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Volume 2
Project Reviews G-P

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

November 8, 1974

Honorable Dixy Lee Ray
Chairman
U.S. Atomic Energy Commission
1717 H Street, N.W.
Washington, D.C. 20545

Subject: CONCEPTUAL DESIGN FOR PROTOTYPE GAS-COOLED FAST BREEDER REACTOR (GCFBR)

Dear Dr. Ray:

At its 142nd meeting on February 3-5, 1972, its 173rd meeting on September 5-7, 1974, its 174th meeting on October 10-12, 1974, and a special meeting on October 31-November 2, 1974, the Advisory Committee on Reactor Safeguards reviewed a conceptual design and proposed design bases for a prototype 300 MW(e) Gas-Cooled Fast Breeder Reactor (GCFBR). Subcommittee meetings were held on July 21, 1971, in Denver, Colorado, December 1, 1971, in La Jolla, California, September 11-12, 1973, in La Jolla, California, and January 9, February 6, and August 6, 1974, in Washington, D.C. During its review the Committee had the benefit of discussions with representatives of General Atomic Company, the AEC Regulatory Staff and of the documents listed.

The purpose of this review was to acquaint the Committee with the current status of the conceptual design and proposed design bases and to enable it to identify those areas which the Committee believes require further technological development, or which it currently considers unacceptable. The information available, however, was not sufficient to permit the Committee to determine if all areas important to safety have been identified.

The reactor concept utilizes helium cooling of stainless steel clad oxide fuel elements whose design is similar in many respects to those used in liquid metal fast breeder reactors. The reactor core, three primary coolant loops and three auxiliary coolant loops are completely contained in a cylindrical prestressed concrete reactor vessel (PCRV). The core occupies the central cavity. The steam generators, primary helium circulators and the auxiliary coolant circulators and heat exchangers are located in cavities in the PCRV wall. A conventional low-leakage containment building, similar to those used for PWRs and proposed for HTGRs, is provided.

The Committee recognizes that the GCFBR has certain advantageous safety characteristics relative to other types of fast reactors. These include:

(1) The reactivity effect associated with the helium coolant is small;

(2) Potential chemical reactions between the primary coolant and the secondary steam are eliminated because helium is chemically inert;

(3) Maintenance access problems tend to be less severe because the helium coolant is subject to limited radioactivation.

Certain safety disadvantages unique to the GCFBR, as well as some safety problems common to all fast reactors, are discussed below.

A significant problem area, requiring substantial additional study, is the reliability of core cooling capability. Special emphasis needs to be given to partial or total loss of core cooling without depressurization and to a spectrum of loss-of-coolant accidents with various rates of depressurization. Sensitivity studies in these areas are necessary, including coolant compositions ranging from helium alone to helium plus various concentrations of hydrogen, water vapor and air. Because reliability of helium circulators is essential, problems such as common mode failures affecting the primary circulators, auxiliary circulators, or both, must be addressed more extensively. The reliability of valve operation in the primary circuit requires additional careful scrutiny. Further work is required on thermal and mechanical parameters influencing fuel damage within the spectrum of accidents which potentially could lead to some fuel melting to determine the impact of fuel damage on core cooling reliability.

Because the cooling efficiency during a depressurization accident is a function of the back pressure in the containment, various aspects of design relevant both to the containment and to the core cooling system capability in the depressurized condition should be evaluated further. Sensitivity studies should be made covering the spectrum of containment pressure from the assumed maximum to zero gage. Other features affecting containment systems and filter design such as the presence of combustible gases, e.g. hydrogen, the creation and release of plutonium aerosols, and the response to post-accident heat generation, should be investigated more extensively.

Postulated core disruptive accidents should be examined as a potential design basis for the GCFBR. Analyses should be conducted in detail on the GCFBR, as is being done on LFMBRs, taking into account possible reassembly and potential autocatalytic phenomena, to permit a better understanding of PCRV and containment response to such accidents.

Potential sources of these accidents include a loss-of-coolant flow, depressurization, or a rapid reactivity insertion with failure of timely scram.

A desirable approach, for this prototype plant relates to the ability to maintain containment in the unlikely event of melting of fuel.

The Committee recognizes that two independent reactor shutdown systems represent a desirable step toward reducing the probability of an anticipated transient without scram. Efforts should be continued to improve the reliability of these shutdown systems.

While the ACRS recognizes that there are some advantages in a PCRV, the world-wide experience with PCRFs is still too limited to provide meaningful reliability statistics. Because the GCFBR operating pressures are substantially higher than those in most previous PCRVs, additional analytic and experimental studies are needed to establish possible failure mechanisms under a variety of accident conditions.

A critical component of the PCRV is the thermally insulated liner, which is similar to that proposed for HTGRs. While the GCFBR design provides greater accessibility to the insulation and liner for inservice inspection than exists in an HTGR, there are still problems on inspection techniques, the liner response to loss of thermal insulation and the impact of loss of insulation on system operation, fuel, etc. These problems should be investigated further.

The various core internals, including the fuel, are subject to variable loads at temperatures at which creep, stress rupture, and creep-fatigue interactions may be critical. Since the proposed core materials are sensitive to parameters of time, temperature, modes of loading, and environment, it is essential that sufficient engineering data be obtained to permit prediction of component behavior throughout life, including normal, upset, emergency, and faulted conditions.

It is important that the applicant maintain adequate flexibility of design for purposes of modifying or supplementing presently contemplated safety features until the major safety questions and design criteria are resolved.

This is an interim letter for the purpose of aiding in the identification of major problem areas. Other items may prove to be equally significant, requiring extensive evaluation. The Committee will continue its review as viable alternates or acceptable justification of the existing proposed systems are provided.

Sincerely yours,

W. R. Stratton
Chairman

References attached

References

1. General Atomic Company (formerly Gulf General Atomic) "Gas-Cooled Fast Breeder Reactor - Preliminary Safety Information Document" Volumes I and II

2. Supplements 1 through 10 to the Preliminary Safety Information Document" (PSID)

3. Supplements I and II to the PSID

4. Amendments 1 through 6 to the PSID

5. Regulatory Staff, U.S. Atomic Energy Commission, "Preliminary Report to the ACRS - Gas-Cooled Fast Reactor (GCFR)" dated June 14, 1971

6. Regulatory Staff, U.S. Atomic Energy Commission, "Report to the ACRS Gas-Cooled Fast Reactor Conceptual Design Review" dated November 19, 1971

7. Regulatory Staff, U.S. Atomic Energy Commission "Preapplication Safety Evaluation of the Gas-Cooled Fast Breeder Reactor" dated August 1, 1974

8. General Atomic Company letters dated May 23, 1973, regarding the Regulatory Staff's report of a meeting held on March 13-14, 1973, at which accidental positive reactivity insertion mechanisms were discussed; dated October 10, 1974, regarding the definition of the design basis accidents; and dated October 11, 1974, commenting on the Regulatory Staff's Safety Evaluation Report

9. GA-A12934 "Reactivity Insertion Mechanisms in the GCFBR" by Torri and Driscoll, dated April 10, 1974

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: GENERAL ELECTRIC TEST REACTOR

Dear Mr. Strauss:

This letter constitutes the report of the Advisory Committee on Reactor Safeguards on the application for a construction permit by the General Electric Company, Docket No. 50-70, in accordance with Section 182 of the Atomic Energy Act of 1954, as amended.

The application is for a test reactor designed to operate at power levels up to 30 megawatts of heat. It is to be located at the Vallecitos Atomic Laboratory site situated in Pleasanton Township, California.

The purpose of the reactor is to provide a facility to irradiate at high neutron fluxes fuel elements and other components for proposed nuclear power plants for developmental testing.

The Committee is of the opinion that the proposed reactor and the experimental program as generally described in the application can be operated at the site selected with an acceptably low risk of any injury to the health and safety of the public.

The Committee, in reaching its opinion, has been influenced primarily by the following considerations:

a. The containment proposed by General Electric appears adequate;

b. The leakage rate from this container can be periodically checked;

c. The site appears adequate for this reactor, particularly because of the low population density in the surrounding area. Further, this location has already been found acceptable for the operation of reactors of comparable power;

d. The technology of the type of reactor proposed is fairly well understood.

The Committee, of course, cannot pass judgment at this time on the conduct of particular experiments in this facility since the experimental program is only described in general terms. However, the Committee does believe that the kind of experimental program outlined can be conducted safely in the proposed facility with appropriate restrictions.

Sincerely yours,

/s/

C. Rogers McCullough
Chairman
Advisory Committee on
Reactor Safeguards

Orig. & 2 copies sent to Chairman

cc: ACRS Members
R. H. Grahams
11/15/57

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

July 12, 1958

Dr. Willard F. Libby
Acting Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: GENERAL ELECTRIC TEST REACTOR (GETR)

Dear Dr. Libby:

The Advisory Committee on Reactor Safeguards reviewed, at its Eighth Meeting, July 10-12, 1958, the proposal of the General Electric Company to operate the General Electric Test Reactor. The Committee had previously offered advice on this reactor at its Second Meeting, November 1-3, 1957, in connection with the General Electric request for a construction permit. In its current review, the Committee had access to the reports referenced below and discussed the proposal with representatives from both the General Electric Company and the Hazards Evaluation Branch.

The GETR is a pressurized water reactor operating at 33 Mw (Thermal) located at the General Electric Vallecitos Atomic Laboratory, Pleasanton, California. A large body of information and experience exists on the nuclear, hydraulic, and mechanical behavior of the components of pressurized water reactor systems. The primary area of uncertainty, with regard to reactor safety, now concerns the transient response of this type of reactor to rapid additions of excess reactivity. Pertinent information is now being obtained as part of the SPERT program. However, in this interim period, considerable guidance as to reactor dynamics can be obtained from existing Borax and SPERT data. The ACRS believes that operation of this reactor, considered separately from the intended experimental program, presents no greater hazard than many other reactors now approved for operation.

A judgment as to the continuous safe operation of this reactor including its testing function presents an additional problem because of the inability to define precisely the specific characteristics of the future experimental program. Relatively more dependence must be placed upon the sound judgment of the operators of test reactors than upon that of operators of reactors for which less flexibility is required. While it is hoped that in the future more flexible definitions of the areas for the independent action on the part of operators for testing reactors will be developed, the ACRS believes that the General Electric Company has proposed reasonably acceptable limitations within which the GETR staff may take action independent of prior AEC approval.

The Advisory Committee on Reactor Safeguards thus advises that the GETR may be operated as a testing reactor as proposed by the General Electric Company without undue hazard to the public.

Sincerely yours,

/s/

C. Rogers McCullough
Chairman

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

References:

Amendment No. 3 to License Application for GETR, 2/26/58

Amendment No. 4 to License Application for GETR for: Experimental Facilities 5/15/58

Amendment No. 5 to License Application for GETR, 6/18/58

HEB Staff Report on GETR, 6/27/58

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

July 22, 1966

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

Subject: REPORT ON GENERAL ELECTRIC TEST REACTOR

Dear Dr. Seaborg:

At its seventy-fifth meeting, on July 14-16, 1966, the Advisory Committee on Reactor Safeguards considered the application of the General Electric Company (GE) to increase the power level of the General Electric Test Reactor (GETR) from 33 to 50 MW(t) and to operate the GETR under a ten-year license. The Committee had the benefit of discussion with representatives of GE and the AEC Regulatory Staff, as well as the documents listed. An ACRS Subcommittee reviewed this project on July 1, 1966.

The GETR is a light-water moderated reactor with enriched uranium-aluminum alloy fuel plates, operated since 1959 at GE's Vallecitos Atomic Laboratory in California. In connection with the proposed increase in power, GE has made plans for updating the facility in accordance with present safety standards. Significant changes will include the following:

1. Provision of redundant containment isolation valves and improved containment testing.

2. Improvement of safety instrumentation to eliminate potential loss of function due to single failures.

3. Installation of an emergency water recirculation system to maintain water in the reactor and the pool in the unlikely event of loss of water through certain breaks in the pool or reactor piping or nozzles.

4. Provision of a secondary shutdown system using injection of gadolinium nitrate into the primary coolant.

The Committee was assured that these changes will be made expeditiously.

In discussions regarding proposed in-pile experiments, GE stated that the inventory of liquid metals would be limited to 1 kg per experiment. Experiments will be arranged so that rupture of one such experiment will not induce failure of another. The Committee recommends that the AEC Regulatory Staff review carefully the basis for any future proposed modification of the 1 kg limit or other criteria for liquid-metal experiments.

The Committee concludes that the GETR can be operated as proposed at power levels up to 50 MW for a 10-year period without undue risk to the health and safety of the public.

Sincerely yours,

/s/ David Okrent

David Okrent
Chairman

References:

1. Application for Amendment to Facility License for General Electric Test Reactor, dated October 29, 1965.

2. APED-5000-A, Facility Description and Safety Analysis Report for the General Electric Test Reactor, Volumes I and II, dated July 1965.

3. Amendment 19 to License Application for General Electric Test Reactor, dated June 3, 1966, with attachment.

4. Supplement to Amendment 19, undated, received June 24, 1966.

5. TWX dated June 23, 1966 from General Electric Company to AEC.

6. TWX dated July 12, 1966 from General Electric Company to AEC.

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

November 12, 1980

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

SUBJECT: REPORT ON THE RESTART OF THE GENERAL ELECTRIC TEST REACTOR

Dear Dr. Ahearne:

During its 247th meeting, November 6-8, 1980, the ACRS reviewed a request by the General Electric Company to restart and operate the General Electric Test Reactor (GETR) at power levels up to its rated power of 50 MWt. A tour of the facility was made by members and consultants in connection with November 14, 1979 and June 16 and 17, 1980 meetings of the Subcommittee and the matter was further considered at a Subcommittee meeting on November 4, 1980. During its review, the Committee had the benefit of discussions with representatives and consultants of the Licensee and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

The GETR, which was granted an operating license in January, 1959, was shut down on October 24, 1977 in accordance with a Commission order. This order followed discovery of a fault near the location of GETR and brought into question the plant's capability to withstand the effects of an earthquake that might occur on or near the newly discovered fault.

After extensive study of the geology and seismology of the site, and of the region nearby, the NRC Staff concluded that in order to operate the plant it must be shown that it can sustain a ground level acceleration, unaccompanied by surface offset under the foundation, of 0.75g and that it must also be shown to be capable of withstanding a ground level acceleration of 0.6g simultaneous with a surface displacement of one meter of reverse-oblique net slip along a fault plane having a dip between 10 and 45 degrees. The ACRS agrees that these criteria are sufficiently conservative.

In order to achieve compliance with these criteria, the General Electric Company has proposed some plant modifications and has performed an extensive analytical investigation to demonstrate that the modified plant can survive an earthquake having the characteristics of the Staff's criteria. The NRC Staff has reviewed and approved the analyses and the modifications.

The ACRS agrees that the plant as modified should be able to withstand the postulated seismic events with no significant release of radioactive material.

The NRC Staff has yet to resolve one issue of seismic loading which is dependent on the characteristics of the soil beneath the GETR foundation. The Staff and the Licensee are both confident, however, that this issue can be resolved after further calculation by the Licensee and review by the Staff. The ACRS recommends that this issue be resolved to the satisfaction of the Staff.

Subject to resolution of the above issue, the ACRS believes that the GETR, as modified, can be restarted and operated at its rated power level of 50 MWt, without undue risk to public health and safety.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. General Electric Company, Vallecitos Nuclear Center, "GETR Safety Analysis Report," NEDO-12622, June 1977.
2. Letter, E. G. Case, NRC, to R. Darmitzel, General Electric Company (GE), regarding the Order to Show Cause, dated October 12, 1977.
3. General Electric Company, "Updated Response to NRC Order to Show Cause Dated October 24, 1977," June 1978.
4. Engineering Decision Analysis Company, Inc., "Seismic Analysis of Reactor Building, General Electric Test Reactor, Phase 2," prepared for General Electric Company, EDAC 117-217.03, June 1978.
5. Engineering Decision Analysis Company, Inc., "Seismic Analysis of Primary Coolng System and Reactor Pressure Vessel, General Electric Test Reactor," Prepared for General Electric Company, EDAC 117-217.05, June 1978.
6. Engineering Decision Analysis Company, Inc., "Seismic Analysis of Primary Heat Exchange, General Electric Test Reactor," prepared for General Electric Company, EDAC 117-217.06, June 1978.
7. Engineering Decision Analysis Company, Inc., "Seismic Analysis of Reactor Pressure Vessel and Pool Drain Lines and Poison Injection Line, General Electric Test Reactor," prepared for General Electric Company, EDAC 117-217.07, June 1978.
8. Engineering Decision Analysis Company, Inc., "Seismic Analysis of Fuel Flooding System, General Electric Test Reactor," prepared for General Electric Company, EDAC 117-217.08, June 1978.
9. Engineering Decision Analysis Company, Inc., "Qualification of Safety-Related Valves, General Electric Test Reactor," prepared for General Electric Company, EDAC 117-217.09, June 1978.
10. Structural Mechanics Associates, "Structural Analysis of New Fuel Storage Tanks and Support System, General Electric Test Reactor," prepared for General Electric Company, June 1978.

References:

11. Structural Mechanics Associates, "Structural Analysis of Third Floor Missile Impact System, General Electric Test Reactor," prepared for General Electric Company, June 1978.
12. Letter, R. W. Reid, NRC, to R. Darmitzel, GE, on the review of Geological Investigation, Phase II, by Earth Sciences Associates, dated June 8, 1979.
13. Engineering Decision Analysis Company, Inc., "Probability Analysis of Surface Rupture Offset Beneath Reactor Building, General Electric Test Reactor," prepared for General Electric Company, EDAC 117-217.13, April 12, 1979.
14. Letter, R. Darmitzel, GE, to R. W. Reid, NRC, "Structural Modifications for the General Electric Test Reactor," July 9, 1979.
15 Letter, H. Denton, NRC, to R. Darmitzel, GE, regarding Show Cause Proceeding, Geosciences Branch Safety Evaluation Report Input, GE Test Reactor Site, Vallecitos Nuclear Center, dated September 27, 1979.
16. Letter, R. Darmitzel, GE, to D. Eisenhut, NRC, regarding Response to Questions Raised by the GETR Subcommittee of the ACRS consultants," dated April 14, 1980.
17. Letter, R. E. Jackson, NRC, to J. F. Devine, USGS, transmitting report entitled, "Seismicity of the Livermore Valley in Relation to the General Electric Vallecitos Plant, by B. Bolt and R. Hanson," dated April 17, 1980.
18. Letter, D. L. Gilliland, GE, to D. G. Eisenhut, NRC, "Part I Response to NRC Questions, Structural Issues," April 23, 1980.
19. Letter, R. W. Darmitzel, GE, to D. G. Eisenhut, NRC, regarding Analysis of Slip Rate of Shear Surfaces at the General Electric Test Reactor (GETR) Site," dated April 29, 1980.
20. Letter, R. W. Darmitzel, GE, to D. G. Eisenhut, NRC, regarding General Electric Test Reactor Foundation Excavation Photographs, dated April 29, 1980.
21. Letter, R. W. Darmitzel, GE, to D. G. Eisenhut, NRC, regarding Responses to NRC Questions on Additional Probability Analysis of Surface Rupture Offset Beneath Reactor Building - General Electric Test Reactor, dated April 30, 1980.
22. Letter, R. W. Darmitzel, GE, to D. G. Eisenhut, NRC, regarding Responses to NRC Questions - Structural Issues - Part II, Recent Investigation, dated May 8, 1980.
23. Letter, D. G. Eisenhut, NRC, to R. W. Darmitzel, GE, regarding Safety Evaluation by the Office of Nuclear Reactor Regulation, for the General Electric Reactor, General Electric Company, Docket No. 50-70, dated May 23, 1980.
24. Letter, D. L. Gilliland, GE, to D. G. Eisenhut, NRC, regarding Landslide Stability Investigation of the General Electric Test Reactor (GETR) Site, dated July 25, 1980.
25. Letter, R. W. Darmitzel, GE, to D. G. Eisenhut, NRC, regarding General Electric Test Reactor (GETR) Landslide Stability Analysis, dated August 29, 1980.
26. Letter, R. W. Darmitzle, GE, to D. G. Eisenhut, NRC, regarding Responses to Additional Information Request Regarding Seismic Scram System for the General Electric Test Reactor, dated October 13, 1980
27. Letter, D. G. Eisenhut, NRC, to R. W. Darmitzel, GE, regarding the Safety Evaluation by the Office of the Nuclear Reactor Regulation, for the General Electric Test Reactor, General Electric Company, Docket No. 50-70, dated October 27, 1980.

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

March 18, 1966

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

Subject: REPORT ON BROOKWOOD NUCLEAR STATION UNIT NO. 1

Dear Dr. Seaborg:

At its seventy-first meeting, March 10-12, 1966, the Advisory Committee on Reactor Safeguards considered the proposal by the Rochester Gas and Electric Corporation to build a 1300 MW(t) pressurized-water reactor at its Brookwood site. The Committee had the benefit of discussions with representatives of the applicant, the Westinghouse Electric Corporation, Gilbert Associates, Inc., and consultants to the applicant; with the AEC Staff and its consultants; and of the documents listed. The Committee had previously reviewed some features of the plant at its seventieth meeting in February 1966. A subcommittee of the ACRS visited the site on July 16, 1965, and met with the applicant to review the proposal on January 27, 1966 and March 9, 1966.

The reactor system will be housed in a concrete containment building of novel design, with tensile forces carried by a combination of reinforcing steel and pre- and post-stressed tendons. The containment is an important engineered safeguard and should be accorded careful study commensurate with the importance and novelty of the structure.

The Committee believes that the following action should be taken before design of the containment is set:

1. Detailed design criteria and general specifications should be formalized by the applicant, and reviewed by the Staff and its consultants to assure that the design will take into account not only the ACI Code for conventional structures but also European experience with design, construction, and testing of prestressed-concrete nuclear pressure vessels. A high degree of conservatism should be reflected in the design to allow for uncertainties in the state of the art.

2. The Committee calls attention to the potential problem of loss of strength or failure of tendons by corrosion over a 40-year life, and since the applicant proposes to use nonreplaceable tendons, the Committee recommends that this problem be given close attention. Provision for a surveillance program may be appropriate or even necessary. The Committee notes that there is some difference of opinion among experts in the field concerning the use of grouted versus ungrouted tendons and suggests that the applicant review the advantages and disadvantages associated with each approach and provide means for coping with any shortcomings of the selected approach to assure the reliability of the containment during its lifetime.

3. Quality control and inspection procedures for construction should be formalized, including a statement of the authority and prequalification of inspectors.

4. Criteria for testing the containment and evaluation of test results should be developed as far as necessary to assure that desired embedded instrumentation will be available during the test.

5. The desirability of model testing should be reconsidered for regions that do not lend themselves to reliable analysis; testing to destruction may be desirable to establish failure modes. As an alternative to model testing, difficult design areas should be appropriately instrumented during construction so that relevant data can be obtained at the time of the pressure test.

The Staff and its consultants should follow the above items closely and be satisfied as to the adequacy of the approaches adopted. The applicant has already agreed to work out details of test instrumentation, testing procedures, and acceptance standards for the containment.

The pressure test of the containment will be conducted at 69 psig and the leak test at 60 psig. The applicant states that the 60 psig test can be repeated as necessary over the life of the containment.

The applicant has agreed to provide additional specified redundancy or independence in the containment spray system, the fan and filter systems of the auxiliary building, and the service water supply. Additional measures will be taken, if found necessary, to preclude any credible possibility of the containment pressure exceeding 60 psi. Additional control room shielding will also be provided. The reactor may be subject to low-frequency xenon oscillations, and the applicant has stated that, if further analysis shows such to be necessary, he will take measures to control the instability. The postulated accident involving sudden ejection of a control rod will be analyzed by the applicant during detailed design, and suitable measures will be taken to limit the consequences of the accident, if necessary. The Committee believes that these problems can be resolved during construction.

The applicant described a program of improved quality control in the fabrication of the reactor vessel and also described a program for surveillance of the increase in nil-ductility transition temperature over the life of the vessel; the Committee attaches considerable importance to these programs. The Committee suggests that the applicant give further consideration to the development and use of improved methods of in-service inspection of the reactor vessel.

It is the opinion of the ACRS that, with due regard to the above considerations, a satisfactory containment of the proposed type can be designed and constructed, and the Brookwood Unit No. 1 can be built at the proposed site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/

David Okrent
Chairman

References attached

References: Brookwood

1. Rochester Gas and Electric Corporation, Brookwood Nuclear Station Unit No. 1, Preliminary Facility Description and Safety Analysis Report, Volume 1, Volume 1 - Appendices, Volume 2 - Part A, and Volume 2 - Part B, transmitted by Le Boeuf, Lamb & Leiby letter dated November 1, 1965.

2. First Supplement to: Preliminary Facility Description and Safety Analysis Report, dated January 17, 1966.

3. Second Supplement to Preliminary Facility Description and Safety Analysis Report, undated, received January 27, 1966.

4. Third Supplement to: Preliminary Facility Description and Safety Analysis Report, dated February 28, 1966.

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

May 15, 1969

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

Subject: REPORT ON ROBERT EMMETT GINNA NUCLEAR POWER PLANT UNIT NO. 1

Dear Dr. Seaborg:

During its 109th meeting, May 8-10, 1969, the Advisory Committee on Reactor Safeguards completed its review of the application by the Rochester Gas and Electric Corporation for a license to operate the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 at power levels up to 1300 MWt. The Committee had previously met with the applicant during its 103rd meeting, October 31 to November 2, 1968, to review an important change in the design of the large penetrations of the containment, and again during its 108th meeting, April 10-12, 1969, for a partial review of the application. During the review, Subcommittee meetings were held on October 24, 1968 (at the site); January 23, 1969; March 5, 1969; and May 1, 1969. In the course of the review, the Committee had the benefit of discussions with representatives of the applicant, the Westinghouse Electric Corporation, Gilbert Associates, Inc., and their consultants; of discussions with the AEC Regulatory Staff and its consultants; and of the documents listed. The Committee reported to you on the construction permit application for this plant on March 18, 1966.

The reactor primary fluid system, containment, and engineered safety features all incorporate important developments from the design of previously licensed pressurized water reactors. The developments reflect both economic and safety considerations, and the plant represents the first of the line of Westinghouse reactors currently being licensed for construction.

The applicant is re-examining his estimate of the appropriate design flood level, including still water level, wave action, and wave runup. In the event of disagreement with the AEC Regulatory Staff, he will assure plant protection consistent with the flood estimates by the Staff consultants.

The applicant has agreed to install a strong-motion accelerograph if considered necessary. The Committee believes that at least one strong-motion accelerograph should be installed and, in addition, wishes to point out that a strong-motion accelerograph could minimize the possibility of a lengthy shutdown for inspection in the event that a significant earthquake of otherwise undetermined intensity at the site should occur.

The high thermal performance demanded of the fuel in the Ginna reactor, and the potential for axial xenon oscillations, requires that the spatial power distribution in the reactor core and the positions of the control rods be dependably known. In the proposed design all alarms related to control-rod malpositioning are derived from the on-line computer. The Committee believes good information regarding possible anomalies in the power distribution is important and that, as a minimum, the power should be reduced appropriately, or adequate alternative measures should be taken, when the computer is inoperative.

The applicant and the Regulatory Staff are not in agreement on the radioactivity that might be released and the off-site dose that could result from dropping a spent fuel assembly in the storage pit. The applicant will attempt to reconcile the disagreement but, if necessary, will take corrective measures to satisfy safety criteria in accordance with the Staff model for this postulated accident. The applicant will not handle irradiated fuel until this matter is resolved.

The applicant calculates that the reactor pressure vessel wall will be exposed to a fairly large fast neutron fluence (about 3.7×10^{19}) over the reactor life. This will lead to a sizeable increase in the nil ductility transition temperature and to some degradation in fracture toughness properties. Prior to the accumulation of a peak fluence of 10^{19}, the Regulatory Staff should reevaluate the continued suitability of the currently proposed reactor vessel startup, cooldown and operating conditions, as well as the assurance of vessel integrity despite thermal shock in the unlikely event of a loss-of-coolant accident.

The Committee understands that the applicant is providing means for preoperational monitoring of the pressure vessel and other parts of the primary system for signs of excessive internal vibration or structural damage. The Committee believes the applicant should give consideration to a program of monitoring during the service life of the plant.

The Committee believes the applicant's proposal of an in-service inspection program for the reactor pressure vessel and other portions of the primary system covering the first five years of operation, with a commitment to review the program after that period in the light of then-existing inspection technology, is satisfactory. The applicant has modeled his inspection program on the draft USA code dealing with in-service inspection; the Committee concurs in this approach.

Several Westinghouse reports pertinent to Ginna and other Westinghouse reactors have recently been received and others are expected. Some matters relating to Ginna consequently remain to be resolved by the Staff either before plant operation or on an acceptable time scale subsequent to initial operation. These matters include assurance of long-term compatibility of the containment spray solution with the exposed materials in the containment and verification of the performance of the hydrogen recombiners that may be necessary in the unlikely event of a loss-of-coolant accident; evaluation of the probability and consequence of systematic instrument failures. A more detailed analysis of the dynamic response of a portion of the system piping to an earthquake is also being prepared by the applicant for review by the Staff. The Committee believes that these matters will be resolved satisfactorily.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 can be operated at power levels up to 1300 MWt without undue risk to the health and safety of the public.

Additional remarks of Dr. David Okrent are attached.

Sincerely yours,

/s/

Stephen H. Hanauer
Chairman

Attachments:
1. Additional Remarks of
   Dr. David Okrent
2. References

Dr. David Okrent makes the following additional remarks:

"In view of the great importance of pressure vessel integrity to the health and safety of the public, I believe that for welds in the pressure vessel wall that will receive a large integrated fast neutron irradiation over the reactor life it would be prudent for the applicant to commit himself to a more thorough and extensive in-service, non-destructive, volumetric testing program by such means as are or become practical. In particular, within the framework of currently anticipated technology, I would recommend a commitment to 100% ultrasonic inspection of such a weld every ten years. Consideration should also be given to non-destructive, volumetric inspection or monitoring of those steel forgings making up the vessel wall that will be highly irradiated."

References - Robert Emmett Ginna Nuclear Power Plant Unit No. 1

1. Preliminary Facility Description and Safety Analysis Report, Volume 1, Appendices.

2. Amendments 6-17 and Amendment 19 to Application for Licenses.

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 ROBERT EMMETT GINNA NUCLEAR POWER PLANT UNIT NO. 1

Dear Dr. Schlesinger:

At its 140th meeting, December 9-11, 1971, the Advisory Committee on Reactor Safeguards reviewed the request by Rochester Gas and Electric Corporation for an increase in the licensed power level of its Robert Emmett Ginna Nuclear Power Plant Unit No. 1 from 1300 MW(t) to 1520 MW(t). A Subcommittee had previously met with the licensee on December 6, 1971. During its review the Committee had the benefit of discussions with representatives of the licensee, the Westinghouse Electric Corporation, 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 operating license application on May 15, 1969, and a Provisional Operating License was issued on September 19, 1969, authorizing operation at steady-state power levels up to 1300 MW(t). New analyses have been submitted to show that the plant will perform satisfactorily at 1520 MW(t). The Ginna Unit is essentially the same as Point Beach Nuclear Plant, Units 1 and 2, which have been authorized for operation at 1518 MW(t).

Changes in Technical Specifications have been proposed to assure safe operation at the higher power. The licensee has also applied the Ginna operating experience to make improvements in the plant and mode of operation. The Committee believes the licensee should continue to work towards solutions of problems that have been identified by the Regulatory Staff and ACRS as common to large water reactors, including tolerance to anticipated transients with failure to scram. These matters can be resolved wih the Regulatory Staff on an appropriate time schedule, not necessarily before commencing operation at the higher power.

The monitoring of iodine released with gaseous wastes at the Ginna plant has not provided reliable evidence of satisfactory performance of the iodine removal system. The Committee believes that attention should be given to improving iodine monitoring methods such that assurance can be provided that total offsite doses remain within appropriate limits.

The licensee will maintain a peak linear power density at full power not exceeding 16.0 kw/ft. Analyses of postulated loss-of-coolant accidents indicate acceptable low peak clad temperatures at the proposed power level of 1520 MW(t).

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, there is reasonable assurance that the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 can be operated at power levels up to 1520 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Spencer H. Bush

Spencer H. Bush
Chairman

References

1. Rochester Gas & Electric Corporation (RG&E) Proposed Technical Specifications and Bases, R. E. Ginna Nuclear Power Plant Unit No. 1, received April 25, 1969

2. RG&E Report, Significant Plant Problems, dated February 5, 1970

3. RG&E Petition Requesting Amendment of License and Extension of Expiration Date of Provisional Operating License, with Technical Supplement, received February 22, 1971

4. Amendments 1-4 to Petition and Technical Supplement

5. RG&E Performance Report for Ginna Plant No. 1, received September 13, 1971

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

August 18, 1982

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 R. E. GINNA NUCLEAR POWER PLANT

During its 267th meeting, July 8-10, 1982, the ACRS reviewed the results of the Systematic Evaluation Program, Phase II, as it has been applied to the R. E. Ginna Nuclear Power Plant. These matters were also discussed during a Subcommittee meeting in Washington, D.C. on June 3, 1982. During our reviews, we had the benefit of discussions with representatives of the Rochester Gas and Electric Corporation (Licensee) and the NRC Staff. We also had the benefit of the documents listed below. We completed our report regarding this matter during the 268th meeting, August 12-14, 1982.

Our first review of Phase II of the Systematic Evaluation Program (SEP) was carried out in connection with its application to the Palisades Plant. Our findings from that review were addressed in a letter to you dated May 11, 1982. Our continuing review of the SEP, in relation to the Ginna Plant, has resulted in no changes in our previous findings and comments as they relate to the SEP program in general. Mr. William J. Dircks responded to some of those comments in a letter dated June 7, 1982. We find his response acceptable.

The remainaer of this letter relates specifically to the SEP review of the Ginna Plant.

Of the 137 topics to be addressed in the SEP, 21 were not applicable to the Ginna Plant, and 24 were deleted from the review because they were being reviewed generically under either the Unresolved Safety Issues (USI) program or the TMI Action Plan. Of the 92 topics addressed in the Ginna Plant review, 58 were found to meet current NRC criteria or to be acceptable on another defined basis. Seven topics were later added to this category as a result of modifications made or committed to by the Licensee during the review. We have reviewed the assessments and conclusions of the NRC Staff relating to these topics and have found them appropriate.

For all or part of the remaining 27 SEP topics, the Ginna Plant was found not to meet current criteria. These topics were addressed by the Integrated Assessment and have been resolved to various degrees and in various ways.

The Integrated Assessment has not yet been completed for portions of seven topics, for which additional information must be provided by the Licensee. This information includes the results of studies, calculations, and evaluations that are required by the NRC Staff for its assessments and decisions. Six of these topics relate to structural design and the Licensee has proposed a coordinated program for their resolution. The NRC Staff has agreed to this program. The resolution of these topics will be addressed by the NRC Staff in a supplemental report that will be available for review in connection with the application for a Full-Term Operating License (FTOL) for the Ginna Plant.

For portions of ten topics included in the Integrated Assessment, the NRC Staff concluded that no backfit is required. We concur.

For the remaining topics for which the assessment has been completed, the NRC Staff requires the addition or modification of structures or equipment, or the development or modification of procedures or technical specifications. Except for the three topics discussed below, the Licensee has agreed to the resolution required by the NRC Staff.

One area of disagreement relates to the groundwater level and the associated hydrostatic pressures that the structures below grade must withstand. The plant was designed assuming a groundwater elevation of 250 ft. Although limited observations from borings have shown the groundwater to be near that elevation, there has been no program of continuing measurement to demonstrate that the level does not exceed 250 ft. during periods of prolonged precipitation. In the absence of such a program, the NRC Staff has determined that the effects of groundwater should be evaluated for an assumed elevation at the surface of the ground, approximately 270 ft. for the structures of interest. We believe that such an evaluation should be made. We recommend that acceptability of the structures be based on "no loss of function" and not on arbitrary limits of stresses computed using linear-elastic assumptions.

A second topic for which resolution has not been reached relates to flooding of the site by Deer Creek, a small stream flowing into Lake Ontario in the vicinity of the plant. Flooding from Deer Creek was not considered when the plant was originally licensed; Lake Ontario was the only source of flooding considered by the Applicant and the AEC Staff at that time. Neither the NRC Staff nor the Licensee consider this question to be resolved, nor do we. Since flooding is an important matter that may have implications for other operating plants, we plan to continue our review of flood criteria, both for the Ginna Plant and on a more generic basis, and to provide our comments or recommendations when that review is completed.

The third topic for which agreement has not yet been reached concerns several containment isolation valves that do not satisfy the requirements of General Design Criterion No. 57. In view of the generally acceptable and well-considered manner in which the NRC Staff has evaluated the numerous other topics related to isolation valves, we believe that this topic should be resolved in a manner satisfactory to the NRC Staff.

As was the case for the Palisades Plant, a plant-specific Probabilistic Risk Assessment (PRA) was not available for the Ginna Plant. In its absence, the NRC Staff made careful and conservative use of a limited and essentially qualitative risk assessment, based in part on the Reactor Safety Study, for a three-loop Westinghouse plant and in part on the Interim Reliability Evaluation Program PRA for the Crystal River Plant, a two-loop Babcock & Wilcox plant. From even this limited use of a PRA, it is clear that many of the decisions involved in the SEP could be made much more rationally if plant-specific PRAs were available.

Our conclusions can be summarized 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 Ginna Plant and should be achieved for the remaining plants in Phase II of the program.

2. The actions taken thus far by the NRC Staff in its SEP assessment of the Ginna Nuclear Power Plant are acceptable.

3. The ACRS will defer its review of the FTOL for the Ginna Plant until the NRC Staff has completed its actions on the remaining SEP topics and the USI and TMI Action Plan items.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. U.S. NRC Draft Report, "Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant," NUREG-0821, dated May 1982.
2. NRC Staff Consultants' Review of the R. E. Ginna Nuclear Power Plant Integrated Plant Safety Assessment Report including Consultant Reports from R. J. Budnitz, S. H. Bush, J. M. Hendrie, H. S. Isbin, and Z. Zudans.
3. R. E. Ginna SEP Topic, Safety Evaluation Reports, Volumes 1 through 3, dated May, 1982.
4. U. S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, dated November 1980

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

April 9, 1984

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

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON FULL-TERM OPERATING LICENSE FOR THE R. E. GINNA NUCLEAR POWER PLANT

During its 288th meeting, April 5-7, 1984, the Advisory Committee on Reactor Safeguards completed its review of the application by the Rochester Gas and Electric Corporation (Licensee) for conversion of the provisional operating license (POL) for its R. E. Ginna Nuclear Power Plant to a full-term operating license (FTOL). This application was considered also during a Subcommittee meeting in Washington, D.C. on November 16, 1983 and during the 283rd ACRS meeting, November 17-19, 1983. Issues related to flood, severe wind, and earthquake hazards were reviewed in depth during meetings of the Subcommittee on Extreme External Phenomena on October 21-22, 1982 and April 4, 1984. During our review, we had the benefit of discussions with representatives of the Licensee and the NRC Staff. We also had the benefit of the documents referenced. The Committee most recently discussed and reported on this plant in a letter dated August 18, 1982 relating to the Systematic Evaluation Program (SEP) review of the Ginna Plant.

The Ginna Plant received a POL in September 1969 and began commercial operation in December of the same year. The Licensee applied for an FTOL in a timely fashion in August 1972, but review of this application was deferred by the NRC Staff in 1975, along with several other FTOL reviews. In 1978, the Ginna Plant was included in Phase II of the SEP because much of the review needed for the FTOL was similar in scope to that for the SEP.

In the Committee's letter reporting on the results of the SEP as applied to the Ginna Plant, the ACRS indicated that its review of the FTOL would be deferred until the NRC Staff had completed its actions on the SEP issues that were still pending and on the Unresolved Safety Issue (USI) and TMI Action Plan items. The SEP issues have been resolved to the satisfaction of the NRC Staff in the manner reported in Supplement No. 1 to the Integrated Plant Safety Assessment Report for the Ginna Plant. The status of the USI and TMI Action Plan items for the Ginna Plant has been discussed by the NRC Staff in its Safety Evaluation Report related to the FTOL for the Ginna Plant.

Although all of the actions proposed or committed to as a result of the SEP review have not yet been completed, we believe that the procedures and schedules that have been agreed to are satisfactory. A large proportion of the TMI Action Plan items have been completed and those remaining are in a status acceptable to the NRC Staff, and to us. A similar situation exists with regard to those USI items for which a resolution has been reached by the NRC Staff.

The Licensee has proposed to modify the plant to decrease its vulnerability to tornado winds and missiles. These modifications will be based on a tornado having a design wind velocity of 132 mph. Modifications to the steel structures will be based on criteria that will ensure no significant yielding at wind speeds up to 132 mph, and no instability or collapse that might affect components or systems needed for safe shutdown at wind speeds up to about 200 mph. It appears from the Licensee's analyses that the cost of plant modifications would increase sharply if design basis tornadoes significantly higher than 132 mph were used. The NRC Staff believes that these planned modifications will upgrade the plant design such that tornadoes will not be a dominant contributor to the risk of core melt. We believe that this is an adequate approach, but recommend that the NRC Staff consider further the measures proposed or needed to assure operability of the diesel generator during the reduced pressure transient accompanying a tornado.

We concur with the process used by the NRC Staff and the Licensee to assure that the plant is adequately protected from the effects of external floods. The procedures used by the NRC Staff to evaluate the seismic adequacy of the plant are reasonable and are similar to procedures used in seismic reevaluation of other SEP plants.

We do not believe that any of the pending actions related to the SEP, USI, or TMI Action Plan items would be accelerated by withholding an FTOL at this time.

In connection with our review of the SEP, we have considered the operating experience at the Ginna Plant and have found nothing that would preclude granting an FTOL at this time. We have also reviewed the most recent Systematic Assessment of Licensee Performance (SALP) Report for the Ginna Plant, for the period June 1, 1982 through May 31, 1983, and note that all activities reviewed were classed in either Category 1 or 2. We find this encouraging.

The Committee believes that there is reasonable assurance that the R. E. Ginna Nuclear Power Plant can continue to be operated at power levels up to 1520 MWt under a full-term operating license without undue risk to the health and safety of the public.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

References:
1. Rochester Gas and Electric Corporation, "Final Safety Analysis Report, R. E. Ginna Nuclear Power Plant," Volumes 1-3 and Supplements 1-12
2. U. S. Nuclear Regulatory Commission, "Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant," USNRC Report NUREG-0821, dated December 1982 and Supplement 1 dated August 1983
3. Letter from H. Denton, Director, Office of Nuclear Reactor Regulation to P. Shewmon, Chairman, ACRS, dated September 17, 1982, Subject: Staff Response to the ACRS Report on the Systematic Evaluation Program Review of the R. E. Ginna Nuclear Power Plant
4. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Full-Term Operating License for R. E. Ginna Nuclear Power Plant," USNRC Report NUREG-0944, dated October 1983
5. U.S. Nuclear Regulatory Commission, "NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant," USNRC Report NUREG-0909, dated April 1982
6. Letter dated September 26, 1983 from T. Murley, NRC Regional Administrator, to John E. Maier, Rochester Gas & Electric Corp., Subject: Systematic Assessment of Licensee Performance (SALP) Report
7. Institute for Disaster Research, Texas Tech University, "A Methodology for Tornado Hazard Probability Assessment," prepared for USNRC by J. R. McDonald, NUREG/CR-3058, dated October 1983

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

May 15, 1974

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

Subject: REPORT ON GRAND GULF NUCLEAR STATION, UNITS 1 AND 2

Dear Dr. Ray:

At its 169th meeting, on May 9-11, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application of the Mississippi Power and Light Company for a permit to construct the Grand Gulf Nuclear Station, Units 1 and 2. The Committee also considered this application during its 166th meeting on February 7-9, 1974, and its 167th meeting on March 7-9, 1974. Subcommittee meetings were held on this project in Los Angeles, California, on October 25, 1973, at Jackson, Mississippi, on December 21-22, 1973, at San Jose, California, on January 17-18, 1974, and in Washington, D. C., on March 6, 1974, and May 3-4, 1974. The site for the proposed station was visited by Committee members on December 21, 1973. In its review, the Committee had the benefit of discussions with representatives of the applicant, his consultants and contractors, and representatives of the Regulatory Staff and its consultants, and of the documents listed.

The Grand Gulf Nuclear Station will employ the BWR/6 nuclear system on which the Committee reported on September 21, 1972, and the Mark III containment concept on which the Committee reported on January 17, 1973.

The site of the Grand Gulf Nuclear Station is located in Claiborne County, Mississippi, on the east bank of the Mississippi River. The nearest population center with more than 25,000 persons is Vicksburg, Mississippi, 25 miles north-northeast of the site.

The history of seismic activity in the tectonic province including the Grand Gulf site is dominated by the three Modified Mercalli Intensity XII earthquakes which occurred near New Madrid, Missouri, in 1811-1812. The applicant's studies support a conclusion that the New Madrid earthquake zone is confined to a region extending northward from near Memphis, Tennessee, and the Regulatory Staff and its consultants concur that possible future major earthquakes in this tectonic province should be so confined. On this basis a safe shutdown earthquake ground acceleration of 0.15g in the Catahoula formation at the site, and 0.2g for those Category I structures founded in formations above the Catahoula formation, has been selected. The Committee finds this seismic design basis to be acceptable. However, the Committee recommends that, in the design of the plant, the applicant give careful attention to the possible effects of long duration, low frequency ground shaking.

The General Electric Company is pursuing an analytical and experimental program intended to provide more detailed knowledge of the behavior of the Mark III containment system and to confirm the design bases of the Grand Gulf Station. Among the phenomena for which further information will be obtained are vent-clearing, vent-interaction, pool stratification, and dynamic loads on suppression-pool and other containment structures. A well-defined and well-executed experimental program is of great importance to the validation of the Mark III concept and should be pursued diligently and expeditiously. Should any results indicate a significant deviation from current predictions of the designer, the Committee wishes to be informed.

The Regulatory Staff is continuing its review of the criteria for, and the preliminary design of, guard pipes around process lines traversing the region between the drywell and the containment. In view of the importance of the guard pipe function, special care, including use of conservative design stresses and achievement of an independent design check, should be taken. Because these pipes constitute a part of containment, it also is important that appropriate precautions be taken to assure the integrity of any penetrations incorporated, such as inspection hand holes. These matters should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant reported a marked reduction in the use of non-metallic insulation within the drywell which might, if displaced, plug screens or otherwise lead to a short or long term degradation of the efficacy of the heat removal systems required in the unlikely event of a loss-of-coolant accident. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant reported plans to utilize means to monitor for loose parts in the reactor pressure vessel during operation.

The applicant reported calculated peak cladding temperatures of 1515°F using interim acceptance criteria evaluation models, including densification. He also reported that he anticipated about 100°F or less increase in calculated peak cladding temperatures when the evaluation model for the recently adopted ECCS Acceptance Criteria is implemented. The Committee believes that such improvements are appropriate for reactors whose construction permits are requested after January 7, 1972, as noted in the Committee's report of September 10, 1973 on Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors.

To meet Regulatory Guide 1.7 the applicant has proposed a combustible gas control system in which a high-capacity recirculation system is available to mix the gases in the drywell and surrounding containment building beginning ten minutes after a postulated loss-of-coolant accident, should the hydrogen generation be as large as assumed in this guide. The proposed combustible gas-control system includes recombiners, is redundant, and is designed to meet engineered safety system requirements. However, the mixing system is relatively complicated and would require careful attention to reliability considerations.

The applicant has described an alternative system for the control of combustible gas, based on hydrogen generation resulting from only one percent metal-water reaction as compared to the five-percent figure required by Regulatory Guide 1.7. The Committee believes that the design of this plant, including the reactor core, the ECCS, and the containment system, are such that the assumption of one percent metal-water reaction is sufficiently conservative, and that use of the alternative system is preferable.

The applicant has stated that the station will be designed to deal with main steam line isolation valve leakage in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed of the resolution of this matter.

The Regulatory Staff is continuing to review several matters relating to the reactor instrumentation and control system, including system response to a turbine trip and the possible operation of control rods in groups. The Committee wishes to be kept advised of the resolution of these matters.

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 Grand Gulf Nuclear 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.

Additional comments by Dr. S. H. Bush and Dr. D. Okrent are attached.

Sincerely yours,

/s/ W. R. Stratton

W. R. Stratton
Chairman

References Attached.

Additional Comments by S. H. Bush

I believe the use of guard pipes is inappropriate in most, if not all instances, in nuclear designs. Industrial experience with such systems has not been satisfactory. There have been failures due to moisture entrapment, limited in-leakage and differential thermal loads. Such designs make visual inspection and volumetrc inspection difficult. A similar guard pipe design was suggested at the Brunswick construction permit and a suitable inspection program was substituted. While I do not dissent on this specific item, I do believe that approval of this feature for a class of reactors is undesirable. I urge that alternate approaches be considered for future BWR/6 Mark III plants.

Additional Comments by D. Okrent

Although I agree that the proposed safe shutdown earthquake for the Grand Gulf Station appears to be equivalent in level of safety to that utilized for most recent nuclear stations east of the Rockies, I find little basis for judging that the probability of exceeding the safe shutdown earthquake is less than 10^{-7} or event 10^{-6} per year. To say the least, the uncertainty in any such prediction is very large. In view of this situation I believe it would be prudent to provide some additional margin in the seismic design bases at this site and for most other future nuclear plants sited east of the Rockies.

I would also like to note specifically that, in addition to the large margins between calculated peak clad temperatures and acceptance criteria limits for a LOCA and to the diversity and stated reliability of the ECCS, an important consideration in applying the assumption of 1% clad-water reaction as an acceptable design basis for the combustible gas control system is the evaluation of the applicant that the drywell can accept the rapid burning of substantial quantities of hydrogen in the post-blowdown period without adversely affecting any vital safety function.

References

1. Preliminary Safety Analysis Report, Grand Gulf Nuclear Station, Units 1 and 2, Volumes 1 through 11.

2. Amendments 1 through 18 to the PSAR.

3. Directorate of Licensing letter to the Executive Secretary, ACRS, dated January 12, 1974 forwarding Safety Evaluation of the Grand Gulf Nuclear Station, Units 1 and 2 by the USAEC Directorate of Licensing, January 1974.

4. Directorate of Licensing letter to the Executive Secretary, ACRS, dated April 12, 1974 forwarding Supplement No. 1 to the Safety Evaluation by the USAEC Directorate of Licensing, April 12, 1974.

5. Mississippi Power & Light Company letter dated January 2, 1973 regarding fuel densification.

6. Mississippi Power & Light Company letter dated May 10, 1973 regarding maximum allowed thermal power.

7. Mississippi Power & Light Company letter dated October 17, 1973 regarding seismic survey program.

8. Mississippi Power & Light Company letter dated November 30, 1973 regarding miscellaneous additional information.

9. Mississippi Power & Light Company letter dated December 4, 1973 regarding proprietary seismic data.

10. Mississippi Power & Light Company letter dated December 11, 1973 regarding additional proprietary seismic data.

11. Mississippi Power & Light Company letter dated December 12, 1973 regarding additional proprietary seismic data.

12. Mississippi Power & Light Company letter dated December 12, 1973 regarding other proprietary information.

13. Mississippi Power & Light Company letter dated December 18, 1973 requesting an exemption to proceed with construction.

14. Mississippi Power & Light Company letter dated January 9, 1974 regarding additional information.

15. Mississippi Power & Light Company letter dated January 10, 1974 regarding ATWS.

16. Mississippi Power & Light Company letter dated January 29, 1974 regarding request for exemption to proceed with construction.

17. Mississippi Power & Light Company letter dated February 6, 1974 regarding additional information.

18. Mississippi Power & Light Company letter dated April 8, 1974 regarding guard pipes and blowdown from a recirculation line.

19. Mississippi Power & Light Company letter dated April 11, 1974 regarding seismic design.

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

October 20, 1981

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

SUBJECT: INTERIM REPORT ON GRAND GULF NUCLEAR STATION UNIT 1

Dear Dr. Palladino:

During its 258th meeting, October 15-17, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of the Mississippi Power and Light Company (MP&L), Middle South Energy, Inc., and the South Mississippi Electric Power Association for a license to operate the Grand Gulf Nuclear Station Units 1 and 2. The units are to be operated by the Mississippi Power and Light Company. A Subcommittee meeting was held in Jackson, Mississippi on September 17-18, 1981 to consider this project. A tour of the facility was made by members of the Subcommittee on September 17, 1981. During its review, the Committee had the benefit of discussions with representatives of the Applicant, 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 this station in its report dated May 15, 1974.

The Grand Gulf Station is located in Claiborne County, Mississippi on the east side of the Mississippi River about 25 miles south of Vicksburg, the nearest city having a population in excess of 25,000 persons.

Each Grand Gulf unit is equipped with a General Electric BWR-6 nuclear steam supply system with a rated power level of 3833 MWt and a Mark III pressure suppression containment system with a design pressure of 15 psig. Construction of Unit 1 is over 90% complete while Unit 2 is about 20% complete and construction of it has been temporarily suspended.

Because of the extended schedule for Unit 2, the Committee does not believe it appropriate to report on operation of Unit 2 at this time.

The Committee review included the management organization, capability, and operator training of MP&L. This is the first nuclear power plant to be operated by this utility. While the plant staff has a reasonable amount of nuclear background, the ACRS agrees with the NRC Staff on the need for additional personnel with BWR experience, at least during the first year or two of operation. MP&L also needs to fill certain senior technical personnel positions in its management organization. The Committee recommends that the MP&L Nuclear Safety Review Board include two or more experienced voting members from outside MP&L having appropriate backgrounds.

During this meeting, the NRC Staff identified a large number of license conditions and confirmatory matters, and several outstanding issues which remain to be resolved. Except for the two issues identified below, the ACRS is satisfied with progress on the other topics and believes that they should be resolved in a manner satisfactory to the NRC Staff.

We have not completed our review of the following outstanding issues identified in the NRC Staff Safety Evaluation Report:

- dynamic loads on structures above the Mark III suppression pool due to froth impact

- hydrogen control

The ACRS will complete its review of the full power operating license when the Staff and the Applicant have made sufficient additional progress in resolving these items. In the interim, the ACRS believes that if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, it would be acceptable for Grand Gulf Nuclear Station Unit 1 to be operated at power levels up to 5% of full power.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:
1. Mississippi Power and Light Company, "Final Safety Analysis Report, Grand Gulf Nuclear Station Units 1 and 2," Volumes 1-21 and Amendments 25-50
2. U. S. Geological Survey Professional Paper by T. G. Hildenbrand, M. F. Kane, and J. D. Hendricks, "Magnetic Basement in the Upper Mississippi Embayment Region - A Preliminary Report," received August, 1981
3. Report by S. W. Hatch, Sandia National Laboratories and P. Cybulskis and R. O. Wooton, Battelle Columbus Laboratories for Office of Nuclear Regulatory Research, NRC, "The Reactor Safety Study Methodology Applications Program Results for the Grand Gulf #1 BWR Power Plant," NUREG/CR-1659, Vol. 4, SAND80-1897/4, Draft Received 2/6/81

4. Letter from M. D. Houston, Project Manager, Division of Licensing, NRR, to H. Alderman, ACRS, Subject: Staff Responses to Questions asked by ACRS at the Grand Gulf Subcommittee Meeting, September 17-18, 1981, dated October 14, 1981
5. Letters from L. F. Dale, Mississippi Power and Light Company to USNRC, dated August 27, 1981, August 27, 1981, August 26, 1981, August 24, 1981, August 21, 1981, August 21, 1981, August 21, 1981, August 19, 1981, August 18, 1981, August 18, 1981
6. Letter from C. Stewart, Jacksonians United for Livable Energy Policies (JULEP) to R. F. Fraley, ACRS, dated October 8, 1981
7. Letter from K. Lawrence, JULEP, to ACRS Grand Gulf Subcommittee dated September 18, 1981
8. Statement by K. Lawrence, JULEP, to ACRS Grand Gulf Subcommittee dated September 17, 1981
9. Letter from C. Dana, et al., member of public, to ACRS Subcommittee on Reactor Safety dated September 16, 1981
10. Anonymous letter to H. Alderman, ACRS Staff, regarding quality assurance concern, postmarked September 18, 1981

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

August 18, 1982

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

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON GRAND GULF NUCLEAR STATION UNIT 1

During its 268th meeting, August 12-14, 1982, the Advisory Committee on Reactor Safeguards completed its review of the application of the Mississippi Power & Light Company (MP&L), Middle South Energy, Inc., and the South Mississippi Electric Power Association for an operating license for the Grand Gulf Nuclear Station Unit 1. The unit is to be operated by MP&L. The Committee provided an Interim Report, dated October 20, 1981, on the operation of Grand Gulf Unit 1. In completing its review, the Committee had the benefit of Subcommittee meetings on July 29-30, 1982 and on August 11, 1982, discussions with representatives of the Applicant and the NRC Staff, and of the documents listed.

In its Interim Report, the ACRS listed two outstanding issues:

- dynamic loads on structures above the Mark III suppression pool due to froth impact

- hydrogen control

Our Interim Report concluded that, with due consideration to the recommendations of that report and subject to the satisfactory completion of construction, staffing, and preoperational testing, it would be acceptable for Grand Gulf Nuclear Station Unit 1 to be operated at power levels up to 5% of full power.

The NRC Staff has stated that the matter of dynamic loads on structures above the suppression pool is now resolved. The ACRS is satisfied with the resolution of this matter. Since October 1981, several additional detailed questions have been raised concerning suppression pool performance and resulting loads. The Committee has reviewed this matter and is satisfied with the manner in which the NRC Staff is handling the questions involved.

Hydrogen control systems for Mark III containments are being developed by the Hydrogen Control Owners Group. Efforts by the Owners Group are being directed toward the development of a hydrogen ignition system which makes use of distributed ignition sources. In addition, MP&L has performed plant-specific analyses of hydrogen control measures for Grand Gulf. Although some questions remain concerning the optimum number and location of ignitors, the NRC Staff has reached the interim conclusion that MP&L has shown that the hydrogen ignition system will provide reasonable assurance of protection against breach of containment following the generation of a substantial quantity of hydrogen for several significant postulated accident scenarios. We agree with the Staff.

A final evaluation of the hydrogen control system remains to be completed. The ACRS expects to review the final NRC Staff position regarding acceptability of this approach on a generic basis and requests that the NRC Staff arrange for such a review at the appropriate time.

The NRC Staff has indicated that MP&L has an adequately competent staff to operate the Grand Gulf Station when enhanced by supplemental advisory staff with relevant BWR experience. During the first year of operation, the ACRS believes MP&L should continue to strengthen its nuclear plant management and its technical support capability.

MP&L has proposed to include in the Grand Gulf Emergency Procedures a provision for venting the containment in the unlikely event of buildup of pressures above the design basis. The NRC Staff has not completed its review of this proposal. The ACRS wishes to be advised when the NRC Staff has reached a position on this matter and to have an opportunity to comment generically or specifically.

If due consideration is given to the items mentioned above and to those mentioned in our Interim Report of October 20, 1981, the ACRS believes there is reasonable assurance that the Grand Gulf Nuclear Station Unit 1 can be operated at power levels up to 3833 MWt without undue risk to the health and safety of the public.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:
1. Mississippi Power & Light Company, "Final Safety Analysis Report, Grand Gulf Nuclear Station Units 1 and 2," Volumes 1-22, with Amendments 25-52
2. Letters from L. F. Dale, Mississippi Power & Light Company to H. R. Denton, U. S. Nuclear Regulatory Commission:
   a. 7/15/82 regarding Regulatory Guide 1.97 compliance license condition 2.C(23)
   b. 7/15/82 regarding action plans for resolution of Mr. J. H. Humphrey's concerns

3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Grand Gulf Nuclear Station Units 1 and 2," USNRC Report NUREG-0831, dated September 1981 with Supplement No. 1 dated December 1981, Supplement No. 2 dated June 1982, and Supplement No. 3 dated June 1982
4. Letter from Mr. J. M. Humphrey to A. Schwencer, NRC regarding comments on containment design, dated June 17, 1982

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

October 12, 1977

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

SUBJECT: REPORT ON GREENE COUNTY NUCLEAR POWER PLANT

Dear Dr. Hendrie:

During its 210th Meeting, October 6-8, 1977, the Advisory Committee on Reactor Safeguards completed its review of the application of the Power Authority of the State of New York (Applicant) for a permit to construct the Greene County Nuclear Power Plant. A Subcommittee meeting was held in Catskill, New York on September 21, 1977 and the plant site was visited by members of the Subcommittee the same day. The Committee had the benefit of discussions with representatives and consultants of the Applicant, Babcock and Wilcox Company, Stone and Webster Engineering Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

The Greene County Plant will utilize a 3600 MW(t) Babcock & Wilcox pressurized water reactor, enclosed in a steel-lined reinforced concrete containment. The basic design of the Nuclear Steam System (NSS) is similar to designs for the Washington Public Power Supply System Nuclear Projects, WNP 1 and 4, the Bellefonte Nuclear Power Plant, Units 1 and 2 and the Pebble Springs Nuclear Plant, Units 1 and 2 reported on in Committee letters of June 11, 1975, July 16, 1974 and February 11, 1976, respectively. The NSS design is also similar to the 3800 MW(t) Babcock-205 Standard NSS design on which the Committee reported in its letter of August 18, 1977. The balance-of-plant design is similar to the Stone and Webster standard balance-of-plant design for Westinghouse reactors on which the Committee previously reported in its letter of August 18, 1976.

The proposed Greene County Plant will be located on a 190 acre site on the west bank of the Hudson River approximately 35 miles south of Albany, New York and 13 miles north-northeast of Kingston, New York (the nearest population center, 1970 population 25,500). The minimum exclusion distance is 1500 feet from the center of containment and the radius of the low population zone is 2 1/2 miles.

The Applicant and the Staff have agreed on a horizontal ground acceleration of 0.2g for the safe shutdown earthquake and 0.1g for the operating basis earthquake. The Committee considers these values acceptable for this plant.

The 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 Staff. The Committee believes these items can be resolved prior to the issuance of a construction permit.

The Committee has concerns about the substantial quantities of explosives used near the site, and believes this should be given special consideration in the development of security measures.

With regard to generic problems cited in the Committee's report "Status of Generic Items Relating to Light Water Reactors: Report No. 5," dated February 24, 1977, items considered relevant to the Greene County Plant are: II-2, 3, 4, 5 (loose parts monitor resolved), 6, 7, 10; IIA-3, 4, 5, 6, 7; IIB-1, 2; IIC-1, 2, 3, 4, 5, 6; IID-2. These problems should be dealt with by the Staff and the Applicant as solutions are found.

The Advisory Committee on Reactor Safeguards believes that if due consideration is given to the foregoing, the Greene County Nuclear Power Plant can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely,

M. Bender

M. Bender
Chairman

References
1. Greene County Nuclear Power Plant Preliminary Safety Analysis Report, Volumes 1 through 12 and Supplements 1 through 19.

