unit 1 and after further review by the Regulatory Staff and the ACRS of proposed power increases. However, the Regulatory Staff has requested an earlier review of a transition to higher power, based at least in part on a letter from the Federal Energy Administration, dealing with possible power shortages.

As of early December, 1974, Zion Unit 1 has operated about three months at 2760 MW(t); Unit 2 has recently reached the same level. In general, steady-state core and system performance measurements conform with prediction. More than the normal amount of power shape monitoring has been performed as part of an augmented startup program for Unit No. 1.

An evaluation of the compliance of these units with the requirements of 10 CFR 50.46 has not been completed; however, it is anticipated that total power peaking factors less than 2.3 would be required for operation at 3250 MW(t). An axial power distribution monitoring system (APDMS) is to be implemented on both units; however, experience with automatic operation of APDMS at Zion, including further evaluation of both reliability and accuracy aspects, remains to be obtained. Also, operation at 3250 MW(t) would make the average fuel linear heat generation rate for the Zion units higher than that for any other operating Westinghouse PWR.

A number of incidents having significance to the reliability of engineered safety features have occurred at the Zion Station, including some in recent months. The Applicant has recently made organizational and administrative changes in an effort to improve operational quality assurance and to minimize problems of a repetitive nature.

Various generic items, including monitoring for vibration or loose parts, anticipated transients without scram, instrumentation for determining the course of postulated accidents, and the possibility of reactor coolant pump-flywheel overspeed in the unlikely event of a downstream pipe break, remain to be resolved for Zion Units No. 1 and 2.

In view of the above, unless there exists an overriding national need for additional power from these units, the ACRS reaffirms its recommendation that neither Unit 1 nor 2 be operated at power levels greater than 85% of full power until after the first refueling of Unit 1. The Regulatory Staff and the ACRS should review any proposals for changes in maximum power levels during subsequent operation, taking into account the further operational experience, progress made in understanding of the performance and potential improvement of emergency core cooling systems, and the implementation of resolution of generic items.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

Attachments:
References

References:

1. Safety Evaluation by the Directorate of Licensing, U. S. Atomic Energy Commission (DL), in the Matter of Commonwealth Edison Company (CECO), "Startup Program Results Up to 85% of Rated Power Level for the Zion Station Unit 1," dated December 3, 1974.

2. Letter, dated September 3, 1974, CECO to DL, concerning postulated accident analysis and proposed changes to Technical Specifications.

3. Letter, dated September 6, 1974, CECO to DL, concerning postulated accident analysis and proposed changes to Technical Specifications.

4. Letter, dated November 22, 1974, CECO to DL, concerning postulated accident analysis and proposed changes to Technical Specifications.

5. Letter, dated June 25, 1974, DL to CECO, concerning DL review and acceptance of Westinghouse topical report, WCAP-8218, "Fuel Densification, Experimental Results and Model for Reactor Application," dated Oct. 31, 1973.

6. Letter, dated November 5, 1974, DL to CECO, concerning DL review of Westinghouse topical report, WCAP-8377, "Revised Clad Flattening Model," dated July, 1974.

7. Westinghouse topical report, WCAP-8183, Revision 1, "Operational Experience with Westinghouse Cores," dated July, 1974.

8. "Westinghouse PWR Operating Experience," Handout at ACRS Meeting, November 14-16, 1974.

9. Westinghouse topical report, "Power Distribution Control and Load Following Procedures," dated September, 1974, WCAP-8385.

10. Letter, dated November 18, 1974, DL to CECO, concerning DL review and acceptance of topical report, WCAP-7912-L, "Power Peaking Factors."

11. Report, dated June, 1974, by the U. S. Atomic Energy Commission, Office of Operations Evaluations, "Diesel Generator Operating Experience at Nuclear Power Plants," OOE-ES-002.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUNE 9, 1976

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20005

SUBJECT: REPORT ON ZION STATION UNITS 1 AND 2

Dear Mr. Rowden:

During its 194th meeting, June 3-5, 1976, the Advisory Committee on Reactor Safeguards completed its review of the proposal to increase the maximum reactor power of the Zion Station Units 1 and 2 from 2760 MWt (85% of full power) to the rated power of 3250 MWt. The Committee had previously discussed operation of the Zion Station, in its reports of August 17, 1972, May 17, 1973 and December 9, 1974. A Subcommittee meeting on the current proposal was held in Kenosha, Wisconsin on May 27, 1976, subsequent to a visit to the site on May 26, 1976. During its review, the Committee had the benefit of discussions with representatives of Commonwealth Edison Company, Westinghouse Electric Corporation, the Nuclear Regulatory Commission Staff and of the documents listed, as well as comments from members of the public.