2. Safety Evaluation Report related to construction of Greene County Nuclear Power Plant, NUREG-0283, September 1977.

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

August 13, 1974

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

Subject: REPORT ON THE GREENWOOD ENERGY CENTER, UNITS 2 AND 3

Dear Dr. Ray:

At its 172nd meeting, August 8-10, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application of the Detroit Edison Company for a permit to construct the Greenwood Energy Center, Units 2 and 3. This application had been considered previously during a Subcommittee meeting in Port Huron, Michigan on July 24, 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 Detroit Edison Company, the Bechtel Corporation, the Babcock and Wilcox Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Greenwood Energy Center is located on a 3,620 acre tract in St. Clair County, Michigan about 10 miles inland from Lake Huron and approximately 15 miles northwest of Port Huron, Michigan. An oil-fired electric generating plant is under construction on the site.

The Greenwood Energy Center consists of two nuclear units, each using a Babcock and Wilcox two-loop pressurized water nuclear steam supply system having a design power level of 3600 MW(t). The reactor core will use 205 Babcock and Wilcox Mark C (17 x 17) 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 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 improvement in the capability and reliability of the ECCS. 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. RPS-II incorporates a single-chip central processor unit as a microcomputer for the more complex trip functions. The applicant proposes to qualify this system by a series of environmental, reliability, and *in situ* tests prior to its use in Greenwood 2 and 3. The matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Committee agrees with the position of the Regulatory Staff that the prestressed concrete containment structures for the Greenwood Units are different from those that have been tested previously as prototypes under the provisions of Regulatory Guide 1.18. Unless a similar structure will be tested as a prototype, tests should be made on the containment for Unit 2 in accordance with the requirements of Regulatory Guide 1.18. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant has provided, as an emergency heat sink, an Emergency Cooling Reservoir. The applicant proposes careful control of the compaction procedures for the fill portions of the embankment. The Committee recommends that the compaction specifications should include strength tests as well as *in situ* density tests to assure that the soil strength is adequate.

The Staff analysis of the decay heat removal system proposed by the applicant concluded that it does not meet the single failure criterion. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Committee believes the applicant and the Regulatory Staff should continue to review Greenwood Units 2 and 3 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."

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 Greenwood in this regard. The Committee recommends that resolution of this matter be expedited. The Committee wishes to be kept informed.

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 Greenwood Energy Center, Units 2 and 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,

W. R. Stratton

W. R. Stratton
Chairman

References

1. The Detroit Edison Company Application for Construction Permit for the Greenwood Energy Center, Units 2 and 3, with Preliminary Safety Analysis Report, Vols. 1-7 (Vols. 8 and 9 received with subsequent Amendments to the Application)
2. Amendments 1-8, 10, 11 to the Application
3. Directorate of Licensing letter, July 17, 1974, transmitting Safety Evaluation Report

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

January 13, 1966

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

Subject: REPORT ON THE GROUND TEST REACTOR (GTR)

Dear Dr. Seaborg:

At its sixty-ninth meeting, January 6-8, 1966, the Advisory Committee on Reactor Safeguards reviewed the proposed increase from 3 to 10 MW(t) in the power level of the Ground Test Reactor (GTR). During the review, the Committee had the benefit of the documents listed below and of discussions with representatives of General Dynamics-Fort Worth, and the AEC Regulatory Staff. A visit to the reactor facility was made by a member of the Committee on November 17, 1965. An ACRS Subcommittee meeting was held in Washington, D. C. on December 10, 1965.

The GTR is a light-water moderated and cooled pool-type reactor utilizing MTR-type fuel elements. It is operated by General Dynamics-Fort Worth for the U. S. Air Force as part of the Nuclear Aerospace Research Facility (NARF). The GTR has been used to carry out research programs for the Air Force and Army and, most recently, radiation-effects experiments at cryogenic temperatures in support of NASA's nuclear rocket engine (NERVA) program. Initial operation of the GTR was begun in 1952 with a maximum power level of 10 KW. The maximum power level was progressively increased to its present level of 3 MW(t) during the period from 1952 to 1957.

Modifications to be made to the GTR to accommodate the higher power level include: modifications to accommodate new control rods of increased worth, adjustment of reflector geometry to equalize flux in the three available irradiation positions, increase in cooling system capability, installation of a liner to protect against possible pool leakage in the event of an accident, installation of a test-cell ventilation-and-filter system, and incorporation of an additional start-up channel. In addition, means were described for providing redundancy in the scram circuits of the safety system so that a single line fault could not cause loss of scram capability. The General Dynamics representatives stated that procedures and equipment would be provided for periodic testing to determine that redundant circuits were operating as designed.

General Dynamics presented analyses showing that the proposed operation of GTR would not introduce unacceptable radiation doses to the public under normal or accident conditions including any effects resulting from a possible hydrogen detonation.

The Committee concludes that the GTR can be operated at power levels up to 10 MW(t) as proposed without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

David Okrent
Chairman

References.

1. 10 Mw GTR Hazards Summary, FZK-241, dated April 30, 1965.
2. Supplement to 10-Mw GTR Hazards Summary, dated September 10, 1965.
3. Supplement No. 2 to 10-Mw GTR Hazards Summary, dated September 24, 1965.
4. Supplement No. 3 to 10-Mw GTR Hazards Summary, dated November 29, 1965.
5. Additional Information Concerning 10-Mw GTR Hazards Summary, dated December 22, 1965.
6. General Dynamics Letter to Division of Reactor Licensing, dated September 11, 1965.
7. General Dynamics Letter to Division of Reactor Licensing, dated September 28, 1965.
8. General Dymamics Letter to Division of Reactor Licensing, dated November 30, 1965.
9. Special Safety Study Report on the Operation of the GTR at 10 Megawatts, USAF NRSSG 65-1, dated July 1965.

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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: HALLAM NUCLEAR POWER FACILITY (HNPF)

Dear Mr. McCone:

The Advisory Committee on Reactor Safeguards considered the design of the Hallam Nuclear Power Facility at its Seventeenth Meeting on July 24, 1959.

The proposed design is described in reports cited below. The Committee has had the benefit of meetings with the contractor, Atomics International, at its March and July 1959 meetings, a visit by a Subcommittee to the SRE, a prototype at Santa Susanna, meeting with the contractor there, and the Hazards Evaluation Branch analysis and discussion.

This is a 240 thermal megawatt, sodium-graphite power reactor to be located in a sparsely settled region of southeastern Nebraska. It is similar in design to the 20 thermal megawatt SRE at Santa Susanna. This prototype has been operated by the contractor without serious difficulty.

The system is contained in a number of interconnected steel lined concrete cavities believed by the Committee to be capable of containing fission products that might be released accidentally. An improved filter system for collecting radioactive fumes from a sodium fire, should one occur, will be required.

Considering SRE experience, the isolated site with adequate exclusion distance, and the design proposed, the Committee believes that this proposed reactor 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. Rogers McCullough
Chairman

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

<u>References:</u>

1) NAA-SR-3379 - Preliminary Safeguards Report Based on Uranium-Molybdenum Fuel for the Hallam Nuclear Power Facility, issued on February 10, 1959.

2) NAA-SR-3379 - Supplement I - Supplement to the Preliminary Safeguards Report Based on Uranium-Molybdenum Fuel for the Hallam Nuclear Power Facility, April 1959.

3) NAA-SR-MEMO-4067 - Additional Safeguards Evaluation for the U-Mo Fueled Core of the Hallam Nuclear Power Facility, issued July 7, 1959.

4) Division of Licensing and Regulation Report to the ACRS on the Hallam Nuclear Power Facility, February 27, 1959.

5) Division of Licensing and Regulation Report to the ACRS on the Hallam Nuclear Power Facility, April 28, 1959.

6) Division of Licensing and Regulation Report to the ACRS on the Hallam Nuclear Power Facility, July 6, 1959.

7) Office Memorandum from the Director of the Division of Biology & Medicine on the Hallam Nuclear Power Facility, July 17, 1959.

February 8, 1960

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

Subject: HALLAM NUCLEAR POWER FACILITY (HNPF)

Dear Mr. McCone:

At its twenty-third meeting on January 28-30, 1960, the Advisory Committee on Reactor Safeguards considered the Hallam Nuclear Power Facility (HNPF). A letter was addressed to you on this subject July 25, 1959.

This letter indicated that a portion of its safety evaluation was based upon the Sodium Reactor Experiment experience. In December 1959, Report NAA-SR-4505, "Safeguards Evaluation of Recent SRE Experience Applicable to HNPF," was distributed to the Committee. This report has been reviewed by an ACRS Subcommittee in conjunction with a review prepared by the Hazards Evaluation Branch. It appears the SRE experience has been utilized to produce new design features which will be incorporated in the HNPF.

Based upon the report of the ACRS Subcommittee and the review of the HEB, the ACRS considers no revision of its opinion relative to the construction permit as recorded in the letter of July 25, 1959, is required at this time.

Sincerely yours,

/s/

Leslie Silverman
Chairman

cc:A.R.Luedecke, GM
W.F.Finan, OGM
HH.L.Price,DL&R
ACRS Members & Dr. Duffey
bc: L.K.Olson, GC

References
NAA-SR-4504 - Safeguards Evaluation of Recent SRE Experience Applicable to HNPF (undated)
DL&R Report to the ACRS ON Hallam Nuclear Power Facility, January 12, 1960

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

October 28, 1961

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

Subject: REPORT ON HALLAM NUCLEAR POWER FACILITY - DRY CRITICAL EXPERIMENTS

Dear Dr. Seaborg:

At its thirty-seventh meeting on October 26-28, 1961, the Advisory Committee on Reactor Safeguards reviewed the application of Atomics International to conduct dry critical and dry excess loading tests at this facility. These are critical experiments with no fission product build-up. The reports listed below were available. Atomics International representatives and the AEC staff participated. An ACRS subcommittee visited the plant on August 4, 1961.

It is the opinion of the Committee that the dry critical and the excess loading tests may be conducted without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

T. J. Thompson
Chairman

References:
(Attached)

References:

1. NAA-SR-5700, Final Summary Safeguards Report for the Hallam Nuclear Power Facility, issued April 15, 1961.

2. NAA-SR-5700, Errata, issued July 21, 1961.

3. 61AT4121, letter to USAEC from Atomics International, Items for Safeguards Report, dated May 17, 1961.

4. 61AT4094, letter to USAEC from Atomics International, HNPF Reactor Vessel Bellows, with attachments, dated May 16, 1961.

5. NAA-SR-5700, Supplement 1, Safeguards Report on Dry, Zero-Power Experiments in HNPF, issued September 22, 1961.

6. NAA-SR-5700, Revision for Section 5, issued September 1961.

7. NAA-SR-5700, Supplement 2, Additional Information on Dry, Zero-Power Experiments in HNPF, issued September 1961.

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

November 1, 1961

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

Subject: REVIEW OF CLEAN CRITICAL EXPERIMENTS

Dear Dr. Seaborg:

Under the provisions of Section 29 of the Atomic Energy Act of 1954, as amended, the Advisory Committee on Reactor Safeguards is charged with the responsibility of reviewing safety studies and facility license applications referred to it and to advise the Commission in reports with respect to the hazards involved therein.

In discharging this responsibility, the Committee recognizes that the protection of the health and safety of the public should receive the primary attention. Analyses of the consequences of possible reactor malfunctions have shown that the next serious widespread effects are the result of dispersal of fission products. In the case of clean critical experiments in which no significant fission product burden is present, the health and safety of the public is usually not placed in jeopardy. The AEC staff is qualified to judge the adequacy of the precautions taken in critical facilities and has been taking the responsibility for them without formal reference to this Committee. We suggest that this procedure be extended to clean critical experiments carried out in facilities which will ultimately be used as power or test reactors.

At its thirty-seventh meeting on October 26-28, 1961, the Committee was asked to review the safety of a series of clean critical experiments to be conducted in the Hallam Nuclear Power Facility. While a comment on this application is the subject of a separate letter, it is the opinion of the ACRS that future actions of this type need not be referred to it. The Committee understands that the Commission staff will continue to keep it informed in regard to proposed experiments of this type.

Sincerely yours,

/s/
T. J. Thompson
Chairman

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

February 15, 1962

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

Subject: REPORT ON HALLAM NUCLEAR POWER FACILITY (HNPF) - WET CRITICAL EXPERIMENTS AND OPERATION AT PARTIAL POWER

Dear Dr. Seaborg:

The Advisory Committee on Reactor Safeguards at its thirty-ninth meeting February 8-10, 1962, considered the request of Atomics International to operate the Hallam Nuclear Power Facility.

The reports referenced below were available. Representatives of Atomics International and the AEC staff participated in the presentation.

The Committee letter of October 28, 1961 covered operation of the facility through the dry critical and dry excess loading tests. This phase is now completed.

There remain several technical areas within which data are required to resolve questions propounded by the ACRS and the AEC Regulatory Staff. These significant problems include: fuel handling operations, reactivity coefficients, halogen releases, and primary system testing. The applicant has partially completed work upon these items but some of the data required must be developed through actual operation of the plant during the wet critical and wet excess loading tests, with some reactor operation at a low power level.

The applicant should insure that under all conceivable conditions a negative pressure can be continuously maintained in all areas within which significant releases of radioactivity may occur.

The present stack installation is such that the prevention of overexposure to plant personnel depends on administrative control.

With more detailed consideration of actions to be taken and suitable provisions made for control during emergencies such as fuel handling malfunctions or sodium fires, the wet critical and the wet excess loading tests may be conducted safely. A suitable halogen collection system should be provided before partial power operation, not to exceed 15% of full power, is undertaken. This power level is required to permit completion of primary system testing. With the above provisions in effect, it is the opinion of the ACRS that operations may be conducted without undue hazard to the health and safety of the public.

The Committee believes the AEC Regulatory Staff is fully cognizant of the situation and will continue to follow it closely. The ACRS sees no need for further Committee review until operation above 15% of full power is desired.

Dr. John P. Howe did not participate in the review or discussion of this project.

Sincerely yours,

Sgd/ F. A. GIFFORD, Jr.

F. A. Gifford, Jr.
Chairman

References Attached.

References:
1. NAA-SR-5700, Supplement 3 - Additional Safeguards Information for Hallam Nuclear Power Facility, issued Nov. 1961.
2. NAA-SR-5700, Supplement 4 - Additional Safeguards Information for Hallam Nuclear Power Facility, dtd Dec. 1, 1961.
3. American Air Filter Co., Inc. Report - Project 1544, "Glass Fabric Swatch Tests on Sodium and NaK Fumes for Atomics International", dtd June 9, 1961.
4. Letter-62AT477 from Atomics International to AEC, dtd Jan. 19, 1962 transmitting "HNPF Primary Pipe Tunnel Leak Test".
5. Letter-62AT604 from Atomics International to AEC, dtd Jan. 23, 1962, subject: "HNPF Hot Sodium Circulation Test, AI-P-1167, Rev. to Supplement 4.
6. Letter - 62AT885 from Atomics International to AEC, dtd Feb. 1, 1962 transmitting "Proposed Technical Specifications for Operation at Power for Hallam Nuclear Power Facility", dtd Feb. 2, 1962.
7. Letter - 62AT1019 w/att. from Atomics International to AEC, dtd Feb. 7, 1962, subject: "Report on Final Design of HNPF Dry Scrubber".
8. Letter - 62AT1048 from Atomics International to AEC, dtd Feb. 7, 1962, subject: "Information Presented at Meeting Jan. 9 and 10, 1962, Lincoln, Neb."

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

February 13, 1963

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

Subject: REPORT ON HALLAM NUCLEAR POWER FACILITY

Dear Dr. Seaborg:

At its 46th meeting, January 31 - February 2, 1963, the Advisory Committee on Reactor Safeguards reviewed the report of operation of the Hallam Nuclear Power Facility through the wet critical phase and a partial power phase. The proposal to operate up to full power was considered. Representatives of Atomics International, the Consumers Public Power District, and AEC staff participated in the discussion. The reports listed were available.

The Committee letter of February 15, 1962 covered the proposed operation up to 15% of full power. In this letter several questions in technical areas were cited upon which data were required. It appears that these questions have been resolved satisfactorily.

During low power operation, below 20 MWt, the following problems have appeared: (a) possible carburization of stainless steel due to carbon in the sodium; (b) fuel element orifice defects; (c) helium entrainment occurring in the secondary loops and in the primary pumps; and, (d) high sodium oxide content in the coolant. It appears the foregoing have been or will be controlled or corrected satisfactorily.

Leakage has appeared in one tube in an intermediate heat exchanger. This leak is so recent that data as to cause are not yet available. The tube has been removed for study and analysis and the tube sheet plugged. Problems of leakage in heat exchangers are common in industrial practice. Leakage in intermediate heat exchangers of liquid metal reactor systems is a cause for some concern since conceivably radioactive sodium could be released into the atmosphere. The applicant and the Regulatory Staff are conducting studies which should determine the magnitude of this problem and develop adequate measures for its solution. The Committee is of the opinion that such measures, together with appropriate liquid level sensors and alarms such as the applicant has installed, coupled with operational vigilance, will afford adequate protection from a safety standpoint.

It was reported that no nuclear problems appeared during the wet critical phase or subsequent operation up to a power level of 20 MWt or 8% of full power. The reactor reached 15% of full power on January 30, 1963. Operation at this power level is planned to continue for approximately 30 days.

The Advisory Committee on Reactor Safeguards believes that, if continued operation at the 38 MWt power level produces no additional problems which are not resolved to the satisfaction of the Regulatory Staff, operation of the reactor up to full power level (256 MWt) may be conducted without undue hazard to the health and safety of the public.

Dr. John P. Howe did not participate in the Committee's consideration of this project.

Sincerely yours,

/s/

D. B. Hall
Chairman

References:

1. Letter 62AT1032 dated February 19, 1962, subject: "HNPF Primary--Intermediate Heat Exchanger Cells Leakage Tests", w/enclosures.
2. Letter 62AT1853 dated March 12, 1962, with two enclosures: AI-P-1155, "Preoperational Test Completion Report, Dry Criticality; and AI-P-1163, "Preoperational Test Interim Report, Dry Excess Loading."
3. Letter 62AT1869 dated March 9, 1962, subject: "Additional Information for Safety Review of HNPF," w/enclosures, 3 drawings, D-793575, D-795188, D-79306.
4. Letter 62AT2027 dated March 16, 1962, subject: "Safety Review of HNPF - SRE Experience."
5. Letter 62AT2028 dated March 16, 1962, subject: "Safety Review of HNPF - Building Exhaust System High Efficiency Filters."

References: HALLAM NUCLEAR POWER FACILITY

6. Letter 62AT2029 dated March 16, 1962, subject: "Safety Review of HNPF--Reactivity Coefficients."
7. Letter 62AT2182 dated March 22, 1962, subject: "Safety Review of HNPF -- Building Exhaust System Halogen Removal Unit."
8. Letter 62AT2714 dated April 11, 1962 transmitting "Additional Errata for Final Summary Safeguards Report for the HNPF," dated March 1, 1962.
9. Letter 62AT2799 dated April 17, 1962 w/enclosures: Supplement I and Errata for Dry Excess Loading, AI-P-1163.
10. Letter 62AT2800 dated April 20, 1962 w/enclosures.
11. Letter 62AT2663 dated April 23, 1962 w/enclosures: "HNPF - Primary Service and Fill Tank Test Reports."
12. Letter 62AT3090 dated April 27, 1962, subject: "Safety Review of HNPF - Primary Piping Inspection."
13. Letter 62AT3152 dated April 25, 1962, subject: "Safety Review of HNPF - Steam Generator Room Nitrogen."
14. Letter 62AT3585 dated May 16, 1962, subject: "HNPF Reactor Cavity Test Report."
15. Letter 62AT3542 dated May 10, 1962, subject: "HNPF Sodium Draining and Source Relocation Tests."
16. Letter 62AT3414 dated May 15, 1962, subject: "HNPF Steam Generator Feedwater Line Leak Basin."
17. Letter 62AT4356 dated July 7, 1962 transmitting "AI-P-1167, HNPF Preoperational Test Completion Report, Hot Sodium Circulation Test."
18. Letter 62AT5964 dated August 14, 1962, subject: "HNPF Technical Specifications."
19. Letter 62AT5587 dated August 21, 1962, additional information to 62AT2182 dated March 22, 1962.
20. Letter 62AT8022 dated November 26, 1962, subject: "Modifications to HNPF," w/enclosures as indicated.
21. Letter 62AT8386 dated November 26, 1962, subject: HNPF Low Power Testing and Future Plant Surveillance," w/enclosure.
22. Letter 62AT8412 dated November 27, 1962, subject: "HNPF Testing."
23. Letter 62AT8465 dated November 30, 1962, subject: "HNPF Test Summaries," w/enclosure.
24. Letter 62AT8468 dated December 4, 1962, subject: "HNPF Zero Power Test Summary," w/enclosure.
25. Letter 63AT25 dated January 14, 1963, subject: "Summary of HNPF Low Power Test Results," w/enclosure.
26. TWX dated January 22, 1963 re Primary System Sodium Purity.
27. Letter 63AT27 dated January 21, 1963, subject: "Errata for Enclosures Atomics International letter 63AT25, dated January 14, 1963," w/enclosures.

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

October 14, 1963

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

Subject: REPORT ON HALLAM NUCLEAR POWER FACILITY

Dear Dr. Seaborg:

At its forty-ninth meeting, September 5 and 6, and at its fiftieth meeting on October 10 and 11, 1963, the Advisory Committee on Reactor Safeguards considered the application of the Consumers Public Power District to assume the operating responsibility for the Hallam Nuclear Power Facility. The Committee's letter of February 13, 1963 commented on the full power operation of this reactor by Atomics International. In the present review, the Committee had the benefit of discussions with representatives of the Consumers Public Power District, Atomics International, and the AEC staff. In addition, the documents listed were available.

It was reported that no unsolved nuclear or mechanical problems have developed during all operational test phases including operation at full power. This reactor was originally designed to operate with uranium-molybdenum alloy fuel elements. Tests of developmental uranium carbide fuel elements are underway with ten elements of this type now in the core. The ACRS will be interested in the result of this experimental program and assumes that its progress will be reviewed by the AEC Regulatory Staff.

In the area of nuclear plant operation, it appears to the Committee that the operating organization still contains only a minimum of fully qualified supervisory personnel. The Committee believes that the personnel situation as now described by the applicant should be considered as the minimum acceptable.

Assuming that at least the minimal requirement of trained competent personnel will always be maintained, it is the opinion of the Advisory Committee on Reactor Safeguards that this reactor can be operated by Consumers Public Power District without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

D. B. Hall
Chairmam

References:
1. Letter from Wilson & Barlow, dated April 24, 1963, subject: "Hallam Nuclear Power Facility, Docket 115-3, Application for Assignment of Operating Authorization to Consumers Public Power District", with enclosures.
2. Letter from Wilson & Barlow, dated July 24, 1963, subject: "Hallam Nuclear Power Facility, Docket 115-3, Amended Application for Operating Authorization", with enclosures.
3. Letter from Wilson & Barlow, dated September 20, 1963, subject: "Hallam Nuclear Power Facility, Docket 115-3, Amendments to Amended Application for Operating Authorization", with enclosures.
4. Letter from Wilson & Barlow, dated September 30, 1963, subject: "Hallam Nuclear Power Facility - Docket 115-3, Amendment to Amended Application for Operating Authorization", with enclosures.

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

October 19, 1972

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

Subject: REPORT ON HANFORD NO. 2 NUCLEAR POWER PLANT

Dear Dr. Schlesinger:

At its 150th meeting, October 12-14, 1972, the Advisory Committee on Reactor Safeguards reviewed the application of the Washington Public Power Supply System for a permit to construct the Hanford No. 2 Nuclear Power Plant. This project was considered at a Subcommittee meeting on September 30, 1972, at the plant site. During its review, the Committee had the benefit of discussions with representatives of the Washington Public Power Supply System, Burns and Roe, Incorporated the General Electric Company, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed.

The Hanford No. 2 Plant will be located in the State of Washington on the U. S. Atomic Energy Commission's Hanford Reservation, three miles west of the Columbia River and approximately 12 miles north of Richland, Washington, the nearest population center (1970 population 26,290). The makeup water intake structure will be located on the west bank of the river. The low population zone (LPZ) radius is three miles and the minimum exclusion area radius is 1.2 miles. Both of these areas are within the Hanford Reservation and have zero permanent population. The Fast Flux Test Facility will be the only installation within the LPZ. It has an expected normal day shift of about ninety persons. By 1980, the resident population is projected to be 528 within 10 miles.

The Hanford No. 2 Plant will utilize a General Electric boiling water reactor to be operated at power levels up to 3323 MW(t). It is of a design similar to that of the LaSalle County Station units, previously approved for construction.

Waste heat is to be rejected to the atmosphere by mechanical draft cooling towers to which makeup water will be supplied from the Columbia River. Two seismic Category I spray ponds will be provided and will have sufficient capacity to maintain the plant in a safe shutdown condition for 30 days independent of water makeup.

The containment system includes the primary containment which utilizes the pressure suppression concept, and secondary confinement provided by a low-leakage reactor building. The primary containment consists of a conical drywell and cylindrical wetwell, the two separated by a reinforced concrete floor penetrated by 102 vent pipes. The entire structure is a free-standing steel pressure vessel designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The applicant is developing a seal design for the peripheral joint between the drywell floor and the steel containment in order to preclude deck bypass leakage which would affect the pressure suppression capability of the containment system. Four vacuum breakers provide a return flow path from the suppression chamber to the drywell. These provide another potential bypass path which could impair the performance of the pressure suppression system. The design of both the seal and vacuum breakers should be such as to avoid excessive bypass leakage. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

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.

Active pumps and valves of the reactor coolant pressure boundary required to perform safety functions will be designed to deformation limits for which the calculated primary stresses will be in the elastic range. Acceptable design criteria for inactive pumps and valves are yet to be established. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant has proposed to install a sealing system to ensure minimal leakage through the main steam line isolation valves following a postulated loss-of-coolant accident and has in progress a study to establish the design of such a system. The Committee believes that a sealing system should be installed. This matter should be resolved in a manner satisfactory to the Regulatory Staff prior to completion of construction of the plant.

Analyses of postulated control-rod drop accidents occurring in similar cores during certain portions of the fuel cycle indicate unacceptable results. Studies of provisions to reduce the probability of this accident to negligible levels are underway. This matter should be resolved in a manner satisfactory to the Regulatory Staff prior to completion of construction.

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 for a suitable backup to the control rod system for this type of event. The Committee believes that this approach represents a substantial improvement and should be provided for the Hanford No. 2 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 plant.

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 Hanford No. 2 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, Hanford 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/ C. P. Siess

C. P. Siess
Chairman

References

1. Washington Public Power Supply System letter dated August 10, 1971 transmitting PSAR, Volumes 1 through 6 to Hanford No. 2 Nuclear Power Plant

2. Amendments 1 through 9 and 12 to the License Application for Hanford No. 2 Nuclear Power Plant

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

March 8, 1972

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

Subject: REPORT ON SHEARON HARRIS NUCLEAR POWER PLANT

Dear Dr. Schlesinger:

At its 143rd meeting, March 2-4, 1972, the Advisory Committee on Reactor Safeguards considered the proposal of the Carolina Power and Light Company for a single review of its application to construct four reactors at the Shearon Harris Nuclear Power Plant site. The applicant's request was considered at a Subcommittee meeting on February 23, 1972, in Washington, D. C. During these meetings, the Committee had the benefit of discussions with the applicant and his consultants, and with the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The Carolina Power and Light Company proposes to build the Shearon Harris Units 1, 2, 3, and 4 at a location about 20 miles from Raleigh, North Carolina. Each unit will have a core thermal power output of 2775 MWt. The four pressurized water nuclear units will be similar to the Virgil C. Summer reactor, now under review.

The applicant stated that the four reactors and the associated auxiliary structures and components will be arranged in a compact plan which requires almost simultaneous construction of foundations and sequential, but closely coupled, construction schedules for the four units. It is planned that the four units will go into operation at one-year intervals during the period 1977-1980.

For multiple, sequentially constructed units, such as proposed for the Shearon Harris plant, a considerably longer than normal period exists between issuance of the construction permit and the beginning of operation of the final unit. The Committee reiterates its belief that, at the time of completion of the construction permit review, there should be a minimum number of problems, the proper resolution of which could be affected significantly because construction and major component procurement had proceeded too far. In response to the Committee's concern, the applicant has made the following statement with regard to inclusion of new developments affecting plant safety:

> "Carolina Power and Light Company recognizes that during the period of the pre and post construction permit there may be developments which further enhance the safety of nuclear power plants. We wish to emphasize that CP&L will incorporate AEC required safety improvements in these units, although we may suffer a schedule penalty in so doing. We will also actively evaluate the feasibility of incorporating other significant improvements which may not be AEC requirements. Furthermore, we wish to strongly emphasize that our 1979 and 1980 units <u>will</u> represent the same quality of safety incorporated in other units which become operational during that time period."

Subject to the above, the Advisory Committee on Reactor Safeguards has no objection to conducting a single review of the application to construct the four units of the Shearon Harris Nuclear Power Plant.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

References:

1. Carolina Power and Light Company letter dated September 7, 1971; License Application dated June 3, 1971, Preliminary Safety Analysis Report, Volumes 1 through 5

2. Carolina Power and Light Company letter dated January 12, 1972; re: Appropriateness of single review proceeding for issuance of CP for four units

3. Carolina Power and Light Company letter dated February 9, 1972; Amendment No. 1 to PSAR dated February 9, 1972

4. Carolina Power and Light Company letter dated February 15, 1972; Providing additional information re: appropriateness of single review proceeding for issuance of CP for four units

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 SHEARON HARRIS NUCLEAR POWER PLANT, UNITS 1, 2, 3, AND 4

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 Carolina Power and Light Company for a permit to construct the Shearon Harris Nuclear Power Plant, Units 1, 2, 3, and 4. The project was considered at Subcommittee meetings held at the plant site on October 19, 1972, and in Washington, D. C. on January 10, 1973. During its review, the Committee had the benefit of discussions with representatives and consultants of Carolina Power and Light Company, the Westinghouse Electric Corporation, Ebasco Services Incorporated, and the AEC Regulatory Staff, and of the documents listed. The Committee reported to the Commission on March 8, 1972, its acceptance of the applicant's proposal for a single review of the application to construct four reactors at the Shearon Harris Nuclear Power Plant site.

In its report of March 8, 1972, the Committee noted that, in response to its concern, the applicant emphasized that his 1979 and 1980 units (representing Units 3 and 4) "will represent the same quality of safety incorporated in other units which beco operational during that time period." The Committee believes that the Regulatory Staff should follow closely the development of design details and the construction of the Shearon Harris plant so that appropriate improvements in safety-related systems can be incorporated in a timely manner.

The Shearon Harris Nuclear Power Plant is to be located in a sparsely populated region in Wake County, North Carolina, about 16 miles southwest of Raleigh (population 124,000). The exclusion radius is to be 7000 feet (2133 meters) and the low population zone radius has been selected to be three miles. The applicant stated that an underground liquefied petroleum gas pipeline which now traverses the proposed exclusion area will be relocated to be outside the exclusion area.

Each of the four units for this 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 essentially identical to the 3-loop Westinghouse system to be provided for the Summer Nuclear Station, Unit 1 which has slightly higher reactor power, coolant flow rate, and coolant temperature than previous 3-loop Westinghouse systems and on which the Committee has reported recently. 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 main reservoir of approximately 10,000 acres with a normal average depth of approximately 27 feet which will be created by constructing a Seismic Category I earthen dam on Buckhorn Creek about 2½ miles north-northeast of its confluence with the Cape Fear River. The main reservoir will serve as the principal source of plant cooling water. The Cape Fear River will be used as a supplemental source, when necessary. An auxiliary reservoir is to be formed by constructing a Seismic Category I dam across an arm of the main reservoir adjacent to the plant site. The auxiliary reservoir will serve as an emergency source of service water. Design details of these dams and related spillways are under development and should be reviewed by the Regulatory Staff prior to construction.

The applicant plans to design the Shearon Harris Nuclear Power Plant to withstand a bedrock acceleration of 0.15 g for the safe shutdown earthquake (SSE) and an acceleration of 0.075 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 these reactors, 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, 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 the Shearon Harris Plant, 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 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 on the local performance characteristics of high power density cores. 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 flexibility should be retained.

The Committee finds that the applicant's estimates of the probability of generation of large high-energy missiles, in the unlikely event of a turbine failure, are significantly smaller than those that others have derived from existing world experience. The Committee believes that the applicant has not, as of now, demonstrated that the probability of an intolerable accident arising from turbine missile generation is acceptably low, and recommends that, unless the applicant can demonstrate this probability to be acceptably low, further measures both to reduce the probabilities and the potential consequences of turbine missile generation be studied and implemented. Analytical and experimental work on the penetration of reinforced concrete by missiles of the type of interest is an example of the kinds of data important to evaluation of this problem.

The Committee believes that protection against pipe whip should be provided in accordance with criteria being developed by the AEC Regulatory Staff.

The applicant has proposed criteria for means to mitigate safety-related consequences of a possible main steam line or feedwater line rupture outside the containment building. This matter should be resolved in a manner satisfactory to the Regulatory Staff; 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 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.

Emergency onsite a-c power for this plant will be provided by two sets of diesel-driven generators -- one set assigned to Units 1 and 2 and the other set to Units 3 and 4. Each set would consist of three diesel generators, one for each unit and one to be shared such that its power can be directed to either unit. The applicant stated that he proposes to proceed on this basis for Units 1 and 2 but that he will maintain flexibility to make modifications to the design of the onsite emergency a-c system for Units 3 and 4 pending formulation of AEC criteria for sharing of electrical systems of multi-unit plants. The Committee believes that this approach is satisfactory. The details of the onsite power system for all four units should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant reported that the criteria for design of safety-related items in the instrumentation and control system will meet the requirements of IEEE-279 (1971). Details should be resolved in a manner satisfactory to the Regulatory Staff.

The Committee believes it desirable for the applicant and the Regulatory Staff to review further the Shearon Harris Nuclear Power Plant for design features, in accordance with Safety Guide No. 17, that should reduce the possibility of sabotage.

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 Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items, the Shearon Harris Nuclear Power Plant Units 1, 2, 3, and 4 can be constructed with reasonable assurance that these units can be operated without undue risk to the health and safety of the public.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References attached

<u>References</u> - <u>Shearon Harris</u>

1. Amendments 3-4, 6-10, 12-13, 15, and 17-20 to the License Application

2. Carolina Power & Light Company (CP&L) letter dated October 26, 1972

3. Safety Evaluation by Directorate of Licensing dated December 22, 1972

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

August 19, 1977

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

Subject: REPORT ON SHEARON HARRIS NUCLEAR POWER PLANT

Dear Dr. Hendrie:

During its 208th meeting, August 11-13 1977, the Advisory Committee on Reactor Safeguards completed an updated review of the application of Carolina Power and Light Company for a permit to construct the Shearon Harris Nuclear Power Plant, Units 1, 2, 3, and 4. The application was first reviewed by the Committee in late 1972-early 1973 and reported on in its letter of January 17, 1973. Subsequently (May 8, 1975) the Applicant announced a three to six year delay in the project and an interruption of licensing activities. The principal matters of this review are: (1) the applicability of new significant safety issues to the Shearon Harris plant and (2) the updating of previously reviewed matters to current requirements. These matters had been considered at a Subcommittee meeting with the Staff and the Applicant in Raleigh, N. C. on August 6, 1977, following a site visit the preceding day. The Committee had the benefit of discussions with representatives and consultants of the Carolina Power and Light Company, the Westinghouse Electric Corporation, Ebasco Services, Inc., and the Nuclear Regulatory Commission Staff (Staff). The Committee also had the benefit of the documents listed.

Each Shearon Harris unit will utilize a 2775 MWt three loop Westinghouse pressurized water reactor (with 17x17 fuel assemblies) enclosed in a steel lined concrete containment. The basic design of the nuclear steam supply system is similar to designs used for Virgil C. Summer, Unit 1, reported on in the Committee's letter of November 15, 1972 and Koshkonong Nuclear Plant Units 1 and 2, reported on in ACRS letters of January 15, 1976 and May 12, 1976.

The safe shutdown earthquake acceleration for the Shearon Harris plant is 0.15g and that for the operating basis earthquake is 0.075g.

The Applicant has developed conservative seismic design response spectra and other seismic design bases in agreement with the latest NRC Regulatory Guides. The Staff and the Committee concur that the bases for design of Category I structures, systems and components are appropriate. The Applicant made a comprehensive investigation into the history of movement along the geological fault, discovered in 1974, in the excavation for the Waste Processing Building. Results from a series of diverse radioactive dating methods indicated that the last movement of the fault had occurred a minimum of 2.5-35 million years ago. Based upon other geological considerations, the Applicant concluded that the last movement had occurred at least 150 million years ago. The Staff reviewed the information developed by the Applicant and agreed that the radiometric test results were minimum age assessments. The Staff concluded from other geological considerations that the last movement took place more than 136 million years ago. The Committee concurs with the conclusion of the Applicant and Staff that the fault is not capable.

The Applicant has reviewed the Shearon Harris safety design to assure that design, equipment, materials, fabrication and construction meet or will be upgraded to meet current requirements. Safety systems undergoing major modifications include: reactor core, reactor coolant, emergency core cooling, residual heat removal and waste processing systems, and Category I plant structures. The Applicant and the Staff concur that the Shearon Harris plant, to the extent details have been developed at this stage of the project, conforms to current requirements. Both the Staff and the Applicant need to continue to apply appropriate quality assurance measures to ensure that such compliance continues throughout construction with particular attention paid to problems which could arise as a consequence of the unusual length of construction.

Two safety issues remain to be resolved prior to the Staff recommendation for issuance of a Construction Permit. These issues are confirmation of the "worst case" break for emergency core cooling system performance evaluation and the methodology and acceptance criteria for containment subcompartment analysis.

These matters should be resolved in a manner satisfactory to the Staff.

The Committee believes that the items mentioned above can be resolved during construction.

With regard to generic problems cited in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 5," dated February 24, 1977, items considered relevant to the Shearon Harris Nuclear Power Plant Units 1, 2, 3, and 4 are: II-2, 3, 4, 5, 6, 7, 9, 10; IIA-4, 5, 7; IIB-2; IIC-1, 2, 3, 5, 6; IID-2. These problems should be dealt with by the Staff and the Applicants as solutions are found.

The design and construction of the four units at the Shearon Harris Station will span almost two decades. The commitment by the Applicant to participate in the timely resolution of generic matters identified by the NRC Staff and by the ACRS and the appropriate implementations are of major significance. The ACRS recommends that the Applicant provide the Staff with annual reports on these matters. The reports should include the safety programs in which the Applicant participates, evaluations made to improve reliability and effectiveness of engineered safety features, and design improvements incorporated into the units.

The Advisory Committee on Reactor Safeguards believes that if due consideration is given to the foregoing, the Shearon Harris Nuclear Power 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,

M. Bender

M. Bender
Chairman

References

1. Shearon Harris Nuclear Power Plant Units 1, 2, 3, and 4, Preliminary Safety Analysis Report, Volumes 1-9

2. Amendments 1-62 to the Preliminary Safety Analysis Report

3. Safety Evaluation Report, related to the construction of the Shearon Harris Nuclear Power Plant, Units 1, 2, 3, and 4, Supplement Nos. 1-3.

4. Letter from J. A. Jones, Carolina Power and Light Company to E. Case, U. S. Nuclear Regulatory Commission, on Fault Investigation, Shearon Harris Nuclear Power Plant, Units 1, 2, 3, and 4, dated March 7, 1975.

5. Letter from J. A. Jones, Carolina Power and Light Company to B. C. Rusche, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, concerning responses to NRC questions on the geological fault investigation, dated June 1975.

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

October 11, 1977

Mr. Lee V. Gossick
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: SHEARON HARRIS NUCLEAR PLANT, INQUIRY REGARDING RESOLUTION OF ACRS GENERIC ITEMS

Dear Mr. Gossick:

The ACRS has been informed by the NRC Staff that, during the Shearon Harris pre-hearing conference on June 19, 1977, Atomic Safety and Licensing Board Member Dr. J. V. Leeds, Jr. requested guidance from the ACRS regarding which items in the Committee list of Generic Items for Light-Water Reactors must be resolved prior to the issuance of a construction permit, and which must be resolved after construction permit issuance, but prior to issuance of an operating license.

The Unresolved Generic Items listed by the ACRS have the following characteristics:

a) They are items of concern to the ACRS for which neither the ultimate solution nor its implementation for reactors in various stages of licensing, construction or operation have yet been determined.

b) They are applicable not only to a given plant or license application but also to a class of plants or, in some cases, to all light-water reactors.

In the ACRS review of a particular application, it may be decided that certain of the Generic Items should be resolved prior to issuance of a construction permit or, more likely, prior to operation of the plant. In such cases, a recommendation to this effect is made specifically in the body of the ACRS letter.

Those Generic Items referred to, and now listed explicitly, in the penultimate paragraph of the ACRS letter, are intended to be considered generically, outside the scope of the particular licensing action. It is the intent that, when solutions are found, a determination will be made by the NRC Staff and the ACRS as to their implementation on all plants for which they are applicable and necessary.

Sincerely yours,

M. Bender

M. Bender
Chairman

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

January 16, 1984

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

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE SHEARON HARRIS NUCLEAR POWER PLANT

During its 285th meeting, January 12-14, 1984, the Advisory Committee on Reactor Safeguards reviewed the application of Carolina Power & Light Company (CP&L) and the North Carolina Eastern Municipal Power Agency (the Applicants) for an operating license for the Shearon Harris Nuclear Power Plant. The Shearon Harris Nuclear Power Plant will be operated by CP&L which also operates three other nuclear units. The project was considered during an ACRS Subcommittee meeting in Apex, North Carolina on January 3-4, 1984. Members of the Subcommittee toured the facility on January 3, 1984. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, Westinghouse Electric Corporation, Ebasco Services, Inc., the NRC Staff, and a member of the public. The Committee also had the benefit of the documents referenced. The Committee commented on the application for a permit to construct the Shearon Harris Plant in reports dated March 8, 1972, January 17, 1973, and August 19, 1977. On October 11, 1977 the Committee provided a response to an inquiry regarding the resolution of ACRS Generic Items related to the Shearon Harris Nuclear Power Plant.

The Shearon Harris Nuclear Power Plant is located in Wake County, North Carolina, approximately 16 miles southwest of the nearest boundary of Raleigh, North Carolina. Originally the Shearon Harris Nuclear Power Plant was to comprise four units. However, only Unit 1 will be completed, with an estimated fuel load date of June 1985. Units 3 and 4 were cancelled on December 18, 1981 and Unit 2 was cancelled on December 21, 1983.

The Shearon Harris Nuclear Power Plant uses a three-loop Westinghouse nuclear steam supply system with a rated core power of 2775 MWt. The containment is a large, dry, reinforced concrete structure.

During the Committee's consideration of this plant, the control room design was reviewed. The Applicants informed us that they intend to perform an operational test of the control room emergency air recirculation system. As a part of this exercise, control room habitability during the recirculation mode will be evaluated. We wish to be kept informed.

The Shearon Harris Nuclear Power Plant uses Westinghouse D-4 steam generators. Steam generators of this design have experienced tube degradation related to flow-induced vibrations in the preheater region. Internal modifications have been developed by Westinghouse which include expanding some steam generator tubes and directing some of the main feedwater flow through the auxiliary feedwater nozzle. We expect to be kept informed regarding the operating experience of these steam generators.

The NRC Staff has previously identified management deficiences in CP&L's nuclear program. These deficiencies are enumerated in the report (May 1983) of the most recent Systematic Assessment of Licensee Performance (SALP) conducted by the NRC Staff to assess CP&L's nuclear operations for the period January 1982 - January 1983. CP&L has taken measures to improve management function and capability. These include restructuring of the corporate organization which will eventually result in a consolidation of CP&L's nuclear organization under one senior manager. The restructuring also provides for a corporate level executive to be located onsite, as a member of involved site management, to ensure greater access to resources and to enhance the ability to initiate new programs from the site. These efforts are expected to correct the past deficiencies. Members of the Region II Staff reported orally during the meeting that significant improvement in performance has been observed since the last SALP inspection. The Committee believes that written evidence of an improvement in CP&L's nuclear operations, which could, for example, be reported in the two scheduled SALP reviews prior to fuel load should be available prior to full power operation. We wish to be kept informed.

Subsequent to the meeting with the Applicants, we have received a letter from a member of the public which makes several allegations concerning quality assurance and other issues. We request that the NRC Staff investigate these allegations and provide a written report to the Committee.

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. In a letter dated January 11, 1983, the ACRS made recommendations concerning a possible broad approach to deal generically with the question of seismic margins. In the meantime, for the Shearon Harris Nuclear Power Plant, we recommend that, in addition to items already considered, specific attention be given to assurance of adequate seismic capability of the emergency AC power supplies, the DC power supplies, and small components such as actuators and instrument lines that are important to the accomplishment of safe shutdown and decay heat removal. We suggest also that specific attention be given to the adequacy of clearances between adjacent buildings.

During this review there was discussion of the reliability and the fracture resistance of the chilled water system. The Applicants and the NRC Staff reported orally that the system is satisfactory in these respects. The ACRS would like to receive a detailed discussion of the chilled water system in a supplement to the Safety Evaluation Report.

One of the confirmatory issues concerning this application is "turbine missiles." Because of the nonoptimum orientation of the turbine relative to vital components in this plant, we recommend that a structured test program for evaluating overspeed protection of the turbine be prepared and submitted to the NRC Staff for review and approval before full power operation.

A number of 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.

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 Shearon Harris Nuclear Power Plant can be operated at core power levels up to 2775 MWt without undue risk to the health and safety of the public.

Sincerely,

Jesse C. Ebersole
Chairman

References:

1. Carolina Power & Light Company, "Shearon Harris Nuclear Power Plant Units 1, 2, 3, and 4, Final Safety Analysis Report," Volumes 1-20 and Amendments 1-10
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Shearon Harris Nuclear Power Plant, Units 1 and 2," USNRC Report NUREG-1038, dated November 1983
3. Letter from Wells Eddleman, Intervenor, Subject: Comments on the Shearon Harris Nuclear Power Plant to the Advisory Committee on Reactor Safeguards, dated January 13, 1984

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

MAY 13, 1976

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON HARTSVILLE NUCLEAR PLANTS UNITS A-1, A-2, B-1, AND B-2

Dear Mr. Rowden:

At its 193rd meeting, May 6-8, 1976, the Advisory Committee on Reactor Safeguards reviewed the application of the Tennessee Valley Authority (TVA) for a license to construct the Hartsville Nuclear Plants Units A-1, A-2, B-1, and B-2. This application was previously reviewed at a Subcommittee meeting in Nashville, Tennessee on April 23, 1976, subsequent to a visit to the site on April 22. The Committee also had the benefit of discussions with representatives and consultants of the Tennessee Valley Authority, the Nuclear Regulatory Commission Staff, and the General Electric Company, statements by area residents, and the documents listed.

The Hartsville Nuclear Plants consist of four 3579 MWt reactors of the GESSAR-238 design which uses a BWR-6 boiling water reactor with a Mark III containment. Preliminary design approval for GESSAR-238 (PDA-1) was issued December 22, 1975. This is the first use of a PDA as part of a Construction License Application. PDA-1 covers the nuclear island which consists of the nuclear steam supply system, the reactor building, and associated facilities. The Tennessee Valley Authority will design the turbine island portion and other installations external to the nuclear island for the Hartsville Plants.

The plants will be located in Trousdale and Smith Counties in North Central Tennessee, approximately 40 miles east northeast of Nashville and approximately five miles southeast of Hartsville, Tennessee (1970 population 2,243). The site consists of approximately 1,940 acres on the north bank of the Old Hickory Reservoir of the Cumberland River. The minimum exclusion area distance measured from the edge of the reactor building nearest the site boundary is approximately 4,000 ft. The low population zone has a radius of three miles and includes a population of 625 persons. The nearest population center is Nashville, Tennessee (1970 Metropolitan population 887,000).

In its March 14, 1976, report on GESSAR-238 for the PDA, the ACRS identified four items requiring further consideration by the Committee. Of these, only the matter related to continuous venting of the containment remains to be resolved by the NRC Staff.

The Committee believes that the Applicant and the NRC Staff should review the Hartsville Plants for design features that could significantly reduce the possibility and consequences of sabotage, and that such features should be incorporated into the plant design where practicable. The Committee wishes to be kept informed.

The matter of suitable design loadings for the Mark III containment has been a continuing concern of the Committee and the NRC Staff. The Staff has reviewed the ongoing tests being made by the General Electric Company and has specified what it believes to be loadings that are sufficiently conservative to allow for the uncertainties in the empirical and limited knowledge now available. The ACRS believes that this approach is acceptable at this stage of design and construction, but urges that the tests being made by the General Electric Company should be continued and, if necessary, accelerated in order to assure that the hydrodynamic phenomena important to the design of the Mark III containment will be understood and defined more completely before operation of the first of the Hartsville Units.

The ACRS report on GESSAR-238 also identified a number of generic matters requiring attention prior to final design approval (FDA). In particular, the following generic items should have a specific plan and implementation schedule established prior to issuing a Construction Permit for the Hartsville Plants:

1. Fire protection features required in both the GESSAR-238 and TVA portions of the plant design, taking into account the NRC Staff's new fire protection regulatory requirements.

2. Anticipated transients without scram, if changes in the scram system are anticipated from that presently used in BWRs in order to meet regulatory requirements.

3. A thorough assessment of the adequacy of the provisions to reduce the likelihood of stress corrosion cracking in BWR systems.

4. An assessment of occupational exposures in accordance with the ALARA criteria taking into account the need for improved decontamination capability, personnel access for in-service inspection, and general accessibility for maintenance of installed equipment in both the nuclear island and the turbine island portions of the plants.

5. The adequacy of the planned instrumentation to follow the course of accidents.

The Committee wishes to be kept informed regarding these items.

The NRC Staff should take the necessary steps to assure direct participation of the TVA personnel in the GESSAR-238 Final Design Approval actions in order to make certain that the Applicant is fully aware of the regulatory requirements pertaining to the FDA.

Generic problems relating to large water reactors are discussed in the Committee's April 19, 1976, Status Report, No. 4. These problems should be dealt with in a timely fashion 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 Hartsville Nuclear Plants Units A-1, A-2, B-1, and B-2, can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Mr. J. Ebersole did not participate in the review of this project.

Sincerely yours,

Dade W. Moeller

Dade W, Moeller
Chairman

REFERENCES:

1. Hartsville Nuclear Plants Units A-1, A-2, B-1, and B-2, Preliminary Safety Analysis Report, Volumes 1-4.
2. Amendments 1-17 to the Preliminary Safety Analysis Report.
3. Safety Evaluation Report, NUREG-0014, related to the construction of the Hartsville Nuclear Plants Units A-1, A-2, B-1, and B-2, April 8, 1976.

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

May 15, 1969

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

Subject: REPORT ON EDWIN I. HATCH NUCLEAR PLANT

Dear Dr. Seaborg:

At its 109th meeting, May 8-10, 1969, the ACRS completed its review of the application by Georgia Power Company for authorization to construct the Edwin I. Hatch Nuclear Plant. This project was considered at the 108th ACRS meeting, April 10-12, 1969, a special meeting on May 2, 1969, and at a Subcommittee meeting and site visit on March 27 and 28, 1969. During its review, the Committee had the benefit of discussions with representatives of the Georgia Power Company, General Electric Company, Southern Services, Inc., Bechtel Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed below.

The Edwin I. Hatch Plant will be located in a sparsely populated area in southeastern Georgia, approximately 75 miles west of Savannah, Georgia. The Altamaha River flows through the 2100 acre site with the reactor located on its south bank. A minimum exclusion distance of 4400 feet has been provided. Only 840 persons are located within five miles of the site, and no concentrated areas of population of 2000 or more are within ten miles. Baxley, Georgia, with a population of approximately 4800, is situated eleven miles to the south. A major north-south highway, U. S. Route No. 1, passes through the site near its western boundary.

The nuclear plant will utilize a General Electric boiling water reactor similar to that provided for the Cooper Nuclear Station, which was discussed in the Committee's report dated March 12, 1968. Each reactor is essentially identical to those proposed for the Brunswick Steam Electric Plant, also under review for a construction permit. The Hatch reactor is designed to produce 2436 MWt with a maximum performance rating of 2537 MWt. Primary and secondary containment structures for the nuclear steam system will be similar to those for the Cooper Station. A closed-cycle cooling system employing two banks of cooling towers will be used; makeup water will be supplied from the river.

The geology and meteorology of the site appear favorable. Provision will be made for protection of the plant against earthquakes, floods, hurricanes, and tornadoes.

Several problems unique to boiling 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 Hatch Plant.

The Committee continues to reiterate its interest in an appropriate program for in-service inspection of the reactor primary system. The applicant is conducting a study to establish a more vigorous in-service inspection program than that initially proposed and to specify design provisions to facilitate the new program, particularly with regard to access to the primary system. The applicant stated he will give careful attention to the provisions of the USA draft standard on in-service inspection in this study, and he will complete the study within six to nine months. The Regulatory Staff should review this program and should report the results of its review to the Committee.

In the area of reactor instrumentation, the Committee believes:

(a) that the rod block monitor system can perform an important safety, as well as operational, function and that incorporation of such a system, or its equivalent, is necessary;

(b) that there should be suitable provisions to ensure that low-pressure core cooling capability will be available before the auto-relief depressurization can be initiated;

(c) that the flux scram point should be automatically reduced to an appropriate level as the reactor recirculation flow is reduced below the normal full-power flow;

(d) the systems which perform these functions should be built to meet appropriate protection system criteria. The criteria to be used for each system should be established on a basis acceptable to the Regulatory Staff.

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. In the event of a turbine trip, reliance is placed on prompt control-rod scram to prevent large rises in primary system pressure. The applicant and his contractors have devoted considerable effort to providing a reliable protective system. However, systematic failures due to improper design, operation, or maintenance could obviate the scram reliability. The Committee recommends that a study be made of further means of preventing common failure modes from negating scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients.

For purposes of design of the engineered safety features, the applicant has proposed using a fission-product source term smaller than that suggested in TID-14844, and a treatment of this source within the containment different from that recommended in the same document. The Committee believes that the assumptions of TID-14844 should be used as a design basis for the engineered safety features of the Hatch plant, unless and until the use of a different set of assumptions has been justified to the satisfaction of the Regulatory Staff and the ACRS.

The Committee reiterates its concern that the post-accident cooling system 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.

Engineered safety systems that are required to recirculate water after a loss-of-coolant accident should be designed so that a gross system leak will not result in critical loss of recirculation or in loss of isolation capability. The Committee believes that exception to this general rule may be made in respect to a very short run of pipe from the torus to the first valve, if extremely conservative design of the pipe (and its connection to the torus) is used and suitable remote operability of the valve is provided. The design of these systems also should provide adequate leak detection and surveillance capability.

The applicant has agreed to supply, for review by the Regulatory Staff, preliminary details concerning aseismic design of the supports for the torus and associated piping and of the personnel lock prior to installation of these components.

Studies are continuing on the possible effects of radiolysis of water in the unlikely event of a loss-of-coolant accident. The Committee believes the applicant should evaluate all problems which may arise from hydrogen generation, including various levels of Zircaloy-water reactions which could occur if the effectiveness of the emergency core cooling system were significantly less than that predicted. The matter should be resolved between the applicant and the AEC Regulatory Staff.

The applicant proposes acceptable standards of design, fabrication, and inspection of the steam lines downstream of the second isolation valve. The Committee understands that a simplified dynamic analysis of the turbine building will be made to determine the displacements and forces transmitted to the main steam piping supports in the event of an Operating Basis Earthquake. Consideration should be given to an appropriate program of in-service inspection.

The ACRS 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 Edwin I. Hatch 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 - Edwin I. Hatch Nuclear Plant

1. Letter from Shaw, Pittman, Potts, Trowbridge and Madden, dated May 6, 1968; License Application; Volumes I, II, III, and IV of the Preliminary Safety Analysis Report.

2. Letter from Bechtel Corporation, dated August 9, 1968; Amendment No. 1 to License Application, dated August 6, 1968.

3. Amendment No. 2 to License Application, dated January 24, 1969.

4. Letter from Bechtel Corporation, dated March 10, 1969; Amendment No. 3 to License Application, dated March 7, 1969; Volume V of PSAR.

5. Letter from Bechtel Corporation, dated March 24, 1969; Amendment No. 4 to License Application, dated March 21, 1969.

6. Letter from Bechtel Corporation, dated April 9, 1969; Amendment No. 5 to License Application, dated April 1, 1969.

7. Letter from Bechtel Corporation, dated April 28, 1969; Amendment No. 6 to License Application, dated April 25, 1969.

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

November 13, 1971

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

Subject: REPORT ON EDWIN I. HATCH NUCLEAR PLANT UNIT 2

Dear Dr. Schlesinger:

At its 139th meeting, November 11-13, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application by the Georgia Power Company for authorization to construct a second nuclear power reactor, Unit 2, at its Edwin I. Hatch Nuclear Plant. This project was considered at a Subcommittee meeting in Washington, D. C., on October 28, 1971. During its review the Committee had the benefit of discussions with representatives of the Georgia Power Company, Southern Services, Inc., General Electric Company, Bechtel Corporation, the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents listed below.

The Edwin I. Hatch Plant is located in a sparsely populated area in southeastern Georgia, approximately 75 miles west of Savannah, Georgia. The Altamaha River flows through the 2,244-acre site with the reactors located on its south bank. The minimum exclusion distance is 4300 feet from the plant buildings to the site boundary. Only 840 persons are located within five miles of the site and no concentrated areas of population of 2,000 or more are within ten miles. Baxley, Georgia, with a 1970 population of 3,500 is eleven miles to the south. The nearest population center is Waycross, Georgia, about 48 miles south with a population of 19,000. A major north-south highway, U. S. Route No. 1, passes through the site near its western boundary.

Hatch Unit 2 utilizing a 2537 MWt General Electric single cycle, forced circulation boiling water reactor, will be almost identical with and located immediately adjacent to Hatch Unit 1. Both units will have separate conventional primary containment and secondary confinement structures and separate closed cycle cooling systems each employing three banks of cooling towers. The principal facilities shared by both units are the refueling floor, control room, and main stack. Hatch Unit 1 was discussed in the Committee's report dated May 15, 1969, and its construction is now about one-third completed.

The main steam lines up to the turbine stop valves and all branch lines 2-1/2 inches and larger up to the first valve will be designed to Class I seismic requirements in a manner satisfactory to the Regulatory Staff. In addition, a sealing system, designed to standards applicable to engineered safety features, will be provided to minimize leakage through the main steam line isolation valves; this design should be reviewed by the Regulatory Staff before installation of the system.

The applicant's criteria for protection against damage from pipe whip are that there will be no violation of primary containment and that operation of the ECCS will be assured. Several design changes are being made including an increase in drywell neck radius and rearrangement of steam and feedwater piping to make room for restraints, a heavier steel platform, thicker drywell shell, stronger pedestal for the reactor vessel, and ring girders surrounding the biological shield. These provisions should substantially improve the effectiveness of protection against damage from pipe whip.

The applicant has studied design features to make tolerable the consequences of a 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 Hatch Unit 2 reactor. However, further evaluation of the sufficiency of this approach and specific means for 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.

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 Hatch Unit 2.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items, the Edwin I. Hatch Nuclear Plant Unit 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/ Spencer H. Bush

Spencer H. Bush
Chairman

References attached.

References - Edwin I. Hatch Nuclear Plant Unit 2

1. Georgia Power Company letter dated September 11, 1970, transmitting Preliminary Safety Analysis Report (PSAR) to the Edwin I. Hatch Nuclear Plant Unit 2, Volumes 1 through 5

2. Amendments No. 1, 2, 4, 6 through 13 to the License Application for the Edwin I. Hatch Nuclear Plant Unit 2, Preliminary Safety Analysis Report

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

February 10, 1972

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

Subject: REPORT ON EDWIN I. HATCH NUCLEAR POWER PLANT UNIT 1

Dear Dr. Schlesinger:

At its 142nd meeting, February 3-5, 1972, the Advisory Committee on Reactor Safeguards completed a review of the condition of the reactor pressure vessel for the Edwin I. Hatch Nuclear Power Plant Unit 1. This matter was reviewed by a Subcommittee on February 2, 1972. During its review the Committee had the benefit of discussions with representatives and consultants of the Georgia Power Company (owner of the Hatch Plant), Southern Services, Inc. (the architect-engineer), General Electric Company (supplier of the nuclear system), and Combustion-Engineering, Inc. (manufacturer of the vessel), and the AEC Regulatory Staff. The Committee also had the benefit of the document listed.

The Hatch reactor pressure vessel was manufactured to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components. It met all requirements of Section III, including radiographic examination of the nozzle welds that are the subject of this report, and was delivered to the site as a code-stamped Section III vessel.

In conformity with the ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, the vessel was ultrasonically tested at the site to provide a baseline reference for inservice surveillance over the life of the plant. The ultrasonic tests showed indications of discontinuities around two of the ten approximately 12-inch inside-diameter inlet nozzles of the water recirculation system. The indications appeared to be near the interface between the nozzle-attachment weld and the vessel wall, at mid-wall thickness, and extending circumferentially around the nozzle for a distance of approximately 37 inches in one nozzle and 12 inches in the other. The orientation of the indications is approximately normal to the vessel wall (like a ribbon wrapped around the nozzle) but their character cannot be determined by existing nondestructive techniques and their widths can be expressed only as an upper limit, which was estimated to be 3/4 inch, over a limited distance, in one nozzle and less in the other. Two other independent ultrasonic examinations confirmed, in general, the length and orientation of the indications, but placed much lower upper limits on the width. An independent radiographic examination in the field failed to show indications, which is in agreement with the shop findings during fabrication.

The vessel thus meets the ASME fabrication code, Section III, but the field inspection by a method not required by Section III has revealed linear indications which, depending on their character, might have required repair if they had been found prior to certification.

The applicant has made fracture mechanics analyses assuming, as an extreme case, a full-circumference crack 3/4 inch wide. The analyses show that the calculated stresses in the region are low and that there should be no significant crack growth over the life of the plant. The analyses assume material properties at the lower limit of acceptability.

Notwithstanding the applicant's analyses, with which the Committee does not disagree, the Committee believes it is unacceptable to put this vessel into service with linear indications of incompletely defined character and dimensions. The ACRS therefore believes the vessel should be repaired, unless it can be shown by physical examination of samples obtained from the vessel that the discontinuities present and the relevant physical properties of the metal are within the limits set by Section III of the ASME Code. Any changes in the reactor vessel resulting from sampling should be evaluated analytically to establish the integrity and design life of the vessel will not be significantly impaired. The sampling program, acceptance criteria for discontinuities and metal properties, and analyses of effects of the sampling program on the vessel should be developed in conjunction with and be satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

Sincerely yours,

/s/ C. P. Siess

C. P. Siess
Chairman

Reference:

Georgia Power Company letter dated January 25, 1972, w/Summary of the Detection and Evaluation of Ultrasonic Indications for the Edwin I. Hatch Unit 1 Reactor Pressure Vessel

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

June 12, 1973

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

Subject: INTERIM REPORT ON THE EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

Dear Dr. Ray:

During its 158th meeting, June 7-9, 1973, the Advisory Committee on Reactor Safeguards conducted a review of the application by the Georgia Power Company for authorization to operate the Edwin I. Hatch Nuclear Plant, Unit 1 at power levels up to 2436 MW(t). A Subcommittee made a tour of the partially completed plant on February 27, 1973. The project was considered during a Subcommittee meeting in Washington, D. C. on May 24, 1973. During its review, the Committee had the benefit of discussions with representatives and consultants of the Georgia Power Company, Southern Services Incorporated, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee reported to the Commission on the construction of this unit in a letter dated May 15, 1969 and on the construction of Unit 2 in a letter dated November 13, 1971.

The Hatch Nuclear Plant is located on the south bank of the Altamaha River in a rural area of southeastern Georgia, about 11 miles north of Baxley, Georgia, and about 75 miles west of Savannah. Hatch Unit 2, now under construction, is immediately adjacent to Unit 1.

The Committee reported to you, in a letter dated February 10, 1972, concerning possible defects in the reactor vessel, and recommended that repairs should be made unless proven to be unnecessary by appropriate tests. The applicant has completed the examinations and repairs and presented the results to the Regulatory Staff and the ACRS. The repairs and subsequent inspections have been reviewed by the Regulatory Staff and the ACRS and, subject to satisfactory completion of the hydrostatic test and base line examination, the repairs are considered to be acceptable.

The applicant has developed plans for in-service inspection of accessible portions of the reactor coolant pressure boundary both inside and outside of containment. The Committee recommends that continued attention be given to means for assuring the integrity of those portions of the reactor pressure vessel that are currently inaccessible for inspection.

In the unlikely event that a break occurs in the recirculation pump discharge line, the pump impeller might act as a turbine, causing the pump and motor to overspeed and become potential sources of missiles. The applicant is reviewing means of dealing with this possibility. The Committee believes that this matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Regulatory Staff is developing Technical Specifications for maintenance and testing of the main steamline isolation valves to control leakage rates. The Committee believes that the criteria adopted for frequency of leak testing and for permissible leak rates before and after maintenance should be of such a nature as to assure, at a suitable confidence level, that the leak rate at any time during operation will not exceed the value assumed in the calculation of offsite radiation doses for the postulated main steamline break accident. If these criteria cannot be met during operation of Hatch Unit 1, the Committee believes that a suitable sealing system should be designed and installed on an appropriate time scale.

The applicant has examined the problems that might develop should a main steamline or other high-energy line rupture outside of containment and has concluded that the plant could be shut down safely. The Regulatory Staff is reviewing the applicant's submittal. The Committee recommends that this matter be resolved in a manner satisfactory to the Regulatory Staff.

To avoid possible damage from dropping a spent fuel cask, the applicant has proposed to modify overhead handling equipment in the reactor building to provide appropriate reliability. The modifications will be made prior to the time of first refueling. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Committee believes that the microwave tower, located just north of the electrical feeder lines from the switchyard to the startup transformers, should be relocated so as to eliminate the possibility of its falling on the feeder lines or their supporting structures.

Although details of emergency planning appear to be well developed, questions remain with respect to coordination of these plans with State agencies. Such questions include specification of dose levels at which emergency action is to be implemented, the nature of such action, and administrative responsibilities. These matters should be resolved in a manner satisfactory to the Regulatory Staff.

Reviews are continuing on the problem of fuel densification and whether it might affect the efficacy of the Hatch Unit 1 emergency core cooling system.

The Committee believes that the matters mentioned above can be resolved satisfactorily on a suitable time scale.

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 Committee will report to you further regarding the acceptable power level for this plant after a recommendation has been made by the Regulatory Staff and the appropriate Supplement to the Safety Evaluation has been reviewed by the Committee.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References Attached.

References

1. Final Safety Analysis Report for the Edwin I. Hatch Nuclear Plant, Unit 1 - Volumes I through VII
2. Amendments 10-20, 22-24, 26-33 to the License Application
3. Georgia Power Company letters dated March 7 & 20, 1972 and April 19, 1972 re: program and procedures to remove the ultrasonic reflectors from the reactor vessel
4. Georgia Power Company letters dated June 13, October 9 & 30, and December 21, 1972 re: Reactor Vessel Repairs
5. Georgia Power Company letter dated October 9, 1972 re: Prototype Vibration Monitoring Program
6. Georgia Power Company letter dated December 4, 1972 re: Post LOCA Hydrogen Control
7. Georgia Power Company letter dated January 3, 1973 re: Fuel Densification
8. Georgia Power Company letter dated January 9, 1973 re: Potential for Internal Flooding
9. Georgia Power Company letter dated March 20, 1973 transmitting report "Drywell Air Gap-Removal of Grout and Repair of Concrete Biological Shield"
10. Directorate of Licensing Safety Evaluation Report dated May 11, 1973

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

January 14, 1974

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

Subject: REPORT ON THE EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

Dear Dr. Ray:

During its 165th meeting, January 10-12, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application by the Georgia Power Company for authorization to operate the Edwin I. Hatch Nuclear Plant, Unit 1. A Subcommittee made a tour of the partially completed plant on February 27, 1973. The project was considered during Subcommittee meetings in Washington, D. C., on May 24, 1973, and December 20, 1973, and at the Committee's 158th meeting, June 7-9, 1973, in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives and consultants of the Georgia Power Company, Southern Services Incorporated, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee forwarded an interim report on the operation of this unit to the Commission on June 12, 1973.

Hatch, Unit 1, utilizes a General Electric boiling water reactor similar to those provided for Brunswick Units 1 and 2 and for the Cooper Nuclear Station previously reviewed by the ACRS for operation. The Hatch reactor is designed to produce 2436 MWt.

The applicant, with the General Electric Company, has studied the problem of fuel densification. The analyses indicate that, except in regard to peak clad temperatures in postulated loss of coolant accidents, the effects of expected densification are relatively small. To assure conformance with peak clad temperature and other limits of the Interim Acceptance Criteria, the applicant proposes to operate the Hatch reactor in such manner as to maintain the maximum average planar linear heat generation rate (MAPLHGR) at all times below a specific allowable value. So-called "gamma" curves, developed by the applicant and reviewed and approved (to date, only for the first fuel cycle) by the Regulatory Staff, depict the allowable value as a function of fuel exposure. The Committee believes that the approach described is acceptable. However, the ACRS recommends that the Regulatory Staff assure itself that the detailed procedures to be employed by the reactor operator to accomplish and to demonstrate compliance with the proposed limits on core conditions are adequate. Particular emphasis should be given to the procedure to be followed when the computer normally used for core power distribution calculation is inoperable or unavailable. The Committee wishes to be kept informed.

Re-evaluation of core operating limits will be necessary as a result of the recently promulgated Acceptance Criteria for Emergency Core Cooling Systems. The Committee wishes to be kept informed.

Since the Committee's last report, the applicant has made further 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 such an event. 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 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 those mentioned in its June 12, 1973 Interim Report, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the Edwin I. Hatch Nuclear Plant, Unit 1 can be operated at power levels up to 2436 MWt without undue risk to the health and safety of the public.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References attached.

References

1. Amendments 34 through 40 to the License Application

2. Georgia Power Company letter dated October 25, 1973, furnishing information on Control Rod Drop Accident and Core Thermal Power Level

3. Directorate of Licensing Supplement No. 1 to the Safety Evaluation Report, dated December 10, 1973

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

February 16, 1978

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

Subject: REPORT ON EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

Dear Dr. Hendrie:

During its 214th meeting, February 9-11, 1978, the Advisory Committee on Reactor Safeguards completed its review of the application of the Georgia Power Company, Oglethorpe Electric Membership Corporation, Municipal Electric Authority of Georgia and the city of Dalton, Georgia (the Applicants) for a license to operate the Edwin I. Hatch Nuclear Plant, Unit No. 2. The plant will be operated by Georgia Power Company. The application was reviewed at Subcommittee meetings on January 27 and 28, 1978 in Washington, D.C. During its review, the Committee had the benefit of discussions with representatives and consultants of the Nuclear Regulatory Commission (NRC) Staff; General Electric Company; Southern Company Services, Incorporated; Bechtel Power Corporation; and the Applicants. The Committee also had the benefit of the documents listed.

The Edwin I. Hatch Nuclear Plant is a two-unit station located on the south bank of the Altamaha River approximately 11 miles north of Baxley, Georgia. The two units are virtually identical except that Hatch Unit No. 1 utilizes 7X7 fuel assemblies while Hatch Unit No. 2 will utilize 8X8R (Retrofit) fuel assemblies. The rated thermal power for each unit is 2436 MW(t). Each unit includes a General Electric Company BWR/4 boiling water reactor. The Committee reported on the application for a construction permit for Unit No. 2 on November 3, 1971.

Hatch Unit No. 2 is the first reactor scheduled to use the new General Electric 8X8R fuel on a core-wide basis. This fuel design is a slightly modified version of the General Electric 8X8 fuel assembly design currently in use in a number of boiling water reactors. These modifications include, among others, an increase in fuel length, use of natural uranium at the top and bottom of the fuel rod and the addition of a second water rod to each fuel assembly. These changes improve the shutdown and thermal margins, provide flatter local power distribution, and improve fuel cycle efficiency. Four of the 8X8R fuel assemblies have been operating in Peach Bottom Unit No. 2 since May 1976 and two assemblies have been operating in Vermont Yankee since August 1976. The NRC Staff has concluded that the 8x8R fuel assembly design is acceptable for use in Hatch Unit No. 2. The Committee concurs.

The NRC Staff has identified a number of safety-related items which will require resolution prior to a decision on the issuance of an operating license. These matters should be resolved in a manner satisfactory to the NRC Staff.

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 Edwin I. Hatch Nuclear Plant, Unit No. 2 are: II-1, 4, 5A, 5B, 6, 7, 8, 10; IIA-4; IIB-2, 4; IIC-1, 3A, 3B, 5, 6, 7; IID-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 items mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the Edwin I. Hatch Nuclear Plant, Unit No. 2 can be operated at power levels up to 2436 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

References

1. Edwin I. Hatch Nuclear Plant, Unit No. 2, Final Safety Analysis Report, with Amendments 18 through 41.
2. Report to the Advisory Committee on Reactor Safeguards by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission in the matter of Georgia Power Company, et al, Edwin I. Hatch Nuclear Plant, Unit No. 2, dated January 4, 1978.
3. General Electric Company, "Lattice Physics Methods," NEDE-20913A, January, 1977.
4. General Electric Company, "Lattice Physics Methods Verification," NEDO-20939A, January, 1977.
5. General Electric Company, "BWR Simulator Methods Verification," NEDO-20946A, January, 1977.
6. General Electric Company, "Three-Dimensional BWR Core Simulator," NEDO-20953A, January, 1977.

7. General Electric Company, "BWR/6 Fuel Design," NEDE-20948-P, June, 1976, and Amendment No. 1, November, 1976.
8. General Electric Company, "BWR/4 and BWR/5 Fuel Design," NEDE-20944-P, September, 1976.
9. General Electric Company, "BWR Fuel Channel Mechanical Design and Deflection," NEDE-21354-P, September, 1976.
10. General Electric Company, "BWR/6 Fuel Assembly: Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," NEDE-21175-P, November, 1976 and Amendment 1, April, 1977.

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

November 16, 1959

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

Subject: HEAT TRANSFER REACTOR EXPERIMENT-3A (HTRE-3A)

Dear Mr. McCone:

At its twenty-first meeting, November 12-14, 1959, the Advisory Committee on Reactor Safeguards reviewed the proposal for operation of the HTRE-3A at the National Reactor Testing Station, Idaho Falls, Idaho. Present at this meeting were representatives of the applicant, the Division of Licensing and Regulation, the Division of Reactor Development, and the Idaho Operations Office.

The Committee concluded that at the design power level, with the proposed meteorological restrictions, the experiment should proceed only if a cooling air cleaning system is provided which removes the halogens and bone seekers efficiently.

In view of the necessity of driving the components of this experiment close to their ultimate limits, a meltdown of a fair fraction of the reactor should be considered an operational hazard and not an unlikely accident. An analysis of the possible course of such a meltdown, of its consequence, and of possible corrective measures has been promised by the applicant for review in the near future. The Committee will review this information prior to submitting additional advice on this experiment.

Sincerely yours,

/s/

C. Rogers McCullough
Chairman

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

References

1) XDC 59-9-81 - Preliminary HTRE-3A Hazards Report, August 28, 1959.

2) Division of Licensing and Regulation Report to the ACRS on the Preliminary HTRE-3A Hazards Report, dated October 28, 1959.

3) U. S. Weather Bureau Comments on XDC 58-9-81, "Preliminary HTRE-3A Hazards Report," dated November 3, 1959.

4) Comments of the Office of Health and Safety on HTRE-3A Hazards Report XDC 59-9-81", October 27, 1959.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

United States Atomic Energy Commission

Washington 25, D. C.

November 12, 1958

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

Subject: HEAVY WATER COMPONENTS TESTING REACTOR

Dear Mr. McCone:

At its Eleventh Meeting on November 6, 1958, the Advisory Committee on Reactor Safeguards reviewed the site selection for the Heavy Water Components Testing Reactor. It is proposed to locate a 60-thermal-megawatt, pressurized, heavy water cooled and moderated, testing reactor at the Savannah River Plant. Containment is to be provided.

Data concerning the site were obtained from DPST 58-409 et al, Hazards Evaluation Branch summary report, and through oral presentations by representatives of the contractor and by the Hazards Evaluation Branch.

The Committee considers the site proposed, under the conditions of design tentatively presented, including containment, to be acceptable from the standpoint of health and safety of the public.

It is understood that the detailed design features of the reactor when available will be submitted to the Committee for further review.

Sincerely yours,

/s/

C. Rogers McCullough
Chairman

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: HEAVY WATER COMPONENTS TESTING REACTOR (HWCTR)

Dear Mr. McCone:

Additional information presented by the Hazards Evaluation Branch on the Savannah River Heavy Water Components Testing Reactor was reviewed by the Advisory Committee on Reactor Safeguards at its nineteenth meeting September 10-12, 1959. Safety aspects of this reactor have been reviewed at previous meetings of the Committee based on information contained in the references cited herein.

Additional design details remain to be resolved.

However, it is the conclusion of the Advisory Committee on Reactor Safeguards that a reactor of the type and containment proposed by the du Pont Company at its Savannah River location 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. Rogers McCullough
Chairman

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

References

DPST 58-409 - Preliminary Hazards Evaluation of the Heavy Water Components Testing Reactor (HWCTR), August 1958.

DPST 59-180 - Preliminary Hazards Evaluation of the Heavy Water Components Testing Reactor (HWCTR), March 1959.

DPW-59-292 - HWCTR - Reactor Safeguards, Aug. 13, 1959.

Division of Licensing and Regulation Report to the ACRS on HWCTR dated June 30, 1959.

Division of Licensing and Regulation Report to the ACRS on HWCTR dated August 24, 1959.

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

November 1, 1961

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

Subject: REPORT ON HEAVY WATER COMPONENTS TEST REACTOR

Dear Dr. Seaborg:

At its thirty-seventh meeting on October 26-28, 1961, the Advisory Committee on Reactor Safeguards considered the Heavy Water Components Test Reactor (HWCTR) on the basis of the documents referenced below, a presentation by representatives of the E. I. du Pont de Nemours Company, and discussion with the Staff of the AEC. A subcommittee of the ACRS visited the HWCTR on October 6, 1961. This reactor also was the subject of our letters of November 12, 1958 and September 14, 1959.

In view of the large exclusion radius, the adequate containment of the reactor, negative temperature coefficient of reactivity, and the experienced operating organization, the ACRS concludes that this reactor can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
T. J. Thompson
Chairman

References (Attached)

References:

1. DP-489 - Preliminary Hazards Evaluation of the Isolated Coolant Loops in the HWCTR, dated July 1960.
2. Letter dated December 14, 1960 from E.I. du Pont de Nemours & Co. to Savannah River Operations Office re Safety of the Coolant Loops of the HWCTR.
3. DP-600 - Final Hazards Evaluation of the Heavy Water Components Test Reactor, dated July 1961.
4. DP-600 - Supplement I, Final Hazards Evaluation of the Heavy Water Components Test Reactor, dated Sept. 1961.
5. Special Inspection Brief No. 2325 for Reactor Vessel & Parts, Savannah River Plant, dated 7/23/59.
6. Condition of Reactor Vessel - Preliminary Report Issued Aug. 1961.
7. DPE-2167 - Use of 17-4 PH Stainless Steel in Safety and Control Rod Drives, Part II, issued Aug. 1961.
8. Memo - Nelson to Duff, dated Sept. 8, 1961 - HWCTR Rod Drive - SL-1 Rod Drive Sticking.
9. Purchase Orders, dated Apr. 30, 1959 - Specification for Reactor HWCTR.
10. Info Reply to TWX of Sept. 15, 1961 - Stetson to Worthington, dated Sept. 25, 1961.
11. Fay to Kamack, dated Sept. 22, 1961 - ACRS Review 10/6/61 of Reactor Design Features.
12. DPE-2166 - Use of 17-4 PH Stainless Steel in Safety and Control Rod Drives, Part I, dated April 1961.

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

May 9, 1960

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

Subject: HIGH FLUX ISOTOPE REACTOR (HFIR)

Dear Mr. McCone:

At its twenty-fifty meeting on May 5-7, 1960, the Advisory Committee on Reactor Safeguards considered the problem of site location for the High Flux Isotope Reactor (HFIR) proposed by the Oak Ridge National Laboratory. The conceptual design of this reactor is in a preliminary stage and this review was for a consideration of site only. The proposal is for a 100 tMW pressurized water flux-trap reactor with a high-flux central zone for the production of trans-plutonium isotopes. References available on the proposed reactor are listed herein. In addition to this referenced material the Committee had the benefit of discussions with representatives of the Oak Ridge National Laboratory and the AEC Division of Licensing and Regulation.

The design power of the HFIR is greater by a factor of three or more than the power of any present reactor at ORNL and its design pressure of 1000 psi is in the range of nuclear power plant reactors. It has a design maximum heat flux higher than that of any other reactor. It has a positive void coefficient in the central water zone, which means that any accidental increase in central voids will result in a rising power transient.

The Oak Ridge National Laboratory proposes to locate this reactor adjacent to the present ORNL cafeteria in a building designed for the confinement of fission products but not for containment in the conventional manner. An alternate site is located a short distance away. Because of the relative isolation of ORNL from populated residential centers these sites do not appear to offer any significant hazard to the health and safety of the public outside of ORNL. However, the operation of the proposed reactor at the two suggested sites with the type of building confinement proposed by ORNL appears to the Committee to present some risk to the health and safety of the employees at the Oak Ridge National Laboratory. Therefore, it is the Committee's recommendation that if it is not feasible to house the proposed reactor in conventional gastight containment, an alternate location be sought that will provide a greater degree of isolation from the main body of ORNL employees than the sites presently under consideration.

Drs. Ergen, Newson, and Gifford did not participate in the Committee's consideration of this reactor.

Sincerely yours,

/s/

Leslie Silverman
Chairman

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

References

CF No. 60-3-33 - High Flux Isotope Reactor - A General Description, March 1960.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

July 25, 1960

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

Subject: HIGH FLUX ISOTOPE REACTOR (HFIR)

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 new site proposed for location of the High Flux Isotope Reactor (HFIR). This reactor, its confinement system and its previously proposed site were considered by the Committee at its twenty-fifth meeting. You were advised of our recommendation for an alternate site location or conventional gastight pressure containment at the original site. The Oak Ridge National Laboratory has now selected for consideration an alternate site located in Melton Valley.

Since the environmental characteristics of the new proposed site and its distance from the existing ORNL complex are both favorable, it now appears that the confinement system proposed is acceptable. The Committee believes that the proposed reactor can be constructed in the Melton Valley location with reasonable assurance that it may be operated without undue risk to the health and safety of the public or to the ORNL employees.

Doctors Gifford, Newson and Ergen did not participate in the Committee's consideration of this reactor.

Sincerely yours,

/s/

Leslie Silverman
Chairman

References:

(1) ORNL 60-3-33, "High Flux Isotope Reactor, A General Description," dated March 15, 1960.

(2) Letter from J.A.Swartout, ORNL, to H.M.Roth, ORNL, subject "High Flux Isotope Reactor (HFIR), Request for Site Approval," dated June 13, 1960

(3) Memo from S.H.Sapirie, OROO, to P.W.McDaniel, AEC, subject "HFIR Site Review" dated July 14, 1960 with enclosure; letter from J.A.Swartout, ORNL, to H.M.Roth, ORNL, subject "HFIR Site Review" dated July 5, 1960.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

July 15, 1965

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

Subject: REPORT ON HIGH FLUX ISOTOPE REACTOR (HFIR)

Dear Dr. Seaborg:

At its sixty-fourth meeting, July 8-10, 1965, the Advisory Committee on Reactor Safeguards reviewed the proposed operation of the High Flux Isotope Reactor (HFIR) by the Oak Ridge National Laboratory. The Committee had the benefit of discussions with representatives of the Oak Ridge National Laboratory and the AEC staff. The Committee also had the benefit of the documents referenced below. A Subcommittee meeting in Washington, D. C. was held on June 16, 1965.

The Committee reviewed the conceptual design of this reactor for consideration of the site location at its twenty-fifth and twenty-seventh meetings in May 1960 and July 1960, respectively. However, the Committee did not have the benefit of reviewing the detailed reactor design prior to construction.

The High Flux Isotope Reactor is a light-water-cooled and -moderated, beryllium-reflected, flux trap reactor with a high-flux central zone for the production of transplutonium isotopes. It contains experimental facilities for beam experiments and other irradiation experiments.

The HFIR reactor contains a number of novel features. The reactor core consists of a series of concentric annular regions surrounding a 5-inch diameter cylindrical hole into which the transuranium target is positioned. The fuel region consists of two concentric fuel elements. The fuel elements contain involute-shaped fuel plates composed of U_3O_8-Al cermet clad with type 6061 aluminum. The fuel is highly enriched in the U-235 isotope. To minimize the radial peak-to-average power density ratio, the fuel is nonuniformly loaded along the arc of the involute. A burnable poison, boron carbide, is included in the inner fuel element to further flatten the neutron flux and to reduce the negative reactivity requirements of the control plates. The fuel region is surrounded by a beryllium reflector; exterior to the beryllium is a water reflector. In the axial direction the reactor is reflected by water. The core is approximately 24 inches long and has an active volume of about 50 liters. The length of the typical fuel cycle is two weeks; as a result the core must be refueled frequently.

The design power is 100 MW(t). At this power level the core develops the extremely high average power density of about 2 megawatts per liter, and has a maximum unperturbed neutron flux of about 5.5×10^{15} neutrons per square centimeter per second in the central target region.

The core is controlled by two thin poison-bearing concentric cylinders located between the fuel region and the beryllium reflector. These are driven in opposite directions by drive mechanisms located beneath the reactor. The inner control cylinder is used for shimming and regulation, but has no scram function. The reactivity of the core is increased by downward motion of this control cylinder. The outer control cylinder consists of four separate quadrants, each having an independent drive and safety release mechanism. Reactivity is increased as these outer plates are raised. All five control elements have three regions of different material content, one with europium oxide, another tantalum, and the third aluminum, to minimize the axial peak-to-average power density throughout the core lifetime.

In the design of the reactor control and instrumentation system, considerable emphasis has been placed on providing continuity of operation; as a result, the control system is designed to provide fast response so as to minimize scrams that might lead to lengthy shutdowns due to transient xenon poisoning.

The primary coolant system is completely water-filled, with the operating pressure of 600 psi controlled by the use of pumps and let-down valves. The coolant is appreciably subcooled to avoid boiling at the surface of the fuel plates during normal operation.

A feature that contributes to the over-all safety of the reactor is the submersion of the reactor pressure vessel in a pool of water.

It should be noted that the HFIR shares a common exhaust stack with the Transuranium Processing Plant (TRU) which will process the HFIR targets. The HFIR is provided with a gravity-closing damper in the ventilation system to prevent back-flow from the TRU facility to HFIR when the latter is shut down. It is expected that the TRU facility will be similarly protected.

The Committee feels that additional information and discussion is required about some of the features of this plant which influence its safety. Included among these are: the beam-tube design; the criteria to be used in evaluating future experiments; the extent and consequences of metal-water reactions; reactivity effects associated with displacement of fuel; the behavior of the reactor under severe reactivity transients; the reliability and adequacy of the control and instrumentation system; and alternative ways of limiting any irradiation of the public or of the personnel at the Oak Ridge National Laboratory in the unlikely event of a serious accident.

The expected performance of the iodine adsorbers in the filter system with organic iodine compounds has not been resolved at this time. Further studies are in progress to determine capability of the installed system. The filter and charcoal cleanup systems have been tested recently with DOP and elemental iodine and have met design specifications for these materials.

The Committee believes that the policy of setting the emergency evacuation alarm levels at the low value of 25 mr/hr could lead to a safety problem. Evacuation of essential personnel as the result of a small release of radioactivity could lead to a larger accident, because preventive actions could not be taken.

The representatives of ORNL advised the Committee that they may be ready to begin initial loading to criticality in August 1965 and plan to pursue an experimental program at powers not to exceed 20 MW for several months.

The Committee believes that the HFIR reactor can be operated as proposed at power levels up to 20 MW without undue hazard to the health and safety of the public, while further information is developed on the topics mentioned above. The Committee wishes to review this information as well as the results of the operations carried on during this period before power level is increased beyond 20 MW.

Dr. F. A. Gifford, Dr. S. H. Hanauer, Mr. W. D. Manly and Dr. H. W. Newson did not participate in the review of this project.

Sincerely yours,

/s/David Okrent

David Okrent
Acting Chairman

References Attached.

References (HFIR)

1. The High Flux Isotope Reactor - Volume I, dated May 1964, (ORNL-3572).

2. The High Flux Isotope Reactor - Volume II, dated August 1964, (ORNL-3572).

3. The High Flux Isotope Reactor - Accident Analysis, dated February 1965 (draft) (ORNL-3573).

4. Revisions and Corrections to ORNL 3572, Volume I, dated March 1965.

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

May 11, 1966

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

Subject: REPORT ON HIGH FLUX ISOTOPE REACTOR (HFIR)

Dear Dr. Seaborg:

At its seventy-third meeting, May 5-7, 1966, the Advisory Committee on Reactor Safeguards reviewed the proposed operation of the High Flux Isotope Reactor (HFIR) by Oak Ridge National Laboratory (ORNL) at power levels up to 100 MWt. This matter had previously been considered at the sixty-fourth and seventieth meetings of the Committee. The Committee had the benefit of discussions with representatives of ORNL and the AEC Regulatory Staff. The Committee also had the benefit of the documents referenced below. Subcommittee meetings were held in Washington, D. C. on February 9, 1966 and May 4, 1966.

In its letter of July 15, 1965, the Committee concluded that the HFIR reactor could be operated at power levels up to 20 MWt without undue hazard to the health and safety of the public, while further information was being developed on some of the features of this plant which influence its safety. The Committee also stated that it wished to review the results of the operations carried on during the period before the power level was increased beyond 20 MWt. The foregoing information has since been submitted to and reviewed by the Committee; the results of the review are summarized below.

ORNL representatives reported that the planned program of zero and low power tests up to 20 MWt has been completed and that no unanticipated results of significance were experienced from any of the tests.

The HFIR reactor has some design features which lead to the possibility of large autocatalytic reactivity effects. ORNL presented the results of analyses of nuclear excursions more severe than those previously postulated, including estimates of energy releases from metal-water reactions which might be initiated by such excursions. They also submitted evaluations of the containment capability of the pressure vessel and of the pool and reactor building. The analyses performed by ORNL indicated that the reactor vessel can withstand the effects of such excursions without rupture. They also indicated that, even if the rupture of the vessel took place, the containment features of the pool and reactor building would not be violated.

It was reported to the Committee that a back up shutdown system utilizing cadmium nitrate solution had been developed and is being installed in the HFIR facility. This system is to be made operable prior to reactor operation at power levels above 20 MWt. This will include the development of operating procedures to ensure that the system would be effective at all power levels and operating conditions if the control rods were to be immobilized.

The bolts holding the end caps on the beam tubes have been changed to increase their ability to withstand high pressure. Tests on a spare beam tube indicate that the beam tubes in the reactor can contain the internal pressure resulting from rupture of that end of the tube which is inside the reactor vessel. It was reported that, if simultaneous failure of both ends of a beam tube occurred, the rate of coolant loss would be limited by the internal collimator plug which would be restrained by the beam tube shutter assembly. Representatives of ORNL stated that the plant would not be operated without this or a similar flow restriction in the beam tubes without previously discussing the proposed change with the AEC Regulatory Staff. The ORNL representatives also stated that they had developed a program for exposing surveillance specimens in HFIR at neutron fluxes higher than those experienced by the beam tubes; these specimens are to be evaluated periodically to determine if there is any significant change in mechanical properties.

ORNL recognizes that the experiments performed in this reactor could introduce additional hazards if not properly controlled. They have agreed to establish, in conjunction with the AEC Regulatory Staff, operating limitations on the possible energy releases associated with such experiments. The Committee recommends that any experiments involving the possiblity of large chemical energy releases be referred to the AEC Regulatory Staff for review.

The representatives of ORNL also reported that procedures for the testing of the iodine adsorbers in the filter system are being developed and that the adsorbers are to be tested at least twice a year.

The Committee was informed that stack monitors are being installed and that their installation would be completed during the early part of July, 1966, upon receipt of required components. The Committee urges timely completion of the installation.

The Committee concludes that, subject to the foregoing comments, the HFIR reactor can be operated at power levels up to 100 MWt without undue hazard to the health and safety of the public.

Dr. F. A. Gifford and Dr. S. H. Hanauer did not participate in the review of this project.

Sincerely yours,

/s/
David Okrent
Chairman

References.

1. ORNL-TM-1291, The Release and Absorption of Methyl Iodide in the HFIR Maximum Credible Accident, dated October 1, 1965.

2. ORNL-65-11-29, High Flux Isotope Reactor - Safety Review Questions and Answers, dated November 12, 1965.

3. ORNL-65-11-29, Supplement No. 1, Draft - February 1, 1966, The High Flux Isotope Reactor - Safety Review Questions and Answers.

4. ORNL letter to USAEC, Subject: HFIR Safety Review - Request for Approval for Interim Operation at 50 MW, dated April 1, 1966.

5. ORNL-65-11-29, Supplement No. 2, Draft - High Flux Isotope Reactor - Safety Review Questions and Answers, dated April 19, 1966.

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

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

February 12, 1974

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

Subject: REPORT ON HOPE CREEK GENERATING STATION (FORMERLY NEWBOLD ISLAND GENERATING STATION), UNITS 1 AND 2

Dear Dr. Ray:

At its 166th meeting, February 7-9, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application by the Public Service Electric and Gas Company for a permit to construct the dual-unit Hope Creek Generating Station. The design features of this facility are the same as those for the Newbold Island Generating Station except for certain site-related matters. The Newbold Island facility was considered by the Committee at a number of meetings, and the results of its review reported to the Commission in a report dated August 10, 1971. Further recommendations regarding the Newbold Island facility, some of which are not site-related, were included in the Committee's report to you of July 17, 1973. The Hope Creek project was considered also at a Subcommittee meeting on January 23, 1974 in Washington, D. C. The site was visited by the Subcommittee on January 22, 1974. During its review, the Committee had the benefit of discussions with representatives and consultants of the applicant, the General Electric Company, the Bechtel Power Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The station will be located on a 700-acre site adjacent to the Salem Generating Station on the east bank of the Delaware River, approximately 18 miles southeast of Wilmington, Delaware, 40 miles south of Philadelphia, Pennsylvania, and 7.5 miles southwest of Salem, New Jersey. The nearest population center of 25,000 or more is Wilmington, Delaware. The low population zone, with a radius of 5 miles, has a population of about 1500 (1970 census data). The nearest residence is 2-3/4 miles from the site. The minimum exclusion distance is 2600 feet.

Each of the Hope Creek units includes a boiling water reactor to be operated at 3293 MWt. These units are unchanged from those previously reviewed for the Newbold Island station. Waste heat from the station will be rejected to the atmosphere by natural draft cooling towers. Cooling water for safety-related equipment as well as make-up water for the turbine condenser cooling system will be supplied from the Delaware River.

Re-evaluation of core operating limits will be necessary as a result of the recently promulgated Acceptance Criteria for Emergency Core Cooling Systems.

Although the seismological, geological and foundation conditions at the site are expected to be essentially the same as those at the adjacent Salem station, the applicant is reviewing these features and has underway an extensive program of soil borings and laboratory tests of soil samples, as a basis for selecting the methods of excavation and dewatering, the seismic design bases, and the foundation design. These matters should be resolved in a manner satisfactory to the Regulatory Staff.

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 plant. This study will include, among other things, barge collision with the service water intake structure, spills of oil or of LNG and possible fires, and explosions of ship cargoes. If the probability of such an accident affecting the safety of the plant is not acceptably low, the applicant has agreed to provide suitable protection or make other design changes as required. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

Attention is called to the fact that the additional remarks by H. O. Monson, D. Okrent, and N. J. Palladino, appended to the Committee's report of August 10, 1971, and those by N. J. Palladino, appended to the report of July 17, 1973, were specific to the station proposed at the Newbold Island site and do not apply to the Hope Creek station.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items as well as to the non-site-related items mentioned in previous reports on the Newbold Island station, the Hope Creek Generating Station, Units 1 and 2 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

<u>References</u>

1. Newbold Island Nuclear Generating Station Preliminary Safety Analysis Report, Amendments 13, 14, 15, 16, and 17.

2. Hope Creek Generating Station, Nos. 1 and 2 Units, Preliminary Safety Analysis Report.

3. Supplement No. 2 to the Safety Evaluation Report on the Newbold Island Generating Station by the Directorate of Licensing.

4. Supplement No. 3 to the Safety Evaluation of the Hope Creek Generating Station (formerly Newbold Island Nuclear Generating Station) by the Directorate of Licensing.

5. Public Service Electric and Gas Company letter dated January 4, 1974 regarding additional information and commitments to provide additional information.

6. Public Service Electric and Gas Company letter dated January 11, 1974 regarding anticipated transients without scram (ATWS).

7. Public Service Electric and Gas Company letter dated January 11, 1974 regarding soils liquefaction studies and design heat rejection requirements.

December 18, 1984

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

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE HOPE CREEK GENERATING STATION

During its 296th meeting, December 13-15, 1984, the Advisory Committee on Reactor Safeguards reviewed the application of Public Service Electric and Gas Company (the Applicant), acting on behalf of itself and as agent for the Atlantic City Electric Company, for a license to operate the Hope Creek Generating Station. The ACRS commented on the construction permit application for the Hope Creek Generating Station in a report dated February 28, 1974. Members and consultants of the Hope Creek Subcommittee toured the facility on November 28, 1984 and met in Philadelphia, Pennsylvania on November 28 and 29, 1984 to discuss the application. During our review, we had the benefit of discussions with representatives and consultants of the Applicant, General Electric Company, Bechtel Power Corporation, and the NRC Staff. We also had the benefit of the documents referenced.

The Hope Creek Generating Station consists of one unit and is immediately adjacent to the Salem Nuclear Generating Station. Both Stations are located on Artificial Island in Salem County, New Jersey, which is approximately 18 miles south of Wilmington, Delaware. The nearest densely populated center of 25,000 or more persons is Newark, Delaware, which is approximately 18 miles northwest of the Stations. Hope Creek uses a boiling water reactor (BWR/4) with a rated power level of 3293 MWt. The nuclear reactor is similar to other previously reviewed BWRs, such as the Limerick Generating Station, the Susquehanna Steam Electric Station, and the Edwin I. Hatch Nuclear Plant. The Hope Creek primary containment is a Mark I steel vessel and the secondary containment is reinforced concrete. The pressure suppression chamber is a torus shaped steel vessel which encircles the drywell at a lower elevation.

During our meeting, the NRC Staff identified a number of open issues that must be resolved prior to the granting of an operating license. We believe that these can be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed.

We heard a report from a representative of the NRC's Region I Office that the construction quality and quality assurance effectiveness at Hope Creek were satisfactory. He indicated that there is good communication at the site and that management attention is evident.

The liquefaction potential of the soils associated with plant-related structures was evaluated by the Applicant. The Applicant indicates that soils surrounding safety-related structures are stable against liquefaction at the design basis earthquake of 0.2g. The NRC Staff agrees that none of these soils will liquefy at levels up to the design basis earthquake. We agree with the NRC Staff.

Because of the nonoptimum orientation of the turbine relative to vital components in this plant, we recommend that a structured test program for evaluating overspeed protection of the turbine be prepared and submitted to the NRC Staff for review and approval before full power operation. We wish to be kept informed.

Although the control room at the Hope Creek Generating Station has been reviewed with respect to human factors, we encourage the NRC Staff to give additional attention to its habitability requirements. This should include evaluations of the potential loss of both trains of the emergency ventilation system and the heat load and rate of temperature rise in the room under a range of HVAC conditions.

We believe that, subject to the resolution of open items identified by the NRC Staff and the items noted above, and subject to the satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Hope Creek Generating Station can be operated at power levels up to 3293 MWt without undue risk to the health and safety of the public.

Additional comments by ACRS Member Jesse C. Ebersole are presented below.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

<u>Additional Comments by ACRS Member Jesse C. Ebersole</u>

The Applicant has indicated that there will be an investigation of the current proposals by some BWR owners and by the General Electric Company to provide a simplified system to:

1. Provide an independent means to depressurize the primary coolant system.

2. Provide low pressure feedwater from a variety of sources using a small engine-driven pump or pumps.

3. Provide containment venting of steam after scrubbing through the suppression pool.

The minimum instrumentation for this system would be simple level indicators. The current GESSAR II design refers to this system as UPPS; however, the actual configuration of the system is still being considered.

The apparent overall simplicity and modest cost of this system and, if appropriately designed, the potential flexibility of the system to protect both core and containment cooling against a large number of accidents and system malfunctions would appear to justify careful consideration by both the Applicant and the NRC Staff as to its applicability to this plant.

References:

1. Public Service Electric and Gas Company, "Final Safety Analysis Report, Hope Creek Generating Station Unit 1," Volumes 1-20 and Amendments 1-8
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Hope Creek Generating Station," USNRC Report NUREG-1048, dated October 1984
3. Letter dated November 23, 1984 from Richard W. Starostecki, NRC Region I to Chester Siess, ACRS, enclosing NRC Region I Evaluation of Construction Quality at Hope Creek Generating Station as of November 1984, Presented to ACRS Subcommittee November 28-29, 1984
4. Letter dated December 12, 1984 from Bruce A. Preston, Public Service Electric & Gas Co., to C. P. Siess, ACRS, attaching responses to questions from the ACRS Subcommittee meeting of November 28-29, 1984

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

March 14, 1960

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

Subject: HUMBOLDT BAY POWER PLANT - PACIFIC GAS AND ELECTRIC COMPANY

Dear Mr. McCone:

At its twenty-fourth meeting, March 10-12, 1960, the Advisory Committee on Reactor Safeguards reviewed the proposed 200 MW (thermal) boiling water reactor and vapor suppression containment for the Pacific Gas and Electric Company at Humboldt Bay, California.

This reactor and its containment concept had previously been reviewed by the ACRS at its September and November, 1959, meetings and by the ACRS Subcommittee meetings of October 29, 1959, and February 25, 1960. The Committee reviewed the Preliminary Hazards Summary Report and subsequent Amendments Nos. 1 to 6, referenced below. The Committee had the benefit of advice from the AEC Staff and others.

Presupposing continued generally favorable experience with boiling water reactors of this type, it is the opinion of the Committee that the conceptual design of this boiling water reactor is adequate for this site with conventional pressure vessel type of containment.

Because of the high population density relatively close to this site and other unfavorable site factors, it is essential that the reactor be well contained.

It is not clear how much of the total reactor system will be housed within the vapor suppression chamber. The information so far provided does not demonstrate the suitability of the steam condensing system. Further tests are necessary.

The Advisory Committee on Reactor Safeguards believes that while the concept has merit it has not yet been demonstrated that the vapor suppression system proposed can be relied upon to protect the health and safety of the public at this site.

Sincerely yours,

/s/

Leslie Silverman
Chairman

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

References:

1) Preliminary Hazards Summary Report - Humboldt Bay Power Plant Unit No. 3, April 15, 1959.

2) Amendment No. 1 to Application of Pacific Gas and Electric Company, July 20, 1959.

3) Amendment No. 2 to Application of Pacific Gas and Electric Company, July 1959.

4) Addenda A and B - Amendment No. 3 to Application of Pacific Gas and Electric Company, September 1959.

5) Amendment No. 4 to Application of Pacific Gas and Electric Company, November 1959.

6) Amendment No. 5 to Application of Pacific Gas and Electric Company, November 30, 1959.

7) Amendment No. 6 to Application of Pacific Gas and Electric Company, January 29, 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: HUMBOLDT BAY POWER PLANT - PACIFIC GAS AND ELECTRIC COMPANY

Dear Mr. McCone:

As part of its 26th meeting, at Moss Landing, California, on June 23, 1960, the Advisory Committee on Reactor Safeguards reviewed the Pacific Gas and Electric Company's Humboldt Bay reactor project. The Committee had been supplied with Amendments 7 and 8 and opinions from the AEC staff and others on these amendments. The Committee stated in its March 14, 1960, letter that the site was suitable for a 200 MW (thermal) power reactor of the boiling water type with conventional containment but recommended further testing of the suitability of the steam suppression system. At previous meetings the Committee reviewed the preliminary hazards report and Amendments 1 through 6 pertaining to the preliminary reactor design and its vapor suppression system.

At the present stage there are several design features which the applicant is still evaluating. Among these features is the Zircaloy-2 fuel element cladding, where some concern exists regarding self-propagation of small defects, but where stainless steel could be substituted. Another is the control rod system, which adds reactivity if a rod falls downward under the influence of gravity. Consequently, special reliability of the rod-positioning devices and rod-position indication is required.

At Moss Landing the Committee witnessed a full-scale pressure suppression system test of a 1/48-segment. It appears from the actual test and on the basis of the reported series of measurements, that a satisfactory suppression system can be designed for this reactor. Application of these data to other designs would require further analysis.

The Committee believes that baffles in the pool between each vent pipe would make proof test and full-scale conditions comparable. The Committee concurs with the proposed installation of baffle plates in front of the 40-inch vent pipes and of the 40-inch ring header.

It is the Committee's opinion that the proposed suppression system adequately protects those parts of the primary system housed within the dry well, which includes the pressure vessel and approximately 20 percent of the primary piping. However, the major portion of the primary piping is outside the dry well and is uncontained against loss of coolant in the event of a pipe rupture or failure of the isolation valve. The Committee believes that double isolation valves will provide adequate protection and believes that as much of the primary piping system as possible outside of the dry well should be shrouded and the shroud vented to the pressure suppression system.

With the modifications suggested above, the Committee believes that a 200 MW (thermal) boiling water reactor of the design and features proposed can be adequately contained and that it may be constructed with reasonable assurance that it can be operated with the proposed pressure suppression system at the site selected without creating undue hazard 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

References:

(1) Amendment No. 7 to Application of Pacific Gas & Electric Company, May 6, 1960
(2) Amendment No. 8 to Application of Pacific Gas & Electric Company, May 27, 1960

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: HUMBOLDT BAY POWER PLANT - PACIFIC GAS AND ELECTRIC COMPANY

Dear Mr. McCone:

At its twenty-seventh meeting, in Washington, D. C., on July 20-22, 1960, the Advisory Committee on Reactor Safeguards met with and was advised by the applicant of its proposals to resolve the recommendations made in our letter of June 27, 1960.

As indicated in General Luedecke's letter to me dated July 21, 1960, the applicant proposed to install double isolation valves welded in tandem outside the dry well. These valves will be located immediately outside the dry well and the first valve will connect with a shroud tube which is an extension of the dry well.

It is the opinion of the Advisory Committee on Reactor Safeguards that this proposed arrangement will meet the recommendations contained in the next to last paragraph in our letter dated June 27, 1960.

Sincerely yours,

/s/

Leslie Silverman
Chairman

Reference:
Letter from A.R.Luedecke to L.Silverman dated July 21, 1960

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

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

April 4, 1962

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

SUBJECT: REPORT ON THE HUMBOLDT BAY REACTOR OF THE PACIFIC GAS AND ELECTRIC COMPANY

Dear Dr. Seaborg:

At its fortieth meeting, the Advisory Committee on Reactor Safeguards met with the applicant and AEC staff to review the request of the Pacific Gas and Electric Company for consideration for an operating license. The Committee had the benefit of a previous review with the applicant at its thirty-eighth meeting in December 1961 in regard to the proposed operation following a visit of the AEC staff and an ACRS subcommittee to the Humboldt Bay site. The Committee also reviewed the reports referenced below.

The Committee discussed with the staff and the applicant shutdown margins, burnout correlations, burnout ratios, incore monitors, safety circuits, and the detection of possible control rod separations. The Committee recognizes that several of these problems are common to a number of reactors and it is actively considering them on a general basis at present. The Committee suggests that the shutdown margin be initially set at 0.01 Δk with any rod wholly out of the core and completely unavailable. The Committee is currently reviewing this problem for a number of related reactors and suggests this value on the basis of consistence with other reactors for interim use pending a final resolution of the problem. The Committee believes that the burnout correlation and burnout ratios should remain rather conservative at present pending study of the information currently being accumulated by various groups. We recommend that, for the present, a minimum burnout ratio of 2, calculated using the applicant's suggested correlation, be specified. The Committee believes that it is not necessary to provide an automatic shutdown capability on signal from the incore monitors but that the signal from incore monitors should be displayed or checked often where this is possible, and that these monitors be maintained well and used as an essential means for gathering information.

The Committee has some doubts concerning the use of extensive neutron monitor signal smoothing circuits in conjunction with the reactor safety instrumentation. It recognizes that all circuits contain some smoothing features, but the extensive use of these circuits with their inherent loss of detailed information and time response delays has safety implications. The Committee suggests that the applicant be cautious in the use of such circuits and that the staff and the applicant consider this problem more carefully--particularly with regard to making detailed nuclear flux information available at once to the reactor operator.

The Committee suggests that the applicant adopt some method of checking for control rod separations each time the reactor is brought to critical and whenever major control rod movements are made. This check should occur as early as feasible during the rod withdrawal and should probably be dependent on neutron flux signal changes.

The Committee also reviewed in detail the question of routine and accidental releases of radioactive gases in relation to the significance of the topography and meteorology of Humboldt Bay.

After its numerous reviews and consideration of the detailed staff analysis, the Committee believes there are no serious unsolved problems in regard to design and construction existing in this reactor. The Committee is aware, however, that there are still problems in the case of radioactivity control, operating and supervisory staff, and start-up operating and emergency procedures. We believe that these are matters of concern for the AEC staff rather than subjects for further review by the ACRS.

When the above matters on operating staff and procedures are resolved between the staff and the applicant, it is the opinion of the ACRS that this reactor can be operated without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

F. A. Gifford, Jr.
Chairman

References attached

References:

1. Amendment #11 - Final Hazards Summary Report, Humboldt Bay Power Plant, Unit #3, dated September 1, 1961.
2. Amendment No. 12 - Revisions to Final Hazards Summary Report, dated Oct. 16, 1961.
3. Technical Specifications, Humboldt Bay Power Plant, Unit #3, dated Oct. 16, 1961.
4. Amendment #13 - Addendum A to Final Hazards Summary Report - Humboldt Bay Power Plant, Unit #3 with revisions to Technical Specifications, dated Feb. 26, 1962.
5. Amendment #14 - Revisions to Appendix I of Addendum A of Final Hazards Summary Report, dated March 16, 1962.
6. PG&E letter to AEC re: Information on Reactor Vessel, dated October 23, 1961.

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

September 12, 1963

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

Subject: REPORT ON HUMBOLDT BAY POWER PLANT, UNIT NO. 3 -- PACIFIC GAS AND ELECTRIC COMPANY

Dear Dr. Seaborg:

At its forty-ninth meeting on September 5 and 6, 1963, the Advisory Committee on Reactor Safeguards met with the applicant and AEC Staff to review the request of the Pacific Gas and Electric Company to carry out a fifteen-day test program, which involves a stepwise rise in power to 230 Mw(t). The Committee also reviewed the documents listed below.

A series of experiments and calculations conducted by the applicant has shown favorable behavior of the reactor at presently approved maximum power levels of 165 Mw(t). The applicant, by extrapolation, has indicated the reliability and safety of reactor operation at a proposed power level of 230 Mw(t). With a stepwise approach to power, as indicated in the application, the test can be terminated at any point where proposed limits are attained.

In its previous report on this reactor, the ACRS expressed the opinion that the heat flux relative to burnout should remain conservative pending further study. Since that time a comprehensive correlation has been made which appears conservative and is based on sufficient experimental data to justify a minimum burnout ratio limit of 1.5 as proposed rather than 2.0. Calculations and values from previous tests indicate that the minimum burnout ratio is expected to be 1.84 at the 230 Mw(t) level. Operation at 230 Mw(t) is consistent with the original design basis for this plant.

The Committee believes that the proposed test program may be conducted safely and does not constitute an undue hazard to the health and safety of the public.

Sincerely yours,

/s/

D. B. Hall
Chairman

References:

1. PG&E letter to AEC dated July 3, 1963 transmitting report: "Power Operation Testing of the Humboldt Bay Power Plant Unit No. 3", dated June 25, 1963.
2. PG&E letter to AEC dated July 18, 1963 transmitting Proposed Change No. 12, dated July 19, 1963.

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

March 17, 1965

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

Subject: REPORT ON PACIFIC GAS AND ELECTRIC COMPANY-HUMBOLDT BAY POWER PLANT, UNIT NO. 3

Dear Dr. Seaborg:

At its sixty-second meeting on March 11, 12, and 13, the Advisory Committee on Reactor Safeguards met with the applicant and AEC Regulatory Staff to review the proposal to operate the Pacific Gas & Electric Company-Humboldt Bay Power Plant, Unit No. 3 at a power level of 240 MW(t). Also discussed in detail were the proposals for use of a continuous leak rate monitoring system, the proposed integral leak testing schedule, and a proposed change in leakage specifications. The Committee had the benefit of the documents referenced below.

A series of stepwise power-increase experiments carried out by the applicant has indicated the reliability and safety of operation of this reactor up to 230 MW(t) at pressures of 1020 and 1130 psig. Steady-state and transient measurements were made of steam behavior, core thermal and hydraulic performance, and turbine plant performance. It was reported that no significant variations from predictions were observed in these tests. The plant was originally designed for operation of 230 MW(t). Extrapolation of test results to 240 MW(t) does not appear to introduce any safety problems.

The continuous leak rate monitor which is installed and will be evaluated during the coming year is considered to be a desirable adjunct for this reactor. The Committee believes that results from this system, when properly evaluated, may provide the information needed to detect certain changes in containment integrity during operation. Until experience has been gained from testing, however, the Committee is unable to assess the value of this system in measuring leakage rates.

The Committee does not wish at this time to consider any increase in containment leakage limits until the results of the continuous leak rate monitor evaluation and the proposed integral leak test to be conducted before December 1965 are available.

The Committee believes that, in light of the calculated low dose levels that would result from an MCOA with present specified leak rates, a failure to meet specifications during continuous leak rate tests or the December 1965 integrated leak test, would not constitute a serious immediate problem, so long as the difference is by a modest margin.

The Committee discussed with the applicant the question of whether the Humboldt Bay Reactor was adequately protected against tsunamis. The applicant described the present use of the U. S. Coast and Geodetic Survey warning system and its procedures to protect the plant. The Committee believes that further study of this problem is warranted.

The ACRS believes that the Humboldt Bay Power Plant, Unit No. 3 may be operated continuously at power levels of 240 MW(t) and with leak rate monitoring as proposed without creating undue hazard to the health and safety of the public.

Sincerely yours,

/s/

W. D. Manly
Chairman

References attached.

References:

1. Amendment No. 23, Pacific Gas and Electric Co., dated April 13, 1964.

2. Letter dated April 9, 1964 from Robert H. Gerdes, P.G. & E., to Director, Division of Reactor Licensing, Proposed Change No. 14.

3. Report on the Operation of Humboldt Bay Power Plant Unit No. 3, Covering the Period of August 16, 1963 through February 15, 1964, dated May 13, 1964.

4. Letter dated August 19, 1964, from S. L. Sibley, P.G. & E., to Director, Division of Reactor Licensing, Addendum A to Proposed Change No. 14.

5. Report on the Operation of the Humboldt Bay Power Plant Unit No. 3, Covering the Period of February 16 through August 15, 1964, dated September 30, 1964.

6. Amendment No. 24, Pacific Gas and Electric Co., dated December 17, 1964.

7. Letter dated December 17, 1964 from S. L. Sibley, P.G. & E., to Director, Division of Reactor Licensing, Proposed Change No. 16.

8. Letter dated September 11, 1963, from Robert H. Gerdes, P.G. & E., to Director, Division of Licensing and Regulation, Proposed Change No. 13.

9. Letter dated July 24, 1964, from S. L. Sibley, P.G. & E., to Director, Division of Reactor Licensing, Addendum A to Proposed Change No. 13.

10. Letter dated January 20, 1965, from S. L. Sibley, P.G. & E., to Director, Division of Reactor Licensing, Addendum B to Proposed Change No. 13.

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

August 10, 1965

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

Subject: REPORT ON PACIFIC GAS & ELECTRIC COMPANY -- HUMBOLDT BAY POWER PLANT, UNIT NO. 3 -- CORE II

Dear Dr. Seaborg:

At its sixty-fifth meeting, at Augusta, Georgia, and the Savannah River Laboratory, on August 5-7, 1965, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Pacific Gas & Electric Company to operate the Humboldt Bay Power Plant, Unit No. 3, with Type II fuel. The Committee had the benefit of discussions with representatives of the Pacific Gas & Electric Company, the General Electric Company, and the AEC Regulatory Staff and of the documents referenced below. The Committee had previously reviewed a number of proposed modifications to this reactor and its engineered safeguards at its sixty-fourth meeting, on July 8-10, 1965. A Subcommittee of the ACRS met with the applicant on July 7, 1965.

Unit No. 3 of the Humboldt Bay Power Plant has a boiling water reactor with pressure suppression containment. It has been approved for operation at powers up to 240 MW(t). It was originally fueled with stainless steel clad, enriched uranium oxide and has had nearly two years of successful operation. In the Final Hazards Summary Report it was stated that a core with Zircaloy-2 clad would be considered at a later date.

The applicant now proposes to reload the Humboldt Bay Reactor in four stages by substituting Zircaloy-2 clad fuel assemblies for the original stainless steel clad fuel. In addition, sixteen control rod assemblies will be replaced. Approximately one-quarter of the core would be reloaded as soon as possible after the plant modifications described in Proposed Change No. 17 and Addenda A and B are completed. The remainder of the core would be loaded at approximately 8 to 12 month intervals until a complete fuel change has been accomplished.

The applicant proposes to modify the post-accident core cooling so that it will consist of three distinct systems. One of these is the originally installed core spray; the second is the feedwater and condensate system; the third is an additional source of core flooding which is to be provided by connecting the yard fire system to the reactor. The applicant states that, in the unlikely event of a major rupture of the primary coolant system, any one of these sources is capable of providing adequate after-heat removal for the core.

The applicant has calculated the consequences of zirconium-water reactions resulting from an unlikely loss-of-coolant accident, assuming adequate emergency core cooling on the one hand, and a failure of all emergency core cooling, on the other. Appreciable metal-water reaction may occur for the latter conditions, producing hydrogen and leading to increased pressure within the containment. The applicant proposes to inert the dry well and the suppression pool chamber with nitrogen during reactor operation to prevent a subsequent hydrogen-oxygen reaction. The applicant also proposes to strengthen the suppression pool chamber in accordance with applicable construction codes. This will allow an increase in design pressure of the suppression pool chamber from 10 psig. to 25 psig. After this modification, the applicant plans to test the containment to 28.75 psig.

It is the opinion of the Advisory Committee on Reactor Safeguards that the proposed system modifications presented by the applicant provide reasonable assurance that Type II fuel can be installed in the Humboldt Bay Unit No. 3 reactor, as proposed, and operated at the approved maximum power level without undue hazard to the health and safety of the public. In view of some current uncertainties about the extent and character of metal-water reactions, the Committee believes it is desirable that the applicant and the AEC Staff continue to evaluate the metal-water reaction problem which could result from the unlikely loss-of-coolant accident. The Committee expects that such studies will be completed before the third quarter of Type II fuel is loaded and would like to be informed of the results of these studies.

Sincerely yours,

/s/

W. D. Manly
Chairman

References attached.

REFERENCES - HUMBOLDT BAY

1. Letter dated April 9, 1965 from S. L. Sibley, Pacific Gas & Electric Co., to Director, Division of Reactor Licensing, Proposed Change No. 17.

2. Letter dated June 15, 1965 from S. L. Sibley, Pacific Gas & Electric Co., to Director, Division of Reactor Licensing, Addendum A to Proposed Change No. 17.

3. Letter dated July 26, 1965 from S. L. Sibley, Pacific Gas & Electric Co., to Director, Division of Reactor Licensing, Addendum B to Proposed Change No. 17.

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

March 12, 1968

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

Subject: REPORT ON PACIFIC GAS AND ELECTRIC COMPANY - HUMBOLDT BAY POWER PLANT, UNIT NO. 3

Dear Dr. Seaborg:

At its ninety-fourth meeting, on February 8, 9, and 10, 1968, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Pacific Gas and Electric Company to have their provisional operating license No. DPR-7 converted to a full-term, 40-year operating license. The Committee had the benefit of discussions with representatives of the Pacific Gas and Electric Company, General Electric Company, AEC Regulatory Staff and their consultants, and of the documents listed below. A Subcommittee of the ACRS met with the licensee on January 29, 1968, to discuss this matter.

Humboldt Bay Power Plant, Unit No. 3, a direct-cycle boiling-water reactor with pressure-suppression containment, began commercial power operation in August, 1963. After two years of operation with Core I, of stainless-steel clad, uranium oxide fuel, a reloading with Zircaloy-2 clad fuel was carried out in four stages. This reloading was completed in October, 1967. The operating history of the unit has been generally satisfactory; its availability factor has averaged about 85%.

The licensee described the results of a study of the effect of tsunamis on the unit as requested by the ACRS in its report of March 17, 1965. Further analysis of the effect of a tsunami on the end wall of the suppression pool is planned.

It is desirable that means be developed and provided to guide or implement decisions concerning reactor operation in the event of a large earthquake in the region of the site.

The AEC Regulatory Staff has requested of the licensee: an updated seismic study of the site; preparation and implementation of an improved primary system in-service inspection program, including non-destructive examination of selected portions of piping; and, a single-failure mode analysis of the protection system. The ACRS recommends that these studies be completed expeditiously and that the resulting program, together with any system modifications found appropriate, be implemented as soon as practicable. The Committee also recommends that the licensee give further consideration to quantitative aspects of primary system leak detection and to appropriate operating procedures in response to leak detection signals.

To reduce the possibility of undesirable core and primary system damage in the unlikely event of a rod dropout reactivity insertion accident, the Regulatory Staff has recommended that the Technical Specifications be revised to limit the maximum in-sequence control rod worth to a value more conservative than that proposed by the applicant. The Committee agrees that the more conservative approach is preferable.

Based on its present review, the ACRS reaffirms its previous conclusion, that the Humboldt Bay Unit No. 3 reactor can be operated without undue hazard to the health and safety of the public, and recommends conversion of the present provisional operating license to a full-term operating license.

Sincerely yours,

/s/

Carroll W. Zabel
Chairman

References attached.

References - Humboldt Bay

1. Proposed Change No. 21 to Technical Specifications, Pacific Gas and Electric Company, dated February 24, 1966

2. Proposed Change No. 22 to Technical Specifications, Pacific Gas and Electric Company, dated April 16, 1966

3. Letter from Pacific Gas and Electric Company, dated June 8, 1966; Humboldt Bay Power Plant Unit No. 3, Report on Tsunamis

4. Letter from Pacific Gas and Electric Company, dated August 22, 1966; Environmental Radiological Monitoring Program.

5. Proposed Change No. 23 to Technical Specifications, Pacific Gas and Electric Company, dated September 10, 1966

6. Addendum A to Proposed Change No. 22, Pacific Gas and Electric Company, dated October 3, 1966

7. Addendum B to Proposed Change No. 22, Pacific Gas and Electric Company, dated October 31, 1966

8. Addendum C to Proposed Change No. 22, Pacific Gas and Electric Company, dated February 28, 1967

9. Addendum A to Proposed Change No. 21, Pacific Gas and Electric Company, dated March 16, 1967

10. Letter from Pacific Gas and Electric Company, dated May 17, 1967; Emergency Core Cooling Systems

11. Letter from Pacific Gas and Electric Company, dated August 23, 1967; Refueling with Zircaloy Clad Fuel

12. Addendum D to Proposed Change No. 22, Pacific Gas and Electric Company, dated December 15, 1967

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

March 12, 1970

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

Subject: REPORT ON HUTCHINSON ISLAND PLANT UNIT NO. 1

Dear Dr. Seaborg:

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

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

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

The proposed pressurized water reactor has a design power level of 2440 MW(t) and is similar to the previously reviewed Maine Yankee and Calvert Cliffs reactors (ACRS reports dated July 19, 1968 and March 13, 1969). The containment system consists of a steel containment vessel enclosed within a reinforced concrete building, with the annular space maintained at a slight negative pressure and exhausted through filters. The applicant has stated that the containment and other structures and systems important to safety will be designed to meet the same tornado design criteria as have been used for other recently reviewed plants, and that protection of vital components will be provided against the probable maximum hurricane-induced flood and runup level as estimated by the Coastal Engineering Research Center.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Sincerely yours,

/s/

Joseph M. Hendrie
Chairman

References attached.

References - Hutchinson Island Plant Unit No. 1

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

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

3. Amendments 1 - 8 to License Application.

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

June 8, 1960

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

Subject: IMPROVED CYCLE BOILING WATER REACTOR (ICBWR) - DEPARTMENT OF WATER AND POWER OF THE CITY OF LOS ANGELES

Dear Mr. McCone:

At a special meeting of the Advisory Committee on Reactor Safeguards held in Boston on June 7, 1960, the site for the improved cycle prototype boiling water reactor (ICBWR) for the Department of Water and Power of the City of Los Angeles was considered. Members of the staff of the Atomic Energy Commission met with the Committee to discuss this problem. Drs. Silverman and McCullough, together with Mr. John Newell of the Hazards Evaluation Branch, had visited the site and made and aerial survey on Saturday, June 4th. The Committee had for review a site report from the Department of Water and Power of the City of Los Angeles and, in addition, had the benefit of comments from the Atomic Energy Commission staff and others.

It is the opinion of the Committee that before a reactor is constructed at this site a detailed environmental survey on its hydrological and meteorological aspects should be completed, the latter stressing the atmospheric dispersion characteristics peculiar to this site.

The Committee is aware that the Department of Water and Power of the City of Los Angeles is contemplating the use of this site "for future Dresden size plants" in addition to the proposed 50 MWE prototype boiling water reactor. It is the Committee's understanding that its opinion is being sought only on the suitability of this site for the 50 MWE reactor. On the basis of the preliminary data now available, the Committee concludes that this site is suitable for a 50 MWE boiling water reactor which follows existing technology. The Committee has doubts whether this site can be safely expanded into a large power complex.