In its previous letters relating to the Zion Station, the Committee listed several concerns which, in its opinion, mitigated against the full power operation of these large reactors at a site having a significantly larger than average population density. Three of these concerns, relating to fuel behavior, control of core power distribution, and reliability of diesel generator startup, have been resolved to the satisfaction of the NRC Staff, as reported in its Safety Evaluation Report dated May 20, 1976.

The ACRS believes that with the resolution of these three matters it is acceptable for the Zion Station reactors to be operated at full power. However, the Committee believes that other matters should be dealt with in a timely fashion if the Zion reactors are to continue to be operated at full power over the lifetime of the plant. The ACRS recommends that these matters, as set forth below, be addressed by the Applicant and the NRC Staff during the next year:

(1) A review of the entire Station for systems interaction that might lead to significant degradation of safety.

(2) A review of the Station with regard to differences from current criteria, and judgments concerning possible back-fitting requirements.

(3) The implementation of instrumentation to provide early information concerning the course of a full range of postulated serious accidents, and procedures for interpreting and relating this information to emergency plans.

(4) Installation of a loose-parts monitoring system as soon as practicable.

(5) Evaluation and prompt implementation of improvements in fire protection capability, as necessary.

(6) Timely implementation of modifications required in connection with the resolution of ATWS.

(7) Continued studies directed to enhancement of the capability and reliability of the Emergency Core Cooling Systems.

(8) Demonstration of the reliability of the diesel generators to operate with load for extended periods of time.

(9) Assessment of the safety significance of the large number of reportable events experienced at the station, and prompt implementation of significant improvements in operational quality assurance.

(10) Prompt implementation of improvements in industrial security as appropriate.

Other generic problems relating to large water reactors are discussed in the Committee's report dated April 16, 1976. As solutions to these problems are found, both the Applicant and the NRC Staff should give a high priority to their prompt implementation at the Zion Station.

The ACRS wishes to review the status of these matters by June of 1977.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, there is reasonable assurance that the Zion Station Units 1 and 2 can be operated at full power, 3250 MWt, without undue risk to the health and safety of the public. As noted above the Committee will again review the operation of the Zion Station in approximately one year.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

REFERENCES

1. Safety Evaluation Report on Zion Station, Units 1 and 2, dated May 20, 1976.
2. Supplement No. 1 to the Startup Test Report Issued November 1974 on Zion Nuclear Power Station, Unit 1, dated April 16, 1976.
3. Letter from E. Jenkins to ACRS concerning full power operation of the Zion Station, dated June 1976.
4. Letter from D. Comey to E.G. Case, concerning reliability of emergency diesel generators at Zion Stations, dated August 28, 1974.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 17, 1977

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON THE ZION STATION, UNITS 1 AND 2

Dear Mr. Rowden:

During its 206th meeting, June 9-10, 1977, the Advisory Committee on Reactor Safeguards continued its review of the operation of the Zion Station, Units 1 & 2. The Committee had previously discussed operation of the Zion Station in its reports of August 17, 1972, May 17, 1973, December 9, 1974, and June 9, 1976. A Subcommittee meeting was held in Kenosha, Wisconsin, on May 17, 1977, following a tour of the station by Committee members. During its review, the Committee had the benefit of discussions with representatives of the Commonwealth Edison Company (Licensee), Westinghouse Electric Corporation, and the Nuclear Regulatory Commission Staff. The Committee also had the benefit of the documents listed.

In its June 9, 1976 report, the Committee identified ten unresolved matters that should be dealt with in a timely fashion if the Zion reactors are to continue to be operated at full power over the lifetime of the plant. The ten matters identified are as follows:

(1) A review of the entire Station for systems interaction that might lead to significant degradation of safety.

(2) A review of the Station with regard to differences from current criteria, and judgments concerning possible backfitting requirements.

(3) The implementation of instrumentation to provide early information concerning the course of a full range of postulated serious accidents, and procedures for interpreting and relating this information to emergency plans.

(4) Installation of a loose-parts monitoring system as soon as practicable.

(5) Evaluation and prompt implementation of improvements in fire protection capability, as necessary.

(6) Timely implementation of modifications required in connection with the resolution of ATWS.