Sincerely yours,

/s/

Leslie Silverman
Chairman

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

Reference

Description of Site for Nuclear Power Plant by Department of Water and Power of the City of Los Angeles, undated, received by ACRS on May 17, 1960.

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

June 13, 1960

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

Subject: SPECIAL MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS HELD IN BOSTON, MASSACHUSETTS, ON JUNE 7, 1960.

Dear Mr. McCone:

At the Commission's request, the Advisory Committee on Reactor Safeguards convened a special meeting of the full Committee which was held in Boston, Massachusetts, on June 7, 1960.

A Proposal by the Department of Water and Power of the City of Los Angeles to locate an improved cycle boiling water reactor near Saugus, California, was reviewed by the Committee and discussed with representatives of the Division of Reactor Development and the Hazards Evaluation Branch. A separate letter expressing the Committee's opinion on the suitability of this site has been sent to you.

The Committee heard a presentation by members of the Oak Ridge Operations Office and its contractors on the Small Size Pressurized Water Reactor proposed for location in the City of Jamestown, N. Y. The Committee has had insufficient time to study the changes in the engineering safeguards recently proposed for this reactor and was unable to reach a conclusion on the suitability of the site selected for the reactor. The ACRS Subcommittee on this project plans further study of this new information and the full Committee will review this case at its scheduled meeting on June 22, 1960 at Livermore, California.

Because of the Commission's request for a special ACRS meeting to provide advice on the Improved Cycle Boiling Water Reactor and the Small Size Pressurized Water Reactor, it became necessary for the ACRS Environmental Subcommittee to cancel its planned meeting of June 7th to consider the site criteria problem. In view of the Committee's other business, it is not certain when in the immediate future the Subcommittee will be able to meet for further study of this problem.

In the interim between the Committee's regular meeting May 5-7 and the special meeting of June 7, there was a meeting of the PRDC Subcommittee (Enrico Fermi Reactor) on May 13 in Washington. The Subcommittee, together with members of the Hazards Evaluation Branch, heard a presentation on the status of fuel element design, detection of melted fuel, cover and waste gas systems, and the building ventilation system. Further study of this reactor system is planned by the Subcommittee for July 13-14 at Cambridge, Massachusetts.

Sincerely yours,

/s/

Leslie Silverman
Chairman

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

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

June 27, 1960

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

Subject: IMPROVED CYCLE BOILING WATER REACTOR (ICBWR) DEPARTMENT OF WATER AND POWER OF THE CITY OF LOS ANGELES

Dear Mr. McCone:

It is our understanding that more clarification is desired by the Commission in regard to statements in Paragraph 2 and Paragraph 3 of our June 8th letter, subject as above. This letter is also in reply to Mr. Finan's letter of June 14, 1960.

The detailed environmental survey recommended in Paragraph 2 of our letter is required primarily for the purpose of determining the necessary "engineered safety features: of the reactor such as particularly low containment leak rates, holdup systems for gaseous and liquid effluents and recirculating cleaning units. The information presented to date supports the conclusion "that this site is suitable for a 50 MWE boiling water reactor which follows existing technology".

The improved cycle refers essentially only to modifications proposed for the steam generation equipment; and that the proposed reactor will not differ significantly from previous boiling water reactors. The modifications proposed for the steam generating equipment for the ICBWR would fall within "existing technology" if the applicant can show that these modifications have no significant effect on the nuclear safety of the reactor.

Sincerely yours,

/s/

Leslie Silverman
Chairman

cc: A.R.Luedecke, GM
W.F.Finan, OGM
W.L.Price, DL&R

Reference: Letter from W.F.Finan to L.Silverman dated June 14, 1960

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: IMPROVED CYCLE BOILING WATER REACTOR

Dear Mr. McCone:

At its twenty-seventh meeting in Washington, D. C., on July 20-22, 1960, the Advisory Committee on Reactor Safeguards considered two alternate sites proposed for the improved cycle prototype boiling water reactor (ICBWR). These are the Upper San Francisquito Canyon site and the Haskell Canyon site; both are within 2.5 miles of the site originally proposed.

On the basis of the preliminary site information that has been supplied and referenced below these sites appear to be essentially equivalent to the site originally proposed. The Committee, accordingly, concludes that either is suitable from the safety standpoint for a 50-MWE boiling water reactor as described in our previous letters dated June 8, 1960, and June 27, 1960. The opinion of the Committee concerning the necessity for a detailed environmental survey of the original site expressed in the previous letters applies equally to the proposed alternate sites.

Sincerely yours,

/s/

Leslie Silverman
Chairman

References:
(1) Memo - K.A.Dunbar to F.K.Pittman, dated July 14, 1960, "Evaluation of Haskell Canyon Site Proposed by the Department of Water and Power - City of Los Angeles, California, for the Improved Cycle Boiling Water Reactor"
(2) Memo - K.A.Dunbar to F.K.Pittman, dated July 14, 1960, "Evaluation of Upper San Francisquito Canyon Site Proposed by the Department of Water and Power - City of Los Angeles, California, for the Improved Cycle Boiling Water Reactor"

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

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

January 14, 1961

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

Subject: IMPROVED CYCLE BOILING WATER REACTOR

Dear Mr. McCone:

The letter from Mr. William S. Peterson, General Manager of the Los Angeles Department of Water and Power, to Mr. Price, referred to the Advisory Committee on Reactor Safeguards by Mr. W. F. Finan, contains four questions relative to expansion of the Haskell Canyon site which has been approved for a 50 MW(e) boiling water reactor (ICBWR) into a large reactor complex. Each question will be considered separately.

1. The ACRS letter of June 8, 1960, after concluding that the site is suitable for "a 50 MW(e) boiling water reactor which follows existing technology," states - "The Committee has doubts whether this site can be safely expanded into a large power complex." The selection of a reactor manufacturer does not alter this conclusion.

2. The reservations in connection with a large reactor complex at Haskell Canyon are related to the site. The use of pressure suppression design probably would not affect the acceptability of the site for a possible 300 MW(e) reactor.

3. The letter of June 8 stated - "It is the opinion of the Committee that before a reactor is constructed at this site, a detailed environmental survey on its hydrological and meteorological aspects should be completed, the latter stressing the atmospheric dispersion characteristics peculiar to this site." Again on July 25, 1960, the ACRS letter stated - "The opinion of the Committee concerning the necessity for a detailed environmental survey of the original site expressed in the previous letters applies equally to the proposed alternate sites." Thus, the

ACRS reservations even on a 50 MW(e) reactor at Haskell Canyon were expressed because of unknowns about the site. It is extremely doubtful that clarification of these unknowns would make it prudent to use this site for a larger power complex. The precise number of additional reactor units would not materially affect our position.

4. The experience in the operation of a 50 MW(e) nuclear plant on a site being considered for a larger reactor probably would not be a significant factor in a decision on the suitability of the site for a larger reactor.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

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

References:
Letter from Wm. S. Peterson to H. L. Price, dated December 28, 1960
Letter from W. F. Finan to T. J. Thompson, dated January 11, 1961

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 DAIRYLAND POWER COOPERATIVE SITE FOR IMPROVED CYCLE BOILING WATER REACTOR

Dear Dr. Seaborg:

At the request of the Director of the Division of Licensing and Regulation the Advisory Committee on Reactor Safeguards at its thirty-fourth meeting, May 18-20, 1961, considered the Dairyland Power Cooperative Site near Genoa, Wisconsin for the 50 MWe (approximately 175 MWt) Improved Cycle Boiling Water Reactor.

The same Dairyland site was considered and found suitable for a 200 MWt organic-cooled prototype reactor by the Committee at its thirty-third meeting on April 6-8, 1961.

A preliminary description of the Improved Cycle Boiling Water Reactor was presented to the Committee in 1960 for siting in California.

It is the opinion of the Advisory Committee on Reactor Safeguards that the presently proposed Dairyland site is suitable for a reactor of this general type and power level.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

(References attached)

References:

1. Letter R. L. Kirk to T. J. Thompson, dated May 18, 1961.

2. Letter T. J. Thompson, Chairman, ACRS to Hon. Glenn T. Seaborg, dated April 10, 1961; Subject: Report on Sites for A 200 MW Thermal Organic Cooled Prototype Reactor.

3. Letter T. J. Thompson, Chairman, ACRS to Hon. John A. McCone, dated January 14, 1961; Subject: Improved Cycle Boiling Water Reactor.

4. Letter L. Silverman, Chairman, ACRS to Hon. John A. McCone, dated July 25, 1960; Subject: Improved Cycle Boiling Water Reactor.

5. Letter L. Silverman, Chairman, ACRS to Hon. John A. McCone, dated July 25, 1960; Subject: Dairyland Site Near Genoa, Wisconsin, for the Small Pressurized Water Reactor (SPWR).

6. Letter L. Silverman, Chairman, ACRS to Hon. John A. McCone, dated June 27, 1960; Subject: Improved Cycle Boiling Water Reactor (ICBWR) Department of Water and Power of the City of Los Angeles.

7. Letter L. Silverman, Chairman, ACRS to Hon. John A. McCone, dated June 8, 1960; Subject: Improved Cycle Boiling Water Reactor (ICBWR) - Department of Water and Power of the City of Los Angeles.

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

January 12, 1959

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

Subject: CONSOLIDATED EDISON REACTOR

Dear Mr. McCone:

At its twelfth meeting, December 11-13, 1958, the Advisory Committee on Reactor Safeguards reviewed the containment proposed for the nuclear power station being built for the Consolidated Edison Company at Indian Point, New York. Members of the Hazards Evaluation Branch and representatives of the Consolidated Edison Company, Babcock & Wilcox Company, and the Vitro Corporation of America participated in the discussion. Documents pertinent to this proposal are referenced below. The Committee considered this proposal further at its thirteenth meeting, January 8-10, 1959.

The proposed steel sphere and the heavy-walled concrete building enclosing it will provide the most nearly complete containment presented to the Committee in any reactor project to date. The Committee is satisfied that this containment will provide protection to the public. This favorable opinion relates to the proposed containment structures only. Information regarding the safety of the reactor itself and its operation will be considered later.

Sincerely yours,

C. Rogers McCullough
Chairman

cc: Alvin R. Luedecke, GM
Harold L. Price, DLR

References:

1) Evaluation of Potential Radiation Hazard Resulting from Assumed Release of Radioactive Wastes to Atomosphere from the Proposed Buchanan Nuclear Power Plant, April 1957.

2) Core Design and Characteristics for the Consolidated Edison Reactor, August 18, 1958.

3) Report on Hazards Analysis and Design for Containment Vessel for the Consolidated Edison Reactor, August 29, 1958.

4) Division of Licensing and Regulation Report to the Advisory Committee on Reactor Safeguards on Consolidated Edison Company of New York Indian Point Reactor, December 3, 1958.

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 CONSOLIDATED EDISON REACTOR DESIGN

Dear Dr. Seaborg:

At its thirty-first meeting on January 12-14, 1961, and its thirty-second meeting on March 2-4, 1961, the Advisory Committee on Reactor Safeguards considered the design of the Consolidated Edison 585 MW thermal uranium oxide thorium oxide fueled pressurized water reactor under construction at Indian Point near Peekskill, New York. The Committee had the benefit of discussions with the applicant, the AEC staff and their consultants. The Committee explored features of the design with the applicant, the applicant's contractors and the applicant's consultants.

Many questions raised by the Committee have been answered to the Committee's satisfaction. Although the applicant has not yet documented his proposals on control rod mechanisms and core information system, an adequate oral description was given to the Committee and the following is based on that presentation.

The question of integrity of control rod mechanism of water reactors has been raised by recent failures in other reactors of 17-4 PH stainless steel parts attributed to stress corrosion. The Consolidated Edison control rod mechanisms have been fabricated and are now undergoing an extensive testing program. The Committee notes that Consolidated Edison has placed on order a duplicate set of control rod drive shafts heat treated at 1100°F., instead of 900°F. specified for the present rods, in accordance with optimum specifications as indicated by the recent extensive survey of experience in other reactors. The Committee believes that the design of these units and the applicant's proposed modifications in fabrication will not result in a hazard to the health and safety of the public. The Committee suggests that the staff follow the fabrication details of these units.

In response to the several discussions between the HEB staff, the Committee and Consolidated Edison concerning the absence of in-core monitoring devices or other means of measurement of local power levels, Consolidated Edison proposed on March 3 that they will remove and gamma scan a portion of the fuel elements. This should provide a reasonable check on precalculated core performance at an intermediate point or points during the core life. Consolidated Edison expressed confidence that this scanning, and other operating procedures yet to be worked out in detail, will enable them to operate the core without in-core monitors and without excessive heat and neutron fluxes in any part of the reactor. The frequency and time in core life when these measurements will be taken will be discussed with the applicant as a part of the review of operating procedures. The Committee believes that there is considerable assurance that the reactor, as designed, can operate at designed power. However, the question of whether full power operation can actually be reached without in-core monitors will depend upon data which can be obtained only during the initial operation of the reactor at power levels less than full power.

The Committee's attention thus far has been directed solely at design of the Consolidated Edison facility. Staffing, operating procedures, start-up program, etc., will be considered following a review of the applicant's reports not yet furnished.

The ACRS finds the Consolidated Edison reactor design such that it may be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

References:

1. Report on Hazards Analysis and design for Containment Vessel, dated Sept. 18, 1959.
2. Hazards Summary Report, dated Jan. 1960.
3. Amendment #11, dated April 21, 1960, to the Application for Licenses.
4. Amendment No. 7 to Application for Licenses, undated, received April 27, 1959.
5. Reactor Vessel Internal Components Design (BAW-136), dated July 1960.

6. Fuel Element Structural Design and Manufacture (BAW-133), dated Sept. 1960.
7. Design of the Movable and Fixed Control Components (BAW-147), dated Aug. 1960.
8. Irradiation Test Program (BAW-134), dated Aug. 1960.
9. Thermal and Hydraulic Design (BAW-132), dated July 1960.
10. Physics Design (BAW-120, Rev. 1), revised July 1960.
11. Critical Experiments with Oxide Fuel Pins (BAW-119, Rev. 1), dated July 1960.
12. Hot Exponential Experiment (BAW-116, Rev. 1), revised June 1960.
13. Geometric and Temperature Effects in Thorium Resonance Capture (BAW-144), dated June 1960.
14. Control Rod Drive Line Testing, dated Aug. 1960.
15. Supplementary Information on Plant Design of Con-Ed Nuclear Steam Generating Station, dated Aug. 1960.
16. Functional Design Analysis of the Pressurizer (BAW-41, Rev. 1), revised June 1960.
17. The Effects of Fuel Rod Fission Product Leakage (BAW-85, Rev. 1), revised June 1960.
18. Corrosion Product Activity Distribution Across the Chemical Processing System (BAW-142, Rev. 1), revised Aug. 1960.
19. Control System Design (BAW-138), dated Aug. 1960.
20. Amendment #14, dated Nov. 23, 1960 and attachments to the Application for Licenses.
21. Amended and Substituted Application for Licenses, dated Nov. 30, 1960 and exhibits.
22. Amendment #1, dated Dec. 9, 1960, to the Amended and Substituted Application for Licenses.
23. Amendment #2 and attachments, dated Feb. 14, 1961, to the Amended and Substituted Application for Licenses.

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.

November 1, 1961

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

Subject: REPORT ON CONSOLIDATED EDISON REACTOR

Dear Dr. Seaborg:

At its thirty-sixth and thirty-seventh meetings on September 7-9, and October 26-28, 1961, the Advisory Committee on Reactor Safeguards considered previously unresolved questions regarding the design and operation of the Consolidated Edison 585 MW(t) reactor at Indian Point, New York. The Committee had the benefit of discussions with the applicant and the AEC staff.

It its final review, the Committee considered results of the critical experiments performed at Lynchburg, Virginia and their correlation with calculations of predicted reactor performance. The Committee has also reviewed the organization and procedures to which the applicant is committed.

In our letter of March 4, 1961 to you, the Committee discussed the applicant's proposal to gamma scan fuel elements at an intermediate point in the core life. Consolidated Edison has now made this proposal part of the application.

In view of the satisfactory degree of confirmation of the applicant's power distribution calculations afforded by the Lynchburg tests, the ACRS does not consider necessary its formal review of results of specific steps in the normal step-wise procedure according to which this reactor will be brought to power. However, as for all new reactors, the Committee would like to be kept informed of the performance of this reactor.

It is the opinion of the ACRS that this reactor can be operated without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

T. J. Thompson
Chairman

References (attached)

References:

1. Amendment #4, dated May 4, 1961, with enclosures, to the Amended and Substituted Application for Licenses.

2. Amendment #5, dated August 11, 1961, with enclosure, to the Amended and Substituted Application for Licenses.

3. Amendment #6, dated August 24, 1961, with enclosures, to the Amended and Substituted Application for Licenses.

4. Amendment #7, dated September 14, 1961, with enclosures, to the Amended and Substituted Application for Licenses.

5. Amendment #8, dated September 26, 1961, with enclosure, to the Amended and Substituted Application for Licenses.

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

May 20, 1965

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

Subject: REPORT ON CONSOLIDATED EDISON INDIAN POINT REACTOR CORE B

Dear Dr. Seaborg:

At its sixty-third meeting, May 13-15, 1965, the Advisory Committee on Reactor Safeguards considered the application of the Consolidated Edison Company of New York, Inc., to replace its present core containing thorium oxide (Core A) with a low-enrichment uranium oxide core (Core B), and to increase the maximum steady state power level of the reactor plant from 585 to 615 Mw(t). The Committee had the benefit of discussions with representatives of the Consolidated Edison Company, the Westinghouse Electric Corporation, and the AEC Staff. The Committee also had the benefit of the documents referenced.

The proposed Core B consists of three concentric regions each containing 40 fuel assemblies. The initial loading will utilize enrichments of 2.86, 3.26 and 4.08 weight per cent respectively in the central, intermediate and outer regions. The UO_2 is in the form of sintered pellets which are contained in cold-worked type 304 stainless steel cladding. The fuel rods are supported within perforated stainless steel box assemblies which are to be installed into the present core structure as modified to reduce by-pass flow.

Reactivity control for Core B will be accomplished by a combination of control rods and chemical shim. New control rods fabricated from stainless steel tubes filled with silver-indium-cadmium alloy and having Zircaloy-2 followers are to be installed in Core B. In addition, boric acid is to be used in the primary coolant to provide for long term reactivity changes. The applicant stated that the boron concentration will be sufficient to maintain a shut-down margin of at least 0.5% delta k/k with the highest worth control rod fully withdrawn.

The applicant stated that the maximum allowable reactivity anomaly in the reactor would be kept below 1.25% delta k/k. The applicant also stated that the void coefficient, over-all and locally, of Core B utilizing borated water would be negative at all times. If boron concentrations leading to local positive void coefficients are contemplated, the Committee recommends that studies of the influence of such coefficients on reactor safety be made prior to their use.

To provide for operation of the reactor at 615 Mw(t) and for utilization of boric acid in the coolant, several changes have to be made in the present plant. Chief among these are the increase of primary and secondary flow, alteration of certain control and safety set points, addition of scram trips to isolate the steam boilers, and installation of independent supply lines from each boron storage tank to the two high pressure boron injection pumps.

In evaluating the safety of the plant when operating with Core B, the applicant used a 1% per day containment leak rate instead of the original design rate of 0.1% per day. In order to show that the higher leak rate would be acceptable, the applicant took credit for the annular space between the containment and the biological shield, which space is exhausted to the 400-foot superheater stack; for purposes of analysis it was assumed that the stack was cold.

The question of contaminating potable water supplies in the unlikely event of a severe accident accompanied by rainout was discussed. The applicant assured the Committee that within ten miles from the reactor there are no water supply reservoirs for which there are no alternates.

The Committee does not wish to consider any increase in containment leakage limits until the results of the continuous leak rate monitor evaluation and the proposed integral leak test, to be conducted at core changeover, are available.

With the above reservations, the ACRS believes that the Consolidated Edison - Indian Point Reactor with Core B can be operated at 615 Mw(t) without creating undue hazard to the health and safety of the public.

Sincerely yours,

/s/

W. D. Manly
Chairman

References Attached.

References (Consolidated Edison)

1. Final Hazards Summary Report for the Consolidated Edison Indian Point Reactor Core B.

2. Supplement to Final Hazards Summary Report for Consolidated Edison Indian Point Reactor Core B (Appendix B).

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

August 16, 1966

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

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Dear Dr. Seaborg:

At its seventy-fifth meeting, July 14-16, 1966, and its special meeting on August 4-5, 1966, the Advisory Committee on Reactor Safeguards completed its review of the application of Consolidated Edison Company of New York, Inc. for authorization to construct Indian Point Nuclear Generating Unit No. 2. This project had previously been considered at the seventy-second and seventy-third meetings of the Committee, and at Subcommittee meetings on March 30, May 3, and June 23, 1966. During its review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants and with representatives of the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents listed.

The Indian Point 2 plant is to be a pressurized water reactor system utilizing a core fueled with slightly enriched uranium dioxide pellets contained in Zircaloy fuel rods; it is to be controlled by a combination of rod cluster-type control rods and boron dissolved in the primary coolant system. The plant is rated at 2758 MW(t); the gross electrical output is estimated to be 916 MW(e). Although the turbine has an additional calculated gross capacity of about 10%, the applicant has stated that there are no plans for power stretch in this plant.

The Indian Point 2 facility is the largest reactor that has been considered for licensing to date. Furthermore, it will be located in a region of relatively high population density. For these reasons, particular attention has been given to improving and supplementing the protective features previously provided in other plants of this type.

The proposed design has a reinforced concrete containment with an internal steel liner which is provided with facilities for pressurization of weld areas to reduce the possibility of leakage in these areas. The containment design also includes an internal recirculation containment spray system and an air recirculation system consisting of five air handling units to provide long-term cooling of the containment without having to pump radioactive liquids outside the containment in the event of an accident. Even though the applicant anticipates negligible leakage from the containment, two independent means of iodine removal within the containment have been provided. These are an air filtration system using activated charcoal filters, and a containment spray system which uses sodium thiosulfate in the spray water as a reagent to aid removal of elemental iodine.

The reactor vessel and various other components of the system are surrounded by concrete shielding which provides protection to the containment against missiles that might be generated if structural failure of such components were to occur during operation at pressure. This includes missile protection against the highly unlikely failure of the reactor vessel by longitudinal splitting or by various modes of circumferential cracking. The Committee favors such protection for large reactors in regions of relatively high population density.

The Indian Point 2 plant is provided with two safety injection systems for flooding the core with borated water in the event of a pipe rupture in the primary system. The emergency core cooling systems are of particular importance, and the ACRS believes that an increase in the flow capacity of these systems is needed; improvements of other characteristics such as pump discharge pressure may be appropriate. The forces imposed on various structural members within the pressure vessel during blowdown in a loss-of-coolant accident should be reviewed to assure adequate design conservatism. The Committee believes that these matters can be resolved during construction of these facilities. However, it believes that the AEC Regulatory Staff and the Committee should review the final design of the emergency core cooling systems and the pertinent structural members within the pressure vessel, prior to irrevocable commitments relative to construction of these items.

The applicant stated that, even if a significant fraction of the core were to melt during a loss-of-coolant accident, the melted portion would not penetrate the bottom of the reactor pressure vessel owing to contact of the vessel with water in the sump beneath it.

The applicant also proposes to install a backup to the emergency core cooling systems, in the form of a water-cooled refractory-lined stainless steel tank beneath the reactor pressure vessel. The Committee would like to be advised of design details and their theoretical and experimental bases when the design is completed.

In order to reduce still further the low probability of primary system rupture, the applicant should take the additional measures noted below. The Committee would like to review the results of studies made by the applicant in this connection, and consequent proposals, as soon as these are available.

1. Design and fabrication techniques for the entire primary system should be reviewed thoroughly to assure adequate conservatism throughout and to make full use of practical, existing inspection techniques which can provide still greater assurance of highest quality.

2. Great attention should be placed in design on in-service inspection possibilities and the detection of incipient trouble in the entire primary system during reactor operation. Methods of leak detection should be employed which provide a maximum of protection against serious incidents.

Attention should also be given to quality control aspects, as well as stress analysis evaluation, of the containment and its liner. The Committee recommends that these items be resolved between the AEC Regulatory Staff and the applicant as adequate information is developed.

The applicant has made studies of reactivity excursions resulting from the improbable event that structural failure leads to expulsion of a control rod from the core. Such transients should be limited by design and operation so that they cannot result in gross primary-system rupture or disruption of the core, which could impair the effectiveness of emergency core cooling. The reactivity transient problem is complicated by the existence of sizeable positive reactivity effects associated with voiding the borated coolant water, particularly early in core life. In addition, the course of the transients is sensitive to various parameters, some of which remain to be fixed during the final design. Westinghouse representatives reported that the magnitude of such reactivity transients could be reduced by installation of solid burnable poisons in the core to permit reduction of the soluble boron content of the moderator, thereby reducing the positive moderator coefficient. The Committee agrees with the applicant's plans to be prepared to install the burnable poison if necessary. The Committee wishes to review the question of reactivity transients as soon as the core design is set.

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 Indian Point site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
David Okrent
Chairman

References:

1. Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2, Preliminary Safety Analysis Report, Volume 1, and Volume 2, Parts A & B, received December 7, 1965.
2. First Supplement to Preliminary Safety Analysis Report, dated March 31, 1966.
3. Second Supplement to Preliminary Safety Analysis Report, received June 2, 1966.
4. Errata Sheets for Preliminary Safety Analysis Report and First Supplement thereto, received June 13, 1966.
5. Third Supplement to Preliminary Safety Analysis Report, received June 22, 1966.
6. Fourth Supplement to Preliminary Safety Analysis Report, received July 28, 1966.
7. Fifth Supplement to Preliminary Safety Analysis Report, received July 28, 1966.

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

January 15, 1969

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

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

Dear Dr. Seaborg:

At its 105th meeting, January 9-11, 1969, the Advisory Committee on Reactor Safeguards completed its review of the application of Consolidated Edison Company of New York, Inc., for authorization to construct Indian Point Nuclear Generating Unit No. 3. This project had previously been considered at the 103rd meeting of the Committee, and at Subcommittee meetings on October 22, 1968, and December 28, 1968. During its review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants, and with representatives of the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents listed.

Indian Point Unit No. 3 includes a four-loop nuclear steam supply system with a design power rating of 3025 MW(t). The design is very similar to that of Unit No. 2 except for differences in power level and some of the engineered safety features. The peak values of core heat flux and linear heat generation rate are slightly lower than those proposed for the reactors of the Zion Station.

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 vessel cavity walls will be designed to withstand the mechanical forces which would result if a highly unlikely vessel split were to occur with the primary system pressurized. Design of the system will be such as to permit annealing of the reactor pressure vessel, if this should become necessary.

The applicant proposes to install flame recombiners to cope with potential hydrogen concentration buildup from various sources in the unlikely event of a loss-of-coolant accident. He has described a research and development program to ascertain the need for a recombiner, to study other types of recombiners, and to confirm acceptable performance. The applicant also described measures to be taken in the design and operation to prevent inadvertent introduction of hydrogen into the containment.

The on-site emergency power supply for Unit No. 3 employs four 480 V buses energized (upon loss of normal power) by three diesel generators, two of which are required to furnish energy to engineered safety features. The applicant proposes an automatic system of cross-connecting sources and loads. The Committee believes that the on-site power sources should have a greater independence than in the proposed system, at least to the extent that they cannot be connected together with automatically operated devices. An appropriate modification should be developed by the applicant and the matter resolved with the Regulatory Staff.

The main-coolant-pump flywheels represent a potential source of missiles within the containment, and the applicant has described measures taken to assure conservative design and high quality fabrication to minimize the possibility of flywheel failure. Additional steps may be warranted to assure the integrity of the flywheel assembly, and the Committee recommends that details concerning the adequacy of design, the material characteristics, quality assurance, and in-service inspection requirements be resolved between the applicant and the Regulatory Staff.

In the event that an irradiated fuel assembly is dropped or otherwise damaged during transit from the reactor vessel to the spent fuel pit, the cladding on the fuel rods may be ruptured with a consequent release of radioactivity. In view of the relatively high population density close to the Indian Point site, the applicant should review the assumptions made in analysis of a refueling accident to see whether additional conservatism is warranted in assessing its effects and the provisions to cope with the accident. The matter should be resolved with the Regulatory Staff.

Part-length control rods and special full-length rods are provided to control spatial neutron flux oscillations. Provision will be made for installation of permanent in-core detectors, should such detectors be required to assure adequate measurement of the power distribution.

Means will be provided for early detection of abrupt gross failure of a fuel element.

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 by the applicant and the Regulatory Staff.

The Committee calls attention to matters previously identified as warranting careful consideration with regard to all large, water-cooled power reactors of high power density.

The Committee also emphasizes the importance of independent action by the applicant to assure quality in the construction of the facility.

The ACRS believes that the items mentioned can be resolved during construction, and that, if due consideration is given to the foregoing, nuclear Unit 3 proposed for Indian Point 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 attached.

References - Indian Point Nuclear Generating Unit No. 3

1. LeBoeuf, Lamb & Leiby letter, dated April 26, 1967; Consolidated Edison Company of New York, Inc., Application for Licenses, dated April 25, 1967; Preliminary Safety Analysis Report

2. LeBoeuf, Lamb, Leiby & MacRae letter, dated August 30, 1968, Amendment No. 1 to Application for Licenses

3. LeBoeuf, Lamb, Leiby & MacRae letter, dated September 16, 1968; Amendment No. 2 to Application for Licenses

4. LeBoeuf, Lamb, Leiby & MacRae letter, dated October 18, 1968; Amendment No. 3 to Application for Licenses

5. LeBoeuf, Lamb, Leiby & MacRae letter, dated October 31, 1968; Amendment No. 4 to Application for Licenses

6. LeBoeuf, Lamb, Leiby & MacRae letter, dated November 4, 1968; Amendment No. 5 to Application for Licenses

7. LeBoeuf, Lamb, Leiby & MacRae letter, dated November 25, 1968; Amendment No. 6 to Application for Licenses

8. LeBoeuf, Lamb, Leiby & MacRae letter, dated December 9, 1968; Amendment No. 7 to Application for Licenses

9. Amendment No. 8 to Application for Licenses

10. LeBeouf, Lamb, Leiby & MacRae letter, dated January 3, 1969; Amendment No. 10 to Application for Licenses

11. LeBoeuf, Lamb, Leiby & MacRae letter, dated January 6, 1969; Amendment No. 11 to Application for Licenses

September 23, 1970

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

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Dear Dr. Seaborg:

At its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by Consolidated Edison Company of New York, Inc., for authorization to operate the Indian Point Nuclear Generating Unit No. 2. This project had previously been considered at the Committee's 95th, 98th, 122nd, and 124th meetings, and at Subcommittee meetings on August 23, 1969, March 13, 1970, April 25, 1970, May 28, 1970, July 28-29, 1970, and September 15, 1970. Subcommittees also met at the site on December 28, 1967 and May 11, 1970. The Committee last reported on this project to you on August 16, 1966. During the review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants, and with representatives of the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Indian Point site is located in Westchester County, New York, approximately 24 miles north of the New York City limits. The minimum radius of the exclusion area for Unit No. 2 is 520 meters and Peekskill, the nearest population center, is approximately one-half mile from the unit. Also at this site are Indian Point Unit 1, which is licensed for operation at 615 MWt, and Unit 3, which is under construction.

The applicant has re-evaluated flooding that could occur at the site in the event of the probable maximum hurricane and flood, in the light of more recent information, and has concluded that adequate protection exists for vital components and services.

Additional seismic reinforcement being provided for the Indian Point Unit No. 1 superheater building and removal of the top 80 ft. of the superheater stack will enable the stack to withstand winds in the range of 300-360 mph corresponding to current tornado design criteria. Since the reinforcement of the superheater building, which supports the stack, enables the stack to resist wind loads of a magnitude most likely to be experienced from a tornado, the Committee believes that removal of the top 80 ft. of the stack, to enable it to resist the maximum effects from a tornado, may be deferred until a convenient time during the next few years, but prior to the commencement of operation of Indian Point Unit No. 3. The applicant has stated that truncation of the stack will have no significant adverse effect on the environment.

The Indian Point Unit No. 2 is the first of the large, four-loop Westinghouse pressurized water reactors to go into operation, and the proposed power level of 2758 MWt will be the largest of any power reactor licensed to date. The nuclear design of Indian Point Unit No. 2 is similar to that of H. B. Robinson with the exception that the initial fuel rods to be used in Indian Point Unit No. 2 will not be prepressurized. Part-length control rods will be used to shape the axial power distribution and to suppress axial xenon oscillations. The reactor is designed to have a zero or negative moderator coefficient of reactivity, and the applicant plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses related to these tests.

Unit 2 has a reinforced concrete containment with an internal steel liner which is provided with facilities for continuous pressurization of weld and penetration areas for leak detection, and a seal-water system to back up piping isolation valves. In the unlikely event of an accident, cooling of the containment is provided by both a containment spray system and an air-recirculation system with fan coolers. Sodium hydroxide additive is used in the containment spray system to remove elemental iodine from the post-accident containment atmosphere. An impregnated charcoal filter is provided to remove organic iodine.

Major changes have been made in the design of the emergency core cooling system as originally proposed at the time of the construction permit review. Four accumulators are provided to accomplish rapid reflooding of the core in the unlikely event of a large pipe break, and redundant pumps are included to maintain long-term core cooling. The applicant has analyzed the efficacy of the emergency core cooling system and concludes that the system will keep the core intact and the peak clad temperature well below the point where zircaloy-water reaction might have an adverse effect on clad ductility and, hence, on the continued structural integrity of the fuel elements. The Committee believes that there is reasonable assurance that the Indian Point Unit No. 2 emergency core cooling system will perform adequately at the proposed power level.

The Committee concurs with the applicant that the reactor pit crucible, proposed at the time of the construction permit review, is not essential as a safety feature for Indian Point Unit No. 2 and need not be included.

To control the concentration of hydrogen which could build up in the containment following a postulated loss-of-coolant accident, the applicant has provided redundant flame recombiner units within the containment, built to engineered safety feature standards. Provisions are also included for adequate mixing of the atmosphere and for sampling purposes. The capability exists also to attach additional equipment so as to permit controlled purging of the containment atmosphere with iodine filtration. The Committee believes that such equipment should be designed and provided in a manner satisfactory to the Regulatory Staff during the first two years of operation at power.

The applicant plans to install a charcoal filter system in the refueling building to reduce the potential release of radioactivity in the event of damage to an irradiated fuel assembly during fuel handling. This installation will be completed by the end of the first year of full power operation.

The reactor instrumentation includes out-of-core detectors, fuel assembly exit thermocouples, and movable in-core flux monitors. Power distribution measurements will also ordinarily be available from fixed in-core detectors.

The applicant has proposed that a limited number of manual resets of trip points, made deliberately in accordance with explicit procedures, by approved personnel, independently monitored, and with settings to be calibrated and tested, should provide an acceptable basis for the occasional operation of Indian Point Unit No. 2 with only three of the four reactor loops in service. The Committee concurs in this position.

The applicant stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefulness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

The reactor includes a delayed neutron monitor in one hot leg of the reactor coolant system to detect fuel element failure. Suitable operability requirements will be maintained on the several sensitive means of primary system leak detection.

A conservative method of defining pressure vessel fracture toughness should be employed that is satisfactory to the Regulatory Staff.

The applicant stated that existing experimental results and analyses provide considerable assurance that high burnup fuel of the design employed will be able to undergo anticipated transients and power perturbations without a loss of clad integrity. He also described additional experiments and analyses to be performed in the reasonably near future which should provide further assurance in this regard.

The Committee has, in recent reports on other reactors, discussed the need for studies on further means of preventing common failure modes from negating scram action, and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has provided the results of analyses which he believes indicate that the consequences of such transients are tolerable with the existing Indian Point Unit No. 2 design at the proposed power level. Although further study is required of this general question, the Committee believes it acceptable for the Indian Point Unit No. 2 reactor to operate at the proposed power level while final resolution of this matter is made on a reasonable time scale in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept advised.

Other matters relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS letters should, as in the case of other reactors recently reviewed, be dealt with appropriately by the Staff and the applicant in the Indian Point Unit No. 2 as suitable approaches are developed.

The ACRS believes that, if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit No. 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References attached.

References - Indian Point Nuclear Generating Unit No. 2

1. Amendment No. 9 to Application of Consolidated Edison Company of New York for Indian Point Nuclear Generating Unit No. 2, consisting of Volumes I - IV, Final Safety Analysis Report, received October 16, 1968

2. Amendments 10 - 20 to the License Application
3. Amendments 22 - 24 to the License Application

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

NOV 14 1973

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

Subject: INTERIM REPORT ON INDIAN POINT NUCLEAR GENERATING STATION UNIT NO. 3

Dear Dr. Ray:

At its 163rd meeting, November 8-10, 1973, the Advisory Committee on Reactor Safeguards completed an interim review of the application of Consolidated Edison Company of New York, Inc., for authorization to operate Indian Point Nuclear Generating Station Unit No. 3. The project has been previously considered at Subcommittee meetings on July 11, 1973, October 10, 1973 and November 7, 1973. A tour of the facility was made by Committee members on November 2, 1973. In this review, the Committee had the benefit of discussions with representatives and consultants of Consolidated Edison, their contractor, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for construction of Indian Point Unit No. 3 on January 15, 1969.

Indian Point Unit No. 3 includes a four-loop Westinghouse nuclear steam supply system with a design power rating of 3025 MW(t). The design is similar to that of Unit No. 2 which has a power rating of 2760 MW(t). The three-unit Indian Point Nuclear Generating Station is located approximately 2-1/2 miles southwest of Peekskill, New York, and 24 miles north of the New York City boundary line.

The Committee's report of January 15, 1969, called attention to various matters including the following: consideration of thermal shock to the pressure vessel in the unlikely event of a loss-of-coolant accident (LOCA); measures to deal with possible hydrogen concentration buildup in the containment following a LOCA; greater independence in the on-site power system; main-coolant-pump flywheels as a potential source of missiles; protection against potential effects of a fuel-handling accident; and the possible effects of systematic or common mode failures. Most of these items are generic, not unique to Indian Point Unit No. 3.

Acceptable measures have been taken on Indian Point Unit No. 3 with regard to the on-site power system, hydrogen concentration buildup, and postulated fuel-handling accidents. Studies are still underway on the potential for missile generation from gross reactor coolant pump overspeed in the event of certain postulated LOCAs; this matter should be resolved in a manner satisfactory to the Regulatory Staff. It is believed that resolution of the thermal shock matter can await the development of further information from the Heavy Section Steel Technology Program and other studies. With regard to anticipated transients without scram, the Committee recommends that the recently announced Regulatory Staff position be implemented for Indian Point Unit No. 3 in timely fashion.

Because there is limited operating experience with very large, high power density reactors, the ACRS believes that initial operation should be limited to power levels no greater than 2760 MW(t) and that further review by the Committee is appropriate before higher power levels are permitted. The Committee believes that, in the consideration of the operation of Unit No. 3 at higher power levels, several factors are pertinent, including the following: satisfactory experience in Unit No. 3 and other similar reactors; adequate knowledge of fuel performance; extent to which an independent confirmation of LOCA-ECCS analysis has been made by the Regulatory Staff; further resolution of relevant generic matters; and consideration of the possibility of improvements in ECCS effectiveness.

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.

The Applicant stated that he will apply and utilize suitable equipment to enable periodic testing of the proper positioning of check valves intended to isolate low pressure systems connected to the primary system. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

Studies are underway with regard to the reliability of the service water distribution to the diesel-generators. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The original turbine design has been found by the Applicant to have the possibility of overspeed somewhat beyond the manufacturer's design condition if the turbine should trip at or near the design power. The Applicant is preparing design modifications to eliminate this condition, and will propose appropriate power limitations until acceptable modifications have been made. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Committee believes that several considerations are appropriate in the further development of the Technical Specifications, as follows: operating heatup and cooldown pressure-temperature curves as conservative as practical with respect to 10 CFR Part 50, Appendix G; appropriate baseline inspection and periodic in-service inspection of the steam generator shells; startup of an idle loop at power; acceptable cumulative limits on downtime of protection systems and engineered safety features; and continuing availability of core outlet thermocouples.

The Committee also believes that further consideration should be given to augmented use of movable in-core detectors, appropriate in-service inspection of nozzles in the primary head of the steam generators, and to the detailed specification of administrative controls intended to prevent overpressurization of the reactor vessel below operating temperatures.

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 December 18, 1972. Those problems and additional generic problems identified in more recent ACRS reports should be dealt with appropriately by the Regulatory Staff and the Applicant.

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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 Indian Point Nuclear Generating Station Unit No. 3 can be operated without undue risk to the health and safety of the public. The Committee believes that operation should be at power levels no greater than 2760 MW(t) prior to further Committee review.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References Attached

References

1. Final Facility Description and Safety Analysis Report (FSAR) for Indian Point Nuclear Generating Unit No. 3 dated December 4, 1970 (Amendment No. 13 to the Application for Licenses)

2. Supplements Nos. 1 through 22, dated June 30, 1971 through October 10, 1973, to the Indian Point Nuclear Generating Unit No. 3 FSAR

3. Letter, dated September 21, 1973, Directorate of Licensing, USAEC, to ACRS transmitting the Safety Evaluation Report for Indian Point Nuclear Generating Unit No. 3

4. Proposed Technical Specifications and Bases for Indian Point Nuclear Generating Unit No. 3 transmitted to the ACRS from the Directorate of Licensing, USAEC, on November 1, 1973.

5. Letter, dated September 26, 1973, Consolidated Edison of New York, Inc. (Con Ed) to the Directorate of Licensing, USAEC (DRL) concerning review of tanks at Indian Point Unit No. 3 which contain radioactive liquids

6. Letter, dated September 7, 1973, Con Ed to DRL, transmitting additional information concerning the design of Indian Point Unit No. 3 instrumentation, control and electrical systems

7. Letter, dated July 24, 1973, Con Ed to DRL, regarding results of review of control circuits of safety related equipment at Indian Point Unit No. 3

8. Letter, dated June 28, 1973, Con Ed to DRL, regarding the Indian Point Unit No. 3 Quality Assurance program

9. Letter, dated June 8, 1973, Con Ed to DRL, transmitting a report entitled "Dynamic Analysis of a Postulated Main Steam or Feedwater Line Pipe Break Outside Containment" dated May 8, 1973 applicable to Indian Point Unit No. 3

10. Letter, dated May 25, 1973, Con Ed to DRL, regarding motor-operated valves for isolating the Residual Heat Removal System from the Reactor Coolant System in Indian Point Unit No. 3

11. Letter, dated May 14, 1973; LeBoeuf, Lamb, Leiby and MacRae (LLL&M) to DRL; transmitting a report applicable to Indian Point Unit No. 3 entitled "Analysis of High Energy Lines" dated May 9, 1973

12. Letter, dated April 9, 1973, Con Ed to DRL concerning the electrical and mechanical systems design of Indian Point Unit No. 3

13. Letter, dated April 2, 1973, Con Ed to DRL, regarding modifications to the instrumentation, control and electrical systems in Indian Point Unit No. 3

14. Letter, dated January 23, 1973, Con Ed to DRL, concerning design of non-Category I equipment in Indian Point Unit No. 3

15. Letter, dated January 22, 1973, DRL to Con Ed requesting information needed to complete the Indian Point Unit No. 3 Operating License review

16. Letter, dated January 9, 1973, LLL&M to DRL, regarding fuel densification

17. Letter, dated November 6, 1972, DRL to Con Ed, requesting additional information needed to complete the Indian Point Unit No. 3 Operating License review.

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

July 13, 1978

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

SUBJECT: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT No. 3

Dear Dr. Hendrie:

During its 219th meeting, July 6-8, 1978, the Advisory Committee on Reactor Safeguards completed its review of the request by the Power Authority of the State of New York (PASNY) for authorization to increase the power level of Indian Point Nuclear Generating Unit No. 3 from the current maximum authorized power of 2760 MWt to the design power of 3025 MWt. This matter was considered at Subcommittee meetings on April 24, 1978, and June 16, 1978. During its review, the Committee had the benefit of discussions with representatives of PASNY and its consultants, and the Nuclear Regulatory Commission Staff, as well as comments from individuals studying the seismicity of the region. The Committee also had the benefit of the documents listed.

In its interim report, November 14, 1973, on operation of Indian Point Unit No. 3, the Committee recommended that the power be restricted to 2760 MWt because of limited operating experience at that time with very large high-power-density reactors; the Committee also recommended that specified items receive further attention. The Committee made similar recommendations with respect to the Zion reactors. Following suitable periods of operation at the restricted power levels, the Committee recommended that the Zion reactors be permitted to operate up to their design power.

The Committee finds that the specific issues raised in its interim report of November 14, 1973, have been satisfactorily resolved, and also that operating experience accumulated at Indian Point Unit No. 3 and other large plants warrants approval of the operation of Indian Point Unit No. 3 up to the design power level. However, the Committee urges that continuing effort be made to update safety related features to the maximum degree practical. In particular, the Committee believes attention should be given to the following:

1. Review of the Station for systems interactions that might lead to significant degradation of safety.

2. Review of the Station with regard to differences from current criteria, and judgments concerning possible backfitting requirements.

3. Review 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. A selective audit of the capability for safe shutdown and residual heat removal, using only safety grade equipment.

Since the Committee's interim report, ownership and operating responsibility for Indian Point Unit No. 3 have been transferred from the Consolidated Edison Company to PASNY. The Committee finds that the administrative separation and plans for physical separation are satisfactory.

During the review, the Committee considered recent studies concerning the seismicity of the Indian Point region and found insufficient basis for suggesting a change in the current seismic criteria.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, there is reasonable assurance that the Indian Point Nuclear Generating Unit No. 3 can be operated at full power, 3025 MWt, without undue risk to the health and safety of the public.

Additional comments by Members W. Kerr and P. Shewmon are presented below.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

Additional Comments by Members W. Kerr and P. Shewmon

We do not concur in the request for "review. . . for systems interactions. . . ." We consider the request too vague to have a working interpretation.

References:

1. Safety Evaluation by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission in the matter of the Power Authority of the State of New York removal of license condition limiting operation to 91% of rated thermal power for Indian Point Nuclear Generating Unit No. 3, dated April 6, 1978.

2. Supplement 1 to the Safety Evaluation by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission in the matter of the Power Authority of the State of New York removal of license condition limiting operation to 91% of rated thermal power for Indian Point Nuclear Generating Unit No. 3, dated April 17, 1978.

3. Letter from W. J. Cahill, Consolidated Edison Company, to B. Rusche, NRC, Subject: PASNY desire to operate Indian Point Unit No. 3, dated March 11, 1977.

4. Letter from L. R. Bennett, Power Authority of the State of New York, to A. Schwencer, NRC, Subject: ECCS analysis for full power, dated April 13, 1978.

5. Letter from W. J. Cahill, Consolidated Edison Company, to R. Reid, NRC, Subject: Requesting authorization to increase power from 2760 MWt to 3025 MWt, dated April 20, 1977.

6. Letter from G. T. Berry, Power Authority of the State of New York, to A. Schwencer, NRC, Subject: Amendment to Operating License for Cycle 2, Power Control Maneuvers, dated May 19, 1978.

7. Letter from G. T. Berry, Power Authority of the State of New York, to A. Schwencer, NRC, Subject: Analysis of constant axial offset control, dated May 24, 1978.

8. Report by Drs. Y. Aggarwal and L. Sykes, Lamont-Doherty Geological Observatory, Subject: Earthquakes, Faults and Nuclear Power Plants in Southern New York and Northern New Jersey, Science, Vol. 200, dated April 28, 1978.

9. Prepublication draft report by Dr. L. Sykes, Lamont-Doherty Geological Observatory, Subject: Intra-plate Seismicity, Reactivation of Pre-existing Zones of Weakness, Alkaline Magmatism, and Other Tectonism Post-Dating Continental Fragmentation, Undated.

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

October 12, 1979

Mr. Lee V. Gossick
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: SYSTEMS INTERACTIONS STUDY FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

Dear Mr. Gossick:

In a report dated July 13, 1978 concerning operation of the Indian Point Unit No. 3 at its full power level of 3025 MWt, the ACRS made several recommendations, including one that requested, "Review of the Station for systems interactions that might lead to significant degradation of safety."

In its earlier report of June 9, 1976 concerning full power operation of Zion Units 1 and 2, the ACRS had made a similar recommendation for that plant. In response to the recommendation for Zion, Commonwealth Edison arranged to have a study performed of Licensee Event Reports (LERs) covering the period between 1969 and 1977 to determine which indicated a potential systems interaction question. The results of this study were then applied to the Zion station to see if the potential for any of the same systems interactions were present and needed correction.

The ACRS has recently been asked by Consolidated Edison and the NRC Staff whether an LER systems interactions study similar to that performed for Zion would be an adequate response to its recommendation for a systems interactions study for Indian Point Unit No. 3, which, like Zion, was designed and constructed prior to ACRS identification of the generic need to examine the matter of systems interactions (letter to L. M. Muntzing dated November 8, 1974).

The ACRS believes that some types of systems interactions can be identified by an LER study such as that performed for Zion. However, the Committee believes that such an effort can only be considered to represent a treatment of part of the problem and does not recommend that type of study for Indian Point Unit No. 3.

As the Committee has stated in NUREG-0572 (September 1979), "Review of Licensee Event Reports (1976-1978)," a detailed review of LERs cannot be expected to identify all systems interactions. By far, the bulk of the LERs deal with failure of individual components and equipment, with relatively few cascades of failures resulting from an initiating event. It is not to be expected that LERs will include a relatively comprehensive set of examples of low probability events involving the coupled failures of systems where the initiating event itself is unlikely.

Thus, there will be important aspects of systems interactions which are unlikely to be exposed by a study of LERs. The important question is how to uncover vulnerabilities which may have potentially serious effects the first time they occur. In its letter of November 8, 1974 to Mr. Muntzing, the ACRS gave several examples of possible systems interactions to illustrate the matter. Since a question has arisen concerning what constitutes a reasonably appropriate study of systems interactions at Indian Point Unit No. 3, the ACRS has the following additional comments.

There are at least two general areas of investigation of systems interactions which are unlikely to be covered by a review of LERs.

1. There is a possibility of systems interactions within an interconnected electrical or mechanical complex. In such a study, it is necessary to consider failures which may be outside the usual context of failure analysis. For example, a component may run away or it may partly fail and hang up somewhere between its normal and its "failed" state, in either case leading to some excess in whatever service (voltage, frequency, flow, pressure, temperature, etc.) is provided or controlled by the system comlex under consideration. This kind of failure, which usually is less likely than total functional failure of a sub-system, is unlikely to be revealed by LERs. Investigation of such failures generally will require an appropriate application of failure modes and effects analysis with the use of the systems diagrams.

2. There is a possibility of interactions between nonconnected systems due to the physical arrangement or disposition of equipment and to possibilities of transporting damaging influences, such as heat or water, within a given plant or site. Such interactions are likely to be unique to each plant and are unlikely to be revealed by LERs since the probability for such interaction to occur may be modest. There are exceptions to this, of course, and many reductions in the potential for systems interactions resulted from evaluation of the Quad Cities event of June 9, 1972 in which a rupture in the circulating water system flooded the turbine building basement and some safety-related equipment. Generally speaking, however, neither LERs nor a study of plant diagrams and other drawings will consistently reveal the potential for such interactions between nonconnected systems, because such drawings generally show single features or systems; composite drawings which include all systems are difficult to make without their becoming unmanageably complicated. Thus, uncovering the potential for interaction of nonconnected systems will usually require careful, in-situ examination of the physical plant. This examination must consider all features having the potential to damage safety systems, including the safety systems themselves.

   The physical inspection of the plant could be approached by dividing the plant into "compartments" following discernable structures -- such as walls, ceilings, and floors with appraisable strengths and weaknesses. Doors, stairs, ventilation ducts, piping, and other penetrations would be evaluated for potential influence transport (fire, steam, hot air, etc.). Structures, which act as barriers to the flow of a damaging influence, would be assessed for the adequacy of their resistance to such influences.

In each compartment the elements of the safety systems, including such extensions as instrument lines and power or control wiring should be identified on a "train" basis. The physical vulnerability of the safety system elements to nonstandard conditions (temperature, pressure, water, spray, etc.) should be identified. The characteristics of such systems as influence generators under faulted conditions would have to be assessed if such system elements exist as redundant elements within the identified "compartment" boundaries.

The influence potential of all non-safety elements including such items as sewer and drain lines, combustible gas transport and storage, compressors, and heavy-power-circuits and transformers, within the given compartment should be assessed with respect to potential for damaging or disrupting (as with induced electrical noise) critical system(s) within the "compartment" and the "compartment" boundary itself.

The invasion of damaging influences through the barriers or boundaries into the identified compartment would also have to be assessed. This would include consideration of entry of personnel carrying influence generators such as welding equipment.

Special consideration would have to be given to the identification of convergence of safety functions into single compartments and the degree of convergence within the given space. The study of interactions between nonconnected systems would also have to include the possibility of non-visible interactions, such as the possibly adverse effect of failure of one buried pipe on a neighbor due to scouring. A study of plant drawings would be required in connection with this aspect.

The ACRS believes that one practical method to pursue such a systems interactions investigation is by formation of a small but competent interdisciplinary team, perhaps four to six individuals, who would pursue the two areas of investigation described above. The report of the team should identify the detailed approach employed and tabulate the results in a reviewable form.

The Committee believes that the two areas of investigation described above can be used in defining a suitable approach to a systems interactions study for Indian Point Nuclear Generating Unit No. 3 and are generally applicable to such studies on other LWRs.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

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

Dear Dr. Palladino:

Subject: REPORT ON SYSTEMS INTERACTIONS STUDY FOR INDIAN POINT NUCLEAR GENERATING UNIT 3

During its 263rd meeting, March 4-6, 1982, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Power Authority of the State of New York (PASNY) to perform a systems interactions study of the Indian Point Nuclear Generating Unit 3 (Indian Point 3). In its review the Committee had the benefit of a Subcommittee meeting held on February 26, 1982. The PASNY proposal was made in response to prior recommendations by the ACRS in letters dated July 13, 1978 and October 12, 1979 that a systems interactions study should be performed on Indian Point 3.

The ACRS believes that the PASNY proposal is generally responsive to the ACRS recommendations. The Committee agrees with PASNY that for this study it is reasonable to limit the portion that deals with the investigation of control system influences on safety systems to effects of interconnected systems. The ACRS also believes that, in view of prior efforts to review many aspects of possible adverse interactions between safety systems, it is reasonable in this study to place emphasis on the interactions between nonsafety systems and safety systems. However, the ACRS believes that where interactions between safety systems have not received prior study, they should not be ignored in this study.

The ACRS believes that it is time for the Indian Point 3 systems interactions study to begin and recommends that PASNY conduct the proposed "walk-down" phase during the upcoming plant shutdown for refueling.

A partial review of the NRC Staff's preliminary version of a generic approach to systems interactions studies also took place at the Subcommittee meeting. The Committee will complete its review of this matter after the Staff has finished preparation of its proposed plan. However, it is clear that it will be several years before the Staff completes the development of its approach to systems interactions studies for all reactors. In the interim, the ACRS recommends consideration of the potential merits of simplified walk-through systems interactions studies for all operating light-water reactors in order to look for relatively obvious interactions. In addition, the ACRS recommends that a mechanism be developed for early dissemination and evaluation of any systems interactions observations arising from the ongoing studies and having potentially significant generic implications for a family of operating plants.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

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

November 13, 1975

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

SUBJECT: REPORT ON THE JAMESPORT NUCLEAR POWER STATION, UNITS 1&2

Dear Mr. Anders:

During its 187th meeting, November 6-8, 1975, the Advisory Committee on Reactor Safeguards reviewed the application of Long Island Lighting Company (Applicant) for a permit to construct Jamesport Nuclear Power Station, Units 1&2. The site was visited on October 31, 1975, and a Subcommittee meeting was held in Ronkonkoma, Long Island, New York, on October 30, 1975. The Jamesport Station is a replication of the Millstone Nuclear Power Station, Unit No. 3, on which the Committee reported April 16, 1974. During its review of the Jamesport application, 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 Stone and Webster Engineering Corporation. The Committee also had the benefit of the documents listed.

The Jamesport Station will be located on a 555-acre site on the north shore of Long Island in Suffolk County, New York, about 6 miles northeast of the community of Riverhead (1970 population 7585) and about 65 miles east of New York City. The minimum exclusion distance is 655 meters. The low population zone outer boundary radius is 2 miles. The Applicant has designated the community of Riverhead to be the nearest population center (projected population to exceed 25,000 by the year 2020). The population within a 50-mile radius is projected to increase from a 1970 figure of about 3,000,000 to about 7,000,000 by the year 2020.

Each unit of the Jamesport Station will utilize the RESAR-3 Consolidated Version, four-loop pressurized water reactor having a core output of 3411 MW(t).

The Jamesport Station employs a steel-lined reinforced concrete containment with a net free volume of 2.32×10^6 cu.ft. The containment is designed for an internal pressure of 45 psig and a temperature of 280°F.

The NRC Staff has identified several items in the Jamesport application for which its reviews are not yet completed. 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 analysis of the effects of anticipated transients without scram.

3. The evaluation of the plant design to meet the requirements of Appendix I of 10 CFR Part 50.

4. The provision of overpressure protection during conditions of startup and shutdown when the primary reactor system is completely filled with water.

The RESAR-3 Consolidated Version nuclear design utilizes the Westinghouse 17x17 fuel array. Westinghouse has identified an integrated test program to confirm the design margins associated with this design. The RESAR-3 reactor core 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 Jamesport. 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 were filed after January 7, 1972. The Jamesport Station is in this category. These units will use assemblies with a 17x17 fuel array similar to those to be used in Comanche Peak Steam Electric Station, Units 1&2. Although calculated peak clad temperatures in the event of a postulated LOCA are less for fuel assemblies with a 17x17 than with 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 these units.

In conjunction with a presentation of results of analysis of events subsequent to a postulated LOCA in RESAR-3 plants, Westinghouse has made best-judgment calculations for the same class of accidents. Preliminary results indicate that a considerable margin of safety may exist; however, the methodology used has not been subjected to critical evaluation. The Committee recognizes the potential importance of studies of this type in the improvement and optimization of design of safety features and encourages the Applicant and the NRC Staff to accelerate their efforts to this end.

The Committee believes that the Applicant and the NRC Staff should continue to review the Jamesport Station design for features that could reduce the possibility and consequences of sabotage.

The Committee recommends that the NRC Staff and the Applicant review further the design features that are intended to prevent the occurrence of 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.

Arrangements for spent fuel transport have not been completed. This matter should be resolved in a manner satisfactory to the NRC Staff.

The effects of water wave action and scour in the vicinity of the circulating and service water pumphouse have not been adequately evaluated. The back-fill, especially in the region of the pumps and piping should be adequately protected from scour and water wave action during hurricanes. Additional field evaluations of the natural soils along the path of the critical service water piping system are needed to determine the competence of the soils. The NRC Staff should be satisfied that the soils and system geometry can be maintained in the designed condition, and that the service water intake structure will not lose its capability to function as a result of storms or impact by a ship or barge. 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 can be resolved during construction and that, if due consideration is given to the foregoing, the Jamesport Nuclear Power 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,

WKerr

W. Kerr
Chairman

REFERENCES

1. Jamesport Preliminary Safety Analysis Report with Amendment 1 through Amendment 7.

2. RESAR-3 Consolidated Version, Westinghouse Reference Safety Analysis Report with Amendments 1 through 6.

3. Safety Evaluation Report related to construction of Jamesport Nuclear Power Station Units 1&2, Docket Nos. STN 50-516 and STN 50-517, U. S. Nuclear Regulatory Commission, NUREG-75/095, October, 1975.

4. Excerpt from Hearings of the New York State Siting Board on the Jamesport Nuclear Power Station.

5. LILCO letter JNRC-100 from Andrew W. Wofford, Vice-President to Roger S. Boyd, AD, DRL, USNRC, dated October 29, 1975, forwarding additional information.

6. Letters from Mr. Irving Like, Esq., dated September 29, 1975, October 15, 1975, and October 17, 1975.

7. Nuclear Reactor Licensing, A Critique of the Computer Safety Prediction Methods, by Carl J. Hocevar, Union of Concerned Scientists, with Addendum, September, 1975.

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: REPORT ON THE KEWAUNEE NUCLEAR POWER PLANT

Dear Dr. Seaborg:

At its ninety-seventh meeting, on May 9-11, 1968, the Advisory Committee on Reactor Safeguards completed a review of the combined application of the Madison Gas and Electric Company, the Wisconsin Power and Light Company, and the Wisconsin Public Service Corporation to construct a nuclear power station near Kewaunee, Wisconsin. The Wisconsin Public Service Corporation is designated to act for all three organizations and to operate the plant. The project and site location were considered at a Subcommittee meeting and site visit on April 25, 1968. During its review, the Committee had the benefit of discussions with the Wisconsin Public Service Corporation, the Westinghouse Electric Corporation, Pioneer Service and Engineering Company, the AEC Regulatory Staff, consultants of these organizations and study of the documents listed below.

The Kewaunee Nuclear Power Plant will be located on the shore of Lake Michigan in Kewaunee County, about 25 miles southeast of Green Bay, Wisconsin. The surrounding countryside is rural and relatively sparsely populated; the nearest community is Kewaunee, population about 2700, located seven miles north. The Point Beach reactor site, previously reviewed, is four and one-half miles south.

The reactor design is similar to that reviewed for the Prairie Island application (ACRS report dated March 12, 1968). The steam supply system is a two-loop, pressurized-water reactor to be operated at a power of 1650 MWt (559 MWe). The emergency core cooling systems are to be sized to assure sufficient cooling water to maintain the integrity and the original geometry of the core in the unlikely event of the most serious pipe rupture and subsequent loss of coolant.

The containment concept, also similar to that for Prairie Island, consists of a free-standing steel shell which, in turn, is contained within a cylindrical concrete structure. An annular space of about five feet exists between these two structures. In this design, leakage from the primary steel containment would be collected within the concrete shell, circulated through charcoal filters, and then discharged to the atmosphere. The applicant expects that the release of radioactive iodine to the atmosphere in the unlikely event of a major reactor accident would be significantly less than for single containment designs.

The Committee continues to believe that control and protection instrumentation should be separated to the fullest extent practical. There remain questions in this area on the Kewaunee design. The Committee recommends that the Regulatory Staff review the protection system design before its fabrication and installation.

The Committee continues to emphasize the importance of quality assurance in fabrication of the primary system as well as inspection during service life, and recommends that the applicant implement those improvements in quality that are practical with current technology. The Committee also calls attention to those matters previously emphasized, which it deems to be important for all large water-cooled power reactors.

The Committee believes that the various items mentioned can be resolved during construction and that the proposed power plant can be constructed at the Kewaunee 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.

References:

1. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Application for Licenses with transmittal letter, dated August 18, 1967.

2. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Facility Description and Safety Analysis Report, Volumes I, II, III, and IV.

3. Amendment No. 1 to License Application, Wisconsin Public Service Corporation, dated January 2, 1968.

4. Amendment No. 2 to License Application, Wisconsin Public Service Corporation, dated February 12, 1968.

5. Amendment No. 3 to License Application, Wisconsin Public Service Corporation, dated March 15, 1968.

6. Amendment No. 4 to License Application, Wisconsin Public Service Corporation, dated April 1, 1968 and errata sheets, received April 10, 1968.

7. Amendment No. 6 to License Application, Wisconsin Public Service Corporation, dated May 3, 1968.

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 KEWAUNEE NUCLEAR POWER PLANT

Dear Dr. Schlesinger:

During its 148th meeting, on August 10-12, 1972, the Advisory Committee on Reactor Safeguards completed its review of the application of the Wisconsin Public Service Corporation, the Wisconsin Power and Light Company, and the Madison Gas and Electric Company for authorization to operate the Kewaunee Nuclear Power Plant at power levels up to 1650 MW(t). The Wisconsin Public Service Corporation acts for all three organizations and will operate the plant. The project was considered previously by a Subcommittee during a visit to the site on July 27, 1972. During its review, the Committee had the benefit of discussions with the Wisconsin Public Service Corporation, the Westinghouse Electric Corporation, Pioneer Service and Engineering Company, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee previously discussed this project in a construction permit report dated May 15, 1968.

The Kewaunee Nuclear Power Plant is located on the shore of Lake Michigan in Kewaunee County, about 25 miles southeast of Green Bay, Wisconsin. The surrounding countryside is rural and relatively sparsely populated. The Point Beach Nuclear Plant (ACRS report April 16, 1970) is located four and one-half miles to the south.

The containment consists of a free-standing steel vessel within a reinforced-concrete shield building. An annular space of about five feet separates the two structures. Given an accident signal, this space will be evacuated to a pressure slightly less than ambient within a short time and thereby serve as a volume in which to trap leakage from the primary steel containment. The gas in the annulus will be circulated through particulate and charcoal filters with about 5% of the filtered flow released to the atmosphere; the remainder will be returned to the annulus. A region of the auxiliary building (contiguous to the shield building) is designated as a special ventilation zone and contains the outside terminations of penetrations to the interior of the primary containment. The perimeter of this zone is constructed as a medium leakage barrier and the zone is equipped with redundant fan-filter systems either of which can maintain a negative pressure relative to the environment. Thus any leakage to this portion of the secondary containment is subjected to at least single pass filtration prior to release to the environment. The Committee believes that this containment concept is acceptable.

Defects have developed in unpressurized fuel in some plants. The Kewaunee 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.

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.

The Committee reiterates its previous comments on 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. In particular, 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 Emergency Core Cooling System.

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 Kewaunee Nuclear Power Plant can be operated at power levels up to 1650 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

References

1. Amendment No. 7 (Volumes 1-5 of Final Safety Analysis Report (FSAR)), Wisconsin Public Service Corporation letter, dated January 21, 1971

2. Amendments 8-18 to the Application for Construction Permit and Operating License

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

June 12, 1973

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

Subject: REPORT ON KEWAUNEE NUCLEAR POWER PLANT

Dear Dr. Ray:

In the August 17, 1972, report on the Kewaunee Nuclear Power Plant the Committee recommended that the Regulatory Staff and the ACRS review, prior to operation of the plant at appreciable power, the applicant's proposal for operating power limits as related to the possible densification of fuel in this reactor; the Committee also requested the Regulatory Staff to review the applicant's calculations of containment pressure subsequent to postulated rupture of a reactor coolant pipe within containment. In addition, following the review in August, the question of protecting reactor shutdown equipment against the consequences of rupture, external to the containment, of a pipe carrying high energy fluid was presented to the applicant.

These matters were considered during a Subcommittee meeting held in Washington, D. C. on May 22, 1973, and during the 158th meeting of the ACRS, June 7-9, 1973. During its review, the Committee had the benefit of discussions with representatives of the Wisconsin Public Service Corporation, the Westinghouse Electric Corporation, Pioneer Service and Engineering Company, the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents listed.

The possible effects of fuel densification on power distribution have been included in the applicant's analyses of postulated accidents, and he proposes to restrict operation of the core so that total peaking factor (peak maximum-to-average-power ratio) does not exceed 2.59, which corresponds to a non-augmented factor of 2.38 when the allowance for flux peaking due to possible fuel gaps is not included. The Regulatory Staff and the applicant have agreed that part-length control rods will not be used during the first cycle (except for physics tests), and appropriate in-core surveillance will be required when the reactor is operating at high power. The Committee agrees that these conditions are acceptable and recommends that they be implemented to the satisfaction of the Regulatory Staff.

The Committee recommends that the Regulatory Staff confirm the conservatism of the applicant's peaking factor analysis. The Committee also recommends that, if proposed future operation at high power requires controlling the non-augmented peaking factor to values lower than 2.38, the matter should be reviewed by the Regulatory Staff and the ACRS.

In regard to protection against the consequences of rupture of a high energy piping line outside of containment, the applicant described an evaluation carried out in accordance with criteria established by the Regulatory Staff. Appropriate design and construction modifications including encapsulation sleeves, equipment protection, equipment relocation, and impingement barriers are being made and are to be completed prior to operation above five percent of design power. The Committee believes this approach is satisfactory.

The applicant has re-examined, and the Regulatory Staff has reviewed, the calculation of pressurization of the containment building following postulated rupture of a high energy pipe within containment. These studies confirm the adequacy of the containment structure and also confirm the design strength of compartment walls within the containment which might be subjected to dynamic forces during such an unlikely event.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, in addition to those mentioned in its report of August 17, 1972, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the Kewaunee Nuclear Power Plant can be operated at power levels up to 1650 MW(t) without undue risk to the health and safety of the public.

Sincerely,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References Attached.

<u>References</u>

1. Amendments Nos. 21 through 29 to the Application for Construction Permit and Operating License for the Kewaunee Nuclear Power Plant.

2. Supplement No. 1, dated December 18, 1972, to Safety Evaluation Report by the Directorate of Licensing, USAEC, dated July 24, 1972.

3. Supplement No. 2, dated May 10, 1973, to Safety Evaluation Report by the Directorate of Licensing, USAEC, dated July 24, 1972.

4. WCAP-8092, 'Westinghouse Proprietary Class 2 Report,' "Fuel Densification - Kewaunee Nuclear Power Plant," dated March 1973.

5. Letter AEC/DL to ACRS dated May 16, 1973.

6. Draft Report, "Main Steam and Feedwater Line Rupture Study," Pioneer Service and Engineering Co., dated November 22, 1972.

7. Anonymous letter, dated October 25, 1972 regarding unresolved safety related items for Kewaunee and Prairie Island projects.

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

January 15, 1976

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Wahington, DC 20555

SUBJECT: INTERIM REPORT ON KOSHKONONG NUCLEAR PLANT, UNITS 1 & 2

Dear Mr. Anders:

During its 189th meeting, January 8-10, 1976, the Advisory Committee on Reactor Safeguards completed a partial review of the application of the Wisconsin Electric Power Company, Wisconsin Power and Light Company, Wisconsin Public Service Corporation, and Madison Gas and Electric Company (Applicant) for a permit to construct the Koshkonong Nuclear Plant, Units 1 & 2. This project had been previously considered at the Committee's 188th meeting and at Subcommittee meetings in Ft. Atkinson, Wisconsin, on October 17, 1975 and Washington, DC on December 3, 1975. Members of the Committee visited the site on October 17, 1975. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, Westinghouse Electric Corporation, Stone and Webster Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the references listed.

The application to build the the Koshkonong Nuclear Plant is a part of the application, designated the Wisconsin Utilities Project (WUP), for licenses to construct and operate one or more standardized nuclear power plants at one or more sites in the State of Wisconsin, using the duplicate plant option, Appendix N to 10 CFR Part 50. The site-related aspects specific to the Koshkonong plant are contained in a Site Addendum to the WUP Preliminary Safety Analysis Report.

The Koshkonong plant will be located on an 1109 acre site in Jefferson County, Wisconsin, 11 miles northwest of Janesville, the nearest population center (1970 population 46,246). The minimum exclusion distance is 954 meters and the low population zone radius is three miles.

Each unit will utilize a 3-loop Westinghouse pressurized water reactor with 17x17 fuel assemblies to be operated at power levels up to 2775 MW(t). The nuclear steam supply system is similar in design to Virgil C. Summer, Unit 1 reported on by the Committee in its letter of November 15, 1972.

During October 1975 Westinghouse submitted a revised ECCS evaluation model for review by the NRC Staff. This review is nearing completion. The Applicant intends to use the approved revised model for ECCS evaluation of the WUP nuclear steam systems. The calculated reflooding rates and low peaking factor are of particular interest to the Committee. The Committee will continue its review of the WUP ECCS evaluation until the matter is resolved in a manner satisfactory to both the Committee and the NRC Staff.

The Applicant and the NRC Staff both selected the tectonic province approach permitted by Appendix A to 10 CFR Part 100 to establish conservative design values for horizontal ground acceleration. The NRC Staff considered the applicable province to include Anna, Ohio, the site of a 1937 earthquake of intensity VII-VIII (MM). On this basis the NRC Staff believes that the design value for horizontal acceleration for the SSE should be 0.20g and for the OBE 0.10g. The Applicant is now examining proprietary data from oil exploration drilling in the Anna area, which he believes will show that the Anna earthquake was not a random earthquake but rather was associated with a local active fault. The Applicant is also proposing decoupling of the OBE from the SSE. This matter should be resolved to the satisfaction of the Committee and the NRC Staff.

The NRC Staff has not yet completed its review of: (1) the Applicant's analysis of Anticipated Transients Without Scram; and (2) the capability of the liquid and gaseous radwaste systems to meet the design objectives of Appendix I to 10 CFR Part 50. The Committee wishes to be kept informed.

Recent standardized safety designs for nuclear steam systems have included loose parts monitors. The Committee recommends that a similar requirement be made a part of the WUP safety design. The Committee wishes to be kept informed.

The Committee believes that the Applicant and the NRC Staff should review the Koshkonong Plant for design features that could significantly reduce the possibility and consequences of sabotage, and that such features should be incorporated into the plant design where practicable. The Committee wishes to be kept informed.

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. The Committee wishes to be kept informed.

Generic problems relating to large water reactors are discussed in the Committee's report of March 12, 1975. These problems should be dealt with appropriately by the NRC Staff and the Applicant.

The Committee will complete its review of this application when the necessary additional information has been developed.

Sincerely,

Dade W. Moeller

Dade W. Moeller
Chairman

References

1. Koshkonong Nuclear Plant Units 1 and 2, Preliminary Safety Analysis Report (August 1974) with Amendments 1 through 10.

2. Koshkonong Nuclear Plant PSAR Site Addendum (August 1974) with Amendments 1 through 10.

3. Safety Evaluation Report NUREG-75/092 related to construction of the Koshkonong Nuclear Plant Units 1 and 2, October 1975.

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

May 12, 1976

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON KOSHKONONG NUCLEAR PLANT, UNITS 1 AND 2

Dear Mr. Rowden:

During its 193rd meeting, May 6-8, 1976, the Advisory Committee on Reactor Safeguards completed its review of the application of the Wisconsin Electric Power Company, Wisconsin Power and Light Company, Wisconsin Public Service Corporation, and Madison Gas and Electric Company (the Applicants) for a permit to construct the Koshkonong Nuclear Plant, Units 1 and 2. The site was visited on October 17, 1975. The application had been previously reviewed at the Committee's 188th meeting, January 8-10, 1976, and at Subcommittee meetings in Ft. Atkinson, Wisconsin on October 17, 1975 and Washington, DC on December 3, 1975 and May 5, 1976. The Committee issued an Interim Report dated January 15, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, Westinghouse Electric Corporation, Stone and Webster Corporation, the Nuclear Regulatory Commission (NRC) Staff, and of the documents listed.

The application to build the Koshkonong Nuclear Plant is a part of the Wisconsin Utilities Project (WUP), for licenses to construct one or more standardized nuclear power plants at one or more sites in Wisconsin, using the duplicate plant option, Appendix N to 10 CFR Part 50. The Committee is restricting its current review to Koshkonong Units 1 and 2 since the schedule for the other plants is not well specified, and it may be appropriate to incorporate design changes in the plans for the future plants.

The Applicants used the October 1975 Westinghouse emergency core cooling system (ECCS) model as approved by the NRC Staff to demonstrate compliance with Appendix K to 10 CFR Part 50. The limiting peaking factor at full power is 2.18. The Applicants have committed to install an Axial Power

Distribution Monitoring System or otherwise to demonstrate the capability to manage core power distribution within the limiting peaking factor envelope. The NRC Staff considers this resolution of the ECCS evaluation adequate for purposes of issuance of a construction permit. The Committee concurs with this conclusion; however, the Committee recommends aggressive pursuit of possible improvements in the reliability and function of the ECCS for Koshkonong Units 1 and 2.

The Applicants and the NRC Staff have agreed that horizontal ground accelerations of 0.2g and 0.06g are appropriate design values for the safe shutdown earthquake (SSE) and operating basis earthquake (OBE), respectively. The Committee concurs with these values for the Koshkonong Plant. The Applicants selected the OBE on the basis of economics, holding that the minimum value of the OBE is not safety related. The NRC Staff required the Applicants as a part of their economic evaluation to demonstrate that an earthquake equivalent to the OBE would have a reasonably long return interval. Applying a probabilistic analysis to historic data of the tectonic province, the Applicants estimated a return interval of 1,000 years. The NRC Staff accepted this as a reasonable period. In this regard, the Committee urges the NRC Staff to develop general criteria for the determination of an acceptable OBE. The Committee wishes to be kept informed.

The NRC Staff has completed its evaluation of the liquid and gaseous radioactive waste treatment systems and has concluded that these systems are capable of meeting the design objectives of Appendix I to 10 CFR Part 50.

Two outstanding issues remain to be resolved prior to the NRC Staff recommendation for issuance of a construction permit:

(1) The NRC Staff's review of the Westinghouse Analysis of Anticipated Transients Without Scram (ATWS), WCAP-8330, will be completed in the next few weeks and the final implementation plan for the Koshkonong Plant is under development. The Applicants have stated that it will be feasible to accommodate changes in plant design likely to be required by the implementation program. The Committee wishes to be kept informed.

(2) The implementation of the quality assurance program will remain an outstanding issue until the restrictions imposed by the Public Service Commission of Wisconsin on fund expenditures are removed. The Committee recommends that this issue be resolved to the satisfaction of the NRC Staff.

The Committee believes that the Applicants and the NRC Staff should review the Koshkonong Plant for design features that could significantly reduce the possibility and consequences of sabotage, and that such features should be incorporated into the plant design where practicable. The Committee wishes to be kept informed.

Generic problems relating to large water reactors are discussed in the Committee's April 16, 1976 Status Report Number 4. These problems should be dealt with in a timely fashion by the NRC Staff and the Applicants.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above and those of the Committee's letter of January 15, 1976, can be resolved during construction and that, if due consideration is given to the foregoing, the Koshkonong 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.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

References

1. Koshkonong Nuclear Plant Units 1 and 2, Preliminary Safety Analysis Report (August 1974) with Amendments 1 through 10.

2. Koshkonong Nuclear Plant PSAR Site Addendum (August 1974) with Amendments 1 through 10.

3. Safety Evaluation Report NUREG-75/092 related to construction of the Koshkonong Nuclear Plant Units 1 and 2, October 1975.

4. Safety Evaluation Report NUREG-0051 (Supplement to NUREG 75/092) related to construction of Koshkonong Nuclear Plant, Units 1 and 2, April 1976

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

July 13, 1976

The Honorable Les Aspin
United States House of Representatives
Washington, DC 20515

Dear Congressman Aspin:

This is in response to your letter of June 7, 1976, asking that the Advisory Committee on Reactor Safeguards (ACRS) outline specific changes in the Koshkonong plant design to improve the Emergency Core Cooling System (ECCS) and to prevent sabotage. As you know, Sections 29 and 182(b) of the Atomic Energy Act of 1954, as amended, set forth the legislative basis for ACRS functions (Attachment A). In connection with specific power reactors, the responsibility of the ACRS is to review critically and report on applications for construction permits and operating licenses. The ACRS believes that the necessary objectivity and lack of commitment to any particular system would be compromised if the ACRS itself were to become involved in the development of specific designs.

Please be assured that the Committee believes, as stated in its report of May 12, 1976, that the Koshkonong Nuclear Plant can be constructed with reasonable assurance that it can be operated without undue risk to the public health and safety.

The ACRS has made recommendations with regard to improved ECCS and increased protection against sabotage in a number of reports outside the context of specific power reactor applications. A few examples are given in Attachments B, C, and D. These are areas in which, with continuing effort, new insights and improvements may be realized and would be very worthwhile. Therefore, these matters warrant special attention in the development of the final design of the Koshkonong plant and future plants of this type.

NRC Regulatory Guide 1.17 provides interim guidance regarding access control for sabotage protection, and the ACRS is working actively with the NRC Staff in the development of an improved Guide.

The ACRS has been urging that further consideration be given in nuclear reactor plant design and layout to make still more unlikely the chance that sabotage could adversely affect the public health and safety. Several ideas and concepts have become available. These concepts reflect different approaches to the overall problem, and the ACRS believes that considerable effort will be required by all parties concerned before judgments on appropriate design approaches can be made. However, the ACRS believes such efforts should be given high priority.

With regard to ECCS, the ACRS has for several years been recommending that improvements in function and reliability be developed (Attachment B). A recently developed fuel assembly design, which will be utilized in the Koshkonong plant, is calculated to give reduced peak clad temperatures during postulated loss-of-coolant accidents. However, the ACRS continues to believe that increased reflooding rates and increased reliability of ECC systems should be pursued vigorously.

Finally, it should be noted that before an operating license is issued, an additional review will be made of all aspects of the Koshkonong plant by the NRC Staff and the ACRS.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Attachments:
A. Excerpt from the Atomic Energy Act of 1954, as amended
B. "Report on Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors," September 10, 1973
C. "Report on Evaluation Models for Commission Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors," November 20, 1974
D. "Report on Industrial Sabotage," October 14, 1975

Excerpt from the Atomic Energy Act of 1954, as amended
(Atomic Energy Legislation through 93d Congress, 2nd Session)

Committee on Reactor Safeguards. 42 U.S.C. sec. 2039.

"SEC. 29. ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.—There is hereby established an Advisory Committee on Reactor Safeguards consisting of a maximum of fifteen members appointed by the Commission for terms of four years each. The Committee shall review safety studies and facility license applications referred to it and shall make reports thereon, shall advise the Commission with regard to the hazards of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards, and shall perform such other duties as the Commission may request. One member shall be designated by the Committee as its Chairman. The members of the Committee shall receive a per diem compensation for each day spent in meetings or conferences, or other work of the Committee, and all members shall receive their necessary traveling or other expenses while engaged in the work of the Committee. The provisions of section 163 shall be applicable to the Committee.[24b]

[24b] Public Law 85-256 (71 Stat. 576) (1957), sec. 5, added sec. 29.

ACRS Report.
Sec. 182

"b. The Advisory Committee on Reactor Safeguards shall review each application under section 103 or section 104 b. for a construction permit or an operating license for a facility, any application under section 104 c. for a construction permit or an operating license for a testing facility, any application under section 104 a. or c. specifically referred to it by the Commission, and any application for an amendment to a construction permit or an amendment to an operating license under section 103 or 104 a., b., or c. specifically referred to it by the Commission, and shall submit a report thereon ~~which shall be made part of the record of the application and available to the public except to the extent that security classification prevents disclosure.~~[6]

[6] Public Law 85-256 (71 Stat. 576) (1957), sec. 6, added subsec. b, and relettered former subsecs. b. and c. as subsecs. c. and d. Public Law 87-615 (76 Stat. 409) (1962), sec. 3, amended subsec. b. Before amendment, it read: "b. The Advisory Committee on Reactor Safeguards shall review each application under section 103 or 104 b. for a license for a facility, any application under section 104 c. for a testing facility, and any application under section 104 a. or c. specifically referred to it by the Commission, and shall submit a report thereon, which shall be made part of the record of the application and available to the public, except to the extent that security classification prevents disclosure."

Proposed amendment under H.R. 9285, 6/21/71

... and shall submit a report thereon:

provided, however, that unless the Commission specifically requests a review and report on an application or portion thereof, the Committee may dispense with such review and report by notifying the Commission in writing that review by the Committee is not warranted. Any report or notice required by this subsection shall be made part of the record of the application and available to the public except to the extent that security classification prevents disclosure.

To Moeller ltr to Aspin
dated July 13, 1976

FOR ATTACHMENT B, SEE PAGES 2052-2055, VOLUME IV

ATTACHMENT C, SEE PAGES 2058-2061, VOLUME IV

ATTACHMENT D, SEE PAGE 3363, VOLUME VI

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

December 15, 1962

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

Subject: REPORT ON LA CROSSE BOILING-WATER REACTOR (LACBWR)

Dear Dr. Seaborg:

At its forty-fifth meeting on December 13-15, 1962 at Oak Ridge, Tennessee, the Advisory Committee on Reactor Safeguards reviewed the proposed construction of the 165 MW (t) La Crosse Boiling-Water Reactor (LACBWR) to be located near Genoa, Wisconsin. The Committee had the benefit of the referenced report, and discussions with representatives of the Allis-Chalmers Company, the Dairyland Power Co-operative, and the AEC staff.

The applicant proposes to construct a direct cycle, forced circulation, boiling water reactor with internal steam separation. The site area was previously considered by the ACRS at its thirty-fourth meeting on May 18-20, 1961, and reported to be suitable for a reactor of this general type and power level.

Many features of the reactor design have not been determined at this time. The applicant has proposed an extensive research and development program to provide information for the final design. Among the topics to be explored in this program and in the final design are: (1) Performance and mode of operation of the bottom entry control rod drives and poison elements; (2) specific operating limits for the fuel elements; (3) use of low-alloy steel in portions of the primary system; (4) feasibility of obtaining satisfactory load control by automatic regulation of the primary coolant flow; and (5) the required engineered safeguards for containment. The Committee desires to be kept informed of the progress of this research and development program as it relates to the final design.

With satisfactory completion of the above program, the Advisory Committee on Reactor Safeguards believes that a boiling water reactor of the proposed general type and power level can be constructed at this site with reasonable assurance that it can be operated without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

F. A. Gifford, Jr.
Chairman

Reference:

1. ACNP-62574 - Hazards Summary Report for Construction Authorization of the La Crosse Boiling-Water Reactor, dated October 1962.

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

January 17, 1964

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

Subject: REPORT ON LA CROSSE BOILING WATER REACTOR (LACBWR)

Dear Dr. Seaborg:

At its fifty-second meeting on January 9-10, 1964, the Advisory Committee on Reactor Safeguards reviewed Amendment No. 3 to an approved construction authorization, CAPR-5, dated March 29, 1963, for the La Crosse Boilding Water Reactor (LACBWR). The Committee considered the construction of the LACBWR at its forty-fifth meeting and, in its letter of December 15, 1962, concluded that, subject to satisfactory resolution of five specific considerations, a boiling water reactor of the general type and power level proposed could be constructed at this site.

In considering Amendment No. 3 the Committee had the benefit of discussions with representatives of the Allis-Chalmers Manufacturing Company and the AEC staff, and of the documents referenced below.

Amendment No. 3 describes a reduction in the shielding to be located on the inside walls of the containment building. The reduction in shielding as proposed would increase the calculated whole body radiation dosage that would be received outside the containment vessel in the unlikely event of a complete core meltdown. However, the calculated increased dosage does not indicate an unacceptable hazard to the health and safety of the public.

Amendment No. 3 does not introduce any health and safety factors that have not been considered by the Committee previously. The Committee, therefore, affirms the conclusion expressed in its previous letter.

Sincerely yours,

/s/ Herbert Kouts

Herbert Kouts
Chairman

Reference: ACNP-63584, "Amendment 3 to the Application for Construction of the La Crosse Boiling Water Reactor", dated August 1963.

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

November 17, 1966

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

Subject: REPORT ON LA CROSSE BOILING-WATER REACTOR (LACBWR)

Dear Dr. Seaborg:

At its seventy-ninth meeting, on November 10-12, 1966, the Advisory Committee on Reactor Safeguards reviewed proposed operation of the La Crosse Boiling-Water Reactor (LACBWR) by the Allis-Chalmers Manufacturing Company under a provisional operating authorization. The Committee had the benefit of discussion with representatives of the Allis-Chalmers Manufacturing Company, the Dairyland Power Cooperative, and the AEC Staff, and of the documents listed. A Subcommittee of the ACRS met to review this project at Genoa and La Crosse, Wisconsin on December 7 and 8, 1963, and in Washington, D. C. on August 19, 1966 and October 22, 1966. The Committee previously commented on this project in letters to you dated December 15, 1962 and January 17, 1964.

The LACBWR plant is located in Vernon County, Wisconsin, along the east bank of the Mississippi River approximately one mile south of the village of Genoa, Wisconsin and nineteen miles south of the City of La Crosse, Wisconsin.

The reactor plant consists of a direct-cycle, variable-flow forced-circulation boiling-water system which is to be operated at power levels up to 165 MWt. The plant, except for the turbine and connecting piping, is housed in a 60-foot diameter steel cylindrical containment shell having a hemispherical dome. The containment is designed to withstand an internal pressure of approximately 52 psig at 280°F with a design leak rate of 0.1% per day. Double isolation valves are provided in the steam line to prevent leakage from the containment in the unlikely event of a pipe rupture. A manually operated containment spray system is provided to help control containment pressure in the unlikely event of a loss-of-coolant accident.

The core consists of 72 fuel assemblies made up of stainless steel clad fuel elements containing 3.63% enriched uranium dioxide fuel pellets. Each of the fuel assemblies is contained in a shroud can of Zircaloy or stainless steel. Reactivity control is provided by 29 cruciform control rods that operate between the fuel shroud cans. The control rods consist of Inconel-600 tubes filled with B_4C and sheathed in stainless steel.

A high pressure core spray system is provided as an engineered safeguard to cool the core in the unlikely event of a major loss-of-coolant accident. In addition, a low pressure, high flow, alternate core spray system will be installed before operation at power to provide redundancy in emergency core cooling; this sytem also provides means for flooding the containment building up to the height of the top of the core.

A diesel generator has been installed to assure the availability of electrical power for operation of engineered safeguards and for shutdown heat removal; the pumps for the alternate core spray system are to have their own independent diesel drives.

During construction, a number of modifications were proposed to improve the safety of the plant. These have been identified in a series of amendments to the final safeguards report and are currently being added to the plant. The Committee believes that the modifications should be followed closely by the AEC Staff.

This plant is designed for automatic load-following. The Committee believes, however, that the plant should not be operated with automatic load-following until appropriate experience has been obtained with manual operation and the results of such experience reviewed with the AEC Regulatory Staff.

Prior to operation at power, the following items should be resolved with the AEC Regulatory Staff.

1. Appropriate limits on reactivity and flux anomalies during operation.
2. Methods and procedures for detection of leaks in the primary system and for operator action if leaks are detected.
3. Procedures for use in the event of tornado warnings including identification of circumstances under which limitations are to be placed on the operation of the plant or the plant is to be shut down.

4. A program for periodic inspection of the integrity of the plant stack which is adjacent to the containment. Similar attention should also be given to the tall stack planned for construction nearby.

The Committee believes that, by the end of the first year of operation, a program for periodic inspection of primaty system components should be developed and implemented. The Committee also believes that appropriate records regarding design, fabrication and operation should be preserved so as to be available to the operator for reference purposes during the life of the plant. Action on these items should be followed and reviewed by the AEC Staff. The Committee may wish to examine aspects of the periodic inspection program, particularly the frequency and extent of inspections, at the time of review for a full-term operating authorization.

It is the opinion of the ACRS that, if due attention is given to the foregoing items, LACBWR can be operated by Allis-Chalmers under provisional authorization at power levels up to 165 MWt without undue hazard to the health and safety of the public.

Mr. Harold Etherington did not participate in the Committee's review of this project.

Sincerely yours,

/s/

David Okrent
Chairman

References Attached

REFERENCES

1. ACNP-62614, Amendment 1, dated December 20, 1962.
2. ACNP-63582, Quarterly Technical Report No. 1 (Amendment No. 2), dated July 1963.
3. ACNP-63624, Quarterly Technical Report No. 2 (Amendment No. 4), dated November 1963.
4. ACNP-64529, Quarterly Technical Report No. 3 (Amendment No. 5), dated March 1964.
5. ACNP-64572, Quarterly Technical Report No. 4 (Amendment No. 6), dated June 1964.
6. ACNP-65542, Quarterly Technical Report No. 6 (Amendment No. 10), dated June 1965.
7. ACNP-65543, Amendment 11, Volumes I and II, dated June 1965.
8. ACNP-65544, La Crosse Boiling-Water Reactor, Safeguards Report for Operating Authorization, Volumes I and II, dated July 1965.
9. ACNP-64604, Quarterly Technical Report No. 5 (Amendment No. 7), dated August 1964.
10. ACNP-64628, Quarterly Technical Report No. 6 (Amendment No. 8), dated November 1964.
11. ACNP-65517, Quarterly Technical Report No. 7 (Amendment No. 9), dated February 1965.
12. ACNP-65611, Amendment 12, dated December 1965.
13. ACNP-66501, "Answers to Questions (Group I) about LACBWR Received from the Division of Reactor Licensing on September 8, 1965," dated January 1966.
14. ACNP-66505, Amendment No. 14, dated January 1966.
15. ACNP-66509, Amendment No. 15, dated January 1966.
16. ACNP-66510, Amendment No. 16, dated February 1966.
17. ACNP-66512, Amendment No. 17, dated February 1966.
18. ACNP-66518, Amendment No. 18, dated February 1966.
19. ACNP-66517, Amendment No. 19, dated March 1966.
20. Allis-Chalmers letter dated March 17, 1966 to AEC Division of Reactor Licensing with attachment, ACNP-66523, Amendment No. 20, dated March 1966.
21. Allis-Chalmers letter dated March 18, 1966 to the AEC Division of Reactor Licensing, with enclosures.
22. ACNP-66525, Amendment No. 21, dated March 1966.
23. ACNP-66530, Amendment No. 22, dated April 1966.
24. ACNP-66531, Amendment No. 23, dated April 1966.
25. ACNP-66541, Amendment No. 24, dated June 1966.
26. ACNP-66546, Amendment No. 25, dated June 1966.
27. Allis-Chalmers letter dated June 30, 1966 to AEC Division of Reactor Licensing with attachment, ACNP-66548, Amendment No. 26, dated June 1966.

28. Allis-Chalmers letter dated July 12, 1966 to AEC Division of Reactor Licensing with attachment, ACNP-66549, Amendment No. 27, dated July 1966.
29. Allis-Chalmers letter dated August 8, 1966 to AEC Division of Reactor Licensing with attachment, ACNP-66556, Amendment No. 28, dated August 1966.
30. Allis-Chalmers letter dated September 30, 1966 to AEC Division of Reactor Licensing with attachment, ACNP-66564, Amendment No. 29, dated September 1966.
31. Allis-Chalmers letter dated October 14, 1966 to AEC Division of Reactor Licensing with attachment, ACNP-66564.1, Supplement to Amendment No. 29, dated October 1966.
32. Allis-Chalmers letter dated October 28, 1966 to AEC Division of Reactor Licensing with attachment, ACNP-66572, Amendment No. 39, dated October 1966.

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 THE LA CROSSE BOILING WATER REACTOR (LACBWR)

Dear Dr. Seaborg:

At its ninety-third meeting, January 11-13, 1968, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Dairyland Power Cooperative (DPC) to assume responsibility for operation of the La Crosse Boiling Water Reactor (LACBWR), taking over from the presently authorized operator, Allis-Chalmers Manufacturing Company. The Committee had the benefit of discussion with representatives of Dairyland Power Cooperative, Allis-Chalmers Manufacturing Company, Hittman Associates, United Nuclear Corporation, the AEC Division of Reactor Development and Technology, the Chicago Operations Office, and the AEC Regulatory Staff, and of the documents listed. A Subcommittee met in Wisconsin on January 4, 1968. The Committee last reported to you on this project in a letter dated November 17, 1966.

The proposed turnover of responsibility would take place after completion of testing of the plant and a 28-day warranty power run. That time would be toward the end of March, 1968, according to the current schedule. The Technical Specifications would be continued in force with only minor changes.

The DPC staff would be essentially the same as at present. The experience being gained by the staff should be valuable in the future. There will be a continuing special need, in a plant with a small staff such as LACBWR, for maintaining continuity and competence in key reactor personnel.

Technical support for operation of the LACBWR will be provided by United Nuclear Corporation (UNC) under contract to the AEC. Representatives of DPC, UNC, and AEC all stated that prompt, direct communication would be available between the applicant and UNC on any safety questions.

The Committee believes that the turnover of operating responsibility for the LACBWR, and its operation by DPC under provisional authorization, will not result in undue hazard to the health and safety of the public.

Mr. Harold Etherington did not participate in the Committee's review of this project.

Sincerely yours,

/s/

Carroll W. Zabel
Chairman

References attached.

References - La Crosse

1. Letter from Dairyland Power Cooperative dated October 4, 1967; Application for Transfer of Provisional Operating Authorization DPRA-5 for La Crosse Boiling Water Reactor

2. Letter from Dairyland Power Cooperative dated November 22, 1967; Amendment No. 1 to Application for Transfer of Provisional Operating Authorization DPRA-5 for La Crosse Boiling Water Reactor

3. Letter from Allis-Chalmers Manufacturing Company to Division of Reactor Licensing, dated December 29, 1967; Erratic Behavior of Forced Circulation Loop Discharge Rotovalves

4. Letter from Allis-Chalmers Manufacturing Company to Division of Reactor Licensing, dated January 4, 1968

5. Letter from Dairyland Power Cooperative to Division of Reactor Licensing, dated January 8, 1968; Control Rod Nozzle Attachments

6. Letter from Allis-Chalmers Manufacturing Company to Division of Reactor Licensing, dated January 9, 1968; Steam Separator Assessment

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

May 17, 1983

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

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE SYSTEMATIC EVALUATION PROGRAM REVIEW OF THE LA CROSSE BOILING WATER REACTOR

During its 277th meeting, May 12-14, 1983, the Advisory Committee on Reactor Safeguards reviewed the results of Phase II of the Systematic Evaluation Program (SEP) as it has been applied to the La Crosse Boiling Water Reactor. These matters were also discussed during a Subcommittee meeting in Washington, D. C. on May 6, 1983. During our review, we had the benefit of discussions with representatives of the Dairyland Power Cooperative (Licensee) and the NRC Staff. We also had the benefit of the documents referenced.

The La Crosse plant is the third in Group 2 of the SEP to be reviewed; our review of the Yankee plant was reported in our letter dated April 19, 1983, and our review of the Haddam Neck plant is reported in our letter dated May 17, 1983. The La Crosse plant is unique in several respects. It includes a boiling water reactor, designed and built by the Allis-Chalmers Company as part of the Atomic Energy Commission's Second Round Demonstration Program and was subsequently turned over to the current Licensee. It has been in commercial operation since 1969 but, like several other plants in the SEP, has not yet been issued a Full-Term Operating License (FTOL). Of particular interest is the fact that, with an electrical power output of 50 MWe, it is the smallest commercial power reactor in operation in the United States.

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 La Crosse 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, 36 were not applicable to the La Crosse plant and 18 were deleted because they were being reviewed generically under either the Unresolved Safety Issues Program or the Three Mile Island Action Plan. Of the 83 topics addressed in the NRC Staff's review, 52 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 31 remaining topics involved 70 issues relating to areas in which the La Crosse plant did not meet current criteria. These issues were addressed by the Integrated Plant Safety Assessment and various resolutions have been proposed.

For 27 of the 70 issues included in the Integrated Assessment, the NRC Staff concluded that no backfit is required. We concur.

For 21 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 6 remaining issues for which the assessment has been completed, the Licensee has proposed hardware backfits for their resolution and the NRC Staff has found these proposals acceptable.

As has been the case for the other plants in the SEP, the Integrated Assessment has not been completed for a number 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.

Many of the issues still being evaluated by the Licensee relate to the effects of extreme environmental phenomena such as earthquakes, floods, and tornadoes, since the La Crosse plant was not designed to resist these phenomena at the levels that would be required by current criteria.

Use was made of a limited Probabilistic Risk Assessment (PRA) in connection with the NRC Staff's evaluations. Since a plant-specific PRA was not available for the La Crosse plant, the techniques used were similar to those used in similar circumstances for other plants in the SEP. As in those other cases, we believe that the NRC Staff's use of PRA was appropriate and that suitable use was made of the results.

Our conclusions regarding the SEP review of the La Crosse plant are as follows:

1. The SEP has been conducted in such a manner that the stated objectives have been achieved for the most part for the La Crosse Plant.

2. The actions taken thus far by the NRC Staff in its SEP assessment of the La Crosse plant are acceptable.

3. The ACRS will defer its review of the FTOL for the La Crosse plant until the NRC Staff has completed its actions on the remaining SEP topics and the Unresolved Safety Issues and TMI Action Plan items.

Mr. Harold Etherington did not participate in Committee consideration of this matter.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Acting Chairman

References:

1. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Integrated Plant Safety Assessment Systematic Evaluation Program, La Crosse Boiling Water Reactor," Draft Report, NUREG-0827, dated April 1983.
2. U. S. Nuclear Regulatory Commission, Safety Evaluation Reports, La Crosse Boiling Water Reactor, Volumes 1-3, received April 15, 1983.

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 LA SALLE COUNTY 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 from Commonwealth Edison Company for a permit to construct the La Salle County Station, Units 1 and 2. Unit 2 is scheduled for operation about one year after Unit 1. This project was considered at Subcommittee meetings on November 23, 1971, at the plant site, and on December 8, 1971, in Washington, D. C. During its review, the Committee had the benefit of discussions with the applicant, Sargent and Lundy, the General Electric Company, and the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed.

The La Salle County Station will be located in north-central Illinois, in a rural area of large farms about 65 miles southwest of Chicago. The nearest towns to the site are Seneca, 5 miles distant, and Marseilles, 6 miles from the site, with estimated 1975 populations of 2,210 and 5,360 respectively. Several towns with populations of 10,000-20,000 are within 30 miles of the site. Joliet, with an estimated 1975 population of 123,800 is 37 miles to the northeast.

The minimum exclusion distance from the stack is 667 meters (2,190 ft.) and from the center of the reactor building is 515 meters (1,690 ft.). The low population zone radius is 4 miles. This zone had a 1970 population of less than 720.

The La Salle County Station will contain two General Electric boiling water reactors, each to be operated at a power level of 3293 MWt. These reactors are similar to the smaller capacity reactors recently reviewed for the Zimmer Station and for the Bailly Station.

The La Salle County Station site has an area of about 7,000 acres. Of this, about 4,500 acres will be converted to a cooling lake, contained in part by dikes up to about 45 feet in height. A portion of the lake will be designed as a 75-acre emergency cooling pond contained by a Class 1 submerged dike. Water to maintain the lake will be taken from the Illinois River about 4 miles to the north and blowdown will be returned to the Illinois River. The cooling lake will have an elevation of 700 feet MSL. The Illinois River has an elevation of about 485 feet MSL. The applicant states that the cooling lake and the connections to the river are designed to accommodate the future addition of two large generating units to the La Salle County Station.

The cooling lake and the plant site are underlain by glacial till some 165 feet thick. The till is compact and impervious. The plant foundations will be on concrete mats supported by the till.

Current analysis indicates acceptably low peak clad temperatures following a postulated loss-of-coolant accident. A blowdown research program, which was recently begun under the auspices of General Electric 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 minimum critical heat flux ratio and the level swell process, should also be performed during construction of the plant. The results of these studies should be reviewed by the Regulatory Staff.

The applicant proposes not to provide vacuum relief valves between the containment and the reactor building because of the capability of the concrete containment structure to withstand substantial external pressure. The elimination of the need for vacuum relief valves is desirable. The containment design for external pressure should be carried out in a conservative manner.

The applicant has stated that he will comply with AEC Safety Guide 7, Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident.

The applicant proposes to install a sealing system, designed as an engineered safety feature, to minimize leakage through the main steam line isolation valves. The Committee believes that this sealing system should be installed, and in addition, that the main steam lines should be designed and analyzed in a manner which assures their integrity during a design basis earthquake. 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 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 La Salle County Station. 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.

Analyses are being made to determine whether the effectiveness of the ECCS will be decreased if the recirculation control valves or the pump discharge block valves should close following a break in a recirculation line. If significant adverse effects on the ECCS effectiveness are revealed by the analyses, circuits and interlocks, designed to meet the IEEE-279 requirements, should be provided to assure that these valves will remain in the "as-is" condition.

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 La Salle County 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 La Salle County 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,

Spencer H Bush

Spencer H. Bush
Chairman

References:

1) Commonwealth Edison Company letter dated November 3, 1970 transmitting Preliminary Safety Analysis Report, Volumes 1 through 5 to the La Salle County Station, Units 1 and 2

2) Amendments 1, 4, 5, 6, to the License Application of Commonwealth Edison Company for the La Salle County Station, Units 1 and 2

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

April 16, 1981

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

SUBJECT: REPORT ON LA SALLE COUNTY STATION UNITS 1 AND 2

Dear Dr. Hendrie:

During its 252nd meeting, the ACRS completed its review of the application of the Commonwealth Edison Company (Applicant) for a license to operate the La Salle County Station Units 1 and 2. A subcommittee meeting was held in Morris, Illinois on April 3-4, 1981 to consider this project. A tour of the facility was made by members of the Subcommittee on April 3, 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 James R. Schlesinger dated December 17, 1971.

The La Salle County plant is located in La Salle County, Illinois about 70 miles southwest of downtown Chicago. The nearest population center is Ottawa, Illinois about 11 miles northwest of the site.

The La Salle plant uses GE BWR-5 nuclear steam supply systems with a rated power level of 3323 MW(t) each. The La Salle plant has a Mark II pressure suppression containment with a design pressure of 45 psig. The La Salle plant is one of three plants included in the Mark II Owners Group lead plant program. The NRC Staff has concluded review of the lead plant program and has issued Supplements 1 and 2 to NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," which specify generic acceptance criteria. The Staff has concluded that the La Salle facility satisfies the criteria. We concur in this finding.

The Applicant described the organization of the plant staff, including maintenance, engineering, operations, and health physics personnel. The safety review functions and training programs were also discussed. The Applicant is emphasizing plant staffing and personnel training. The Committee believes that efforts to improve staff capabilities should continue, particularly in the area of health physics.

The Applicant and the Staff have under consideration a recently issued AEOD report (Reference 4) concerning the risk potential for pipe breaks in the BWR scram system. This report is being reviewed by the NRC Staff and by a BWR Owners Group. We believe that this issue should be treated generically and need not be resolved prior to operation of the La Salle plant.

The NRC Staff proposes to require the installation of core thermocouples in the La Salle plant 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. In a letter to Commissioner Gilinsky dated July 16, 1980, the Committee recommended that careful examination of the feasibility of the use of core outlet or core subassembly thermocouples, and the pros and cons of such use, be undertaken. We recommend that such a study be completed for the La Salle plant before a decision is reached on this requirement. The Committee wishes to be kept informed.

The NRC has identified a number of additional outstanding issues. We believe that these can be resolved in a manner acceptable to the NRC Staff.

The ACRS 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 La Salle County Station Units 1 and 2 can be operated at power levels up to 3323 MW(t) each without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. Commonwealth Edison Company "La Salle County Station Final Safety Analysis Report," Volumes 1-12 and Amendments 1-55.
2. U.S. Nuclear Regulatory Commission "Safety Evaluation Report Related to the Operation of La Salle County Station Units 1 and 2," USNRC Report NUREG-0519, dated March 1981.
3. U.S. Nuclear Regulatory Commission, Supplements 1 and 2 to NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," dated September 1980 and February 1981.
4. U.S. Nuclear Regulatory Commission "Safety Concerns Associated With Pipe Breaks in the BWR Scram System," Office for Analysis and Evaluation of Operational Data, March 1981.

ADVISO~Y COMMITTEE ON REACTOR S~EGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

August 10, 1971

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

Subject: REPORT ON LIMERICK GENERATING STATION UNITS 1 AND 2

Dear Dr. Seaborg:

At its 136th meeting, August 5-7, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application from the Philadelphia Electric Company for a permit to construct the two-unit Limerick Generating Station. The project was considered at Subcommittee meetings on November 10, 1970 at the plant site, and on March 31 and July 29, 1971, in Washington, D. C. During its review the Committee had the benefit of discussions with representatives and consultants of the applicant, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The Limerick Station will be located in Pennsylvania on a 587-acre site on the Schuylkill River about midway between Philadelphia and Reading. The nearest population center is Pottstown (1960 population - 26,000; year 2000 predicted population - 55,000) with its nearest boundary 1.7 miles to the northwest. The low population zone radius is 1.3 miles. The estimated population in 1968 was 500 persons within one mile and 5,200 persons within two miles. The minimum exclusion distance is about 2,500 feet, which extends to the west bank of the Schuylkill River and includes a small uninhabited island owned by the State of Pennsylvania. The City of Philadelphia is 20.7 miles to the southeast with a 1970 census population of about 2,000,000.

Each unit of the Limerick Station includes identical boiling water reactors to be operated at a power level of 3293 MWt. The core designs, power densities, and other features of the nuclear steam supply systems are essentially identical to the Browns Ferry units of the Tennessee Valley Authority and Peach Bottom Units 2 and 3 currently under construction by the Philadelphia Electric Company. Waste heat is rejected to the atmosphere by two natural draft cooling towers. The normal cooling water requirement of 74 cfs, including 54 cfs for consumptive use, is supplied from the Schuylkill River. To provide another source during drought periods arrangements are being made to obtain water from the Delaware River.

The containment is of the over-under pressure suppression type similar to that of the Shoreham Nuclear Power Station. The drywell is a reinforced concrete, steel-lined truncated cone; the wetwell is a cylinder of similar construction. The drywell and wetwell are separated by a 3-1/2 foot thick reinforced concrete floor penetrated by 85 vent pipes. A low-leakage, Class I reactor building surrounds both units which share a single compartment above the level of the refueling floor and occupy separate compartments below this level. The building is designed to relieve through blow-out panels at an internal pressure of 7 inches of water, an arrangement which the applicant has stated serves to protect engineered safety equipment from excessive steam exposure while still maintaining offsite doses from postulated process steamline failures far below 10 CFR Part 100 guidelines.

The reactor building in-leakage at a differential pressure of 1/4 inch of water will be limited to 50% of the building volume per day. On isolation of the building a recirculation-filtration system starts automatically, continuously processing about 60,000 cfm through HEPA and charcoal filters. A small fraction of the discharge of this system is exhausted to the outside environment through the standby gas treatment system which includes deep-bed, charcoal filters.

The entire length of the main steam lines, up to and including the turbine stop valves, will be designed to Class I seismic standards. The main steam lines from the downstream isolation valve to the turbine stop valve will be designed and fabricated in substantial accordance with the requirements for AEC quality assurance Classification Group B. In addition, the Committee believes it appropriate to design and install all connected piping down to 2-1/2 inches in diameter to Class I seismic standards out to and including the first valve. The applicant has stated that he will install a third steam line isolation valve downstream of the two fast-acting valves or develop an equivalent water-seal system acceptable to the Regulatory Staff.

The biological shield is to be constructed of magnetite concrete placed between steel plates. The shield will be reinforced near openings to insure integrity for postulated ruptures in the vicinity of nozzles. The Committee believes that the entire biological shield should be designed to have reasonable ability to withstand internal pressure and jet forces.

The emergency core cooling system (ECCS) has been changed in several ways. The high pressure coolant injection (HPCI) system has been modified to inject water directly into the core through the spray sparger rather than into the downcomer region by the feedwater sparger. In addition, the applicant has stated that the turbine driven HPCI pump will also be modified to the extent feasible to increase the volume of water delivered to the core. The low pressure coolant injection (LPCI) system has been changed to inject water inside the core shroud through four vessel penetrations. Each of two pairs of LPCI pumps feed a header serving two nozzles. The applicant has stated that these changes provide increased reliability of these systems and reductions in the peak clad temperatures attained in the unlikely event of a loss-of-coolant accident.

The radioactive waste disposal systems include several features beyond those normally provided in boiling water reactor plants. The liquid waste system permits the recycling of equipment and floor drain wastes and the evaporation of chemical and laundry wastes before discharge to the environment. The gaseous waste system provides for the recombining of hydrogen and oxygen, condensing the vapor, hold-up for decay of short-lived isotopes, and cryogenic separation of the noble gases. Krypton and xenon may be stored for periods sufficiently long that krypton-85 becomes the only significant remaining radioisotope. Provisions will be made to utilize non-radioactive steam in the turbine gland seals and to process containment purge gases when deinerting. The Committee believes that these waste management systems are capable of limiting releases of radioactivity to the environment to levels that are as low as practicable.

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 Limerick reactors. 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 applicant has stated that 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 proposed system is not capable of coping with hydrogen generation rates in accordance with current AEC criteria unless the primary containment is inerted. Therefore, 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 AEC Safety Guide 7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident."

The applicant has selected a value of 0.12 g for the acceleration representing the maximum ground motion at the site and on which Class I seismic design is to be based. The Committee recommends a minimum acceleration of 0.15 g be used for the design basis earthquake for this site.

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 Limerick 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 Limerick Generating Station Units 1 and 2 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H Bush

Spencer H. Bush
Chairman

References

1. Philadelphia Electric Company Preliminary Safety Analysis Report (Volumes 1 through 5), for Limerick Generating Station Units 1 and 2

2. Amendments 1, 2, 3, 4, 6, 7, 9 & 10 to the License Application of Philadelphia Electric Company for the Limerick Generating Station Units 1 and 2

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

October 18, 1983

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

Dear Dr. Palladino:

SUBJECT: ACRS INTERIM REPORT RELATED TO THE OPERATING LICENSE APPLICATION FOR THE LIMERICK GENERATING STATION, UNITS 1 AND 2

During its 282nd meeting, October 13-15, 1983, the Advisory Committee on Reactor Safeguards reviewed the application of the Philadelphia Electric Company (Applicant) for a license to operate the Limerick Generating Station, Units 1 and 2. There was a tour of the facility by members of the Subcommittee on the morning of October 7, 1983. A Subcommittee meeting was held in Pottstown, Pennsylvania on October 7 and 8, 1983 to consider this application. During its review the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff and an oral presentation by a member of the public before the Subcommittee. The Committee also had the benefit of the documents referenced. The Committee commented on the application for a permit to construct this Station in a report dated August 10, 1971.

The Limerick facility is located near the Schuylkill River about 1.7 miles southeast of the limits of the borough of Pottstown, Pennsylvania. The site is about 21 miles northwest of the nearest boundary of Philadelphia. The Limerick Generating Station uses BWR 4 boiling water reactors supplied by the General Electric Company. The pressure suppression containment system uses the General Electric Mark II design. The power rating of each unit is 3293 MWt. Bechtel Power Corporation is providing architectural, engineering, construction, and startup services. Construction on Unit 1 is about 90 percent complete, and construction on Unit 2 is about 30 percent complete.

The nuclear steam supply system and the containment system are almost identical to those of the Susquehanna Steam Electric Station which was reviewed for an operating license with an ACRS report issued on August 11, 1981.

Because of the uncertain schedule for Unit 2, the Committee does not believe it appropriate to report on Unit 2 at this time.

Our review included an evaluation of the management organization, the operational staff, and the training program for the operating and maintenance staff. The tour of the facility by ACRS members included the power plant simulator and the teaching laboratories housed in the training center which is located near the Limerick site and used extensively in the training of plant personnel.

The Limerick Generating Station is the Applicant's second nuclear station. The Applicant operated Peach Bottom, Unit 1, a gas-cooled reactor, from 1967 to 1975 and has operated Peach Bottom, Units 2 and 3, which are boiling water reactors, since 1974. During our discussions, the Applicant demonstrated an extensive knowledge of the operation, design, and construction features of the plant. We conclude that the Applicant has the necessary technical and management capability to operate the Limerick Generating Station.

Stress-assisted corrosion cracking of primary system components has been observed in a number of operating General Electric nuclear steam supply systems. The materials being proposed for similar components in the Limerick Station are believed to be much improved. We recommend, however, in view of past experiences, that the Applicant develop and maintain a careful surveillance program to identify any factors encountered during plant operation which have the potential for materials damage.

The NRC Staff has not completed its review of the emergency planning for the Limerick Generating Station. We expect to review this subject in later meetings with the NRC Staff and the Applicant. We also plan to review the security plan for the Limerick Station.

In response to a request from the NRC Staff, the Applicant submitted a probabilistic risk assessment (PRA) in March 1981. A supplement to this report was submitted in April 1983 in the form of a severe accident risk assessment (SARA) report. In its meetings with the Applicant, the Committee reviewed a number of plant features that had been identified during the PRA and have been modified in order to reduce risk produced by certain hypothesized accidents. The NRC Staff Safety Evaluation Report for the Limerick Station does not make direct use of the information contained in the PRA and in SARA but rather follows the guidelines of the Standard Review Plan. The manner in which the NRC Staff will use the PRA and SARA is described in NRC Staff letters to the Atomic Safety and Licensing Board dated April 13 and May 24, 1983. In these documents the NRC Staff states that the PRA and SARA will be used to compare the risk presented by the Limerick Station with that from other nuclear power plant facilities. If this risk is found to be significantly greater than that associated with other such facilities, the NRC Staff will consider the need to recommend compensatory features. The NRC Staff's review of the PRA and SARA is continuing. We expect to review the PRA and SARA with respect to the methodology, results, and use in the Limerick licensing process. We believe that the demography of the site calls for a careful consideration of the results of the PRA and the SARA.

The Committee has, in several prior operating license reviews, noted the importance of assuring that the seismic contribution to risk is acceptably low, with allowance for lower frequency, more severe seismic events than that considered as the safe shutdown earthquake. This issue is addressed in the SARA report. We intend to explore it further in our continuing review.

We wish to consider further NRC Staff views concerning the failure modes and consequences of the main cooling towers during severe natural phenomena or explosions of materials transported by rail. Our concern is with the close proximity of emergency and residual heat removal service water piping and power supply conduits to the cooling tower basin.

We have not completed our review of the Limerick Generating Station. We do conclude that Unit 1 is well constructed and well managed. As indicated above, matters still to be reviewed are: emergency planning, plant security, margins against less probable but more severe seismic events than that considered as the safe shutdown earthquake, consequences of cooling tower failure, and the PRA and SARA. We will report on these matters in a subsequent letter. However, at this stage of our review, we believe that fuel loading and reactor operation at 5 percent power can be carried out without undue risk to the health and safety of the public.

Mr. J. J. Ray did not participate in the Committee's considerations regarding this matter.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Acting Chairman

References:

1. Philadelphia Electric Company, "Final Safety Analysis Report, Limerick Generating Station, Units 1 and 2," Volumes 1-16, and Amendments 1-21
2. "Safety Evaluation Report Related to the Operation of Limerick Generating Station, Units 1 and 2," USNRC Report NUREG-0991, dated August, 1983
3. Philadelphia Electric Company, "Probabilistic Risk Assessment, Limerick Generating Station," Volumes 1 and 2, dated April, 1982 and PRA Proprietary Volumes, "System Level Fault Trees," dated April, 1982 and "Quantification of Limerick Event Tree Functions," dated June, 1982
4. Report prepared by NUS for Philadelphia Electric Company, "Severe Accident Risk Assessment, Limerick Generating Station," Volumes I-II, dated April, 1983

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

November 6, 1984

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

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE LIMERICK GENERATING STATION

The Committee commented on the application for a permit to construct this Station in a report dated August 10, 1971, and on the application to operate this Station in an interim report dated October 18, 1983. During its 295th meeting, November 1-3, 1984, the Advisory Committee on Reactor Safeguards continued its review of the application of the Philadelphia Electric Company (Applicant) for a license to operate the Limerick Generating Station, Units 1 and 2. This phase of the review was continued during ACRS Subcommittee meetings held on October 9-10 and October 20, 1984. The Committee also had the benefit of the documents referenced. During its review, the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff as well as written and oral statements from members of the public.

The Committee stated in its October 18, 1983 interim report that, because of the uncertain schedule for Unit 2, it was not appropriate to report on Unit 2 at that time. Construction on Unit 2 has been stopped, but may be resumed after the start of operation of Unit 1. We do not believe it is appropriate for the Committee to report on Unit 2 at this time.

The Committee in its October 18, 1983 report stated that it had not completed its review and listed a number of matters yet to be considered. These matters have been discussed at subsequent Subcommittee and Committee meetings, and we conclude that they have been dealt with satisfactorily.

In response to a request from the NRC Staff, the Applicant submitted a probabilistic risk assessment (PRA) in March 1981. A supplement to this PRA report was submitted in April 1983 in the form of a severe accident risk assessment (SARA) report. The NRC Staff has reviewed this study and has used results from this study in the Environmental Statement for this Station. The Applicant has used insights from this PRA/SARA evaluation in the design of and in the development of operational procedures for the Limerick plant. The Applicant, in discussions with the Committee, demonstrated an understanding of the methodology and its uses and a commitment to its application in the operation of the Limerick plant. The Applicant is to be commended for this work.

The Limerick PRA/SARA includes a seismic risk analysis which reflects the state-of-the-art which was used in the Zion and Indian Point PRAs. The results obtained by these methods are characterized by large uncertainties, and are the subject of disagreement in the scientific and engineering communities. We believe that the NRC and the industry should continue to work to develop methods which can be used to quantify seismic risk and to identify any seismic outliers which might exist.

The Committee has previously recommended that the Zion and Indian Point plants be reviewed for systems interactions that might lead to significant degradation of safety. The issue of systems interactions is currently being addressed under the USI A-17, "Systems Interactions in Nuclear Power Plants." Philadelphia Electric Company has already examined many of the possible systems interactions in the Limerick plant. However, in view of the demography of the site, we recommend that Limerick receive special attention in the NRC Staff's consideration of USI A-17.

We believe that, subject to the resolution of open items identified by the NRC Staff and subject to the satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Limerick Generating Station, Unit 1 can be operated at power levels up to 3293 MWt without undue risk to the health and safety of the public.

Additional comments by ACRS Member David Okrent are presented below.

Sincerely,

David A. Ward

David A. Ward
Acting Chairman

Additional Comments by ACRS Member David Okrent

The matter of potential improvements in design either to prevent or to mitigate severe accidents received only limited attention by the NRC Staff during this review. Further studies are in progress which should be completed and evaluated in the next two or three years. At that time, the Limerick Generating Station should be reviewed for the possible desirability and appropriateness of such improvements.

References:

1. Philadelphia Electric Company, "Final Safety Analysis Report, Limerick Generating Station, Units 1 and 2," Revisions 21-36
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Limerick Generating Station, Units 1 and 2," Supplement No. 1, USNRC Report NUREG-0991, dated December 1983

3. BNL Report Prepared for U. S. Nuclear Regulatory Commission, "A Review of the Limerick Generating Station Severe Accident Risk Assessment" - Review of Core Melt Frequency, NUREG/CR-3493 and BNL-NUREG-51711, dated July 1984
4. BNL Report Prepared for U. S. Nuclear Regulatory Commission, "A Review of the Limerick Generating Station Probabilistic Risk Assessment," NUREG/CR-3028 and BNL-NUREG-51600, dated February 1983
5. U. S. Nuclear Regulatory Commission, "Review Insights on the Probabilistic Risk Assessment for the Limerick Generating Station," USNRC Report NUREG-1068, dated August 1984
6. Letter from A. Schwencer, NRC Division of Licensing, to Edward G. Bauer, Jr., Philadelphia Electric Company, Subject: Review of Limerick Severe Accident Risk Assessment, dated June 22, 1984, with attachment, BNL-33835, "Containment Failure Mode and Fission Product Release Analysis for the Limerick Generating Station: Base Case Assessment"
7. Letter from M. Lewis, Member of the Public, to R. Savio, Advisory Committee on Reactor Safeguards, regarding the ACRS review of the Limerick Generating Station, dated October 3, 1984

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

October 11, 1962

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

Subject: REPORT ON LITHIUM COOLED REACTOR EXPERIMENT (LCRE)

Dear Dr. Seaborg:

At its forty-fourth meeting, October 4-6, 1962, the Advisory Committee on Reactor Safeguards considered the 10 MW(th) Lithium Cooled Reactor Experiment (LCRE) to be constructed at the Flight Engine Test (FET) Building at the National Reactor Test Station in Idaho. The review was conducted on the basis of the documents referenced below and discussions with representatives of Pratt and Whitney Aircraft, and the staff of the Atomic Energy Commission.

The Committee received the "Supplement No. 1, PWAC-370, Preliminary Safety Analysis Report for the LCRE," describing containment of the LCRE, too late for consideration at this meeting.

The Committee is of the opinion that a reactor of the proposed type, if adequately contained, can be constructed at the proposed site with reasonable assurance that it may be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
F. A. Gifford, Jr.
Chairman

References:

1. PWAC-370, "Preliminary Safety Analysis Report for the Lithium Cooled Reactor Experiment (LCRE)", dated July 20, 1962, Confidential/RD.

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

August 28, 1964

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

Subject: REPORT ON LOFT FACILITY

Dear Dr. Seaborg:

At its fifty-seventh meeting, on August 24-26, 1964, the Advisory Committee on Reactor Safeguards reviewed the proposal for construction of the LOFT facility at the National Reactor Testing Station, in Idaho. During this review, the Committee held discussions with representatives of the Phillips Petroleum Company, Kaiser Engineers and the AEC Staff. The documents listed below were used. A subcommittee reviewed the LOFT facility on August 10, 1964.

The LOFT facility is to contain a 50 MW(t) pressurized water reactor, and is intended as a device to improve understanding of the course and effects of major loss-of-coolant accidents in reactors of this kind. After a series of preliminary tests, the LOFT reactor is to be operated long enough to develop a near-equilibrium content of halogens in the fuel, and then to be subjected to a planned loss of coolant. The preliminary tests are meant to provide information needed to interpret the loss-of-coolant test, and to establish that it can be performed safely. The test itself is to be instrumented to determine quantitatively the history of the fission products liberated from the fuel elements after the core is no longer cooled.

In its review, the Committee concentrated on some aspects of the proposed design, and on analyses of the radiological effects of the loss-of-coolant test and of potential reactor accidents. The Committee believes that, in the course of design and construction of the LOFT facility, the following items warrant special attention:

(1) Containment vessel temperature.

The integrity of the containment is essential both to safe performance of the planned test and to protection of the public from effects of unlikely major accidents. It is suggested that means be provided for keeping the containment vessel in the ductile temperature range, particularly at times when the reactor is pressurized or the containment is subjected to stresses from high internal pressure.

(2) Containment Leak Rate.

Particular care should be taken to assure that the leak rate of the containment vessel with all its penetrations does not exceed the value proposed (0.2%/day at 24 psig).

(3) Containment Pressure Reduction.

To improve the protection to the public, it is suggested that the proposed engineered safeguards be improved or supplemented to provide the capability for more rapid pressure reduction in the containment vessel.

(4) Additional Emergency Core Cooling.

Means of preventing core melting in case of inadvertent loss of coolant should be provided.

The Phillips staff has analyzed the radiological effects of the loss-of-coolant test and of unlikely major accidents. Analyses were supplied that used conservative assumptions on fission product liberation, transport, and plateout, and conservative estimates of meteorological conditions. These indicate that the radiation exposure limits of AEC Manual Chapter 0524 and 10CFR20 should bot be appreciably exceeded by the planned test, and that a major accident that led to large fission product release and rupture of the containment by a missile would not have consequences that exceed the guideline values in 10CFR100.