(7) Continued studies directed to enhancement of the capability and reliability of the Emergency Core Cooling Systems.

(8) Demonstration of the reliability of the diesel generators to operate with load for extended periods of time.

(9) Assessment of the safety significance of the large number of reportable events experienced at the station, and prompt implementation of significant improvements in operational quality assurance.

(10) Prompt implementation of improvements in industrial security as appropriate.

The current status of these items was the principal matter of this latest review of the Zion Station.

The Committee believes that little progress has been made toward resolution of item 1. Discussion with the Licensee and the NRC Staff suggests that this has been the result, at least in part, of a lack of understanding by the Licensee and the NRC Staff of just what is meant by "systems interaction." In this respect the Committee calls attention to its letter of November 8, 1974, to L. Manning Muntzing, Director of Regulation. The Committee recommends that, within nine months, the Licensee submit to the NRC Staff, as a minimum, the results of a study of systems interaction relating to the possibility that failure of safety and non-safety systems will interfere with the plant operators' ability to accomplish shutdown heat removal, together with a plan and schedule for studies of other system interactions of potential safety significance to the Zion Station.

With respect to item 2, for which little has been done, the Committee recommends that, during the next twelve months, the Licensee review the Zion Station for possibly significant differences from current criteria, and that the NRC Staff evaluate this review and report to the ACRS its conclusion concerning possible backfitting requirements.

The Committee wishes to review the status of items 1 and 2 within the next eighteen months.

Items 3, 5, 6, 7, and 10 are considered by both the ACRS and the NRC Staff to be unresolved matters generic to all operating light water reactors. During the past twelve months, some progress has been made toward the resolution of these generic items and in the planning for the application of appropriate solutions to the Zion Units. The Committee recommends that the NRC Staff and Licensee urgently seek means to expedite solutions to outstanding generic items and the implementation of solutions, when feasible, to the Zion Station.

The Licensee has made a commitment to install loose-parts monitoring systems on the two Zion Station reactors during the 1978 refueling outages. The NRC Staff considers item 4 resolved. The Committee concurs.

In response to item 8, one of the diesel generators was run continuously for seven days at a controlled power output equivalent to the ECCS load. Whereas the unit ran satisfactorily during this period, the significance of the test results in confirming the capability of the emergency power system to perform its intended functions was obscured when, on the seventh day, an operator error led to a large surge in the load and the destruction of the generator by fire. The generator failure was the result of an unanticipated interaction between the main electrical power generating system, the emergency power system, and the loads they were sharing. This unexpected result increases the urgency for a review of the entire station for interactions between electrical generation, distribution, consumption, and control systems that might lead to significant degradation of safety. The Committee recommends that this phase of the review (item 1) be given particular attention. The Committee wishes to be kept informed.

The Licensee has, in response to concerns expressed by the NRC Staff, rewritten his operating procedures, expanded employee training programs, and organized as part of the Quality Assurance program, two independent audit groups: one group to verify procedural compliance and to audit work in progress, and the other group to identify and resolve problems promptly. The NRC Staff has also issued amendments to the Zion license which revise the entire administrative control section of the Technical Specifications. The Committee concludes that these actions of the NRC Staff and Licensee are responsive to item 9 and encourages the Licensee to continue to seek further improvement in these areas.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, there is reasonable assurance that the Zion Station Units 1 and 2 can continue to operate at full power, 3250 MWt, without undue risk to the health and safety of the public.

Sincerely,

/s/ M. Bender

M. Bender
Chairman

References

1. Report to the ACRS by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission in the Matter of Commonwealth Edison Company Zion Station, Units 1 and 2, dated April 22, 1977.

2. Supplement to the Report by the Office of Nuclear Reactor Regulation, NRC, to the ACRS concerning Zion Station, Units 1 and 2, dated June 8, 1977.

3. Letter from Commonwealth Edison Co. to the Office of Nuclear Reactor Regulation, NRC concerning installation of a loose parts monitoring system, dated January 1, 1977.

4. Letter from Commonwealth Edison Co. to the Office of Nuclear Reactor Regulation, NRC, concerning information on gaseous releases from Zion Station, dated April 1, 1977.

5. Letter from Commonwealth Edison Co. to the Office of Nuclear Reactor Regulation, NRC, concerning a summary of the diesel generator test, dated April 6, 1977.

6. Letter from Commonwealth Edison Co. to the Office of Nuclear Reactor Regulation, NRC, concerning a report on fire protection, dated April 29, 1977.

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