The Committee believes that 10CFR20 (or AEC Manual Chapter 0524) should be followed in design for normal performance of the planned tests, and that 10CFR100 should guide the protection of the public from the consequences of unlikely major accidents. The Committee believes that, if the suggestions above are followed, there is reasonable assurance that these safety standards can be met by the LOFT facility and tests.

It is the opinion of the ACRS that, subject to the above conditions, the proposed LOFT facility can be built at the proposed site, with reasonable assurance that it can be operated as planned without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References:
1. IDO-16981, Preliminary Safety Analysis Report, LOFT Facility, dated April 1964.

2. Memorandum from Frank K. Pittman to Edson G. Case, dated May 5, 1964, with attachment.

3. Supplemental Information on LOFT, Preliminary Safety Analysis Report, undated, received August 24, 1964.

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

JUL 16 1975

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

Subject: REPORT ON LOFT FACILITY

Dear Mr. Anders:

At its 183rd meeting, July 10-12, 1975, the Advisory Committee on Reactor Safeguards reviewed the safety of operation of the Loss of Fluid Test (LOFT) facility for the proposed experimental mode involving primary system blowdown into a pressure suppression tank. Subcommittee meetings on this project were held on July 9, 1975, and on July 25, 1973. Members of the Committee visited the facility on July 24, 1973. In its review, the Committee had the benefit of discussions with representatives of the Energy Research and Development Administration (ERDA), Aerojet Nuclear Corporation, and the Nuclear Regulatory Commission (NRC). The Committee also had the benefit of the documents listed. The Committee reported previously on the construction of LOFT on August 28, 1964. At that time, different experimental objectives existed for LOFT.

The LOFT facility is located in southeastern Idaho on the 894 square mile site of the Idaho National Engineering Laboratory, which is approximately 30 miles from Idaho Falls. The test reactor is located in Test Area North, which has a daytime population of about 2100.

The nuclear steam supply system is mounted on the Mobile Test Assembly (MTA), which was initially assembled at the Technical Support Facilities (TSF) and then transported by rail to the LOFT containment vessel. The reactor is rated at 55 MW(t), and has the capability of being transported back to the TSF.

The current purpose of LOFT is to serve as a vehicle for conducting integrated LOCA-ECCS test programs. The NRC Staff has reviewed only that phase of the overall experimental program for which the primary system is caused to blow down into the pressure suppression tank rather than into the containment. Also, the NRC Staff has not evaluated safety matters related to moving the MTA out of the containment.

LOFT was designed and constructed over a considerable period of time and does not meet current NRC requirements in all aspects. The NRC Staff has recommended that several specific areas be addressed by ERDA as part of ERDA's responsibility for the safe operation of the facility. The Committee wishes to call particular attention to the importance of proper requalification of LOFT systems and components after severe transient tests and the establishment of an effective administrative apparatus for review of the continued safe operation of the plant, keeping in mind the needs for the experimental information being sought.

The ACRS recognizes that from the very nature of the facility, the probability of an accident in LOFT may be greater than for a typical commercial reactor. However, the total power is relatively low and operation is intermittent; also, the site is remote and emergency planning is kept in a state of readiness.

The ACRS believes that, in light of the above considerations, operation of LOFT in the pressure suppression mode for the proposed blowdown experiments should not pose an undue risk to the public health and safety. The Committee recommends that other experimental programs for LOFT be reviewed and evaluated for safety by ERDA, and by the NRC Staff if appropriate, maintaining a proper balance between the safety questions arising from a particular proposed experimental program and the need for the information to be gained.

Sincerely yours,

WKerr

W. Kerr
Chairman

REFERENCES:

1. Aerojet Nuclear Company, Final Safety Analysis Report (FSAR), Vols. 1-3, for LOFT Integral Test System, March 29, 1974.

2. Supplements 1 & 1A to FSAR.

3. Aerojet Nuclear Company, LOFT Integral Test System - Design Basis Report, January, 1974.

4. U. S. Nuclear Regulatory Commission, Safety Evaluation of the Loss of Fluid Test Facility, May, 1975.

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

March 14, 1961

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

Dear Dr. Seaborg:

At the request of and in cooperation with the Division of Licensing and Regulation, representatives of the Advisory Committee on Reactor Safeguards plan to visit possible sites which have been proposed for future reactor developments in Los Angeles County, California. A part of the Committee is currently on the West Coast in connection with other ACRS matters and has expressed willingness to undertake a visit to the Los Angeles sites as a part of their trip. Accordingly, I have appointed a Subcommittee for this project consisting of Dr. C. R. McCullough (Chairman), Dr. Leslie Silverman, Dr. Frank A. Gifford, and Dr. C. R. Williams. This Subcommittee together with members of the staff of the Division of Licensing and Regulation plans to meet with representatives of the City of Los Angeles on Thursday, March 16 and Friday, March 17, 1961.

It is our understanding that the sites which we have been asked to consider are for a large power reactor or reactor complex, not necessarily the ICBWR.

Since review of these sites has only recently been requested of the ACRS, I took the liberty of discussing this matter by telephone with Commissioner Olson on Monday afternoon, March 13, 1961.

Sincerely yours,

/s/ by RFF

T. J. Thompson
Chairman

cc: Commissioner Olson

COPY

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

September 11, 1961

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

Subject: REPORT ON SITES FOR REACTORS FOR CITY OF LOS ANGELES DEPARTMENT OF WATER AND POWER

Dear Dr. Seaborg:

At its thirty-sixth meeting on September 7-9, 1961, the Advisory Committee on Reactor Safeguards reviewed eight sites proposed by the Department of Water and Power of the City of Los Angeles for a large multiple reactor complex. The present concept involves the eventual operation of four 300 MW(e) reactors. The proposed sites had been visited by a Subcommittee of the ACRS accompanied by a member of the AEC Staff and a representative of the U. S. Weather Bureau. The Committee was assisted in this review by the AEC Staff and the U. S. Weather Bureau. A representative from the Department of Water and Power was present. Documents listed below were available for reference.

The eight sites can be grouped into three general areas: Area I, San Francisquito Canyon No. 2 (the site of an existing LAWP hydroplant); Area II, Green Valley; and Area III, Fairmount. The first two areas are in the Angeles National Forest. The third is over the ridge of mountains, in the Antelope Valley, a part of the Mojave Desert, near Fairmount.

On the basis of the present population distribution, all three areas provide suitable sites for reactors of conservative design. The Committee received a study of the projected population growth of these areas. It is quite clear from this study and from direct observation that the population of the Newhall-Saugus region will increase. Area I (San Francisquito Canyon No. 2) is nearest to this region. The air flow toward this populated area will be frequent and, under unfavorable meteorological conditions, will give relatively small dilutions. Therefore, reactors at the Area I location will require more engineering safeguards than reactors at the other sites. There is also a substantial projected population growth in the Antelope Valley; however, the location of this growth with regard to the possible reactor site is speculative.

The ACRS recognizes that before a final decision is reached to proceed with the reactor complex at a particular site, additional meteorological and hydrological information based on an extensive study at the site is required. In addition, there would need to be a detailed specification of the exclusion area, an estimate of the population distribution in the vicinity, and the proposed engineering safeguards for the reactors.

With the proper consideration of the above comments, the ACRS believes that it would be possible to locate reactors at these sites without undue risk to the health and safety of the public.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

References:

1. Preliminary Report, Proposed Nuclear Site Areas, April 17, 1961, submitted by Department of Water and Power, City of Los Angeles.
2. Planning for People in North Los Angeles County, Part II. Background for Planning, September 30, 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: REPORT ON LARGE POWER REACTOR SITES PROPOSED BY THE CITY OF LOS ANGELES

Dear Dr. Seaborg:

At its fortieth meeting on March 29-31, 1962, the Advisory Committee on Reactor Safeguards reviewed the suitability of two new sites proposed by the City of Los Angeles Department of Water and Power for a large power reactor complex. The Committee also heard presentations of two reactor concepts by representatives of the Westinghouse Electric Corporation and the General Electric Company. The sites were proposed for ultimate installation of two or more reactors whose individual power levels would be 1600 megawatts thermal. On March 14 and 15 in California, a subcommittee of the ACRS reviewed the sites and reactor concepts proposed. The Committee had the benefit of staff analysis and advice from earthquake and meteorological consultants from the City of Los Angeles Department of Water and Power.

On previous occasions, the ACRS has reviewed several proposed power reactor locations in Southern California including sites proposed by the City of Los Angeles. (See letters referenced below.) Our most recent letter, dated September 11, 1961, approved three inland areas within Los Angeles County for large power reactors with suitable containment.

In its most recent proposal, the City of Los Angeles presented two coastal sites which its representative stated present appreciable economic advantages over the presently accepted sites. These two sites are a southern site now owned by the City, and a western site which could be obtained.

In regard to the two new sites proposed for reactors of the general concepts presented, the Committee has the following comments: Neither of the locations can meet the site criteria guidelines proposed in 10CFR-100 for the power level requested. Both sites are within areas of high and increasing population. In this connection, it should be noted that power reactors of the size proposed have not yet been built and proved. Such reactors would contain larger fission product inventories than any licensed power reactor now operating or under construction.

If the sites proposed are to be considered acceptable, then reliance must be placed on proved engineering safeguards as a means of preventing exposure of significant numbers of people to possible radiation injury. The Committee believes that it is possible with present engineering technology to overcome the potential danger from serious consequences of a major earthquake.

The Committee has the following comment concerning the two reactor concepts proposed, and their respective containments: neither proposal provides proved assurance of satisfactory containment of an accident, such as a serious nuclear excursion, which releases radioactivity simultaneously with the release of pressure. The possibility of such an accident cannot be excluded on the basis of present knowledge.

Of the two coastal sites, the western site is in an area of lower population density and is further removed from large centers of population. Neither site is suitable for either of the proposed reactor facilities. The proposed plant designs might more readily be modified to a form suitable for the western site.

Sincerely yours,

/s/
F. A. Gifford, Jr.
Chairman

Referemces attached

References:

1. Letter dated Oct. 26, 1961, from Los Angeles Department of Water & Power Proposing Two Sites Along Pacific Coast in Vicinity of Los Angeles for Nuclear Power Station.
2. Preliminary Report - Proposed Nuclear Power Plant Ocean Site, dated October 1961.
3. Report - Safety & Site Considerations of 300 Mwe Closed Cycle Water Reactor Power Plant for City of Los Angeles - Revised Nov. 8, 1961.
4. Preliminary Report - Safety Features of the Reactor and Its Containment for a Single Cycle, Forced Circulation Boiling Water Reactor, dated Jan. 10, 1962.
5. Preliminary Report to Department of Water & Power of the City of Los Angeles on "Earthquake Hazards at Site of Proposed Haynes Beach Nuclear Power Generating Plant," dated Feb. 19, 1962.
6. Report - "Foundation Investigation - Units No. 5 and No. 6 Haynes Steam Plant," dated Feb. 13, 1962.
7. Letter from ACRS to AEC, dated March 6, 1960, Subject: Nuclear Power Plants in California.
8. Letter from ACRS to AEC, dated Sept. 11, 1961, Subject: Report on Sites for Reactors for City of Los Angeles Department of Water and Power.
9. Letter from ACRS to AEC, dated June 8, 1960, Subject: Improved Cycle Boiling Water Reactor (ICBWR) - Department of Water and Power of the City of Los Angeles.
10. Letter from ACRS to AEC, dated June 27, 1960, Subject: Improved Cycle Boiling Water Reactor (ICBWR) Department of Water and Power of the City of Los Angeles.
11. Letter from ACRS to AEC, dated July 25, 1960, Subject: Improved Cycle Boiling Water Reactor.
12. Letter from ACRS to AEC, dated Jan. 14, 1961, Subject: Improved Cycle Boiling Water Reactor.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON 25, D. C.

October 12, 1962

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

Subject: REPORT ON CITY OF LOS ANGELES WATER AND POWER DEPARTMENT BOILING WATER AND PRESSURIZED WATER REACTOR PROPOSALS

Dear Dr. Seaborg:

At its forty-third meeting August 23-25, 1962 at Idaho Falls, Idaho, and at its forty-fourth meeting in Washington, D. C., October 4-6, and 12, 1962, the Advisory Committee on Reactor Safeguards reviewed the proposed, approximately 1600 MW(t), boiling water and pressurized water reactors one of which may be constructed and operated by the Department of Water and Power of the City of Los Angeles at a site designated as the "western site". The Committee had the benefit of several subcommittee meetings, the references listed below, and discussions with representatives of the Department of Water and Power of the City of Los Angeles, the General Electric Company, Westinghouse Electric Corporation, Stone and Webster Corporation, and the AEC staff.

The Committee in its reviews has focused its attention on the adequacy of engineered safeguards for the containment of any significant potential releases that might affect the health and safety of the public.

The large pressurized water reactor has, as a proposed engineered safeguard concept, a double containment vessel which completely encloses the primary system. Back pumping and monitored leakage of a porous "popcorn" concrete filled space between the containment walls and of all penetrations are provided. The system depends to some extent on keeping the space between the membranes at negative pressure. Redundancy in the pumping equipment is used to insure against failure. The containment membranes are independent as to leakage, but depend on the porous concrete for strength. The reinforced concrete on the outside augments containment vessel strength and provides shielding. The proposal includes holdup of routine radioactive gaseous release. In the opinion of the Advisory Committee on Reactor Safeguards this containment system is adequate.

The proposed large boiling water reactor has a pressure suppression system surrounded by an additional containment of the dry well and suppression pool. The primary steam line extends beyond this double containment to the turbine building. Containment of fission product release from an accident thus depends upon rapid closure of isolation valves. In view of the stringent requirements imposed by the site, it is the Committee's opinion that the containment as proposed is not adequate in some respects for this reactor at this site. The Committee also believes that holdup of routine gaseous releases will be necessary during unfavorable meteorological conditions.

The Advisory Committee on Reactor Safeguards believes that either reactor if provided with adequate containment of the primary system can be located at the western site with reasonable assurance that such reactor can be operated without undue risk to the health and safety of the public. It is believed also that this site may be adequate for multiple reactors assuming that suitable containment and confinement are provided.

Sincerely yours,

/s/
F. A. Gifford, Jr.
Chairman

References Attached (1 page)

References:

1. Letter dated April 25, 1962 from Los Angeles Department of Water & Power with attachment, NSS-1, Sketch of Proposed Changes.
2. Letter dated May 28, 1962 from Los Angeles Department of Water & Power transmitting Addendum to Preliminary Report - Proposed Nuclear Power Plant Ocean Sites, October 1961.
3. Letter dated June 13, 1962 from Los Angeles Department of Water & Power with attachments as indicated.
4. Preliminary Report on Safety Features of the Reactor and Its Containment for a Single Cycle, Forced Circulation Boiling Water Reactor, Addendum No. 1, dated June 15, 1962.
5. Letter dated August 9, 1962 from Los Angeles Department of Water & Power with attachment, Guillotined Main Steam Line.
6. Letter dated August 10, 1962 from Los Angeles Department of Water & Power with attachment dated August 8, 1962, Memorandum - Pressure Suppression - Nuclear Plant for the City of Los Angeles.
7. Letter dated August 16, 1962 from Los Angeles Department of Water & Power transmitting report, Safety and Site Considerations of a Large Closed Cycle Water Reactor Power Plant for the City of Los Angeles, dated August 8, 1962.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON 25, D.C.

November 14, 1962

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

Subject: REPORT ON CITY OF LOS ANGELES DEPARTMENT OF WATER AND POWER BOILING WATER REACTOR PROPOSAL

Dear Dr. Seaborg:

At a special meeting in Washington, D. C., on November 9-10, 1962, the Advisory Committee on Reactor Safeguards again reviewed the 1600 MW(t) boiling water reactor which may be proposed for construction and operation by the City of Los Angeles Department of Water and Power at a site designated as the "Western Site". The Committee had the benefit of an additional subcommittee meeting on November 2, 1962 with representatives of the City of Los Angeles Department of Water and Power, the General Electric Company, and the AEC Staff. The Committee also had the preliminary documentation referenced below.

The Committee has had several previous reviews of this large boiling water reactor with proposed engineered safeguards for the "Western Site". In its October 12th letter to the Commission the Committee expressed the following opinion:

> "In view of the stringent requirements imposed by the site, it is the Committee's opinion that the containment as proposed is not adequate in some respects for this reactor at this site. The Committee also believes that holdup of routine gaseous releases will be necessary during unfavorable meteorological conditions."

At subsequent meetings the General Electric Company representatives focused their attention on resolving the inadequacies the Committee believes existed in the proposed conceptual engineered safeguards. In particular, attention was directed to these important items: maximum fission product release assumptions; ground level emission values for the postulated worst accident; leakage characteristics of the containment and isolation valves; reliability of valves; probability of primary piping and turbine failures; provisions for piping tunnel ventilation and filtration; and provisions for turbine housing ventilation and filtration.

After evaluation and review of the General Electric Company design concepts and further discussion with ACRS, and the AEC Staff, the City of Los Angeles and the General Electric Company representatives proposed the following engineered safeguards:

1. A vapor suppression system which includes separation of primary and secondary containment.

2. A secondary containment building to withstand 5 psi gauge and having a leakage rate of 1/2% per day or less.

3. A method for rapid detection of fission product release from fuel element failures.

4. Steam line tunnel integral with the secondary containment.

5. Double isolation valves of proven type at least one to be a turbine stop valve protected by steam strainers.

6. Holdup or detention capability for the anticipated noble gas releases to insure that no significant environmental exposures result.

7. A turbine housing provided with controlled ventilation to filter and stack.

The General Electric Company did not evaluate the consequences of a simultaneous release of pressure and fission products, an accident which they believe to be incredible. They propose to substantiate this at the construction permit phase.

The Committee believes that the above additional engineered safeguards should be incorporated into the containment and confinement system for this reactor at this site. In addition, the credibility of the simultaneous release accident should be evaluated before construction.

The Committee wishes to emphasize the following consideration with respect to both the pressurized and boiling water reactors proposed for this site. The Committee has seen only preliminary characteristics of the reactors in either case. Due to their high power level and close proximity to densely populated areas, either of these reactors may require improvements in safety design beyond those features incorporated in existing reactors.

The Advisory Committee on Reactor Safeguards believes that when these items are resolved, a boiling water reactor of the general type proposed, with adequately engineered safeguards, can be located at the "Western Site" with reasonable assurance that such a reactor can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
F. A. Gifford, Jr.
Chairman

Reference:
1. General Electric Company Report "Highlights of Meeting with AEC-ACRS Subcommittee, Washington, D. C. - November 2, 1962" dated November 7, 1962.

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

July 15, 1964

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

Subject: REPORT ON CITY OF LOS ANGELES - MALIBU NUCLEAR PLANT - UNIT NO. 1

Dear Dr. Seaborg:

At its fifty-sixth meeting at Brookhaven National Laboratory on July 9-11, 1964, the Advisory Committee on Reactor Safeguards reviewed the proposal of the City of Los Angeles to construct and operate a 1473 MW(t) pressurized water reactor, Malibu Nuclear Plant - Unit No. 1, at Corral Canyon, twenty-nine miles west of Los Angeles. The Committee had the benefit of discussions with representatives of the Department of Water & Power of the City of Los Angeles, Westinghouse Electric Corporation, Stone & Webster Engineering Corporation, the AEC staff, their consultants, and of a Subcommittee meeting on June 18, 1964. The Committee also had the benefit of the documents listed below.

The proximity of large population centers and the probable growth of population in the vicinity of the proposed reactor site require dependence on engineered safeguards to limit the consequences in the unlikely event of a major credible accident. For this reason, safeguard provisions more extensive than those normally employed in nuclear power reactor plants must be provided in lieu of the distance factor to protect the public.

The applicant has proposed as engineered safeguards a novel containment structure intended to prevent any leakage to the environment, and additional features consisting of:

1. A reinforced concrete containment structure.
2. A containment volume spray system, and
3. An emergency borated-water injection system.

The total containment feature of the building is to be achieved by providing two complete steel liners separated by a layer of porous concrete. The space between the liners will be maintained at a sub-atmospheric pressure by continuously pumping back air to the containment volume. An air recirculating and cooling system is required to remove any heat that is generated within the containment volume. Power and water to assure operation of these systems under all conditions must be provided.

Detailed design of the reactor core has not been established yet, but the general features will be similar to those of other nuclear plants proposed for construction by the same nuclear contractor, and expected to be tested in operation prior to completion of the Malibu plant. Nuclear reactivity coefficients are expected to be negative in this reactor. The probability and effects of control rod ejection require further evaluation. The applicant has suggested several possible means of limiting the consequences of such an accident, and the Committee believes that this question can be resolved satisfactorily during the design stage.

Although stainless steel cladding is planned for the first core, it is anticipated that zirconium alloys may be used in future cores. Complete information on the effect of a possible zirconium-water reaction on the course of accidents is not available. Hence, further review will be needed prior to use of zirconium alloy clad cores.

The Committee was informed that the geology of the site was suitable for the proposed construction. It was reported that no active geological faults are present at the site. Grading of the canyon slopes is proposed to ensure that potential landslide motion does not present a hazard to the plant. It is proposed that critical structures be designed for a suitable response spectrum associated with an earthquake which has a maximum acceleration of 0.3 g. occurring when the containment is under the pressure associated with an accident. The resulting stresses will not exceed 80% of the minimum yield value. Components within the building will be designed to withstand 0.3 g. acceleration acting simultaneously in horizontal and vertical plants.

The ability of the plant to withstand the effects of a tsunami following a major earthquake has been discussed with the applicant. There has not been agreement among consultants about the height of water to be expected should a tsunami occur in this area. The Committee is not prepared to resolve the conflicting opinions, and suggests that intensive efforts be made to establish rational and consistent parameters for this phenomenon. The applicant has stated that the containment structure will not be impaired by inundation to a height of fifty feet above mean sea level. The integrity of emergency in-house power supplies should also be assured by location at a suitable height and by using water-proof techniques for the vital power system. The emergency power system should be sized to allow simultaneous operation of the containment building spray system and the recirculation and cooling system. Ability to remove shutdown core heat under conditions of total loss of normal electrical supply should be assured. If these provisions are made, the Committee believes that the plant will be adequately protected.

The applicant has proposed to deny entrance to the containment while the reactor is operating. This mode of operation does not permit frequent surveillance of equipment and prompt detection of incipient defects. Operating experience at other power plants has demonstrated the value of accessibility for inspection. The Committee suggests that the applicant reconsider this question and explore design modifications which will allow entrance without violating the containment integrity.

As the Committee has commented in its earlier letters, the hold-up of routine gaseous and liquid releases may be necessary during unfavorable conditions. In this connection, it will be necessary to conduct additional pre-operational meteorological and oceanographic survey programs.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be suitably dealt with during construction, and that the proposed Malibu Nuclear Plant can be constructed with reasonable assurance that it can be operated at the site without undue risk to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References Attached.

<u>References</u>:

1. Preliminary Hazards Summary Report, Malibu Nuclear Plant, Unit No. 1, Part B, dated November 1963.

2. General Information in Support of Application for Construction Permit and License, Malibu Nuclear Plant, Unit No. 1, Part A, dated November 1963.

3. Second Amendment to Application for Construction Permit and Facility License for Malibu Nuclear Plant - Unit No. 1, dated May 6, 1964.

4. Third Amendment to Application for Construction Permit and Facility License for Malibu Nuclear Plant - Unit No. 1, dated May 20, 1964.

5. Fourth Amendment to Application for Construction Permit and Facility License, dated June 3, 1964.

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

January 25, 1965

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

Subject: REPORT ON CITY OF LOS ANGELES -- MALIBU NUCLEAR PLANT - UNIT NO. 1

Dear Dr. Seaborg:

At its sixtieth meeting, on December 10-12, 1964, the Advisory Committee on Reactor Safeguards again reviewed the proposal of the City of Los Angeles to construct a 1473 MW(t) pressurized water reactor, Malibu Nuclear Plant - Unit No. 1, at Corral Canyon, twenty-nine miles west of Los Angeles. The Committee had the benefit of discussions with representatives and consultants of the Department of Water and Power of the City of Los Angeles, Westinghouse Electric Corporation, Stone & Webster Engineering Corporation, and the AEC Regulatory Staff and its consultants. The Committee also had available references 1 through 5 listed below. In addition, the applicant and his contractors orally provided design information related to the simultaneous effects on the containment vessel of pressurization and both horizontal and vertical components of the postulated maximum earthquake which was subsequently documented in Amendment 8.

In its report of July 15, 1964, the Committee identified several areas of concern which it believed could be resolved during the construction phase. The applicant has since submitted Amendments 5, 6, 7 and 8. These amendments document earlier presentations to the Committee and the AEC Staff. Final reports of the U. S. Geological Survey and the U. S. Coast and Geodetic Survey documenting the earthquake and tsunami criteria for the Malibu site have also been issued; these reports confirm the information and conclusions given to the Committee earlier concerning this site.

Amendments 5, 6, and 7 submitted by the Applicant provided additional information related to items that the Committee identified in its July 15th report. The proposed reactor facility has been relocated on the site. Protection will be provided against landslides, floods, and tsunamis. The capacity of the on-site emergency diesel generator has been increased to accommodate simultaneous use of both air recirculation and containment-spray systems. The reactor designer has also described a system which is intended to prevent an accident caused by injection of non-borated water.

Since the Committee's July 15th report, some of the more important structural design features of the containment have been developed in further detail. In particular, the design now incorporates conservative factors for: imbedment of reinforcement bars; diagonal strengthening bars; anchoring of the exposed inner membrane; and strengthening at the access hatch penetration. The Committee is of the opinion that the measures being taken by the applicant and listed in Amendment 8 constitute an adequate approach for implementing the desires expressed in the Committee's July 15th report.

With the understanding that items identified in its July 15th report will continue to be considered during construction, the Advisory Committee on Reactor Safeguards reiterates its belief that the proposed Malibu Nuclear Plant can be constructed with reasonable assurance that it can be operated at the site without undue risk to the health and safety of the public.

Sincerely yours,

/s/

W. D. Manly
Chairman

References Attached.

References - Malibu

1. Fifth Amendment to Application for Construction Permit and Facility License for Malibu Nuclear Plant - Unit No. 1, dated July 6, 1964.

2. Sixth Amendment to Application for Construction Permit and Facility License for Malibu Nuclear Plant - Unit No. 1, dated August 21, 1964.

3. Seventh Amendment to Application for Construction Permit and Facility License for Malibu Nuclear Plant - Unit No. 1, dated November 13, 1964.

4. U. S. Coast & Geodetic Survey Report entitled "Report on the Seismicity of the Malibu, California Area", dated November 24, 1964.

5. U. S. Geological Survey Report entitled "Engineering Geology Summary of the Proposed Nuclear Power Plant Site, Corral Canyon, Los Angeles County, California", dated December 1964.

6. U. S. Geological Survey Report entitled "Geologic Report on the Proposed Corral Canyon Nuclear Power Plant Site, Los Angeles County, California", dated December 1964.

7. Eighth Amendment to Application for Construction Permit and Facility License for Malibu Nuclear Plant - Unit No. 1, dated January 8, 1965.

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

September 9, 1961

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

Subject: REPORT ON LOW TEMPERATURE PROCESS HEAT REACTOR

Dear Dr. Seaborg:

At its thirty-sixth meeting on September 7-9, 1961, the Advisory Committee on Reactor Safeguards made a preliminary review of four sites proposed for a 30 to 40 MWT indirect cycle water-cooled process heat reactor. The following four sites were considered:

1. Lincoln, New Hampshire - Site proposed by Franconia Paper Corporation

2. Retsof, New York - Site proposed by International Salt Company

3. Ontonagon, Michigan - Site proposed by Huss Ontonagon Pulp and Paper Company

4. Shawano, Wisconsin - Site proposed by Shawano Paper Mills

The Committee had access to the documents listed below.

Representatives of the Division of Reactor Development stated that these sites were among those submitted by industry in response to their invitation for expressions of interest in a cooperative demonstration project with the AEC for a low temperature process heat reactor. The objective of this preliminary site review by ACRS was to aid the Division of Reactor Development in establishing, before invitations are issued for definite proposals, that suitable sites would be proposed. It is understood that the proposed reactor plant will be a demonstration unit, not an experiment, and will be constructed and operated in a manner similar to those already found to be acceptable.

The ACRS concludes that the Lincoln and Retsof sites named are suitable for the construction and operation of a reactor of the general type and power level proposed. The information available to the ACRS at this time indicates that a more comprehensive study would be required before recommendations can be formulated regarding the last two sites.

Sincerely yours,

/s/ T. J. THOMPSON

T. J. Thompson
Chairman

References:

1. Franconia Paper Corporation, Expression of Interest, dated July 10, 1961, w/attachments.
2. Huss Ontonagon Pulp and Paper Company, Letter to AEC, dated July 11, 1961, w/enclosures.
3. AEC letter to Huss Ontonagon Pulp and Paper Company, dated July 14, 1961.
4. Huss Ontonagon Pulp and Paper Company letter to AEC, dated July 17, 1961.
5. International Salt Company, Inc., Expression of Interest, dated July 13, 1961, w/enclosures.
6. International Salt Company, Retsof, New York - Map of Area Surrounding Proposed Site, received August 11, 1961.
7. Shawano Paper Mills, Expression of Interest - Site Data, dated June 16, 1961, w/attachments.
8. Letter - USAEC, Chicago Operations Office, dated May 5, 1961, Invitation for an Expression of Interest, w/enclosures.
9. Memorandum from F. K. Pittman, Division of Reactor Development to R. Lowenstein, Division of Licensing and Regulation, dated July 31, 1961, "Request for Preliminary Survey of Sites Proposed by Certain Firms for the Low Temperature Process Heat Reactor Project."

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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 MAINE YANKEE ATOMIC POWER STATION

Dear Dr. Seaborg:

At its ninety-ninth meeting, on July 11-13, 1968, the Advisory Committee on Reactor Safeguards completed its review of the application by the Maine Yankee Atomic Power Company for authorization to construct the Maine Yankee Atomic Power Station. The project was previously considered at ACRS Subcommittee meetings held on July 3, 1968 in Washington, D. C., and on July 8, 1968 at the plant site. During its review, the Committee had the benefit of discussions with representatives of the Maine Yankee Atomic Power Company, Combustion Engineering Corporation, Stone and Webster Company, the AEC Regulatory Staff, and the consultants. The Committee also had the benefit of the documents listed below.

The Maine Yankee Atomic Power Station will be located on Bailey Point on the west bank of the Back River in the Town of Wiscasset, Lincoln County, Maine. The area within five miles of the plant is largely forest and farmland; the closest community is the town center of Wiscasset, about four miles northeast of the plant, with a population of about 1800. The closest city is Lewiston, Maine, population 41,000, about 26 miles northwest of the plant. The site comprises about 740 acres, and an exclusion distance of 2000 feet has been established for the plant.

The reactor is a three-loop pressurized water unit with a maximum power of 2440 MWt. The core power density is similar to that of the previously reviewed Fort Calhoun reactor, while the power level is similar to that of the Surry and Palisades reactors, also previously reviewed. The emergency core cooling system includes a high-pressure subsystem with three pumps, and a low-pressure subsystem with two pumps and three injection tanks. 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.

The containment building is a cylindrical, steel-lined reinforced concrete structure with a hemispherical dome and flat foundation mat. The reinforcing steel in the cylindrical portion of the structure is arranged in vertical and circumferential patterns, with the shear forces from possible seismic event being carried by a combination of reinforcing bar dowel action and aggregate interlock in the concrete. No credit is assumed in the design for resistance to the shear forces by the s teel liner. This containment design appears satisfactory in view of the relatively low seismic loads appropriate for this site.

The Committee believes that the system for supplying off-site electrical power to the engineered safeguards equipment should be modified so that no single failure will prevent power from being available from this source.

The control rod drive power supply design has not been completed at this time. The Regulatory Staff should review the design of this portion of the protection system before its fabrication.

The Committee continues to call attention to matters that warrant careful consideration for all large, water-cooled, power reactors.

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 Maine Yankee 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.

References - Maine Yankee

1. Maine Yankee Atomic Power Company letter, dated September 26, 1967; License Application, Preliminary Safety Analysis Report, Volumes I and II.

2. Maine Yankee Atomic Power Company letter, dated January 15, 1968; Hansen, Holley and Biggs report entitled, "Recommended Provisions for Resistance to Tangential Shear Forces Associated with Earthquake Loading -- Containment Shell of Maine-Yankee Nuclear Power Plant," dated December 29, 1967.

3. Maine Yankee Atomic Power Company letter, dated January 15, 1968; Amendment No. 1 to License Application.

4. Maine Yankee Atomic Power Company letter, dated January 15, 1968; Amendment No. 2 to License Application.

5. Maine Yankee Atomic Power Company letter, dated February 5, 1968; Amendment No. 3 to License Application.

6. Maine Yankee Atomic Power Company letter. dated February 5, 1968; Amendment No. 4 to License Application.

7. Maine Yankee Atomic Power Company letter, dated April 8, 1968; Amendment No. 5 to License Application.

8. Maine Yankee Atomic Power Company letter, dated April 17, 1968, Amendment No. 6 to License Application.

9. Maine Yankee Atomic Power Company letter, dated April 30, 1968; Amendment No. 7 to License Application.

10. Maine Yankee Atomic Power Company letter, dated May 10, 1968; Amendment No. 8 to License Application.

11. Maine Yankee Atomic Power Company letter, dated May 14, 1968; Amendment No. 9 to License Application.

12. Maine Yankee Atomic Power Company letter, dated May 22, 1968; Amendment No. 10 to License Application.

13. Maine Yankee Atomic Power Company letter, dated June 19, 1968; Amendment No. 11 to License Application.

14. Maine Yankee Atomic Power Company letter, dated July 1, 1968; Amendment No. 12 to License Application.

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

January 13, 1972

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

Subject: REPORT ON MAINE YANKEE ATOMIC POWER STATION

Dear Dr. Schlesinger:

At its 141st meeting, on January 6-8, 1972, the Advisory Committee on Reactor Safeguards completed its review of the application by the Maine Yankee Atomic Power Company for authorization to operate the Maine Yankee Atomic Power Station at power levels up to 2,440 MW(t). The project was previously considered by the Committee at its 137th meeting, September 9-11, 1971, and at Subcommittee meetings held on July 27, 1971 in Washington, D. C., on August 27, 1971 at the plant site, and on December 20, 1971 in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives of the Maine Yankee Atomic Power Company, Combustion Engineering Incorporated, Stone and Webster Engineering Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The results of the Committee's previous review of this project, at the construction permit stage, are given in its report of July 19, 1968.

The Maine Yankee Atomic Power Station includes a single pressurized water reactor of 2,440 MW(t) power rating. The plant is located on Bailey Point, on the west bank of the Back River in the Town of Wiscasset, Lincoln County, Maine. Condenser cooling water for the unit is drawn from the tidal waters of the river and is returned to the river on the west side of Foxbird Island. The land area around the plant is largely forest and farm land. The town center of Wiscasset, the nearest community, with a present population of about 2,000 is approximately four miles northeast of the plant. The nearest city is Lewiston, Maine, located about 26 miles northwest of the plant.

The applicant stated that he is supplementing the waste treatment systems of the plant by the addition of charcoal filters for radioactive waste gases. This additional equipment should be operative before plant startup and will be used to keep radioactive gaseous waste releases to a small fraction of the 10 CFR 20 limits. The applicant expects that liquid radioactive waste releases in normal operation will be less than the values defined in the recently proposed Appendix I to 10 CFR 50.

The emergency core cooling systems (ECCS) for the reactor have been evaluated using the recently-approved Combustion Engineering Evaluation Model of the AEC "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors." The Committee believes that the indicated performance is satisfactory for the Maine Yankee Reactor. However, the Committee recommends that as the results of additional research and analytical studies become available they should be used by the applicant for evaluation and possible improvement of the system. The applicant's ECCS performance calculations do not as yet include the case of a cold-leg break between the reactor vessel and a closed isolation valve in one loop. The reactor should not be operated at appreciable power levels with a loop isolated until such calculations have been made by the applicant and the results found satisfactory by the Regulatory Staff.

Several welds between austenitic stainless steel sections of the primary system have been found to contain microfissures. Studies by the applicant and his consultants show that microfissures of the type and density found do not impair the serviceability of the welds. Independent studies by the Regulatory Staff and its consultants bear out these conclusions. The Committee believes that these stainless steel weldments in Maine Yankee are acceptable.

The applicant proposes to use a purging technique to control hydrogen build-up 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 power operation. The Regulatory Staff should review the design criteria for the system for conformance to engineered safety feature standards.

The Committee reiterates its previous comments on 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.

The applicant is improving some details of his analysis of the dynamic response of plant elements to possible seismic forces. The results of these calculations should be reviewed by the Regulatory Staff before operation of the plant at appreciable power levels.

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 Maine Yankee Atomic Power Station can be operated at power levels up to 2,440 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

References

1. Amendment 14 to License Application for Maine Yankee Atomic Power Station, dated August 27, 1970: Final Safety Analysis Report, Volumes I and II

2. Amendments 15 through 34, to License Application for Maine Yankee Atomic Power Station

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

June 7, 1978

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

REPORT ON MAINE YANKEE ATOMIC POWER STATION

Dear Dr. Hendrie:

During its 218th meeting, June 1-2, 1978, the Advisory Committee on Reactor Safeguards completed its review of the application by the Maine Yankee Atomic Power Company for authorization to operate the Maine Yankee Atomic Power Station at power levels up to 2630 MW(t). A subcommittee meeting on this matter was held in Washington, D. C. on May 25, 1978. The Committee had previously reported favorably on operation of the Maine Yankee Atomic Power Station at power levels up to 2440 MW(t) in its report of January 13, 1972. During this review, the Committee had the benefit of discussions with representatives of the Maine Yankee Atomic Power Company, Yankee Atomic Electric Company, Combustion Engineering Incorporated, and the Nuclear Regulatory Commission Staff. The Committee also had the benefit of the documents listed.

In the NRC Staff review of the request to increase power, analyses of accidents and transients, physics tests, fuel performance and site meteorology were carried out. Modifications to the Technical Specifications were also considered. In addition, the NRC Staff reviewed the operating history of the plant. In evaluating the proposed power increase in each of these areas, the NRC Staff used current NRC criteria. The NRC Staff has concluded that operation at the proposed power level in accordance with the proposed Technical Specifications is acceptable. The ACRS concurs.

The Advisory Committee on Reactor Safeguards believes that there is reasonable assurance that the Maine Yankee Atomic Power Station can be operated at power levels up to 2630 MW(t), without undue risk to the health and safety of the public.

Sincerely,

Stephen Lawroski

Stephen Lawroski
Chairman

REFERENCES

1. Letter from W. P. Johnson, Maine Yankee Atomic Power Company to NRC, Office of Nuclear Reactor Regulation, concerning a proposed license amendment, on power level increase to 2630 MW(t), dated August 1, 1977.

2. Letter from W. P. Johnson, Maine Yankee Atomic Power Company to the Office of Nuclear Reactor Regulation, modifying the power level increase in two steps, dated December 9, 1977.

3. Safety Evaluation by the Office of Nuclear Reactor Regulation Concerning Power Level Increase of Facility Operating License No. DPR-36, Maine Yankee Atomic Power Company, Maine Yankee Atomic Power Station, Docket No. 50-309, dated January 17, 1978.

4. Letter from D. W. Edwards, Maine Yankee Atomic Power Company, to the Office of Nuclear Reactor Regulation, concerning additional information regarding Maine Yankee power level increase, dated March 1, 1978.

5. Letter from R. H. Groce, Maine Yankee Atomic Power Company, to the Office of Nuclear Reactor Regulation concerning information for the preparation of the SER, dated April 5, 1978.

6. Letter from R. H. Groce, Maine Yankee Atomic Power Company, to the Office of Nuclear Reactor Regulation, concerning additional information on power level increase, dated April 10, 1978.

7. Supplement No. 1 to the Safety Evaluation by the Office of Nuclear Reactor Regulation, concerning Power Level Increase of Facility Operating License No. DPR-36 Maine Yankee Atomic Power Company, Maine Yankee Atomic Power Station, Docket No. 50-309, dated April 11, 1978.

8. Letter from W. P. Johnson, Maine Yankee Atomic Power Company, to the Office of Nuclear Reactor Regulation, concerning Technical Specification changes for power level increase, dated April 28, 1978.

9. Memorandum from Edson Case, Chairman, Regulatory Requirements Review Committee to L. V. Gossick, Executive Director for Operations, dated May 12, 1978, concerning an interim approval of Draft Regulatory Guide, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," dated February 3, 1978, and "Atmospheric Dispersion Model for Accident Evaluations," dated April 18, 1978.

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

October 22, 1976

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON MARBLE HILL NUCLEAR GENERATING STATION, UNITS 1 & 2

Dear Mr. Rowden:

At its 198th meeting, October 14-16, 1976, the Advisory Committee on Reactor Safeguards completed its review of the application of the Public Service Company of Indiana (PSI), the Northern Indiana Public Service Company, the East Kentucky Power Cooperative, and the Wabash Valley Power Association (Applicants) for a permit to construct Marble Hill Nuclear Generating Station, Units 1 and 2. PSI will be responsible for the design, construction, and operation of the Station. The site was visited on October 1, 1976, and a Subcommittee meeting was held in Madison, Indiana, on the same day. The Marble Hill Nuclear Generating Station is a replication of the Byron Nuclear Generating Station on which the Committee reported on May 13, 1975. During its review of the Marble Hill Station application, the Committee had the benefit of discussions with the Nuclear Regulatory Commission (NRC) Staff, representatives of the Applicants, Westinghouse Electric Corporation, and Sargent & Lundy Engineers, as well as comments from members of the public. The Committee also had the benefit of the documents listed.

The Marble Hill Station will be located in Jefferson County, Indiana, on the Ohio River, 11 miles southwest of Madison, Indiana, and 31 miles northeast of Louisville, Kentucky. The minimum exclusion distance is 670 meters; the low population zone radius is 3,218 meters. The Applicants have identified the nearest population center of 25,000 or more persons as Jeffersonville-New Albany, Indiana, which had 1970 populations of 20,000 and 38,000, respectively, and which are located about 26 miles from the site. The total population within 50 miles of the facility, estimated at 1,243,000 in 1970, is projected to grow to 1,880,000 by the year 2020.

The Applicants and the NRC Staff have agreed that horizontal ground accelerations of 0.20g and 0.08g at foundation level are appropriate design values for the safe shutdown earthquake and the operating basis earthquake, respectively.

The Committee considered possible site hazards such as the presence of the Jefferson Proving Grounds and the Indiana Army Ammunition Plant, each about 15 miles away. It agrees that the probability of serious damage from these potential hazards is acceptably low.

The ultimate heat sink will consist of two mechanical draft cooling towers and the makeup system for each tower. The towers will be designed as seismic Category I structures. The cooling tower basin will contain a supply of water adequate to dissipate for 30 days the heat loads resulting from a design basis loss-of-coolant accident (LOCA) in one unit and a shutdown of the other.

Each unit of the Station will utilize the RESAR-3 Consolidated Version, four-loop, pressurized water reactor designed for a thermal output of 3,411 MW. Each unit will employ a steel-lined, prestressed-concrete containment structure with a net free volume of 2,930,000 cubic feet. The structure is designed for an internal pressure of 50 psig and a temperature of 272^{0}F.

The turbines for the Marble Hill Station, like those for the Byron Station, are oriented with their longitudinal axes tangential to the containment building. The Committee has indicated previously its strong preference for a radial orientation as a means of reducing significantly the probability that a missile, resulting from failure of the rotating elements of the turbine, would damage a safety-related portion of the plant. The NRC Staff has accepted the tangential orientation as a necessary consequence of the replication concept, but it has indicated that additional measures will be required to reduce the probability of damage from turbine missiles. The Committee recognizes the advantages and disadvantages of the replication concept and notes that the proposed turbine orientation may be an example of the latter. The Committee is also aware that a change in turbine orientation is only one means of reducing the probability of missile damage. For these reasons the Committee accepts the proposed turbine orientation, but it believes that the probability of damage from turbine missiles should be reduced to an appropriate level, preferably by physical means (e.g., missile shields) rather than by relying solely on means intended to reduce the probability of turbine overspeed. This matter should be resolved in a manner satisfactory to the NRC Staff and the ACRS.

The RESAR-3 Consolidated Version design utilizes the Westinghouse 17 X 17 fuel array. It is expected that operating experience will have been gained from at least six reactors before the Marble Hill Station comes on line.

The NRC Staff has identified only one major technical item for which additional information is needed prior to proceeding to a public safety hearing. This item requires a new LOCA analysis taking into account recent temperature measurements in the upper head of an operating reactor of this type. The Committee wishes to be kept informed of the resolution of this matter, including the peaking factors which result from the analysis.

The Committee recommended in its May 13, 1975 report on the Byron Station, and elsewhere, that significantly improved emergency core cooling system (ECCS) capability be provided for reactors for which construction permit applications were filed after January 7, 1972. If studies establish that significant ECCS improvements can be made, consideration should be given to incorporating them into the Marble Hill Station.

The Committee recommends that the NRC Staff and the Applicants further review the design features that are intended to prevent the occurrence of 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 Committee believes that the Applicants and the NRC Staff should further review the Marble Hill Station for design features that could significantly reduce the possibility and consequences of sabotage, and that such features should be incorporated into the plant design where practical.

The Committee recommends that the matter of anticipated transients without scram be resolved expeditiously and that the plant design maintain adequate flexibility to accommodate any changes which may be required by the program of implementation. The Committee wishes to be kept informed.

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 Marble Hill 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, 10, 11; IIA-1, 4, 5, 6, 7, 8; IIB-2; and IIC-1, 2, 3, 4, 5, 6, 7.

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

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

References:

1. Preliminary Safety Analysis Report (PSAR) for the Marble Hill Nuclear Generating Station, Units 1 and 2, Volumes 1-8 and Amendments 1-12.
2. Report to the Advisory Committee on Reactor Safeguards by the Office of Nuclear Reactor Regulation (NRR) of the Nuclear Regulatory Commission, NUREG-0115, dated September 1976.
3. Letter, Public Service Company of Indiana (PSI) to NRR, concerning request for specific exemption to perform site activities, dated April 21, 1976.
4. Letter, PSI to NRR, concerning policy of replication of base plant, dated May 11, 1976.
5. Written statement received from Shirley Clark on October 1, 1976, representing Save the Valley, Incorporated.
6. Written statement received from Harold Cassidy, representing Save the Valley, Incorporated, dated October 1, 1976.
7. A White Paper II, prepared by Harold Cassidy, representing Save the Valley, Incorporated, dated August 3, 1976.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C.

January 13, 1958

Honorable Lewis L. Strauss
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: THE MIT REACTOR - DOCKET NO. 50-20

Dear Mr. Strauss:

The Massachusetts Institute of Technology has applied for license to operate its nuclear reactor located in Cambridge, Massachusetts, construction of which is now nearing completion. This letter is in reply to a request by the Atomic Energy Commission for the advice of the Advisory Committee on Reactor Safeguards with respect to the safety of the proposed operation of this reactor.

The Committee's advice is based upon information contained in the application and amendments thereto. The former Advisory Committee on Reactor Safeguards reviewed the proposed design of the reactor prior to the issuance of a construction permit and submitted a report, dated March 5, 1956, on this matter to the General Manager.

This is a research reactor designed for one megawatt (thermal) utilizing enriched alloy plate-type fuel elements and D_2O cooling and moderation. It incorporates many of the design principles of the CP-5 and MTR reactors. There are no novel features requiring demonstration and no important changes in design have been made since the review for a construction permit.

The reactor is located in a densely populated area close to public activities. Therefore, it is essential that effective administrative controls and effective operating and emergency procedures be established and maintained. The applicant has indicated provision for such controls and procedures, which appears to the Committee to be adequate.

While the Committee believes that any serious release of fission products is highly improbable, it is important that containment be maintained because of the location of the facility. The containment proposed is generally adequate. However, there is one point of weakness, namely, complete dependence on the reliability of the automatic valve closure mechanism in the ventilation system. The Committee recommends that provision be made for some effective auxiliary means of closing the inlet and outlet lines of the ventilation system. Such a requirement could be met by provision for manual operation of the present valves in addition to the automatic operation already installed or by some other effective means. However, such a requirement is not considered necessary prior to the commencement of the research program.

In the opinion of this Committee, this reactor can be operated with an acceptable degree of risk to the health and safety of the public.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman
Advisory Committee on
Reactor Safeguards

CC: K. E. Fields, GM
H. L. Price, Div. L&R
ACRS Members - except
Dr. Benedict

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: THE OPERATION OF THE MTR WITH PLUTONIUM^{239} LOADING

Dear Mr. Strauss:

On March 7, 1958, the Advisory Committee on Reactor Safeguards reviewed with MTR personnel and the Hazards Evaluation Branch the proposed special Plutonium core for the MTR. In this review the Committee had the benefit of a report by the Hazards Evaluation Branch and report PTR 224 prepared by MTR operating personnel.

The Committee agrees with the conclusion of the MTR personnel that, with the modifications proposed in the control system, the reactor kinetics for the Plutonium loading is not significantly different from that of the current Uranium ^{235}.

It is understood that the Plutonium fuel elements will be subject to the same inspection procedure and specifications as the present Uranium ^{235} element. It is also understood that extensive reactor physics measurements at low power will be carried out in advance of full power operation.

The use of Plutonium in the core does not add significantly to radiological problems associated with the fission product burden at the end of the normal operating cycle.

The Committee therefore believes that there are no reasons involving matters of safety which indicate that the Plutonium loading should not proceed as proposed.

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

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: U^{233} LOADING OF THE MATERIALS TESTING REACTOR (MTR)

Dear Mr. McCone:

The Advisory Committee on Reactor Safeguards reviewed on October 15, 1958, the proposed operation of the Materials Testing Reactor (MTR) with a U^{233} core loading at the request of the Division of Licensing and Regulation. The Committee had access to the report from the Phillips Petroleum Company, PTR-312, and was briefed by the staff of the Division of Licensing and Regulation.

The Committee agrees with the Hazards Evaluation Branch that the proposed operation of the MTR on a U^{233} core will not endanger the health and safety of the public. Also the Committee is of the opinion that such proposed operation will not expose the adjacent reactor operations to an unacceptable hazard.

In the Phillips proposal, it was recognized that the U^{233} loading will reduce the portion of delayed neutrons. To counteract the effect on the reactor kinetics of this decrease, the worth of the control rod and its rate of insertion will be correspondingly reduced. Also the maximum permissible worth of individual experiments will be reduced to a level compatible with the reduced portion of delayed neutrons. In addition, adequate means for the control of environmental hazards are available. The modifications of the MTR due to the U^{233} loading are very similar to those required for the operation of a plutonium core which has already been successfully carried out.

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

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

October 9, 1971

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

Subject: REPORT ON MCGUIRE NUCLEAR STATION UNITS 1 AND 2

Dear Dr. Schlesinger:

At its 137th meeting, September 9-11, 1971, and its 138th meeting, October 7-9, 1971, the Advisory Committee on Reactor Safeguards reviewed the application by the Duke Power Company for a permit to construct the dual-unit McGuire Nuclear Station. The project was considered at Subcommittee meetings at the plant site on June 4, 1971, in Washington, D. C., on July 22-23, 1971, and in Chicago on August 9, 1971 and September 25, 1971. During its review the Committee had the benefit of discussions with representatives and consultants of the applicant, the Westinghouse Electric Corporation and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The station will be located on the southern shore of Lake Norman, approximately eleven miles northwest of the nearest boundary of Charlotte, North Carolina, which is the closest population center. Suitable provisions have been proposed for protection against a major flood. A seismic Class I standby nuclear service water pond will be provided to serve as an ultimate heat sink in the event the normal supply of cooling water from Lake Norman should be unavailable.

Each unit employs an ice-condenser system enclosed within a free-standing steel containment vessel surrounded by a reinforced-concrete shield building. The units will utilize four-loop pressurized water reactor nuclear steam supply systems having an initial power level of 3411 MWt and a design similar to the previously reviewed Sequoyah and Trojan units. A slightly higher coolant inlet temperature and coolant flow rate are proposed for the McGuire units. The Committee believes that appropriate additional evidence regarding the core thermal parameters should be obtained from reactors of similar design prior to implementation of the proposed increase in core thermal performance of the McGuire plant.

The applicant has stated that forces associated with longitudinal and circumferential pipe ruptures will be considered in the design of the supports and restraints for the primary and secondary coolant systems in order to assure the continued integrity of the containment, or other vital components, or engineered safety systems.

The relatively high power density of the McGuire core and the lower containment pressure associated with the ice-condenser system will be taken into account to provide an appropriately conservative design for the emergency core cooling system (ECCS). In order to satisfy the AEC "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors," the applicant has found it necessary to base his loss-of-coolant accident analyses for the cold-leg break on a maximum permissible linear power of 14.9 kw per foot at full power. The Committee believes that, since there is no increase in the average power density, this restriction on maximum permissible linear power should permit a satisfactory mode of operation. However, if this mode of operation is to be employed, the Committee believes that the applicant should be prepared to use appropriate in-core monitoring systems.

The applicant is conducting an experimental and analytical program intended to examine several conservative assumptions in the Interim Criteria to provide an early basis for choice of possible improvements in the ECCS design. The Committee recommends that the results of this program and any changes in ECCS design proposed by the applicant be reviewed prior to installation. This matter should be resolved in a manner satisfactory to the AEC Regulatory Staff and the ACRS.

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 in a manner satisfactory to the Regulatory Staff and the ACRS during construction.

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 McGuire station.

The ACRS believes that the above items can be resolved during construction and that, if due consideration is given to these items, the McGuire Nuclear 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,

Spencer H. Bush

Spencer H. Bush
Chairman

References

1. Duke Power Company Letter dated September 18, 1970; License Application; Preliminary Safety Analysis Report (PSAR), Volumes I, II and III

2. Westinghouse Electric Corporation Letter dated September 11, 1971; Reference Safety Analysis Report (RESAR), Volumes I and II, Unclassified, Volumes I and II, Proprietary

3. Amendments 1, 2, and 4 through 10 to the PSAR and RESAR

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

April 12, 1978

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

SUBJECT: REPORT ON MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

Dear Dr. Hendrie:

During its 216th meeting, April 6 and 7, 1978, the Advisory Committee on Reactor Safeguards completed its review of the application of the Duke Power Company (the Applicant) for a permit to operate the McGuire Nuclear Station, Units 1 and 2. The application was reviewed at a Subcommittee meeting in Charlotte, North Carolina on March 29-30, 1978, and tours of the facility were made on May 17, 1976* and March 28, 1978. During its review, the Committee had the benefit of discussions with representatives and consultants of the Nuclear Regulatory Commission (NRC) Staff, Westinghouse Electric Corporation, and the Applicant. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for the McGuire Nuclear Station on October 9, 1971.

The McGuire Nuclear Station is located on the southern shore of Lake Norman in Mecklenburg County, North Carolina, about 17 miles north-northwest of Charlotte, North Carolina. Each unit will utilize a four loop pressurized water reactor nuclear steam supply system having an initial power level of 3411 MWt. Each unit employs 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 design is similar to that used for the previously reviewed Donald C. Cook Nuclear Plant, but the Applicant has modified the ice condenser system as a result of operating experience gained in the Donald C. Cook Nuclear Plant. The Applicant and the NRC Staff should make plans to monitor the performance of the ice condenser containments at the McGuire Nuclear Station (Generic Item IIA-1 in ACRS Report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977).

Date corrected from that which was
originally sent

The McGuire Nuclear Station 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 which use them. Experience to date has been satisfactory. The Committee wishes to be kept informed of the results of the various 17x17 fuel assembly inspections and test programs now underway (Generic Item IIB-2 in ACRS Report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977).

The Emergency Core Cooling Systems (ECCS) for the McGuire Nuclear Station 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 concurs in the Staff's conclusions. The application of the approved model to McGuire should be made in accordance with the Staff's requirements.

The NRC Staff has identified a number of outstanding issues that will require resolution before the issuance of an operating license. 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. 6," dated November 15, 1977. Those problems relevant to the McGuire Nuclear Station should be dealt with by the NRC Staff and the Applicant as solutions are found. The relevant items are: II-2, 3, 4, 5b, 6, 7; IIA-2, 3, 4; IIC-1, 3a, 3b, 5, 6; and IID-2.

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, there is reasonable assurance that the McGuire Nuclear Station, Units 1 and 2 can be operated at power levels up to 3411 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

REFERENCES:

1. Duke Power Company, "McGuire Nuclear Station, Units 1 and 2 Final Safety Analysis Report," with Amendments 1-48.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of McGuire Nuclear Station, Units 1 and 2," USNRC Report NUREG-0422, March, 1978.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report on Westinghouse Electric Company ECCS Evaluation Model for Plants Equipped with Upper Head Injection," April, 1978.
4. Letter from J. L. Riley, Carolina Environmental Study Group (CESG), to the Advisory Committee on Reactor Safeguards, concerning reactor pressure vessel head bolts, dated March 6, 1977.
5. Letter from W. L. Porter, Duke Power Company, to J. L. Riley, CESG, concerning reactor pressure vessel head bolt test data, dated October 4, 1972.

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

June 18, 1970

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

Subject: REPORT ON MIDLAND PLANT UNITS 1 & 2

Dear Dr. Seaborg:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The Committee recommends that the applicant accelerate the study of means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant stated that the engineering design would maintain flexibility with regard to relief capacity of the primary system and to a diverse means of reducing reactivity. This matter should be resolved in a manner satisfactory to the Regulatory Staff during construction. The Committee wishes to be kept informed.

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

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

Sincerely yours,

/s/
Joseph M. Hendrie
Chairman

<u>References</u>

1) Amendments 1 - 12 to License Application

September 23, 1970

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

Subject: SUPPLEMENTAL REPORT ON MIDLAND PLANT UNITS 1 AND 2

Dear Dr. Seaborg:

At its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards completed its review of amendments to the application by the Consumers Power Company to construct the Midland Plant Units 1 and 2. This project was the subject of a report to you dated June 18, 1970. The review was reopened in consideration of additional submittals by the applicant proposing an increase in the design pressure of the containment structure and the addition of a system of reboilers for the generation of steam to be exported to the Dow Chemical Company. These changes were considered at a Subcommittee meeting held in Washington, D. C. on September 14, 1970. The Committee had the benefit of discussion with representatives and consultants of the Consumers Power Company, Babcock and Wilcox Company, Bechtel Corporation, Dow Chemical Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The applicant has revised downward his estimate of the free volume and internal surface area of the containment structure and has revised upward to 60 psig the calculated peak containment pressure reached in the unlikely event of a loss of coolant accident. The containment design pressure has been raised to 67 psig to provide a suitable margin above the peak accident pressure, and an increased number of prestressing tendons will be provided in the containment structure to accommodate the increased pressure. No changes in the structural design criteria are proposed. The Committee believes these changes are satisfactory.

In the earlier design the export steam was taken from the secondary side of the main steam generators and might contain traces of radioactive leakage from the primary system. The applicant now proposes to use this steam in a system of shell and tube reboilers to generate tertiary steam for export to the Dow Chemical Company. Secondary steam condensate from the reboilers is returned to the turbine condenser hot well while feed water for the tertiary side of the reboilers is supplied by condensate from the tertiary steam which is supplemented as required by demineralized water from Lake Huron. Blowdown from the reboilers is normally routed to the Dow waste treatment system for disposal to the river but may be sent to the radwaste system of the nuclear plant if secondary to tertiary leakage is detected.

The applicant proposes to install monitoring and analytical facilities to determine the levels of radioactivity in the export steam as described in the June 18, 1970, letter; these include an on-line analyzer for gamma activity and sensitive low level beta counting equipment for analysis of samples of the condensed steam. The applicant expects that the tertiary steam delivered to Dow will contain no more radioactivity than the treated make-up water from Lake Huron. Recycling tertiary steam condensate may result in some slight concentration of naturally occurring radioactivity in the reboiler system but is not expected to effect the validity of the comparison between steam and make-up water radioactivity as a sensitive indication of leakage in the reboilers. If detectable leakage occurs, corrective action will be taken in the plant or delivery of export steam will be terminated.

The applicant agrees to demonstrate the analytical equipment and procedures in development programs to be carried forward during construction of the Midland Plant.

The Committee believes that the proposed system of reboilers will provide substantial additional assurance that leakage of primary system radioactivity into the export steam can be maintained at an extremely low and insignificant level and that the export steam can be maintained essentially at natural background levels. The detailed procedures for monitoring and control of the reboiler system should be developed during construction in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The Committee believes that the above items can be resolved during construction and if due consideration is given to these items and to the items referred to in its June 18, 1970 report, the nuclear units proposed for the Midland Plant can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Joseph M. Hendrie

Joseph M. Hendrie
Chairman

<u>References</u>

1) Amendments 14-18 to the License Application

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

November 18, 1976

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: SUPPLEMENTAL REPORT ON MIDLAND PLANT UNITS 1 AND 2

Dear Mr. Rowden:

In response to a request from Chairman D. M. Head of the Midland Atomic Safety and Licensing Board, the Advisory Committee on Reactor Safeguards has reviewed the record pertaining to the Midland Plant Units 1 and 2 as reported in its letter of June 18, 1970. The items listed below are those items referred to in its paragraph on "other problems related to large water reactors" which had been previously "identified by the Regulatory Staff and the ACRS," and which the Committee considered applicable to the Midland Plant. Following each item, the Committee has included an amplifying statement based on ACRS reports on other similar commercial nuclear reactor power plants which had been reviewed during the months prior to the Committee's review of the Midland Plant. Copies of the referenced ACRS reports are attached.

1. Separation of protection and control instrumentation - The Applicant proposed using signals from protection instruments for control purposes. The Committee believed that control and protection instrumentation should be separated to the fullest extent practicable, and recommended that the Applicant explore further the possibility of making safety instrumentation more nearly independent of control functions. (Three Mile Island, 1/17/68).

2. Vibration and loose parts monitoring - The Committee recommended that the Applicant study possible means of in-service monitoring for vibration or 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 found practical and appropriate. (Palisades, 1/27/70).

3. Potential for axial xenon oscillations - The Applicant was continuing studies on the possible use of part-length rods for stabilizing potential xenon oscillations. Solid poison shims were to be added to the fuel elements if necessary to make the moderator temperature coefficient more negative at the beginning of core life. (Three Mile Island, 1/17/68).

4. The behavior of core-barrel check valves in normal operation - The Applicant had 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 had indicated that vibrations would not unseat these valves during normal operation. The Committee desired that this point be verified experimentally. (Three Mile Island, 1/17/68).

5. The potential consequences of fuel handling accidents - The Committee believed that further study was required with regard to potential releases of radioactivity in the unlikely event of gross damage to an irradiated subassembly during fuel handling and the possible need for a charcoal filtration system in the fuel handling building. The Committee recommended that this matter be resolved in a manner satisfactory to the Regulatory Staff. (Hutchinson Island, 3/12/70).

6. The effects of blowdown forces on core internals - The Committee recommended that the Regulatory Staff review the effects of blowdown forces on core internals and the development of appropriate load combinations and deformation limits. (Three Mile Island, 1/17/68).

7. Assurance that LOCA-related fuel rod failures will not interfere with ECCS function - The Committee desired 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. (Three Mile Island, 1/17/68).

8. The effect on pressure vessel integrity of ECCS induced thermal shock - The Committee recommended that the Regulatory Staff review analyses of possible effects, upon pressure-vessel integrity, arising from thermal shock induced by ECCS operation. (Oconee, 7/11/67).

9. Environmental qualification of vital equipment in containment - The Committee recommended that attention be given to the long-term ability of vital components, such as electrical equipment and cables, to withstand the environment of the containment in the unlikely event of a loss-of-coolant accident. (Palisades, 1/27/70).

10. Instrumentation to follow the course of an accident - This item related to the 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. (Hutchinson Island, 3/12/70).

11. Improved quality assurance and in-service inspection of primary system - The Committee continued to emphasize the importance of quality assurance in fabrication of the primary system as well as inspection during service life, and recommended that the Applicant implement those improvements in quality practical with current technology. (Oconee, 7/11/67).

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Attachments:

1. Request from Chairman D. M. Head, AS&LB, dated 10/14/76
[*] 2. Report on Midland Plant Units 1 & 2, dated 6/18/70 — [*] See pages 943-948, Vol. II
3. Report on Hutchinson Island Unit No. 1, dated 3/12/70 — See pages 790-793, Vol. II
4. Report on Palisades Plant, dated 1/27/70 — See pages 1188-1191, Vol. II
5. Report on Three Mile Island Nuclear Station Unit 1, dated 1/17/68 — See pages 1637-1639, Vol. III
6. Report on Oconee Nuclear Station, Units 1, 2, and 3, dated 7/11/67 — See pages 1154-1156, Vol. II

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

RECEIVED

October 14, 1976

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

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

Dear Dr. Moeller:

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

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

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

Very truly yours,

Daniel M. Head

Daniel M. Head, Chairman
Atomic Safety and Licensing Board

Enclosure: ACRS report

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

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

March 16, 1977

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: ADDITIONAL REQUEST FOR INFORMATION FROM THE MIDLAND AS&LB

Dear Mr. Rowden:

The Committee has received an additional request from the Atomic Safety and Licensing Board in the Midland case for further elaboration and "treatment" of matters mentioned in the Committee's Supplemental Report to you of November 18, 1976 and attachments thereto. That report, you may recall, was written in response to a previous request which followed directly from the decision in Aeschliman vs. NRC. A copy of the most recent AS&LB request, dated January 28, 1977, is attached.

Although the Committee is willing to provide reasonable and necessary clarification of its recommendations and opinions, we believe that the Board in this case has misinterpreted the Aeschliman decision and has embarked on a course which, if pursued, could involve the Committee in an unnecessary and potentially unending series of requests for clarification and elaboration of its reports, in connection with not only the Midland proceeding, but other proceedings as well. The Board's "three areas of comment" are addressed below:

I.

The Board notes two specific paragraphs of interest to the Midland proceeding in a set of ACRS meeting minutes (106th ACRS meeting held February 6-8, 1969) during which the Midland project was discussed, and the Board requests "further comment under the rules set forth in the Aeschliman case" regarding these two paragraphs "as well as any other 'matters of concern' (including any matters mentioned in furnished or unfurnished minutes)" and requests that these matters be treated fully by the Committee in accordance with the following excerpt from Aeschliman vs. NRC:

> "At a minimum, the ACRS report should have provided a short explanation, understandable to a layman, of the additional matters of concern to the Committee, and a cross-reference to the previous reports in which those problems, and the measures proposed to solve them, were developed in more detail."

In the opinion of the Committee, the Board has incorrectly concluded that all topics discussed during an ACRS review, and recorded in the meeting minutes, are "matters of concern" to the Committee in the context of the Aeschliman decision. "Items of concern" to the ACRS at the completion of its review are identified in the Committee's report and have been explained in the Committee's Supplemental Report of November 18, 1976, in language "understandable to the layman" as required by the Aeschliman decision. Many other items of interest are documented and discussed during the course of an ACRS review and are not identified as matters of concern in the ACRS report. Some of these items are considered satisfactory or are adequately resolved by amendment of the application or other means during the review process. Some represent points of general information, some represent matters that the Committee explores on a generic basis.

It should be noted that the Aeschliman decision did not address the content of ACRS meeting minutes or other information available to or considered by the Committee but was limited (see Attachment 2) to those matters identified in ACRS reports as items of concern. To require that the ACRS address in its report every item discussed or considered during the course of a review is impractical and unnecessary.

For example, the suitability of the Midland Plant for the proposed Midland site was discussed at length during six Subcommittee meetings held on January 22 and February 4, 1969, and March 24, April 24, June 10, and September 14, 1970, and at five full Committee meetings held on February 6, 1969, and April 9, May 8, June 11-13, and September 17-19, 1970; appropriate safety features were included in the design for this reactor at this site. The minutes of these meetings have been in the public domain since 1974.

II.

This section of the Board's request deals with the substance of the Committee's Supplemental Report of November 18, 1976 and requests that the Committee further clarify one of its recommendations, specifically, that the Committee specify the "danger" that is of concern if instrumentation and control are not separated; further describe the type of separation required (e.g., physical or other); and specify a standard for conformance.

The Board further notes that this illustration is only an example of an area where a problem may exist and further elaboration of other matters may also be required.

The Committee appreciates the Board's desire and interest in understanding the issues identified by the Committee but does not agree with the method being used to develop this understanding. The Committee's Supplemental Report dated November 18, 1976 did provide a brief description of the items considered to have been problems by the Committee and specific cross references to other applicable cases, as required by the Court in Aeschliman vs. NRC.

The desire for additional clarification by the Board with respect to specific questions of this nature is best served by:

* Examination of the record related to the Midland review and the review of other cases specifically cross-referenced by the Committee.

* Discussion with the NRC Staff who participate in the Committee's review process, are thoroughly familiar with the problems and issues involved, and are participants in the hearings.

The example chosen by the Board is itself a case in point. The matter of separation of control and protection instrumentation relates to reducing the probability of failure due to a common cause and is dealt with generically by Section 7.3 of the NRC's Standard Review Plan, which provides guidance to Staff reviewers; the Committee provided a specific reference, in its November 18, 1976 Supplemental Report, to the Three Mile Island Nuclear Station, Unit 1, in response to the Court's order to provide a "cross-reference to the previous reports in which those problems and the measures proposed to solve them were developed in more detail." The July 11, 1973 Safety Evaluation of the then Directorate of Licensing in the matter of Three Mile Island, Unit 1, deals directly with this ACRS concern in Section 7.5, "Separation of Control and Protection Systems" and the Committee's August 14, 1973 report on operation of Three Mile Island, Unit 1, indicates that this matter was no longer of concern for the Three Mile Island case. In the Midland case, the Committee will review the adequacy of the final design as it exists at the time it reviews the Midland Plant for an operating license.

In general, we believe that examination of the implementation of the Committee's advice and of any resulting changes in the application are best left to the NRC Staff which plays a direct role in the hearing, and that any evidence relating to such matters should be sought from them. Indeed, the Court in Aeschliman itself notes, "This is not to say that an ACRS report must contain detailed factual findings of the kind necessary to aid judicial review. Under Commission rules, when ACRS conclusions are controverted, a factual record is compiled anew before the Licensing Board."

The NRC Staff (previously, the AEC Regulatory Staff) has routinely addressed itself to the comments and recommendations in ACRS reports for many years as part of the NRC hearing process. A typical example is to be found in Supplement No. 1 to the Directorate of Licensing's Safety Evaluation for Three Mile Island, Unit 1, dated October 15, 1973. Chapter 4 of that document is addressed entirely to the issues raised in the ACRS report of August 14, 1973.

III.

This section of the Midland Board's most recent request points to perceived "ambiguities" resulting from an examination of several ACRS reports provided as references in the Committee's Supplemental Report of November 18, 1976. The Board notes that those references contain "ambiguities" similar to the ones cited by the Court in Aeschliman and points, by way of example, to the Committee's reference to "other problems" in it's Hutchinson Island report of March 12, 1970. The Board asks that any of the "other problems" which apply to Midland be identified and described as the Court directed.

The Committee's Supplemental Report of November 18, 1976 was provided as ordered by the Court to identify those "other problems" which had been considered applicable to the Midland Plant at the time of the CP review and which were noted generically in the ACRS report of June 18, 1970. Any items not so identified in the Committee's November 18, 1976 report were not considered applicable to Midland during the CP review.

The Committee will be in a position to update this list and address the current status of specific items when it has completed its review for an Operating License for the Midland Plant. This review has not yet been scheduled.

In summary, the Committee believes that the response already provided in its Supplemental Report of November 18, 1976, fully meets the requirements of the Aeschliman Court since:

(1) The Court requested elaboration only of those items referred to in the Committee's original report as "other problems" and no others.

(2) The Committee's Supplemental Report of November 18, 1976, did provide a "short explanation understandable to a layman of the additional matters of concern to the Committee and a cross-reference to the previous reports in which those problems, and the measures proposed to solve them, were developed in more detail" as specifically directed by the Aeschliman decision.

(3) The Committee's Supplemental Report of November 18, 1976, fully identified all additional matters of concern to the Committee during its CP review of the Midland Project.

The ACRS does not feel that any further clarification of its reports on Midland is necessary.

Sincerely yours,

M. Bender

M. Bender
Chairman

Attachments:
1. F. J. Coufal, Chairman, AS&LB letter to M. Bender, ACRS, dated January 28, 1977.
2. Excerpt from the decision in Aeschliman vs. NRC.

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

Attachment 1

January 28, 1977

FEB 1 AM 10 25

Myer Bender, Chairman
Advisory Committee on Reactor
  Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555

RE: MIDLAND PLANT UNITS 1 AND 2

Dear Mr. Bender:

The Board has reviewed the reports in evidence in this case by the Advisory Committee on Reactor Safeguards (ACRS) (Staff Exhibits 1, 2 and 3) and has decided to return those responses to the ACRS for further elaboration. These responses were originally submitted as a result of the decision in Aeschliman vs. NRC ___ F.2d ___, (DC Cir. 1976), slip opinion at 21. The Board has received two responses, both dated November 18, 1976, one including a copy of some minutes of an ACRS meeting discussing Midland and the other having no such enclosure. We have three areas of comment.

I.

The minutes mentioned contain references which we believe require further comment under the rules set forth in the Aeschliman case. Two of these are:1/

> "c. Exclusion area and low population zone - the exclusion area extends 1100 meters from the proposed plant and includes a portion of the Dow plant, including 53 Dow employees; the low population zone extends to three miles and includes all of the Dow plant and part of the City of Midland. The site received a-34 index rating when compared to the hypothetical reference site (considering the maximum population in the Dow complex).

1/ Others may exist. We presently focus on these because of their relationship to current suspension hearings.

"g. Other aspects - ... the Committee mentioned but did not explore in any depth: the suitability of B&W reactors for marginal sites, protection required against reactor vessel splits, cavity flooding systems, and the use of process steam in products to be consumed by people."

Neither the ACRS letter dated June 18, 1970, nor the one dated November 18, 1976, furnished to meet the requirements of Aeschliman, mention these matters. We believe that the court, in the words that are set out in footnote 2 below requires that these matters, as well as any other "matters of concern" (including any matters mentioned in furnished or unfurnished minutes) be treated fully by the Committee.

The significance of the rating system referred to in item (c) and the hypothetical reference site is not apparent nor are there explanatory references cited. Furthermore, the Board does not understand what the ACRS means by "the suitability of the B&W reactors for marginal sites" in item "g."

II.

We are concerned with the adequacy of some responses in the November 18, 1976, letter to meet the Aeschliman test. To illustrate we set out the first of the eleven topics in the letter:

"1. Separation of protection and control instrumentation - The Applicant proposed using signals from protection instruments for control purposes. The Committee believed that control and protection instrumentation should be separated to the fullest extent practicable, and recommended that the Applicant explore further the possibility of making safety instrumentation more nearly independent of control functions. (Three Mile Island, 1/17/68).

2/ "At a minimum, the ACRS report should have provided a short explanation understandable to the laymen of the additional matters of concern to the Committee and a cross-reference to previous reports in which those problems and the measures proposed to solve them were developed in more detail."

It is unclear to the Board what this paragraph means. The danger is not specified and it is unclear as to whether the "separation" mentioned refers to a physical separation of components or to the necessity for separate energy sources for signals and controls or to some other separation. No standard is set for the Applicant's (now Licensee's) conformance. The referenced documentation (Three Mile Island, January 17, 1968) says no more. There is in that document a list of references (some marked ACRS Office Copies Only) which may clarify the matter. But no direction is given as to which of these references is relevant to the particular subject.

This illustration is exemplary only and whether the same infirmity exists in other items is a problem we have not had the opportunity to address. We furnish this now so that the Committee is made aware of our concern and so that further elaboration is not delayed.

III.

The letter of the ACRS to Chairman Rowden, November 18, 1976, referred to other ACRS letters. Those letters contain items which have ambiguities similar to those disapproved in Aeschliman. For example, the March 12, 1970 letter on Hutchinson Island stated:

> "Other problems related to large water reactors have been identified by the Regulatory Staff, and the ACRS and cited in previous ACRS Reports" (p. 3).

Those items, we feel, need to be identified if they apply to Midland and if they do, to be described as the Court directed. See footnote 2 hereof.

* * *

We write this under what we perceive to be out duty under the direction given in the Aeschliman case3/ without waiting to fully identify all of the possible areas

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3/ A "sua sponte" request for elaboration.

of concern relative to the November 18, 1976, letter. We do so because we are in the midst of suspension hearings and will need a resolution of this matter as soon as it may reasonably be furnished.

Respectfully submitted,

Frederic J. Coufal

Frederic J. Coufal, Chairman
Atomic Safety and Licensing
Board

(2) The Court concluded in the Aeschliman decision that:

> The ACRS report in this case must be evaluated in light of the congressional purposes. While the reference to "other problems" identified in previous ACRS reports may have been adequate to give the Commission the benefit of ACRS members' technical expertise, it fell short of performing the other equally important task which Congress gave ACRS: informing the *public* of the hazards. At a minimum, the ACRS report should have provided a short explanation, understandable to a layman, of the additional matters of concern to the committee, and a cross-reference to the previous reports in which those problems, and the measures proposed to solve them, were developed in more detail. Otherwise, a concerned citizen would be unable to determine, as Congress intended, what other difficulties might be lurking in the proposed reactor design. Since the ACRS report on its face did not comply with the requirements of the statute, we believe the Licensing Board should have returned it *sua sponte* to ACRS for further elaboration of the cryptic reference to "other problems." [18]
>
> Turning to the propriety of discovery directed to individual ACRS members and ACRS documents, we conclude it was not error to deny these requests. ACRS' unique role as an independent "part of the administrative procedures in chapter 16 of the act," *supra*, is sufficiently analogous to that of an administrative decisionmaker to bring into play the rule that the "mental processes" of such a "collaborative instrumentalit[y] of justice" are not ordinarily subject to probing. *United States v. Morgan*, 313 U.S. 409, 422 (1941). This rule is particularly apropos in light of ACRS's collegial composition such that no individual may speak for the group as a whole. Where an ACRS report on its face omits material

[18] This is not to say that an ACRS report must contain detailed factual findings of the kind necessary to aid judicial review. Under Commission rules, when ACRS conclusions are controverted, a factual record is compiled anew before the Licensing Board. *See* 10 C.F.R., pt. 2, App. A, V(f)(1) (1976).

information, the appropriate course is not discovery but to return it for supplementation. *Cf. Dunlop* v. *Bachowski*, 421 U.S. 560, 574-75 & n. 11 (1975). We merely hold here that neither the Atomic Energy Act nor general principles of administrative law required the Commission to grant Saginaw's discovery requests.[19]

On remand, the ACRS report should be returned to the ACRS for clarification of the ambiguities noted above.

[19] The case as presented calls upon the court to make no decision whether the Federal Advisory Committee Act, 5 U.S.C. App. I § 10(b) (Supp. III, 1973), entitles a party upon proper request to have access to data which were before the ACRS.

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

May 14, 1982

MEMORANDUM FOR: William J. Dircks, Executive Director for Operations

FROM: R. F. Fraley, Executive Director, ACRS

SUBJECT: FOUNDATION PROBLEMS AND RELATED REMEDIAL ACTIONS AT THE MIDLAND PLANT SITE

Consistent with the request of the Office of Nuclear Reactor Regulation for comments, an Ad hoc ACRS Subcommittee has reviewed the foundation problems and related remedial actions at the Midland Plant Units 1 and 2. These issues were discussed during an April 29, 1982 meeting of the Ad hoc Subcommittee and during the 265th full Committee meeting (May 6-8, 1982). As a result of these meetings, the ACRS accepted the Subcommittee's recommendations that:

1. The ACRS Midland Plant Subcommittee review the adequacy of the seismic input criteria and the Site Specific Response Spectrum and its relation to the proposed permanent site dewatering as a means of reducing the probability of soil liquefaction due to an earthquake.

2. Subject to a finding by the Midland Plant Subcommittee regarding the adequacy of the seismic input criteria, the ACRS recognize the adequacy of the NRC Staff's efforts and consider the proposed remedial measures as a matter that can and should be resolved in a manner satisfactory to the NRC Staff.

3. The EDO be informed at this time that the ACRS has found the Staff's approach to be acceptable, subject to the further review mentioned in Item 1 above.

The seismic related issues at Midland are tentatively scheduled to be discussed during the May 20-21, 1982 Midland Plant Subcommittee meeting in Midland, MI. These issues and others related to the application of Consumers Power Company for a license to operate Midland Plant Units 1 and 2 are tentatively scheduled for review by the full ACRS during its 266th meeting (June 3-5, 1982).

cc:
H. Denton, NRR
E. Goodwin, NRR

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

June 8, 1982

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

Dear Dr. Palladino:

SUBJECT: ACRS INTERIM REPORT ON MIDLAND PLANT, UNITS 1 AND 2

During its 266th meeting, June 3-5, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of Consumers Power Company for a license to operate the Midland Plant, Units 1 and 2. This application was also considered at Subcommittee meetings held on April 29, 1982 in Washington, D. C., on May 20-21, 1982 in Midland, Michigan and on June 2, 1982 in Washington, D. C. On May 20, 1982 members of the Subcommittee toured the plant. In the course of these meetings the Committee had the benefit of discussions with representatives and consultants of Consumers Power Company, Babcock and Wilcox Company, Bechtel Corporation, the Nuclear Regulatory Commission Staff, and members of the public. The Committee also had the benefit of the documents listed below.

The ACRS reported on June 18, 1970 regarding the construction permit application for the Midland Plant; on September 23, 1970 regarding several amendments to the application; and on November 18, 1976 regarding applicable generic matters.

The Midland Plant site is located on the south bank of the Tittabawassee River adjacent to the southern city limits of Midland. The main industrial complex of the Dow Chemical Company lies within the city limits directly across the river from the site. There are about 2000 industrial workers within one mile of the site, and the estimated 1980 population was about 51,400 residents within five miles of the site. This makes the Midland site one of the more densely populated sites at distances close to the Plant.

Each of the two Midland units employs a Babcock and Wilcox designed nuclear steam supply system rated at 2468 MWt with a stretch power rating of 2552 MWt. The Midland Plant is unique in that the heat generated will be used not only to produce electricity but also to produce process steam for the Dow Chemical Company plant via a tertiary system.

The Midland Plant has been the subject of several major problems related to quality assurance during plant construction. One of these problems relates to the soil fill under several safety-related structures. The deficiencies relating to soil fill have led to excessive settlement and some cracking of these structures, and have also introduced questions concerning the adequacy of protection against liquefaction of the granular portions of the fill in the event of strong vibratory motion accompanying an earthquake.

The Applicant has proposed and is implementing, under close surveillance by the NRC Staff, remedial measures with regard to the foundation deficiencies. We are generally satisfied with the approach being taken, subject to confirmation of the overall quality assurance program and the seismic design basis. Both of these items are discussed below.

With regard to quality control of design and construction, the report of the NRC Staff's Systematic Assessment of Licensee Performance (SALP) review for the period July 1, 1980 to June 30, 1981 revealed deficiencies in the installation of piping and piping suspension systems, in the pulling of electrical cables, and in the handling of problems relating to soils and foundation. Deficiencies by the Applicant in the handling of soils-related matters have continued to occur, subsequent to issuance of the SALP report. We believe that the NRC Staff is handling the corrective actions for specifically identified quality assurance deficiencies in an appropriate manner.

In view of the overall concern about Midland quality assurance the NRC should arrange for a broader assessment of Midland's design adequacy and construction quality with emphasis on installed electrical, control, and mechanical equipment as well as piping and foundations. We wish to receive a report which discusses design and construction problems, their disposition, and the overall effectiveness of the effort to assure appropriate quality.

Our reservation concerning seismic design relates to the lack of adequate assurance that the Midland Plant will be capable of accomplishing shutdown heat removal for low probability earthquakes more severe than the safe shutdown earthquake (SSE). The Midland seismic design basis at the construction permit stage corresponded to a MMI VI, peak ground acceleration of 0.12g, employing a modified Housner spectrum. For the operating license review, the NRC Staff has reevaluated the original seismic design basis and the Applicant and the NRC Staff have agreed on the use of site-specific analyses which have led to increases in the design response spectra for frequencies above about 2 cycles/sec.

Historically, no earthquakes stronger than the newly proposed SSE have occurred within 200 miles of the Plant. However, expert opinion differs widely on the exceedance frequency of the proposed SSE and on the severity at the site of earthquakes whose likelihood is less than 1 in 10^4 or 1 in 10^5 per year.

The Applicant is currently reevaluating by selective audit the seismic capability of the plant, as originally designed, to withstand the revised SSE. Measures taken to assure safe shutdown in the event of an earthquake include the use of dewatering to reduce the potential for soil liquefaction. We recommend that all systems and components important to decay heat removal be carefully evaluated for their ability to accomplish necessary functions in the unlikely event of lower-probability, more severe earthquakes in order to provide the necessary degree of assurance. This matter should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed about the resolution of this matter. We believe that any recommendations for changes in the plant resulting from this evaluation should be implemented by the end of the second refueling outage.

The Applicant has agreed to provide core exit thermocouples, a hot-leg-level measurement system, and subcooled margin monitors as instrumentation to detect inadequate core cooling. Consumers Power Company also plans to include a remotely operable vent on top of both inlet loops to the steam generators; however, Consumers has not committed to supply a high point vent on the reactor vessel head. This matter should be resolved in a manner satisfactory to the NRC Staff. The ACRS recommends that the Applicant review further the potential for providing indications of water content or level within the reactor vessel.

The staff of the Applicant includes many personnel who have had nuclear power plant experience. However, operating experience with this B&W type power reactor is limited, and the NRC Staff is requiring that at least one person having experience on a large commercial PWR be included on each shift for one year. We support the NRC Staff position.

The Applicant's experience with the operation of nuclear power plants should, in principle, place Consumers in a favorable position to provide continuing, careful oversight of the operations at the Midland Plant. In view of some prior adverse operating experience at the Palisades Plant however, we recommend that the NRC Staff institute an augmented audit of operations at Midland, at least during the early years of operation at power.

We have reviewed the evaluation made of the tertiary process steam system for use by Dow Chemical Company. This system appears not to impose any unacceptable impacts either on the safe operation of the Midland Plant or on the people working at the Dow Chemical Company.

The Applicant has undertaken an effort to have a probabilistic risk assessment (PRA) performed for the Midland Plant and stated that the results will be available in the fall of 1982. We believe it desirable to have plant-specific PRAs performed for each commercial nuclear power plant and that it is particularly appropriate for the Midland Plant because of its relatively high, close-in population density. We wish to have the opportunity to review the Midland PRA with assistance from the NRC Staff, and to offer comments or recommendations as appropriate. We do not believe that this review need delay licensing of the Midland Plant for operation.

Recently, questions have come to light in connection with B&W plants concerning the availability of natural circulation in the presence of an interrupted or continuing small break loss-of-coolant accident. We wish to see a proposed NRC Staff resolution of this issue.

The Applicant described an extensive systems interactions study being undertaken for the Midland Plant. We wish to be informed of the results of this study.

We believe that, in view of the population density near this plant, additional prudence is appropriate for the Midland Plant in the resolution of the ATWS issue and other Unresolved Safety Issues.

We endorse the participation of Dow Chemical Company plant personnel in emergency procedures developed on the basis of an assumed failure at the Midland Plant. Similarly, there should be active participation by Midland Plant personnel in emergency procedures developed on the basis of an assumed failure at the Dow Chemical plant. The Applicant and the NRC Staff should promote continued coordination of these types of relationships, as well as those involving appropriate state and local groups to assure that the capability for an effective emergency response is developed and maintained.

With regard to the eleven items identified in the ACRS Supplemental Report on Midland Plant, Units 1 and 2 dated November 18, 1976, we have the following comments. The issues related to vibration and loose-parts monitoring, potential for axial xenon oscillations, behavior of core-barrel check valves during normal operation, fuel handling accidents, effects of blowdown forces on core internals, LOCA-related fuel rod failures, and improved quality assurance and in-service inspection for the primary system have all been resolved or are in a confirmatory stage of being resolved. Separation of protection and control equipment has been accomplished in an appropriate manner; however, the safety implications of control systems remains an Unresolved Safety Issue directly applicable to Midland. Resolution awaits completion of the NRC Staff Task Action Plan A-47. The effect of ECCS induced thermal shock on pressure vessel integrity has been resolved in part; however, the Unresolved Safety Issue on pressurized thermal shock will apply. Environmental qualification of equipment remains a generic issue which is under review by the NRC Staff and whose resolution will apply to the Midland Plant. Instrumentation to follow the course of an accident has been resolved in part by the development of revised Regulatory Guide 1.97. We do not believe that licensing of the Midland Plant for operation need await further resolution of any of the eleven issues discussed above.

The various other matters identified by the NRC Staff as open or confirmatory in the Safety Evaluation Report should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept advised concerning resolution of the turbine missile issue.

The ACRS believes that, subject to satisfactory completion of construction and staffing and if due regard is given to the comments above, the Midland Plant, Units 1 and 2 can be operated at power levels up to 5 percent of full power with reasonable assurance that there is no undue risk to the health and safety of the public.

We defer our recommendation regarding operation at full power until we have had the opportunity to review the plan for an audit of plant quality and the proposed resolution of the question regarding natural circulation in the presence of a small break LOCA.

Dr. Kerr did not participate in the Committee's review of this matter.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. Consumers Power Company, "Midland Plant Units 1 and 2 - Final Safety Analysis Report" including Amendments 1-43
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Midland Plant, Units 1 and 2," NUREG-0793, dated May 1982
3. U.S. Nuclear Regulatory Commission, "NRC Licensee Assessments," NUREG-0834, dated August 1981
4. Letter from J. Cook, Consumers Power Company, to J. Keppler, NRC, Subject: Midland Project Response to Draft SALP Report, dated May 17, 1982
5. Letter from J. Cook, Consumers Power Company, to J. Keppler, NRC, Subject: Midland Project Quality Assurance Program Update, dated April 30, 1981

6. Letter from J. Hind, NRC, to J. Cook, Consumers Power Company, Subject: Systematic Assessment of Licensee Performance (SALP), dated April 20, 1982
7. Letter from J. Cook, Consumers Power Company, to H. Denton, NRC, Subject: Summary of Soils-Related Issues at the Midland Nuclear Plant, dated April 19, 1982
8. Letter from K. Drehobl, Consumers Power Company, to D. Fischer, ACRS, Subject: Midland Project Soils Information, dated April 12, 1982
9. Statement of Ms. M. Sinclair to ACRS, dated June 4, 1982
10. Letter from B. Stamiris to Dr. D. Okrent and ACRS Members, Subject: Midland OL Review, dated May 29, 1982
11. Letter from M. Sinclair to Dr. P. Shewmon, ACRS, Subject: Midland OL Review, dated May 28, 1982
12. Statement by Dr. C. Anderson to ACRS Midland Plant Subcommittee dated May 20-21, 1982
13. Statement by Ms. M. Sinclair to ACRS Midland Plant Subcommittee dated May 20-21, 1982
14. Letter from B. Stamiris to D. Fischer and ACRS Members, Subject: Soil Settlement and QA Issues, dated May 20, 1982
15. Letter from M. Sinclair to Dr. C. Siess, ACRS, Subject: Midland Soil Settlement, dated April 26, 1982

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

September 9, 1967

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

Subject: REPORT ON MIDWEST FUEL RECOVERY PLANT

Dear Dr. Seaborg:

At its eighty-sixth meeting, on June 8-10, 1967, and its eighty-eighth meeting, on August 10-12, 1967, the Advisory Committee on Reactor Safeguards reviewed the General Electric Company proposal to build the Midwest Fuel Recovery Plant at a site about a mile from the Dresden Nuclear Power Station near Morris, Illinois. The Committee had the benefit of discussions with representatives of the General Electric Company and the AEC Regulatory Staff and its consultants, and of the documents listed. A Subcommittee of the ACRS met to review this project on June 5, 1967 and on July 31, 1967.

The plant will be designed to process 300 metric tons per year of irradiated uranium in the form of UO_2, clad in stainless steel or zirconium alloy. In the process, fuel bundles are sheared into short lengths and fed to a leacher where uranium, plutonium, neptunium, and fission products are dissolved; the solution is separated from the cladding, and further processed. The major steps include: recovery of plutonium and neptunium by anion exchange, calcination of the uranium process stream to UO_3, fluorination of the UO_3 to UF_6, and UF_6 purification by distillation.

High activity waste streams will be concentrated, reduced to solid form, and packaged in high integrity containers which will be submerged in a water-filled basin for retention. The applicant proposes that low-activity liquid wastes be concentrated and reduced to solid form for storage as an asphalt blend. Development tests and studies are still underway to ascertain the stability of asphalt-waste mixtures stored in large volumes. If the bulk asphalt storage method proves to be unacceptable, alternate methods of storage of low-level wastes are available. The Regulatory Staff should continue to review carefully these aspects of the plant as development and design progress, to assure acceptability of long-term storage procedures for radioactive wastes.

The feasibility of the various operations employed in this plant has been demonstrated. In addition, the General Electric Company is continuing work needed to verify detailed process characteristics and commercial operability. The start-up schedule allows for a six-month period of cold operation to verify performance characteristics and to permit operator training.

The applicant stated that the normal power system, the emergency power system, and related equipment will be designed so that no single failure will interrupt operation of the ventilation system or other vital services. The applicant also stated that the ventilation stack will be designed to preclude excessive restriction of flow in case of stack failure due to tornado winds.

The AEC Regulatory Staff should review significant final design features of vital components and systems of the plant prior to installation.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved by the applicant and the Regulatory Staff during plant construction. The Committee believes that, if due consideration is given to the foregoing comments, the proposed fuel reprocessing 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/
N. J. Palladino
Chairman

References Attached.

References - Midwest Fuel Recovery Plant

1. General Electric Company letter dated November 16, 1966 to AEC Division of Materials Licensing with attached application and Design and Analysis Report.

2. General Electric Company letter dated April 14, 1967 to AEC Division of Materials Licensing transmitting April 17, 1967 Response to AEC Staff Letters.

3. General Electric Company letter dated April 14, 1967 to AEC Division of Materials Licensing with enclosed drawings and schematic.

4. General Electric Company letter dated April 26, 1967 to AEC Division of Materials Licensing, with attachments.

5. General Electric Company letter dated April 14, 1967 to AEC Division of Materials Licensing with attachment.

6. General Electric Company letter dated May 15, 1967 to AEC Division of Materials Licensing, with enclosures.

7. General Electric Company letter dated July 20, 1967 to AEC Division of Materials Licensing, with attached Amendment 3.

8. General Electric Company letter dated August 1, 1967 to AEC Division of Materials Licensing, Amendment No. 4.

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

July 21, 1972

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

Subject: REPORT ON GENERAL ELECTRIC COMPANY - MIDWEST FUEL RECOVERY PLANT

Dear Dr. Schlesinger:

During its 147th meeting, on July 13-15, 1972, the Advisory Committee on Reactor Safeguards completed its review of the application by the General Electric Company for authorization to operate the Midwest Fuel Recovery Plant. The project was considered previously during Subcommittee meetings on March 22, 1972, at the site and on June 30, 1972, in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives of the General Electric Company and of the AEC Regulatory Staff and its consultants, and of the documents listed below. The Committee previously discussed this project in a construction permit report dated September 9, 1967.

The Midwest Fuel Recovery Plant is located in a rural area of Grundy County, Illinois, about eight miles east of Morris, Illinois. The site is south of the Illinois River and adjacent to the Dresden Nuclear Power Station. The applicant has designed and constructed the plant to seismic and tornado criteria which are consistent with those for the Dresden site and which the Committee finds acceptable.

The plant has been designed and constructed to be able to process 300 metric tons per year of irradiated uranium in the form of UO_2, clad in stainless steel or zirconium alloy. The original U-235 enrichment of the fuel to be processed will not exceed five percent and the average burnup will not exceed 44,000 megawatt days per metric ton. The fuel will be stored for a period of time sufficiently long to ensure that the I^{131} content will be less than one curie per metric ton. For fuels of maximum burnup this period will be at least 160 days, implying a reduction of the I^{131} content by a factor of about one million.

In the recovery process, the fuel pins are sheared into short lengths and fed to a leacher tank in which the fuel is dissolved by a nitric acid solution. The cladding hulls are moved to a storage vault and the acid solution is further processed. Plutonium and neptunium are recovered in the form of nitrate by solvent extraction and ion exchange. The uranium process stream is calcined to UO_3, which is then fluorinated to UF_6 followed by purification by distillation.

High activity wastes will be concentrated, calcined and packaged in high integrity containers which will be stored in a water-filled basin. Low activity wastes will be stored in solid form in a monitored, underground vault. Gaseous wastes will be treated by passing through a scrubber, silver zeolite and glass fiber filters and finally a sand filter prior to release. Concentrations of fission products offsite will be substantially below those specified by 10 CFR Part 20. The applicant has stated that no liquid wastes will be released from the plant.

The Technical Specifications for the plant have not been completed and will continue to evolve during startup testing and early operation. Additionally, the Regulatory review of plant physical security measures has not been completed. The Committee recommends that these specifications and this review be completed as appropriate and to the satisfaction of the Regulatory Staff.

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 Midwest Fuel Recovery Plant can be operated without undue risk to the health and safety of the public.

Sincerely yours,

C.P. Siess

C. P. Siess
Chairman

References:
1. General Electric Company letter dated December 31, 1970, forwarding the Final Safety Analysis Report (One Volume) for the Midwest Fuel Recovery Plant

2. Amendments 10 through 26 to the License Application

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

June 14, 1966

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

Subject: REVIEW OF "SECTION 91.b" REACTORS

Dear Dr. Seaborg:

The Advisory Committee on Reactor Safeguards and the AEC's Regulatory Staff review military reactors in accordance with Section 91.b. of the Atomic Energy Act of 1954 as amended and as guided by the Presidential Directive of September 23, 1961, which sets forth the responsibilities of the Department of Defense (DOD) and the Atomic Energy Commission for protecting the health and safety of the public in connection with these projects. As a result of many such reviews and, particularly, of recent attempts by the Committee and the Staff to evaluate the continuing safety status of several military reactors, the Committee has concluded that there exist certain difficulties, mainly of a procedural nature and arising primarily because of the divided nature of the safety responsibility in these cases, which in practice have become obstacles to clear-cut safety review.

Military reactors include at present the various fixed-base reactors and the Army's floating power plant, and the reactors on Navy submarines and surface ships. The Presidential Directive seems to give the DOD the principal safety responsibility for these but states that the AEC is "to participate in the identification and resolution of... (health and safety) problems as a matter of responsibility". DOD is to obtain "advice and assistance...from the AEC on the safety aspects... and in preparation or amendment of safety standards, procedures, or instructions relating to location and operation...and comment or concurrence shall be obtained from the AEC as to their adequacy".

In the Committee's opinion, the Naval Reactors Program complies with this Directive, the AEC's responsibility for reactor operation being exercised through the AEC Division of Naval Reactors and, for nuclear safety review and porting, by the Regulatory Staff. For other military reactors, it is difficult to identify a similarly clear-cut assignment of responsibility for safety review and compliance. AEC field offices sometimes have responsibility over the design contractor, but they have no control once operation is turned over to DOD. The Regulatory Staff does not routinely receive operating reports on all Army and Air Force reactors; and, when it does, its responsibility for action is not clear. The Division of Compiance has investigated some potential safety problems, but does not normally accord the same degree of surveillance to military as to licensed reactors.

The Committee is aware that the AEC is working with DOD to obtain better delineation of safety responsibility and hopes that these efforts will be rewarded with success. Existing nuclear safety groups within the DOD appear to the Committee to serve essentially an "in-house" safety review role. In order to assure a sufficiently experienced and independent safety review, comparable with that accorded licensed reactors, the Committee believes that the AEC should be given the clear responsibility for nuclear safety review of military reactors, except where military considerations are controlling. This review should include all phases; namely, construction, initial operations, operating experience, and significant changes in procedures or facilities.

The Committee believes that the AEC Staff can fulfill these functions and that a clear responsibility should be assigned within the AEC for this purpose. The resulting centralization within the AEC of safety review responsibility for all reactors should have the additional benefit that safety-related information and experience can effectively and rapidly be applied to all reactors. The ACRS would expect to participate only in the review of particularly difficult or novel aspects of these problems.

Sincerely yours,

/s/

David Okrent
Chairman

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

July 19, 1965

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

Subject: REPORT ON THE MILLSTONE POINT REACTOR SITE

Dear Dr. Seaborg:

At its sixty-fourth meeting held on July 8-10, 1965, the Advisory Committee on Reactor Safeguards considered the joint proposal of the Connecticut Light & Power Co., the Hartford Electric Light Co., and the Western Massachusetts Electric Co. for use of a site located at Millstone Point, Waterford, Connecticut upon which these corporations plan to construct a nuclear power plant of approximately 2500 MW(t) capacity. It was indicated that either a pressurized water or a boiling water type reactor will be used but the final selection has not been made as yet. The Committee had the benefit of oral presentations by representatives of the applicants, their consultants, the AEC Regulatory Staff, and of the documents listed herewith. A Subcommittee meeting was held at the site on July 6, 1965.

The proposed site consists of a plot of some 500 acres, somewhat irregular in shape, and surrounded by water on three sides, fronting chiefly on Long Island Sound and the Niantic inlet. Geologic surveys indicate the area to be substantially supported by a solid rock formation and that any expected seismic activity, based on area history, will be negligible. Adjacent to the northeast corner of the site, about 0.5 mile from the reactor center, is located a small housing development with an adjacent privately owned beach.

A flooded quarry on the site is being used under lease as a test station by the Navy Underwater Research Laboratory. On the east shore, in a leased area, the Maxim Division of American Machine & Foundry Corp. operates an experimental desalinization unit. The Committee believes that the continued leased use of these areas presents no significant problem of exposure but the lessees, i.e., the U. S. Navy and the Maxim Division of AMF, should be subject to and agree to necessary restrictions or controls established by the applicants.

Meteorological surveys, local background radiation surveys, and tidal flow studies are being undertaken.

In order for Millstone Point to meet the present site guidelines, reliance must be placed upon engineered safeguards. The Committee believes the added safeguards needed for the protection of the health and safety of the public can be provided.

The Advisory Committee on Reactor Safeguards believes that the Millstone Point site is acceptable for a reactor, either a pressurized water or a boiling water type and of the power level indicated, if adequate containment and associated engineered safeguards are provided.

Dr. T. J. Thompson did not participate in this review.

Sincerely yours,

/s/
W. D. Manly
Chairman

References:

1. Report - Preliminary Site Evaluation, Millstone Point, Waterford, Connecticut, dated May 7, 1965.

2. Preliminary Information on Proposed Millstone Point Generating Station, undated, received June 29, 1965.

3. Population Supplement to Preliminary Site Evaluation, dated June 18, 1965.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON 25, D.C.

March 18, 1966

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

Subject: REPORT ON MILLSTONE NUCLEAR POWER STATION

Dear Dr. Seaborg:

At the seventy-first meeting in Washington, D. C. on March 10-12, 1966, the Advisory Committee on Reactor Safeguards reviewed the proposal of The Connecticut Light and Power Company, the Hartford Electric Light Company, The Millstone Point Company and Western Massachusetts Electric Company to construct the Millstone Nuclear Power Station on the Millstone Point site. The Committee has reported on the Millstone Point reactor site in its letter of July 19, 1965. The applicants now propose a boiling water reactor using pressure suppression containment and designed by General Electric Company. The Committee had the benefit of discussions with representatives of the applicants, the General Electric Company, the AEC Regulatory Staff, and of the documents listed below. An ACRS Subcommittee visited the site on July 6, 1965, and met with the applicants to review the proposal on February 18, 1966.

The nominal thermal power of the Millstone Nuclear Power Station is 1730 MW, but the applicants have reported that all components are to be designed for an anticipated ultimate capability of approximately 2010 MW. It was stated that the General Electric Company has the responsibility to furnish the complete nuclear power station on a "turn key basis". The applicants state that the reactor facility is similar, except for size, to the Dresden Nuclear Power Station - Unit 2 (2255 MWt). Therefore, the development program described by the General Electric Company representatives for answering questions involving jet pump monitoring and system stability, metal-water reactions, instrumentation, and blowdown and emergency cooling for Dresden Unit 2 are expected to be applicable to the Millstone Station. As with Dresden Unit 2, the Committee recommends further studies of pipe-whipping and the generation of missiles which might cause engineered safeguards to be ineffective in the unlikely event of failure of the primary piping system.

It is also recommended that further studies, employing conservative values of significant parameters, be made of the course and consequences of potential reactivity transients.

The Committee urges that particular attention be given to the components in high pressure steam lines and again recommends that special attention be given to insure that no single rupture of the high pressure steam lines can lead to loss of containment. The Committee suggests that a study be undertaken to evaluate possible methods to reduce the escape of fission products from the turbine building in the unlikely event of failure of high pressure steam lines external to the reactor containment.

The Committee was advised that the coastal site of the Millstone Station is vulnerable to flooding during severe hurricanes. The applicants agreed to resolve with the Regulatory Staff the necessary degree of protection from such flooding. The applicants also stated that the stack design and location would be such as to preclude damage to the containment by stack failure.

The Committee notes that the applicants have undertaken a long-term observational program to improve their knowledge of the meteorological and marine biological conditions in the vicinity of the Millstone Point site.

The Committee understands that further consideration is being given by General Electric to additional methods of quality control in the fabrication of the reactor pressure vessel. The Committee also understands that considerable emphasis will be placed on the development and use of in-service inspection methods for ensuring the integrity of the vessel.

It is the opinion of the ACRS that resolution of the above problems can be attained during construction and that the Millstone Station can be constructed at the proposed site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Dr. T. J. Thompson did not participate in the Committee's review of this project.

Sincerely yours,

/s/

David Okrent
Chairman

References attached.

<u>References</u> - Millstone Nuclear Power Station

1. Design and Analysis Report, Millstone Nuclear Power Station, Volumes I and II, received November 18, 1965.

2. Design and Analysis Report, Millstone Nuclear Power Station, Amendment No. 1, received February 7, 1966.

3. Substitute Pages to Design and Analysis Report, Millstone Nuclear Power Station, received March 3, 1966.

4. Design and Analysis Report, Millstone Nuclear Power Station, Amendment No. 3, received March 3, 1966.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

January 15, 1970

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

Subject: REPORT ON MILLSTONE NUCLEAR POWER STATION, UNIT 1

Dear Dr. Seaborg:

During its 117th meeting, January 8-10, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by the Connecticut Light and Power Company, the Hartford Electric Light Company, the Millstone Company, and Western Massachusetts Electric Company for a license to operate Unit 1 of the Millstone Nuclear Power Station, a boiling water reactor power plant, at power levels up to 2011 MW(t). An ACRS Subcommittee meeting with the applicant was held at the site on November 20, 1969, and a second Subcommittee meeting was held in Washington, D. C., on January 7, 1970. During the review, the Committee had the benefit of discussions with the applicant, the General Electric Company, the AEC Regulatory Staff, their contractors and consultants, and of the documents listed.

The Committee reported to you on the Millstone site on July 19, 1965, and on the construction permit application for Unit 1 on March 18, 1966. The Committee's review for the construction permit was based on a proposed power of 1730 MW(t); this report is based on the presently proposed power of 2011 MW(t) which the applicant justifies on the basis of more recent heat transfer correlations and development of the core design. In its March 18, 1966 report the Committee stressed the importance of study of emergency core cooling, metal-water reactions, monitoring of jet pump performance, instrumentation, blowdown problems and system stability. The Committee is satisfied that progress has been made in these areas and that the applicant has been responsive to recommendations made in reports on other applications. Some improvements include substantially improved emergency power supplies, an improved emergency core cooling system, and increased turbine bypass capacity from 50% to 105%.

One design change, however, involved a reduction in the capacity of each of the redundant containment cooling systems. This alteration requires placing greater reliance on the heat capacity of the torus water for temporary storage of heat energy in the unlikely event of the hypothetical loss-of-coolant accident. The increase of the torus water temperature to 203^{o}F under certain degraded conditions is an additional concern because of its potential effects on the performance of the emergency pumps. These include the direct effect of high temperatures on the pumps and the dependence on containment pressure to assure adequate net positive suction head. The applicant stated that this containment cooling system will be designed and qualified for a torus water temperature of 203^{o}F. Confirmatory tests will be performed. The Committee recommends that the Regulatory Staff review the results of these tests and that the applicant resolve with the Regulatory Staff the conditions under which the plant may operate with a portion of the containment cooling system out-of-service.

The General Electric Company has an extensive integrated program for measuring vibration in several reactors. A part of this program involves Millstone Unit 1, but a major fraction of such data important to the Millstone Unit will derive from experiments to be conducted in Dresden Unit 2. In the event that these data are not forthcoming before Millstone Unit 1 is ready to operate or if the data are not clearly favorable, the Committee believes that the matter should be reviewed by the Regulatory Staff before routine full power operation of the Millstone Unit is begun.

The main steam lines are provided with redundant valves that are required to close automatically in the unlikely event of a serious accident. Because experience with these large and special valves is limited, the Committee recommends that their performance be followed closely, and that the applicant make additional provisions to assure the requisite leaktightness if experience should be unfavorable. The Committee wishes to be kept informed of the resolution of this matter.

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 and isolability of these lines and to a program for the periodic examination and testing of the valves in these lines. The adequacy of measures taken with regard to such instrument lines snould be confirmed by the Regulatory Staff.

Continuing research and engineering studies are expected to lead to enhancement of the safety of water-cooled reactors in other areas than those mentioned; for example, by the determination of the extent of the generation of hydrogen by radiolysis and by other sources in the unlikely event of a loss-of-coolant accident, development of instrumentation for in-service monitoring of the pressure vessel and other parts of the primary system for vibration and detection of loose parts in the system, by the development of further means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram during anticipated transients, and evaluation of the consequences of water contamination by structural materials and coatings in a loss-of-coolant accident. As solutions to the problems develop and are evaluated by the Regulatory Staff, appropriate action should be taken by the applicant on a reasonable time scale.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Millstone Nuclear Generating Unit 1 can be operated at a power of 2011 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/
Joseph M. Hendrie
Chairman

<u>References</u>:

1. Letter from The Millstone Power Company, dated July 25, 1967; re: Proposed Design Changes for ECCS and Emergency Power Facilities

2. Letter from Day, Berry and Howard, dated March 14, 1968; Amendment No. 5 to License Application, Application for POL; Volumes 1, 2 and 3 of Final Safety Analysis Report (FSAR)

3. Letter from Day, Berry and Howard, dated May 2, 1968; Amendment No. 6 to License Application, Appendix B to FSAR, "Pre-Operational and Startup Tests"

4. Letters from Day, Berry and Howard; Amendments 8 through 22 to License Application

5. Letter from The Millstone Point Company, dated December 29, 1969; Confirms and clarifies information re: review of application for OL for Millstone Unit 1

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

May 15, 1970

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

Subject: REPORT ON THE MILLSTONE NUCLEAR POWER STATION UNIT 2

Dear Dr. Seaborg:

During its 121st meeting, May 7-9, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by The Connecticut Light and Power Company, The Hartford Electric Light Company, The Millstone Point Company, and Western Massachusetts Electric Company for authorization to construct the Millstone Nuclear Power Station Unit 2. The project was previously considered during an ACRS Subcommittee meeting on May 1, 1970, and the site was visited by an ACRS Subcommittee on November 20, 1969. During its review, the Committee had the benefit of discussions with representatives of the applicant, Combustion Engineering Corporation, Bechtel Corporation, members of 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 Millstone site on July 19, 1965, in regard to the Millstone Nuclear Power Station Unit 1, a 2011 MWt boiling water reactor. Millstone Unit 2, a 2560 MWt pressurized water reactor, will be constructed adjacent to Unit 1. Facilities shared by the two units include the control room, the stack, the switchyard, and fire protection services. During the construction of Unit 2, a security system will be instituted to control access to Unit 1.

The proposed pressurized water reactor is similar in design to the previously reviewed Hutchinson Island, Calvert Cliffs, and Maine Yankee reactors (ACRS reports dated March 12, 1970, March 13, 1969, and July 19, 1968). The power level of Millstone Unit 2, at 2560 MWt, represents an increase of five percent over the 2440 MWt power level of these reactors.

The containment system consists of a steel-lined, prestressed concrete cylindrical structure and a steel-framed enclosure building. The enclosure building provides the capability for collecting the leakage of gases from the concrete structure and for discharging these gases through filters to the existing 375-foot stack. The several emergency core cooling systems are similar to previously reviewed designs.

Further study is required with regard to potential releases of radioactivity in the unlikely event of gross damage to an irradiated fuel assembly in the spent fuel pool. This matter should be resolved in a manner satisfactory to the AEC Regulatory Staff.

The Committee reiterates its interest in active participation by applicants in overall quality assurance programs in order to assure the construction of safer plants.

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

The applicant should accelerate completion of his studies of means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram when required during anticipated transients.

The applicant has stated that turbine-generated missile damage shall not preclude the safe shutdown of the plant. Some questions remain with regard to possible effects of turbine-generated missile damage to Millstone Unit 1. This matter, as well as the adequacy of measures to control turbine overspeed, should be resolved in a manner satisfactory to the Regulatory Staff.

Other problems related to large water reactors have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports. The Committee feels that resolution of these items should apply equally to Millstone Unit 2.

The Committee believes that the above items can be resolved during construction and that, if due consideration is given to these items, this second nuclear unit proposed for the Millstone site can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/

Joseph M. Hendrie
Chairman

References:

1. Letter from Day, Berry and Howard, dated February 26, 1969; License Application: Volumes 1 and 2 of Preliminary Safety Analysis Report

2. Amendments 1 through 8 to the Preliminary Safety Analysis Report

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

June 16, 1970

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

Subject: REPORT ON MILLSTONE NUCLEAR POWER STATION UNIT 1

Dear Dr. Seaborg:

During its 122nd meeting, June 11-13, 1970, the Advisory Committee on Reactor Safeguards met with representatives of the Northeast Utilities Service Company to review proposed changes to the reactor vessel nozzle "safe ends" (stainless steel extensions of the nozzles) of the Millstone Power Station Unit 1. During its review, the Committee had the benefit of discussions with the applicant, the General Electric Company, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported to you on operation of the Millstone Nuclear Power Station Unit 1, on January 15, 1970.

Normal procedures for most reactor pressure vessels have been to joint the austenitic stainless steel safe ends to the nozzles prior to the stress relieving heat treatment. This heat treatment sensitizes the safe ends, which makes the steel less resistant to certain types of corrosion. Sensitized austenitic stainless steels in this condition have given reasonably satisfactory service over many reactor years of operation.

Recently, leaks developed in sensitized safe ends of two operating reactors. The causes of the leaks have been studied exhaustively, and it is concluded by the licensees that they were caused by unusual circumstances that need not have existed. In view of this experience, however, the applicant is making modifications to Millstone Nuclear Power Station Unit 1. These modifications consist of replacing the two sensitized safe ends of the core spray nozzles, and the two outlet and the ten inlet recirculation nozzles, and a number of smaller nozzles. Some other components and attachments in the vessel are also being replaced or overlaid with weld metal cladding of a composition that is resistant to stress corrosion.

The Committee agrees with the applicant that these changes, properly executed, should increase assurance of trouble-free operation. The Committee wishes to call attention to other factors that would further tend to diminish the probability of a failure in a safe end or other piping component. The Committee believes an independent check should be made of stresses in the as-built piping of the primary system; this has been performed. The Committee believes that, in addition, the displacements of the piping system should be observed in the hot condition of the plant. A review should be made of high points in non-flowing parts of the system and means should be provided, where necessary, to vent or otherwise remove gases that could become trapped at such points.

The Committee also believes that the Regulatory Staff should assure itself that the biological shield surrounding the reactor vessel can withstand the pressure that could be developed by loss of integrity of a safe end or nozzle, or that failure of the shield would have no intolerable consequences.

The Committee has on several occasions stressed the importance of in-service inspection and leak detection. It recommends that the Regulatory Staff develop a schedule of inspections for safe ends. The operation of the leak detection and location systems should be reviewed and modified as appropriate to obtain the maximum speed and sensitivity for detection of leaks. In addition, the applicant should study other techniques of detecting leaks.

Subject to these comments, and if due attention is paid to the items discussed in the Committee report of January 15, 1970, the Committee reaffirms its belief that there is reasonable assurance that the Millstone Nuclear Generating Station Unit 1 can be operated at a power of 2011 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/

Joseph M. Hendrie
Chairman

Reference

1) Amendment No. 25 to License Application, Supplementary Information

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

APR 16 1974

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

Subject: REPORT ON MILLSTONE NUCLEAR POWER STATION UNIT NO. 3

At its 168th meeting, April 11-13, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application by the Millstone Point Company et al for authorization to construct the Millstone Nuclear Power Station Unit No. 3. This application had been considered previously at a Subcommittee meeting on March 15-16, 1974, and Committee members visited the site on January 26, 1974. During its review, the Committee had the benefit of discussions with representatives of the applicants and their consultants, the Westinghouse Electric Corporation, the Stone and Webster Engineering Corporation and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The Millstone Nuclear Power Station Unit No. 3 employs a 4-loop pressurized water reactor of 3411 MW(t) rated power. The Millstone Station site is located on the north shore of Long Island Sound about 40 miles southeast of Hartford and about 3.2 miles southwest of New London, Connecticut, the nearest population center (estimated 1970 population of 31,360). The site will be shared with Unit 1, presently in operation, and with Unit 2, now under construction. The exclusion radius of the site is 0.36 miles and the low population zone radius is 2.4 miles.

The applicants' evaluation of seismicity of the site indicated that a 0.17g horizontal ground acceleration value should be used in the analysis of the response of Category I systems to the Safe Shutdown Earthquake. The ACRS has reviewed this evaluation, together with additional information published subsequent to the applicants' studies, and agrees that the value proposed is acceptable for this site.

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 Millstone Unit No. 3 is in this category. This unit 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 applicants, 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 applicants 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 unit.

The containment for the Millstone Unit No. 3, like that of Surry Units 1 and 2, is a subatmospheric design incorporating a steel-lined reinforced concrete vessel and a Supplementary Leak Collection and Release System to better control potential leakage. Reduced containment leakage rates may be required to meet the Part 100 limits. Evaluation of the containment peak pressure and subcompartment differential pressure during accident conditions is continuing. These matters should be resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The proposed offsite power systems for the Millstone Unit No. 3 comply with the requirements of General Design Criteria Numbers 17 and 18 but do not meet the recommendations of Regulatory Guide 1.32 concerning the availability of two, full capacity, immediate-access circuits from the offsite source. The applicants have committed to modifications to upgrade these systems. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Committee recommends that further attention be given by the applicants and the Regulatory Staff to those provisions of Regulatory Guide 1.17 which address design features to prevent or mitigate the consequences of acts of 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 applicants.

The ACRS believes that the above items can be resolved during construction and that, if due consideration is given to these items, the Millstone Nuclear Power 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,

W. R. Stratton

W. R. Stratton
Chairman

References:

1. Millstone Nuclear Power Station Unit 3 Preliminary Safety Analysis Report (PSAR), Volumes 1-V submitted January 29, 1973

2. PSAR Amendments Numbers 1, 4-11, 13-20 dated March 8, 1973 through April 5, 1974

3. Safety Evaluation Report, dated March 13, 1974, by the Directorate of Licensing, U. S. Atomic Energy Commission, in the matter of the Millstone Point Company, et al, Millstone Nuclear Power Station Unit 3, Docket No. 50-423

June 11, 1974

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

Subject: REPORT ON THE MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

Dear Dr. Ray:

During its 170th meeting, June 6-8, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application by the Connecticut Light and Power Company, the Hartford Electric Light Company, the Western Massachusetts Electric Company, and the Northeast Nuclear Energy Company (formerly the Millstone Point Company) for authorization to operate Millstone Nuclear Power Station, Unit No. 2 at power levels up to 2570 MW(t). The application was previously considered at a Subcommittee meeting on May 22, 1974. A tour of the facility was made by Committee members on January 26, 1974. During its review, the Committee had the benefit of discussions with representatives of the applicants, Combustion Engineering Corporation, Bechtel Corporation, and members of the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed below. The Committee reported on the application for construction of Millstone Unit No. 2 on May 15, 1970.

The Millstone site is located on the north shore of Long Island Sound about 40 miles southeast of Hartford and 3.2 miles southwest of New London, Connecticut, the nearest population center (estimated 1970 population, 31,360). The exclusion radius of the site is 0.36 miles and the low population zone radius is 2.4 miles.

When completed, the Millstone Station will be comprised of three nuclear power plants. Unit No. 1 is a 2011 MW(t) General Electric boiling water reactor plant. The Committee reported on the application for authorization to operate this unit on January 15, 1970 and June 16, 1970. Unit No. 3 is to be a 3411 MW(t) Westinghouse pressurized water reactor plant. The Committee reported on the application for construction of this unit on April 16, 1974.

Unit No. 2 uses a Combustion Engineering pressurized water reactor similar in design to Calvert Cliffs Units 1 and 2. The Committee reported on the operating license application for the latter two units on January 14, 1974.

The Millstone Unit No. 2 reactor is located within a steel lined, prestressed concrete containment and the containment, in turn, is enclosed by a steel framed outer building. In the event of an accident signal, the space between the containment and outer building is to be maintained slightly below atmospheric pressure by continuous evacuation of gas inleakage. The evacuated gas will be processed through an air cleaning system prior to venting through a tall stack common to all three units.

The applicants have agreed to limit the peak linear heat generation rate to 17.0 kw/ft for operation during the first fuel cycle. Limits for operations during subsequent fuel cycles are to be established later. The initial limitation of peak linear heat generation rate has been calculated by the applicants on the basis of the Interim Acceptance Criteria and Combustion Engineering evaluation models, incorporating the effects of fuel densification. Initially, maps of core power distribution are to be developed at several power levels based upon readings of incore detectors, and these maps are to be compared with simultaneous readings of excore detectors. After sufficient experience is gained, operations will be based on use of the excore detectors only, with incore mapping done monthly for verification. These proposed operating limits and procedures for control of peak linear heat generation rate during the first fuel cycle have been evaluated by the Regulatory Staff and found satisfactory. The Committee concurs.

The operating limits of Unit No. 2 must be reevaluated in accordance with the recently promulgated Acceptance Criteria for Emergency Core Cooling, 10 CFR Part 50.46. The Committee wishes to be informed of the results of this reevaluation.

The Committee recommends that the Technical Specifications for Millstone Unit No. 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.

The Committee believes that the applicants and the Regulatory Staff should review in greater depth possible sources of debris which might arise in the unlikely event of a LOCA and enter pump suction lines and disable components such as the spray nozzles. The adequacy of the sump screens to hold back, without loss of function, such debris should be determined.

Inservice inspection of the reactor coolant system is to be performed in conformance with Section XI of the ASME Boiler and Pressure Vessel Code, to the extent permitted by the existing design. The Committee believes that appropriate inservice inspection of the outer shell of the secondary side of the steam generators should be utilized to assure continuing integrity. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Committee believes it is essential that plant personnel be provided with those instruments, indicators, and measurements that will define clearly the nature and course of an accident so that offsite emergency plans can be initiated at a level and on a time scale consistent with the severity, or potential severity, of an accident.

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 subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Millstone Nuclear Power Station, Unit No. 2 can be operated at power levels up to 2570 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References

Listed on Page 4

References

1. The Millstone Point Company* letter dated August 10, 1972, Submitting application for operating license for Millstone Nuclear Power Station, Unit No. 2, and Amendment 13, Final Safety Analysis Report (FSAR), Volumes I, II, and III.

2. Amendments 14-20 and 22-31, consisting of revised and additional pages and figures of the FSAR.

3. The Millstone Point Company letter dated January 3, 1973, regarding the effects of fuel densification.

4. The Millstone Point Company letter dated December 31, 1973, regarding anticipated transients without scram.

5. The Millstone Point Company letter dated January 15, 1974, regarding flood protection and shoreline stability.

6. The Millstone Point Company letter dated January 30, 1974, regarding hydraulic shock suppressors.

7. The Millstone Point Company letter dated February 27, 1974 regarding quality assurance program.

8. The Millstone Point Company letter dated March 28, 1974, regarding pre-operational testing of emergency core cooling systems.

9. The Millstone Point Company letter dated April 29, 1974, regarding resolution of items.

10. Directorate of Licensing letter dated May 10, 1974, forwarding the Safety Evaluation of the Millstone Nuclear Power Station, Unit No. 2.

*The Millstone Point Company name was changed to Northeast Nuclear Energy Company (letter dated May 8, 1974).

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

June 18, 1979

Lee V. Gossick, Executive Director for Operations

SUBJECT: POWER LEVEL INCREASE AT MILLSTONE NUCLEAR POWER STATION UNIT 2

During its 230th meeting, the Advisory Committee on Reactor Safeguards considered the proposed increase in licensed power level of the Millstone Nuclear Power Station Unit 2 from 2560 to 2700 MWt. The Committee concluded that it would not object to the NRC Staff's plan to license Millstone Nuclear Power Station Unit 2 to operate at a power level of 2700 MWt.

Raymond F. Fraley
Raymond F. Fraley
Executive Director

cc: ACRS Members
H. Denton, NRR
D. Eisenhut, DOR
D. Ross, DPM
S. Chilk, SECY

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 THE SYSTEMATIC EVALUATION PROGRAM REVIEW OF THE MILLSTONE NUCLEAR POWER STATION, UNIT 1

During its 272nd meeting, December 9-11, 1982, the ACRS reviewed the results of the Systematic Evaluation Program (SEP), Phase II, as it has been applied to the Millstone Nuclear Power Station, Unit 1. These matters were also discussed during Subcommittee meetings in Washington, D. C. on October 27 and November 30, 1982. During our review, we had the benefit of discussion with representatives of the Northeast Nuclear Energy Company (Licensee) and the NRC Staff. We also had the benefit of the documents listed below.

The Committee has reported to you previously on reviews of the SEP evaluations of the Palisades, Ginna, and Oyster Creek plants in letters dated May 11, August 18, and November 9, 1982. The first of these reports included comments on the objectives of the SEP and the extent to which they have been achieved. Our review of the SEP in relation to the Millstone plant has led to no changes in our previous findings regarding this program, as reported in our letter on the Palisades plant.

The remainder of this letter relates specifically to the SEP review of the Millstone plant.

Of the 137 topics to be addressed in Phase II of the SEP, 31 were not applicable to the Millstone plant and 20 were deleted because they were being reviewed generically under either the Unresolved Safety Issues (USI) program or the TMI Action Plan. Of the 86 topics addressed in the Millstone review, 48 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 87 issues relating to areas in which the Millstone plant did not meet current criteria. These issues were addressed by the Integrated Plant Safety Assessment, and various resolutions have been proposed.

The Integrated Assessment has not yet been completed for 42 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. Several relate to structural design, and the Licensee has proposed an integrated structural analysis program for their resolution. The resolution of these issues will be addressed by the NRC Staff in a supplemental report that will be available for review in connection with the application for a full term operating license (FTOL) for the Millstone plant.

For 23 of the issues included in the Integrated Assessment, the NRC Staff concluded that no backfit is required. We concur.

For the remaining issues for which the assessment has been completed, the NRC Staff requires hardware backfits in about half of the cases, and changes in procedures or Technical Specifications in the other half. The Licensee has agreed to make these changes with one exception. Topics XV-16 and 18 relate to the calculated radiological consequences for certain design basis accidents; thyroid doses, calculated in accordance with current criteria, are considerably in excess of the siting criteria. To correct this situation, the NRC Staff has proposed that the radioiodine concentration in the reactor coolant be limited to that permitted by the Standard Technical Specifications for BWRs. The Licensee has proposed to establish plant-specific radioiodine limits based on more realistic dose calculations. We believe that the NRC Staff's proposal is the more appropriate.

We have noted in previous letters on the SEP program that plant-specific probabilistic risk assessments (PRA) were not available for use in connection with the Integrated Assessment. In this case, a plant-specific PRA for the Millstone plant had been developed as part of the Interim Reliability Evaluation Program (IREP), and the results were used in the assessment of 21 of the issues. Contrary to our previous belief (contained in our August 18, 1982 and May 11, 1982 reports on the Ginna and Palisades SEP reviews), it does not appear that the plant-specific IREP PRA for the Millstone plant provided a basis for more definitive assessments than the more limited risk analyses developed for the other plants that we have reviewed.

Our conclusions regarding the Millstone SEP review are similar to those for the plants previously reviewed:

1. The SEP has been carried out in such a manner that the stated objectives have been achieved for the most part for the Millstone plant and should be achieved for the remaining plants in Phase II of the program.

2. The actions taken thus far by the NRC Staff in its SEP assessment of the Millstone plant are acceptable.

3. The ACRS will defer its review of the FTOL for the Millstone Nuclear Power Station, Unit 1 until the NRC Staff has completed its actions on the remaining SEP topics and the USI and TMI Action Plan items.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. U.S. Nuclear Regulatory Commission Draft Report, NUREG-0824, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Millstone Nuclear Power Station, Unit 1," dated November 1982.
2. U.S. Nuclear Regulatory Commission Safety Evaluation Reports, Millstone 1 Systematic Evaluation Program Topics, Volumes 1 and 2, received November 1982.
3. NRC Staff consultants' reports on the Millstone 1 Integrated Plant Safety Assessment Report consisting of consultants' reports from S. H. Bush, J. M. Hendrie, H. S. Isbin, and Z. Zudans, dated November 22, November 29, November 24, and November 24, 1982, respectively.
4. Science Applications, Inc. report number SAI-002-82-BE, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," Volume I, Main Report, Draft dated October 1, 1982.

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

January 11, 1983

Mr. William J. Dircks
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Dircks:

SUBJECT: CLARIFICATION OF COMMENTS IN THE ACRS REPORT ON THE SYSTEMATIC EVALUATION PROGRAM REVIEW OF THE MILLSTONE NUCLEAR POWER STATION, UNIT 1

During its 273rd meeting, January 6-8, 1983 the Advisory Committee on Reactor Safeguards briefly discussed with the NRC staff the interpretation of ACRS comments in its December 13, 1982 report regarding the usefulness of plant-specific probabalistic risk assessments in support of the systematic evaluation program (SEP). Specifically, the following paragraph:

> "We have noted in previous letters on the SEP program that plant-specific probabilistic risk assessments (PRA) were not available for use in connection with the Integrated Assessment. In this case, a plant-specific PRA for the Millstone plant had been developed as part of the Interim Reliability Evaluation Program (IREP), and the results were used in the assessment of 21 of the issues. Contrary to our previous belief (contained in our August 18, 1982 and May 11, 1982 reports on the Ginna and Palisades SEP reviews), it does not appear that the plant-specific IREP PRA for the Millstone plant provided a basis for more definitive assessments than the more limited risk analyses developed for the other plants that we have reviewed."

We provide the comments below with respect to this matter.

The statement in the Millstone letter presumably has been interpreted as saying that plant-specific PRAs are not useful. This was not our intent; the comment related only to the usefulness of a plant-specific PRA, which lacked treatment of external events, in connection with the very limited set of issues to which it was applicable for the SEP Phase II as it has been conducted. Our favorable views regarding the desirability and usefulness of plant-specific PRAs have been expressed several times in the past.

In another sense, the statement in the Millstone letter has been interpreted as arguing against the requirement of a National Reliability Evaluation Program (NREP) PRA for the plants selected for review in Phase III of the SEP. To some extent this is correct. If Phase III is to be conducted in essentially the same manner as Phase II, except for a smaller number of topics, it does not seem that it would be cost-effective to require a plant-specific PRA if its only use were to assist in the Integrated Plant Safety Assessment, unless external events are included in the PRA. There are several reasons for this. One is that the NREP, like IREP, will not include external events, which have represented some of the most important differences in the SEP plants reviewed to date. Another reason is that many of the differences from current criteria are not in areas addressed by PRAs.

Sincerely,

J. J. Ray
Chairman

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

September 10, 1984

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

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE MILLSTONE NUCLEAR POWER STATION UNIT NO. 3

During its 293rd meeting, September 6-8, 1984, the Advisory Committee on Reactor Safeguards reviewed the application of the Northeast Nuclear Energy Company (the Applicant) for a license to operate the Millstone Nuclear Power Station Unit No. 3. This application was considered by the ACRS Subcommittee on the Millstone Nuclear Power Station Unit No. 3 and the Subcommittee on Reliability and Probabilistic Assessment at a combined meeting held on August 28 and 29, 1984 at Windsor Locks, Connecticut. Members and consultants of these Subcommittees toured the facility on August 28, 1984. During our review, we had the benefit of discussions with representatives and consultants of the Applicant, Westinghouse Electric Corporation, Stone & Webster Engineering Corporation, and the NRC Staff. We also had the benefit of the documents referenced. The ACRS commented on the construction permit application for the Millstone Nuclear Power Station Unit No. 3 in a report dated April 16, 1974.

The Millstone Nuclear Power Station Unit No. 3 is located on Long Island Sound, on the east side of Niantic Bay, in Waterford, Connecticut. Millstone Unit No. 3 uses a four-loop pressurized water reactor, with a rated thermal power level of 3411 MW, supplied by the Westinghouse Electric Corporation. It is similar to that of the Comanche Peak Steam Electric Station. The containment for the plant, similar to that of the Surry Power Station Units 1 and 2, is a subatmospheric design, incorporating a steel-lined reinforced concrete structure and a supplementary leak collection and release system.

The Millstone Nuclear Power Station Units 1 and 2, presently in operation, are located on the same site. Unit 1 uses a boiling water reactor, with a rated thermal power level of 2011 MW, and Unit 2 uses a Combustion Engineering pressurized water reactor, with a rated thermal power level of 2700 MW.

Our review included an evaluation of the management organization, the operational staff, and the training program for the operating and maintenance staff. The tour of the plant included the training center located on the Millstone site. A plant-specific simulator for Millstone Unit No. 3 is expected to be installed and in operation well before startup.

During our discussions, the Applicant demonstrated an extensive knowledge of the operation, design, and construction features of the plant. We conclude that the Applicant is well qualified to operate Millstone Unit No. 3.

During our meeting, the NRC Staff identified a number of open issues that must be resolved prior to the granting of an operating license. We believe that these can be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed.

In response to a request from the NRC Staff, the Applicant submitted a Probabilistic Safety Study (PSS) in August 1983. The PSS is now being reviewed by the NRC Staff. In our meeting with the Applicant, a number of plant features were identified that have been modified as a result of the PSS. The NRC Staff is continuing its review of the PSS with special attention being given to resistance to seismic events. At this time, we are not prepared to comment on the suitability of the current NRC Staff confirmatory requirement concerning seismic capability. We expect to continue our review of the PSS and to review the NRC Staff's analyses as they become available. However, this review need not be completed before a decision is made on an operating license for this unit.

We believe that, subject to the resolution of open items identified by the NRC Staff and subject to the satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Millstone Nuclear Power Station Unit No. 3 can be operated at power levels up to 3411 MWt without undue risk to the health and safety of the public.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

References:

1. Northeast Nuclear Energy Company, "Millstone Nuclear Power Station, Unit No. 3, Final Safety Analysis Report," Volumes 1-16 and Amendments 1-8
2. Northeast Nuclear Energy Company, "Millstone Nuclear Power Station, Unit No. 3, Fire Protection Evaluation Report"
3. Northeast Nuclear Energy Company, "Millstone Nuclear Power Station, Unit No. 3, Probabilistic Safety Study," Volumes 1-12 and Amendments 1-2
4. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Millstone Nuclear Power Station, Unit No. 3," USNRC Report NUREG-1031, dated July 1984
5. U. S. Nuclear Regulatory Commission, "Draft Environmental Statement Related to the Operation of Millstone Nuclear Power Station, Unit No. 3," USNRC Report NUREG-1046, dated July 1984
6. Lawrence Livermore National Laboratory, et al., draft report, "A Review of the Millstone-3 Probabilistic Safety Study," dated May 30, 1984

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

September 11, 1961

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

Subject: REPORT ON MOLTEN SALT REACTOR EXPERIMENT

Dear Dr. Seaborg:

At its thirty-sixth meeting on September 7-9, 1961, the Advisory Committee on Reactor Safeguards considered the Molten Salt Reactor Experiment. The Committee had available for review the documents listed below and discussed the reactor facility with representatives of the Oak Ridge National Laboratory and the AEC staff.

The MSRE is a 10 MW(th) reactor experiment to be constructed and operated at the Oak Ridge National Laboratory as a continuing investigation in the study of molten fluoride mixtures and containier materials for circulating fuel reactors. This reactor concept is based upon molten salt experiments and investigations at ORNL which have been underway for several years. The design of the MSRE is about 85% complete at the present time.

The reactor is to be located in the 7300 Area of Oak Ridge about three-fourths of a mile south of the main area of ORNL from which it is separated by a 1000-foot ridge. Although the reactor site is located within a mile of several other existing or proposed reactors, it is several miles from the nearest residential area. The MSRE is to be located in a building which had been used originally for the ARE and later for the ART. (The latter was never operated.) Since the ART was approved for operation at a higher power level at this location, preliminary building modification for the MSRE has already been authorized by the General Manager to expedite the program. The ACRS was asked to consider the reactor and its site at this meeting.

Because of late submission by ORNL, the Committee did not review in detail the additional documents supplied at the meeting. The applicant stated, however, that additional consideration must be given to the possibility that excessive pressures (beyond those originally conceived) may develop in the secondary containment following a major accident. Ways of coping with this problem are under investigation.

The Committee believes that the proposed containment system and the suggested means for preventing excessive pressures are fundamentally sound and should protect the environment and public. We are concerned, however, that the instrumentation and control rod systems are marginal for a reactor being used to evaluate a new concept. In particular, the duplication of important nuclear instrumentation channels is borderline. The worth of the control rods appears to provide inadequate shutdown margin and no fast scram action is available. The Committee recommends that these features be given further study.

It is the opinion of the ACRS that with satisfactory resolution of the above problems, the MSRE can be constructed with reasonable assurance that it can be operated at the site proposed without undue hazard to the health and safety of the public, or to site personnel.

Dr. William K. Ergen did not participate in this review.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

References:

1. ORNL-CF-61-2-46 - "Molten Salt Reactor Experiment Preliminary Hazards Report" - dated February 28, 1961.
2. Addendum to ORNL-CF-61-2-46, dated August 14, 1961.
3. Memorandum - F. K. Pittman, DRD, to R. Lowenstein, DL&R, dated May 24, 1961, "MSRE Preliminary Hazards Review", w/attachments.

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

March 17, 1965

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

Subject: REPORT ON MOLTEN SALT REACTOR EXPERIMENT (MSRE)

Dear Dr. Seaborg:

At its sixty-second meeting on March 11-13, 1965, the Advisory Committee on Reactor Safeguards considered the proposed operation of the Molten Salt Reactor Experiment (MSRE), a 10 MW(t) graphite-moderated, circulating fuel reactor, at Oak Ridge National Laboratory. The Committee had available for review the documents listed below, and it discussed the reactor facility, safety analyses and proposed operation with representatives of the Oak Ridge National Laboratory, the AEC Regulatory Staff, and the Division of Reactor Development and Technology. The Committee had previously considered and reported on construction of this reactor at its thirty-sixth meeting in September 1961. A subcommittee meeting was held at Oak Ridge on December 15, 1964.

Although many novel features are incorporated in the design, and the chemistry of the fuel and its corrosive properties are not completely understood, the reactor characteristics are such that equipment failures, other than those affecting the reactivity control system, are unlikely to present serious safety problems. The fuel is not under any significant pressure, and other sources of stored energy appear to be absent. Leaks in the primary fuel system will decrease reactivity.

Because of the experimental nature of this reactor and the possibility of reactivity anomalies, additional emphasis should be placed on the reactivity control and instrumentation. In particular, provisions should be made to check the reliability of the three control rods by surveillance and by exercising them frequently during operations, since these are the only external means available for rapidly inserting negative reactivity into the reactor. In addition the Committee believes that the reactor should be provided with a suitable positive period scram and that consideration should be given to a negative period scram and to provisions for protection against possible adverse effects of the automatic control system. The Committee also believes that the operating group should establish appropriate allowable limits on reactivity anomalies. These limits should be established before criticality tests begin and should be adhered to during all operations.

It is understood that the MSRE group at Oak Ridge will report on low-power experiments prior to proceeding to a stepwise approach to full power.

With attention to reactivity control and instrumentation as recommended, the Committee believes that the Molten Salt Reactor Experiment may be operated without undue risk to the health and safety of the public.

Dr. F. A. Gifford, Mr. W. D. Manly, and Dr. H. W. Newson did not participate in the review of this project.

Sincerely yours,

/s/ David Okrent

David Okrent
Acting Chairman

References:

1. ORNL-TM-732, MSRE Design and Operations Report, Part V, Reactor Safety Analysis Report, dated August 1964.
2. ORNL-3708, Molten Salt Reactor Program, Semiannual Progress Report for Period Ending July 31, 1964, dated November 1964.
3. ORNL-TM-728, MSRE Design and Operations Report, Part I, Description of Reactor Design, dated January 1965.
4. ORNL-TM-730, MSRE Design and Operations Report, Part III, Nuclear Analysis, dated February 3, 1964.

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

September 15, 1976

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM REPORT ON MONTAGUE POWER STATION, UNITS 1 AND 2

Dear Mr. Rowden:

At its 197th meeting, September 9-11, 1976, the Advisory Committee on Reactor Safeguards completed an interim review of the application of Northeast Nuclear Energy Company, acting as agent for a group of private and municipal utilities (Applicants), for permits to construct the proposed Montague Power Station, Units 1 and 2. This application was reviewed at a Subcommittee meeting at Turners Falls, Massachusetts, on August 26-27, 1976, subsequent to a visit to the site on August 26. The Committee also had the benefit of discussions with representatives and consultants of the Northeast Nuclear Energy Company, the Nuclear Regulatory Commission (NRC) Staff, the General Electric Company, the Stone & Webster Engineering Corporation, and of the documents listed. The Subcommittee also received statements from area residents.

The Montague Station will be located in Franklin County, Massachusetts, 1.2 miles south-southeast of the village of Turners Falls (1970 population: 5,168), and 3.5 miles east-southeast of Greenfield (1970 population: 14,642). The minimum exclusion radius is 2,674 feet. The low population zone has a radius of 2.5 miles and a 1970 population of 4,476. The nearest center of population is Northampton, Massachusetts.

The Montague Units 1 and 2 each utilize a General Electric boiling water reactor (BWR-6) 3579 MWt nuclear steam supply system (NSSS) and a Mark III type containment. The NSSS design is the same as that utilized for the GESSAR-238 Nuclear Island Standard Design. The Mark III type containment design is similar to the Stone & Webster design utilized for the River Bend Station. The latest ACRS reports relative to nuclear generating stations utilizing the BWR-6/Mark III systems are the May 12, 1975 report on the Perry Nuclear Power Plant, the January 14, 1975 report on the River Bend Station, the May 13, 1976 report on the Hartsville Nuclear Plant, and the March 14, 1975 report on the GESSAR-238 Nuclear Island.

The NRC Staff has completed a review related to the construction of the Montague Units based upon current information and current safety considerations. In view of a several year delay in the start of construction of these units, the NRC will require an update of this review, to commence approximately one year before the anticipated date for decision on issuance of construction permits. The ACRS will complete its review of this application at this time. All significant safety considerations identified in the interim will be included in the updated review.

The Mark III containment design has been under continuing review by the NRC Staff and the Committee. Results from tests made to date by the General Electric Company have led to criteria which are believed to be sufficiently conservative to allow for uncertainties in the currently applied empirical design methods. The ACRS anticipates that the remaining tests of the program proposed by the General Electric Company will provide a basis for confirming the adequacy of the Mark III containment design for the Montague Station.

The Committee believes that the Applicants and the NRC Staff should further review the Montague Units for design features that could significantly reduce the possibility and consequences of sabotage, and that such features should be incorporated into the plant design where practicable.

Other generic problems relating to large water reactors are discussed in the Committee's report dated April 16, 1976. Those problems relevant to the Montague Station should be dealt with appropriately by the NRC Staff and the Applicants as solutions are found. The relevant items are: II-1, 2, 4, 5, 6, 7, 8, 9, 10, 11; II-A-1, 2, 4, 6, 8; II-B-2, 3, 4; and II-C-1, 2, 4, 6, 7.

The Advisory Committee on Reactor Safeguards believes that the above items can be resolved by the Applicants and the NRC Staff. Subject to the satisfactory resolution of these items and any new safety considerations that develop between now and the completion of the Committee's review of this application, the Committee believes that the Montague Power 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,

Dade W. Moeller

Dade W. Moeller
Chairman

References:

1. Preliminary Safety Analysis Report (PSAR) for the Montague Nuclear Power Station, Units 1 and 2, Volumes 1-11 and Amendments 1-14
2. Safety Evaluation Report, NUREG-0091, related to construction of Montague Nuclear Power Station, Units 1 and 2, July 1976
3. General Electric BWR/6 Standard Safety Analysis Report, Volumes 1-9 and Question and Response Guide, Volumes 1 & 2 and Amendments 1-44
4. Written statement received from Mr. George O'Brien, representing the International Brotherhood of Electrical Workers Local Union #36, dated August 27, 1976
5. Written statement received from Ms. Elizabeth Bell, dated August 26, 1976
6. Written statement received from Mr. Robert May, dated August 11, 1976
7. Written statement received from Ms. Joanne Katz, dated August 6, 1976
8. Written statement received from Ms. Juanita Nelson, dated August 26, 1976
9. Written statement received from Mr. Wallace F. Nelson, dated August 26, 1976
10. Written statement received from Mr. Robert E. Murphy, dated August 26, 1976
11. Written statement received from Mr. William Hefner, dated August 26, 1976
12. Written statement received from 100 members of the public, dated August 25, 1976
13. Written statement received from 27 members of the public, dated August 25, 1976

Revised Page

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

May 11, 1966

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

Subject: REPORT ON REACTOR SITE FOR NORTHERN STATES POWER COMPANY

Dear Dr. Seaborg:

At its seventy-third meeting on May 5-7, 1966, the Advisory Committee on Reactor Safeguards reviewed a site proposed by the Northern States Power Company (NSP) for construction of a boiling water reactor to be operated initially at about 1469 MW(t). The Committee had the benefit of a Subcommittee meeting on May 4, 1966, of discussion with representatives of NSP, General Electric Company, and the AEC Regulatory Staff, and of the document referenced below.

The proposed site of 1325 acres is on the Mississippi River about three miles northwest of Monticello, Minnesota, approximately 40 miles northwest of the center of the Minneapolis-St. Paul metropolitan area, and 22 miles southeast of St. Cloud, Minnesota. The population density in the general vicinity of the site is low.

A few problems are presented by the maximum and minimum flows of the Mississippi River at this site. The peak flood of record (51,000 cfs) occurred in 1965 and resulted in a crest at elevation 916 feet (MSL) at the proposed site. The 1000-year flood level is estimated at elevation 921 feet. Ground elevations at the site range from 920 to 930 feet, and NSP stated that the reactor and appurtenant structures would be adequately protected against the 1000-year flood.

Another problem relates to minimum river flows. It was reported that the condenser cooling circuit will require about 1000 cfs. River discharges, however, have been as low as 240 cfs and remain below 1100 cfs about 10 percent of the time. For this reason, NSP proposes to utilize cooling towers for recirculation of condenser cooling water during periods of low river flow. During such periods the volume of discharged water into which liquid wastes can be diluted will be greatly reduced.

Water supplies for the cities of Minneapolis and St. Paul are taken from the Mississippi River about 30 miles downstream from the proposed site. NSP and GE propose to include facilities for liquid waste treatment that will be more than adequate to meet the requirements of 10 CFR Part 20. In addition, however, the Committee suggests that NSP consider the desirability of providing supplementary facilities for retention of liquid wastes during periods of low river flow.

The site is underlain by 50 to 80 feet of glacial drift supported on a layer of sandstone 10 to 20 feet thick. Seismic activity in Minnesota has been extremely rare and of minor intensity, but, inasmuch as a detailed seismological report for this site is not yet available, seismic design criteria remain to be established. In addition, consideration should be given to potential damage from tornadoes, which occur frequently in this area.

Transportation of a shop-fabricated pressure vessel of the anticipated size to the proposed site presents many difficulties. NSP and GE are tentatively considering field fabrication of the pressure vessel at the reactor site. The Committee has not reviewed the suitability of a field-fabricated pressure vessel, and, consequently, approval of this site does not imply concurrence with the concept of field fabrication of this component.

The Advisory Committee on Reactor Safeguards believes that the Monticello site is acceptable for a reactor of the general type and power level proposed, if adequate containment and associated engineering safeguards are provided.

Sincerely yours,

/s/

David Okrent
Chairman

Reference:
"Preliminary Information for a Proposed Nuclear Power Plant", Northern States Power Company, March 15, 1966.

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

April 13, 1967

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

Subject: REPORT ON MONTICELLO NUCLEAR GENERATING PLANT, UNIT 1

Dear Dr. Seaborg:

At its eighty-second meeting, on February 8-11, 1967, and its eighty-fourth meeting, on April 6-8, 1967, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Northern States Power Company to construct the Monticello Nuclear Generating Plant, Unit 1 on a site near Monticello, Minnesota. An ACRS Subcommittee met to review this project on February 3 and March 23, 1967. During its review, the Committee had the benefit of discussions with representatives of the applicant, the General Electric Company, Chicago Bridge & Iron Company, Bechtel Corporation, Harza Engineering, General Motors Corporation and the AEC Regulatory Staff and its consultants. The Committee also had the benefit of the documents listed. The Committee had previously conducted a site review of the proposed plant location and had transmitted its comments thereon to you by letter dated May 11, 1966.

The Monticello plant includes a boiling water reactor which the applicant proposes to operate at an initial power of 1469 MW(t) with a design stretch capability for operation at 1674 MW(t). In many respects the plant is similar to the plants proposed for Quad-Cities. However, this plant is the first United States nuclear plant to use a field-erected pressure vessel. Although field erection of large pressure vessels is new to the reactor industry, it is not a new procedure. With the fabrication techniques proposed and with meticulous care and diligence in the quality control program, it is the opinion of the ACRS that a high-quality field-erected pressure vessel for the Monticello plant can be constructed. The Committee recommends that the stress analysis report for the reactor vessel be reviewed by independent experts.

The emergency core cooling systems include a high pressure coolant injection system, a low pressure coolant system, two core spray systems, and a system that will make river water available to the feedwater pumps. In the unlikely event of a steam line rupture external to the reactor containment, steam line isolation valves must close rapidly. It is our understanding that valves of essentially identical design will be tested under simulated accident conditions. It is recommended that the Regulatory Staff satisfy itself with respect to the adequacy of the isolation valve test program and follow the development of the detailed design of the above systems.

It is of great importance that sufficient electrical power is available at the plant to operate emergency core cooling equipment in the unlikely event of loss of normal coolant to the core. Although the reliability of off-site power was stated to be very high, it is the recommendation of the ACRS that the Monticello plant include a second diesel generator of the same capacity as the one proposed.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during construction and that the proposed reactor can be constructed at the Monticello site with reasonable assurance that it can be operated at power levels up to 1469 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/

N. J. Palladino
Chairman

References Attached.

References - Monticello

1. Northern States Power Company letter dated August 1, 1966 to AEC Division of Reactor Licensing.
2. General Electric Company letter dated August 5, 1966 to AEC Division of Reactor Licensing transmitting "NSP-Monticello Nuclear Generating Plant, Monticello, Minnesota, Unit 1, Facility Description and Safety Analysis Report", Volumes I and II.
3. Northern States Power Company letter dated September 8, 1966 to AEC Division of Reactor Licensing transmitting Amendment No. 1.
4. General Electric Company letter dated September 9, 1966 to AEC Division of Reactor Licensing, with enclosures.
5. "Design, Fabrication and Erection of the Reactor Vessel", undated, received November 28, 1966.
6. Northern States Power Company letter dated December 29, 1966 to AEC Division of Reactor Licensing transmitting Amendment No. 3.
7. General Electric Company letter dated December 30, 1966 to AEC Division of Reactor Licensing, with attachments.
8. Northern States Power Company letter dated January 10, 1967 to AEC Division of Reactor Licensing transmitting Amendment No. 4.
9. General Electric Company letter dated January 10, 1967 to AEC Division of Reactor Licensing with attachments.
10. Northern States Power Company letter dated January 19, 1967 to AEC Division of Reactor Licensing transmitting Amendment No. 5.
11. General Electric Company letter dated January 21, 1967 to AEC Division of Reactor Licensing with attachments.
12. Northern States Power Company letter dated March 3, 1967 to AEC Division of Reactor Licensing transmitting Amendment No. 6.
13. "Amendment 6, Answers to AEC Questions", dated March 7, 1967.
14. "Amendment 6, Errata and Addenda Sheet", with attachments, dated March 8, 1967.
15. Northern States Power Company letter dated March 28, 1967 to AEC Division of Reactor Licensing transmitting Amendment No. 8.
16. General Electric Company letter dated March 29, 1967 to AEC Division of Reactor Licensing, with enclosures.

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

January 10, 1970

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

Subject: REPORT ON MONTICELLO NUCLEAR GENERATING PLANT

Dear Dr. Seaborg:

At its 117th meeting, January 8-10, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by the Northern States Power Company for a license to operate Unit 1 of its Monticello Nuclear Generating Plant, a boiling-water reactor unit, at power levels up to 1670 MW(t). A Subcommittee meeting with the applicant was held at the site on November 29, 1969. In the course of the review, the Committee had the benefit of discussions with the applicant, the General Electric Company, and their contractors and consultants; of discussions with the AEC Regulatory Staff; and of the documents listed.

The Committee reported to you on the Monticello site in its report of May 11, 1966, and on the construction permit application in its report of April 13, 1967. The Committee's review for construction was based on initial operation at 1469 MW(t); this report is based on the presently proposed power of 1670 MW(t) which the applicant justifies on the basis of more recent heat transfer correlations and development of the core design. In its April 13, 1967 report, the Committee recommended that the stress analysis report for the field-erected reactor vessel be reviewed by independent experts and that a duplicate diesel generator be installed. Both recommendations have been followed. The Committee is also satisfied that proper attention has been given to other matters referred to in its report. Several recommendations made by the Regulatory Staff and the Committee on recent applications have also been adopted in this plant.

The main steam lines are provided with redundant valves that are required to close automatically in the unlikely event of a serious accident. Because experience with these large and special valves is limited, the Committee recommends that their performance be followed closely, and that the applicant make additional provisions to assure the requisite leak-tightness if experience should be unfavorable. The Committee wishes to be kept informed of the resolution of this matter.

The General Electric Company has an extensive integrated program for measuring vibration in several reactors. A major program of vibration testing is planned for the Dresden 2 reactor and is expected to precede operation of the Monticello unit. The Committee believes that a limited program of vibration monitoring is appropriate for the Monticello reactor during preoperational tests and initial operation. In the event that the Dresden 2 data are not clearly favorable, or are not forthcoming before the Monticello unit is ready to operate, the Committee believes that the matter should be reviewed by the Regulatory Staff before routine full power operation of the Monticello unit.

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 and isolability of these lines and a program for the periodic testing and examination of the valves in these lines. The adequacy of measures taken with regard to such instrument lines should be confirmed by the Regulatory Staff.

Continuing research and engineering studies are expected to lead to enhancement of the safety of water-cooled reactors in other areas than those mentioned, for example, by the determination of the extent of the generation of hydrogen by radiolysis and by other sources in the unlikely event of a loss-of-coolant accident, development of instrumentation for in-service monitoring of the pressure vessel and other parts of the primary system for vibration and detection of loose parts in the system, by the development of further means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram during anticipated transients, and evaluation of the consequences of water contamination by structural materials and coatings in a loss-of-coolant accident. As solutions to the problems develop and are evaluated by the Regulatory Staff, appropriate action should be taken by the applicant on a reasonable time scale.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that Monticello Nuclear Generating Plant Unit 1 can be operated at power levels up to 1670 MW(t) without undue risk to the health and safety of the public.

Mr. Hill did not participate in the review of this project.

Sincerely yours,

/s/

Joseph M. Hendrie
Chairman

References:

1. Final Safety Analysis Report for the Monticello Nuclear Generating Plant Unit 1

2. Amendments No. 10-24 to license application

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

June 15, 1970

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

Subject: REPORT ON MONTICELLO NUCLEAR GENERATING PLANT UNIT 1

Dear Dr. Seaborg:

At its 121st meeting, May 7-9, 1970, and its 122nd meeting, June 11-13, 1970, the Advisory Committee on Reactor Safeguards met with Northern States Power Company to review proposed changes to the reactor vessel nozzle "safe ends" (stainless steel extensions of the nozzles) of the Monticello Nuclear Generating Plant Unit 1. During its review of the changes, the Committee had the benefit of discussions with the applicant, the General Electric Company, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported to you on operation of the Monticello Plant on January 10, 1970.

Normal procedures for most reactor pressure vessels have been to join the austenitic stainless steel safe ends to the nozzles prior to the stress relieving heat treatment. This heat treatment sensitizes the safe ends, which makes the steel less resistant to certain types of corrosion. Sensitized austenitic stainless steels in this condition have given reasonably satisfactory service over many reactor years of operation.

Recently, leaks developed in sensitized safe ends of two operating reactors. The causes of the leaks have been studied exhaustively, and it is concluded by the licensees that they were caused by unusual circumstances that need not have existed. In view of this experience, however, Northern States Power Company is making modifications to Unit 1.

In the Monticello vessel, eight safe ends were sensitized. The modifications consist of replacing six sensitized safe ends with unsensitized material and overlaying the other two with weld metal cladding of a composition that is resistant to stress corrosion. Some other components and attachments in the vessel are also being overlaid or replaced.

The Committee agrees with the applicant that these changes, properly executed, should increase assurance of trouble-free operation. The Committee wishes to call attention to other factors that would further tend to diminish the probability of a failure in a safe end or other piping component. The Committee believes an independent check should be made of stresses in the as-built piping of the primary system, and that displacements should be observed in the hot condition. A review should be made of high points in non-flowing parts of the system and means should be provided, where necessary, to vent or otherwise remove gases that could become trapped at such points.

The Committee also believes that the Regulatory Staff should assure itself that the biological shield surrounding the reactor vessel can withstand the pressure that could be developed by loss of integrity of a safe end or nozzle, or that failure of the shield would have no intolerable consequences.

The Committee has on several occasions stressed the importance of in-service inspection and leak detection. It recommends that the Regulatory Staff develop a schedule of inspections for safe ends. The operation of the leak detection and location systems should be reviewed and modified as appropriate to obtain the maximum speed and sensitivity for detection of leads. In addition, the applicant should study other techniques of detecting leaks.

Subject to these comments, and if due attention is paid to the items discussed in the Committee report of January 10, 1970, the Committee reaffirms its belief that there is reasonable assurance that the Monticello Nuclear Generating Plant Unit 1 can be operated at power levels up to 1670 MW(t) without undue risk to the health and safety of the public.

Mr. Hill did not participate in the review of this project.

Sincerely yours,

/s/ Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References:

1. Amendment 26 to the License Application, dated 5/7/70 -- Proposed modifications to the furnace sensitized stainless steel components attached to the reactor pressure vessel.

2. Amendment 27 to the License Application, dated 5/19/70

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

October 19, 1972

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

Subject: REPORT ON MONTICELLO NUCLEAR GENERATING PLANT, UNIT NO. 1

Dear Dr. Schlesinger:

At its 150th meeting, October 12-14, 1972, the Advisory Committee on Reactor Safeguards reviewed the application by the Northern States Power Company for conversion of its provisional operating license for the Monticello Nuclear Generating Plant, Unit No. 1 to a full-term operating license. This project was considered at Subcommittee meetings on September 11 and 30, 1972, in Washington, D. C. During its review, the Committee had the benefit of discussion with representatives and consultants of the Northern States Power Company, the General Electric Company, and the AEC Regulatory Staff, and of the documents listed. The Committee has reported to the Commission the results of its review of various aspects of this project in reports dated May 11, 1966, April 13, 1967, January 10, 1970, and June 15, 1970.

In its report of January 10, 1970, on the application for a provisional operating license, the Committee stated that the applicant had been responsive to recommendations made in the Committee's construction permit report, but made further specific recommendations relating to main steam line valves, vibration testing, and integrity and isolability of instrument lines. Operating experience suggests that continuing study and surveillance is necessary to assure satisfactory performance of the main steam line isolation valves. The vibration testing program during the preoperational period was satisfactory. The Committee believes the applicant should further evaluate the design of the instrument lines with respect to the Supplement to Safety Guide 11; the Committee wishes to be kept informed.

The Committee also called attention to the need for continuing evaluation and appropriate action with respect to problems common to water-cooled reactors. One of the items mentioned was the problem of hydrogen generation in the unlikely event of a loss-of-coolant accident. The applicant has described his studies for controlling hydrogen buildup, but has not submitted a firm proposal. The Committee believes the applicant should commit himself to completion of design and installation of an acceptable system on a time schedule satisfactory to the Regulatory Staff.

Another item specifically mentioned was the need for design features to make tolerable the consequences of failure to scram during anticipated transients. Studies by the reactor designer indicate that a system modification may accomplish the desired objective, but a final determination has not yet been made. The applicant has indicated that he will make the necessary modifications when a decision has been made on a generic basis.

Analyses of postulated control-rod drop accidents have been revised by the applicant to employ a more realistic rate of reactivity insertion than formerly assumed. These analyses indicate that, for accidents occurring during certain operations and certain portions of the fuel cycle, the results may be unacceptable. The applicant has proposed interim procedures which the Committee believes to be satisfactory. The final resolution should be made in a manner satisfactory to the Regulatory Staff.

Commercial operation of the plant started June 30, 1971. There have since been a number of unscheduled shutdowns caused by equipment or system malfunctions. The Committee recognizes that, during the early stages of operation of a large power plant, some forced shutdowns will occur and corrective action will be necessary. The Committee believes that the number of such events in the Monticello plant has not been excessive. However, the Committee wishes to reiterate its opinion that improvement of the plant and operating procedures to enchance safety should be a continuing process, factoring in technological advances and past and future industry-wide experience.

The Committee believes that the applicant should seek a careful and detailed delineation of responsibilities and authority for determining action levels, implementation, and coordination of the State and local agencies involved in emergency plans.

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, in view of the operating experience to date, and if due regard is given to the items mentioned above, there is reasonable assurance that Monticello Nuclear Generating Plant, Unit No. 1, can continue to operate at power levels up to 1670 MW(t) under a full-term operating license without undue risk to the health and safety of the public.

Mr. Hill did not participate in the review of this project.

Sincerely yours,

C. P. Siess
Chairman

References Attached.

References

1. Final Safety Analysis Report for Monticello Nuclear Generating Plant, Unit No. 1

2. Amendments No. 10-24, 26 and 27 to the license application

3. Northern States Power Company letter dated February 28, 1972 transmitting Six-Month Operating Report No. 2 for the period of July 1 to December 31, 1971

4. Northern States Power Company letter dated June 15, 1972 transmitting an Application to convert Provisional Operating License No. DPR-22 to a Full-Term Operating License for the Monticello Nuclear Generating Plant, Unit No. 1

5. Northern States Power Company letter dated August 30, 1972 transmitting Monticello Nuclear Generating Plant Six-Month Operating Report No. 3 for the period of January 1, 1972 through June 30, 1972

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

February 6, 1963

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

Subject: REPORT ON NASA MOCK-UP REACTOR (MUR)

Dear Dr. Seaborg:

At its forty-sixth meeting on January 31 - February 2, 1963, at the request of the Commission, the Advisory Committee on Reactor Safeguards considered the NASA Mock-up Reactor (MUR) and its relation to the NASA Plum Brook Reactor (PBR). The Committee last commented on the PBR in its letter April 4, 1962. The Committee had the benefit of discussions with representatives of the NASA Plum Brook facility, and the Regulatory Staff, and of reports referenced below.

The MUR is a mock-up of the PBR fueled core. Its operation is intended to determine critical loadings, reactivities, and other physics data for installations of fuel and experiments in the PBR. It is located within the canal in the PBR building but outside the PBR containment vessel. The MUR will operate at a nominal power of 100 KW. There is no significant coupling between the MUR and the PBR. The accidents that have been evaluated by the applicant would not melt any fuel elements of the MUR.

The Committee understands that except for the first slightly irradiated core the present proposal does not include experiments involving irradiated nuclear fuel or significant amounts of radioactive isotopes.

The Committee is of the opinion that the proposed operation of the MUR presents no undue hazard to the health and safety of the public. The Committee also concludes that operation of the MUR does not measurably increase the hazards from operation of the PBR.

Sincerely yours,

/s/

D. B. Hall
Chairman

References:

1. Ltr frm NASA to AEC, dtd Jan. 23, 1963 transmitting Amendment No. 3 to Application for Construction Permit, CPRR-62, Mock-up Reactor.
2. Ltr frm NASA to AEC, dtd Dec. 11, 1962 transmitting:

    (1) Final Hazards Summary Report, Mock-up Reactor, dtd Sept. 1962;

    (2) Technical Specifications for the NASA Mock-up Reactor, Appendix A, dtd Nov. 30, 1962.

3. Ltr from NASA to AEC, dtd Apr. 7, 1961 with enclosure: Amendment 1 to the License Application for NASA Mock-up Reactor.
4. Ltr frm NASA to AEC, dtd June 7, 1961 transmitting Amendment No. 2 to License Application for NASA Mock-up Reactor.
5. Preliminary Hazards Summary, Mock-up Reactor NASA Plum Brook Reactor Facility, dtd Jan. 1961.

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

November 5, 1957

Honorable Lewis L. Strauss
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: NATIONAL ADVISORY COMMITTEE FOR AERONAUTICS (NACA) - DOCKET NO. 50-30.

Dear Mr. Strauss:

This letter constitutes the report of the Advisory Committee on Reactor Safeguards on the application for a construction permit by the NACA Docket No. 50-30, in accordance with Section 182 of the Atomic Energy Act of 1954, as amended.

The application is for a test reactor designed to operate at power levels up to 60 megawatts of heat. It is to be located three miles south of Sandusky, Ohio.

One purpose of the reactor is to test nuclear fuel bearing components to destruction or near destruction. This aspect of the experimental program leads the Committee to be especially concerned with the operation of this reactor at a site so close to a densely populated area.

The Committee is of the opinion that with the proposed container and at the selected site it is possible so to restrict the experimental program that the operation of the reactor will not result in appreciable hazard to the public. However, the necessary restrictions may add materially to the cost of the program and may impose serious time delays. Further, some experiments which fall within the general type of experimental program proposed by NACA may not be permissible at this location.

In view of the above, the Committee believes that the facility proposed would be more useful for the program proposed if it were located at a site less close to a center of population.

It is the opinion of the Committee that NACA is providing reasonable precautions to avoid the escape of radioactivity which is likely to be damaging to the health and safety of the public. Among these precautions are three important items:

1) NACA proposes to place the reactor within a pressure vessel which has as its design criterion a maximum leakage rate of 115 cubic feet per day. Furthermore, the applicant has proposed a variety of measures to check the leak tightness of this container during operations. It is difficult to prove and maintain a leakage rate this low but if such a rate actually can be demonstrated and maintained the Committee believes that it should provide adequate protection to the health and safety of the public.

2) NACA is proposing to enclose each test loop within a secondary tank or container which is designed to contain the possible releases of fission products and other radioactive materials in case of breakdown of the fuel elements and other components being tested. The Committee believes that this would be a valuable additional safeguard but is not convinced that this secondary container can be depended upon under all circumstances.

3) The proposed design includes means to prevent the release directly to the atmosphere of effluents from the operation of the reactor or from the experimental loops. Again, the Committee agrees that this is an important safeguard but does not believe that accidental releases to the atmosphere can be entirely precluded.

The applicant proposes to establish a procedure for reviewing planned experiments in order to minimize the possibility of any failure which would release radioactivity even through the secondary enclosure.

The Committee believes that testing of fuel elements under conditions well within limits of possible failure does not offer a significant potential hazard provided that the experiments are properly designed and operated. However, testing of fuel elements in such a way that they are likely to be destroyed may not be permissible. Since NACA has not defined any specific experiments, the Committee is unable to state a more precise opinion than the above.

The Committee also believes that the operation of a test reactor at a site of this nature requires extensive area monitoring both on and off site so that any release of radioactivity to the environment may be detected as soon as possible and necessary protective or warning measures for the public carried out.

The Committee is aware of the risk that pressure may be brought to bear to permit a loosening of restrictions. This could come about as a result of a false sense of security which might develop from a period of successful operation and as a result of the importance of proposed experimental programs to the national defense. This problem would not be as serious if the proposed reactor were located at a less populated site.

The following are additional remarks by Dr. Abel Wolman:

"While I agree with all that the Committee has stated, I feel that I must add some remarks for purposes of clarifying my own position. In view of the prospect of future continuing debates as to the safety of conducting essential experiments at this site, I would recommend against the site on the information presently available. I believe that the applicant should be required to consider the availability of other sites at which operation of the reactor would be feasible and which would afford a higher degree of protection to the health and safety of the public.

"It is unrealistic to permit operation at this site if experiments of importance to the national defense are likely to have to be curtailed because of the site. The realities of human behavior are such that operation of experiments, the hazards of which may be uncertain, are likely to be permitted if they are important to the national defense.

"I do not believe that we should freeze on a site in a situation like this merely because an applicant has chosen it."

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman
Advisory Committee on
Reactor Safeguards

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

March 14, 1960

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

Subject: NASA PLUM BROOK REACTOR FACILITY

Dear Mr. McCone:

At its twenty-fourth meeting the Advisory Committee on Reactor Safeguards considered the design of this facility. This reactor was previously reviewed at its second meeting and advice given in a letter of November 5, 1957. The Committee had for reference the reports listed below and the benefit of comments from the Staff of AEC and others. An ACRS Subcommittee and members of the Hazards Evaluation Branch visited the site and observed and discussed the nearly completed facility. The proposed operation procedure was also briefly reviewed with the NASA Staff, who designed the reactor and will operate it.

This reactor is a 60 MW thermal test reactor similar to the MTR, designed to accommodate a large number of experiments including loop tests. It has a high degree of containment, and controlled holdup storage for gaseous and for liquid radioactive effluent. NASA proposes a gradual approach to its maximum power density which is higher than that in the MTR.

In its previous report, the Committee concurred in the necessity of controlled release of radioactive gas or liquid effluents as proposed by the NASA. This was deemed necessary due to the proximity of the City of Sandusky, Ohio. For the same reason, proposed experiments will have to be carefully reviewed and appropriate limitation may be necessary at this site.

The Committee considers that the design of the facility is satisfactory for the purposes intended. While not now commenting on the operating procedures or the design of experiments, the Committee believes that this reactor should be capable of being operated without undue hazard 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

<u>References</u>

1) Final Hazards Summary - NASA Plum Brook Reactor Facility, Parts I, II, and III, December 1959.

2) Amendment filed by NASA dated December 23, 1959.

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: NASA PLUM BROOK FACILITY

Dear Mr. McCone:

At its twenty-fourth meeting the Advisory Committee on Reactor Safeguards considered the design of this facility and, as stated in our letter of March 14, 1960, concluded that the design was acceptable for operation without in-pile experiments. At its twenty-seventh meeting, in Washington, D. C., on July 20-22, 1960, the Committee reconsidered the design of the facility in the light of recent test reactor experience and considered the proposed preliminary and normal operating procedure. The plans for in-pile experiments will be submitted later. The Committee had the benefit of the reports referenced below, a Subcommittee meeting with the NASA and AEC staff, and oral presentations by NASA and AEC staff.

The Committee is of the opinion that the modification of the supply piping and the storage tank venting of the primary coolant system, as proposed in Amendment No. 1, are desirable improvements.

While this is a testing reactor, located near Sandusky, Ohio, the Committee is of the opinion that the ability to make a delayed and controlled release of gaseous fission products, the provision of ample reserves of coolant water and the proposed step-wise approach to full power operations, will provide reasonable assurance that this reactor can be operated without undue hazard to the health and safety of the public.

Sincerely yours,

/s/
Leslie Silverman
Chairman

References:

(1) Amendment to Application for Construction Permit CPTR-3 for NASA Plum Brook Reactor, undated, (received July 21, 1960).
(2) Final Hazards Summary, NASA Plum Brook Reactor Facility, Supplement I, dated April 1960.
(3) Final Hazards Summary, NASA Plum Brook Reactor Facility, Supplement II, dated June 1960.

cc: A.R.Luedecke, GM
W.F.Finan, OGM
H.L.Price, 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, D. C.

Subject: NATIONAL AERONAUTICS AND SPACE ADMINISTRATION - (NASA)

Dear Mr. McCone:

At its twenty-fourth and twenty-seventh meetings the Advisory Committee on Reactor Safeguards considered the design and proposed operation of this facility, and in letters of March 14, 1960, and July 25, 1960, concluded that its operation, without in-pile experiments, would be acceptable. An additional amendment to the applicant's request for operating license (referenced below) proposed minor changes and improvements to the facility.

The Advisory Committee on Reactor Safeguards is of the opinion that the changes proposed in this amendment do not increase the previously appraised hazard to the health and safety of the public.

Sincerely yours,

/s/ Leslie Silverman

Leslie Silverman
Chairman

Ref: Amendment III undtd,
recd by ACRS 8/23/60

cc: A.R.Luedecke, Gen Mgr
W.F.Finan, AGM-R&S
H.L.Price, DL&R

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

November 7, 1960

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

Subject: REPORT ON NASA PLUM BROOK FACILITY

Dear Mr. McCone:

At its twenty-ninth meeting on November 3-5, 1960, the Advisory Committee on Reactor Safeguards considered Amendment IV to the NASA application for an operating license for the testing reactor at Plum Brook. Results of AEC staff studies were discussed.

The amendment proposes operation below 100 KW thermal prior to completion of some portions of the facility. This operation is for calibration of controls and checking projected performance. In-pile facilities requiring not more than 1% excess reactivity will also be checked. It is understood that the containment will have been completed and inspected, and will be capable of prompt manual closure in case of an accident.

The Committee suggests that the overpower scram be reset at just above the power limit requested and that one or more temporary instruments be inserted into the core so that water temperature can be observed during the time when the normal instrumentation is not useful and normal coolant flow is not available.

The Committee believes that there is reasonable assurance that the proposed operations can be carried out without undue hazard to the health and safety of the public.

Sincerely yours,

/s/
Leslie Silverman
Chairman

Reference:
Amendment IV to the Application for NASA Plum Brook Reactor, dated October 17, 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.

April 4, 1962

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

SUBJECT: REPORT ON PLUM BROOK FACILITY (NASA)

Dear Dr. Seaborg:

At its fortieth meeting on March 29-31, 1962, the Advisory Committee on Reactor Safeguards again considered the NASA Plum Brook test reactor. Proposed operation of this reactor was considered at the Committee's March 1960 meeting, and subsequent changes were reviewed in July and September of 1960. Advice was given in letters dated March 14, 1960, July 25, 1960, and September 26, 1960.

NASA has since made low power tests with results substantially as predicted. Additional reports and requests for changes, listed below, were available, as well as presentations by NASA and the AEC staff. NASA proposes operation up to full power of 60 megawatts with a step-wise approach to this power, and experiments which do not involve fuel irradiation or loop tests.

The Committee previously reported that operations of the type now proposed in this facility were not considered to cause undue hazard to the health and safety of the public. The changes described and the results of the low power tests do not alter this opinion.

Due to the proximity of the site to the City of Sandusky, the possible consequences of misoperation are somewhat more significant than in the case of many other reactors. The Committee, therefore, recommends that special consideration be given to maintaining the continuity of personnel.

Sincerely yours,

/s/

F. A. Gifford, Jr.
Chairman

References attached:

References:

1. Letter from NASA to AEC, dated June 29, 1961, re: Change #2 to License TR-3.
2. Letter from NASA to AEC, dated June 29, 1961, re: Change #3 to License TR-3.
3. Letter from NASA to AEC, dated July 24, 1961, re: Change #4 to License TR-3.
4. Letter from NASA to AEC, dated Oct. 16, 1961, re: Change #5 to License TR-3.
5. Letter from NASA to AEC, dated Mar. 16, 1962, re: Change #6 to License TR-3.
6. Letter from NASA to AEC, dated Feb. 5, 1962 transmitting report of Low Power Tests and Other Operations Pertinent to Safety for Plum Brook Reactor, dated Jan. 31, 1962.
7. Letter from NASA to AEC, dated Mar. 19, 1962, transmitting Appendix A - Technical Specifications for the NASA Plum Brook Reactor, dated March 19, 1962.
8. Changes to Technical Specifications, dated March 19, 1962 - 1 page.

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

February 19, 1964

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

Subject: REPORT ON NASA PLUM BROOK REACTOR FACILITY

Dear Dr. Seaborg:

At its fifty-third meeting on February 13-15, 1964, the Advisory Committee on Reactor Safeguards again considered operation of the 60 MW(t) test reactor at the NASA Plum Brook station near Sandusky, Ohio. The NASA wishes to convert its present provisional operating license to a ten-year operating license. The Committee heard oral presentations by the PBRF organization and the AEC staff, and had the reports cited below. The Committee last commented on the PBRF on April 4, 1962.

The facility has operated up to full power since April 21, 1963, and has carried out a number of irradiation tests. During the operation, a number of facility modifications have been made, and minor operating problems have been corrected. No major design or operating problems have been reported.

At PBRF, reviews of the design of experiments and of the associated operation are conducted by the facility Safeguards Committee which is made up of eight members: four from PBRF, two from Lewis Research Center, and two consultants who are not regular NASA employees. Ad hoc members are added as needed. The ACRS believes that a careful review of experiments and operations by the full facility Safeguards Committee is an important continuing function and that non-NASA members are necessary to a balanced observation and independent Committee action.

In its letter of November 5, 1957, the Committee noted that, due to site limitations, it might be unsafe to carry out some tests in this facility.

The Committee believes that, with careful planning and operation, the NASA PBRF organization can continue to operate this facility without undue hazard to the health and safety of the public.

Sincerely yours,

/s/
Herbert Kouts
Chairman

References

1. Letter from S. Neil Hosenball, NASA Lewis Research Center, to AEC Director, Division of Licensing and Regulation, dated January 10, 1964.
2. "Report on Approach to Power Test Program and Other Operations Pertinent to Safety for the NASA Plum Brook Reactor", dated May 27, 1963.
3. "Report of Reactor Operations for the NASA Plum Brook Reactor", dated November 18, 1963.

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 NASA PLUM BROOK REACTOR FACILITY

Dear Dr. Seaborg:

At its fifty-fourth meeting on April 2-4, 1964, the Advisory Committee on Reactor Safeguards considered a proposal by the NASA to conduct an encapsulated fuel testing experiment in the Plum Brook Reactor (PBR). This is the first fuel irradiation experiment that has been proposed for this facility. The Committee had the benefit of discussions with members of the PRB facility and the AEC staff, and of the reports listed.

In its letter of February 19, 1964, the Committee pointed out that, because of the limitations of the site, it might be unsafe to carry out some tests in this facility. The Committee is of the opinion that the experiment presently proposed does not present any undue hazard to the health and safety of the public.

Sincerely yours,

/s/
Herbert Kouts
Chairman

References:

1. NASA Change Report No. 20 to Provisional Operating License TR-3, Amendment No. 2, undated, received January 29 and February 11, 1964.
2. "Experiment Design Manual and Hazards Analysis for Irradiation of Refractory Fuel Compounds at High Specific Power to High Burnups", NASA Experiment No. 6215, Westinghouse Report No. X-AMS 91, Revision #2, dated October 31, 1963 (certain pages revised December 31, 1963).

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

October 15, 1964

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

Subject: REPORT ON THE NASA PLUM BROOK REACTOR

Dear Dr. Seaborg:

At its fifty-eighth meeting, October 7-10, 1964, the Advisory Committee on Reactor Safeguards considered the request of the National Aeronautics and Space Administration to conduct fueled and unfueled experiments in the In-Pile Helium Cooled Loop Facility of its Plum Brook Reactor. The Committee had the benefit of oral discussions with representatives of the applicant, the AEC Regulatory Staff, and of the documents listed below.

The Plum Brook Reactor has already been considered many times by the Committee, and reports have been written as listed below. Because the reactor is located three miles from Sandusky, Ohio, the Committee stated in the letter of March 14, 1960 "...proposed experiments will have to be carefully reviewed and appropriate limitation may be necessary at this site." The Committee suggests that general criteria for limitations on experiments be developed by the National Aeronautics and Space Administration. Where specific criteria have not been established, it is necessary to review new proposals for "fuel irradiation or loop tests" (See ACRS Report of April 4, 1962).

The In-Pile Helium Cooled Loop is the first fueled loop experiment to be installed at the Plum Brook Reactor. The proposed loop is multiply enclosed so that any fission products which accidentally escape from the fuel elements should not enter the main containment shell." Eight channels of appropriate instrumentation, which can initiate automatic shutdown (partial or total), are to be installed. The instrumentation, particularly its "fail-safe" nature, is to be tested on an electrically heated mock-up of the experiment.

In view of the proposed safety features, including those intended to prevent fuel failure, it is the opinion of the Advisory Committee on

Reactor Safeguards that the In-Pile Helium Cooled Loop can be operated as proposed in the Plum Brook Reactor without undue risk to the health and safety of the public.

Sincerely yours,

/s/
Herbert Kouts
Chairman

References Attached.

References:

1. Letter from Director, NASA Lewis Research Center, to Director, AEC Division of Reactor Licensing, dated June 25, 1964 with attached Change Report No. 24.

2. Design Manual and Hazards Report, In-Pile Helium Cooled Loop, Exp. No. 62-02, NASA Lewis Research Center, Volume I, Aerojet-General Corporation, dated March 1962, Revised June 1964.

3. Design Manual and Hazards Report for the In-Pile Helium Cooled Loop, Experiment No. 62-02, Appendix A, NASA Lewis Research Center, undated, received August 6, 1964 (C/RD).

4. Letter from Chief, Reactor Division, NASA Lewis Research Center - Plum Brook Station, to Director, AEC Division of Reactor Licensing, dated September 4, 1964, with enclosures.

Previous ACRS Reports

1. Letter from C. Rogers McCullough to Honorable Lewis L. Strauss, dated November 5, 1957.

2. Letter from Leslie Silverman to Honorable John A. McCone, dated March 14, 1960.

3. Letter from Leslie Silverman to Honorable John A. McCone, dated July 25, 1960.

4. Letter from Leslie Silverman to Honorable John A. McCone, dated September 26, 1960.

5. Letter from Leslie Silverman to Honorable John A. McCone, dated November 7, 1960.

6. Letter from F. A. Gifford, Jr. to Honorable Glenn T. Seaborg, dated April 4, 1962.

7. Letter from Herbert Kouts to Honorable Glenn T. Seaborg, dated February 19, 1964.

8. Letter from Herbert Kouts to Honorable Glenn T. Seaborg, dated April 9, 1964.

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

September 11, 1961

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

Subject: REPORT ON NATIONAL BUREAU OF STANDARDS REACTOR

Dear Dr. Seaborg:

At its thirty-sixth meeting, on September 7-9, 1961, the Advisory Committee on Reactor Safeguards considered the National Bureau of Standards Reactor (NBSR) on the basis of the documents referenced below, and discussions with representatives of the National Bureau of Standards and the staff of the AEC.

The NBSR will be a heavy-water moderated, tank-type research reactor with highly enriched MTR-type fuel elements. The design largely follows established precedents. The reactor will be housed in a concrete confinement shell kept at a slightly negative pressure by a filtered-exhaust system. An internal clean-up system will be provided which continuously removes radioactivity from the air inside the confinement shell in case of an emergency. The site will be on the NBS campus one mile from Gaithersburg, Maryland. Approval is being sought for operation up to 10 MW(th).

In the event of a release within the confinement building, the radioactivity exhausted to the atmosphere depends on several factors, the most important of which appear to be:

(a) the volume of air being exhausted to maintain an adequate negative pressure. This volume depends on the building's integrity;

(b) the integrity and efficiency of the filter and clean-up system;

(c) the reliability of the dynamic protection system.

The radioactivity exhausted to the atmosphere must be small enough so that a person located at or beyond the site boundary would not receive excessive radiation exposure, even in the case of a fuel meltdown under adverse meteorological conditions.

The ACRS recommends that the applicant demonstrate experimentally the integrity of the confinement building, the efficiency and adequacy of the exhaust and filter system, and the reliability of the dynamic protection system.

Subject to the above considerations, the ACRS believes that the proposed reactor can be constructed at the proposed location with reasonable assurance that it can be operated at a power up to 10 MW(th) without undue risk to the health and safety of the public.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

References:

1. NBSR-7 - Preliminary Hazards Summary Report, dated January 1, 1961.
2. NBSR-7A- Preliminary Hazards Summary Report, Supplement A, dated April 1, 1961.
3. NBSR-7B- Preliminary Hazards Summary Report, Supplement B, dated August 1, 1961.

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 NATIONAL BUREAU OF STANDARDS REACTOR

Dear Dr. Seaborg:

At its 46th meeting, on January 31 - February 2, 1963, the Advisory Committee on Reactor Safeguards, at the request of the Commission, reviewed certain questions that have been raised concerning the proposed National Bureau of Standards Reactor. In part, these questions were associated with the Committee's letter dated September 11, 1961, which suggested that there be tests of the engineered safeguards of this reactor. They also concerned the adequacy of the criteria that the applicant has tentatively chosen for selection of a general contractor.

The proposed tests of the leak-tightness of the reactor building have been discussed in Supplement C (August 1, 1962) to the Preliminary Hazards Summary Report. Tests of the kind described in this document should prove the desired integrity. The Committee in its earlier letter also suggested the demonstration of the efficiency and adequacy of the exhaust and ventilation system, and the reliability of the dynamic protection system. These demonstrations can be discussed at the operating license stage.

The criteria for contractor selection are contained in references 3 and 4 listed below.

The Advisory Committee on Reactor Safeguards believes that the selection criteria that have been proposed by the applicant are adequate for choosing a general contractor for this reactor and that the proposed reactor can be constructed at the Gaithersburg site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/

D. B. Hall
Chairman

References:
1. NBSR 7C - Preliminary Hazards Summary Report, Supplement C, dated August 1, 1962.
2. Contractor Selection Criteria & Other Special Conditions, undated, received January 22, 1963.
3. Letter from National Bureau of Standards, dated January 15, 1963, with Amendment "Contractor Selection Criteria" dated January 11, 1963.
4. Letter from National Bureau of Standards, dated January 22, 1963, transmitting Amendment to Contractor Selection Criteria, dated January 11, 1963.

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 THE NATIONAL BUREAU OF STANDARDS REACTOR

Dear Dr. Seaborg:

At its eighty-second meeting, February 8-11, 1967, the Advisory Committee on Reactor Safeguards reviewed the request for an operating license for the National Bureau of Standards Reactor (NBSR). This reactor, which is located at the National Bureau of Standards (NBS) facility near Gaithersburg, Maryland, is to be operated at a maximum power level of 10 MWt. The project was previously reviewed by the ACRS at the construction permit stage and discussed in Committee letters, dated September 11, 1961 and February 6, 1963. During the current review, a Subcommittee meeting was held in Washington, D. C. on February 2, 1967, and the Committee had the benefit of discussions with representatives of the National Bureau of Standards, the AEC Regulatory Staff, and of the documents listed below.

The Committee understands that NBS intends to make certain revisions in the reactor prior to start-up. These include the following:

(1) Installation of a safety valve to prevent possible overpressurization of the primary system;

(2) Removal of the helium bubbler backup shutdown system;

(3) Revisions in the protection instrumentation leading to greater redundancy;

(4) Revisions to permit use of radiation monitors in the ventilation ducts during confinement conditions.

The Committee understands that NBS will retain an operations consultant to direct the initial start-up and power operations.

In its construction permit letter of September 11, 1961, the Committee recommended tests to establish the integrity of the confinement building, the efficiency and adequacy of the exhaust and filter system, and the reliability of the dynamic protection system. NBS has conducted several tests covering these items. The Committee believes that these questions have been satisfactorily resolved.

The Committee has reviewed the monitoring facilities for radioactivity in the NBSR in both normal and confinement modes. It is satisfied that radioactivity releases can be monitored within the building and when exiting from the building stack.

NBS recognizes that the experiments performed in this reactor must be properly controlled. They have agreed to establish, in conjunction with the AEC Regulatory Staff, operating limitations on the possible energy releases associated with such experiments. The Committee recommends that any experiments involving the possibility of large chemical energy releases be referred to the AEC Regulatory Staff for review.

The NBS may install a cryogenic facility in this reactor in the future; the Committee has not reviewed NBSR with regard to the operation of this cryogenic facility. It suggests that the AEC Regulatory Staff review this facility at the appropriate time, and take necessary action.

It is the opinion of the ACRS that the reactor can be operated as proposed without undue risk to the health and safety of the public.

Mr. Harold Etherington did not participate in the ACRS review of the National Bureau of Standards Reactor.

Sincerely yours,

/s/

N. J. Palladino
Chairman

References:

1. Letter from National Bureau of Standards, dated April 6, 1966, with Final Safety Analysis Report on the National Bureau of Standards Reactor, NBSR 9.
2. Letter from National Bureau of Standards, dated October 4, 1966, with Supplement A of the Final Safety Analysis Report on the National Bureau of Standards Reactor, NBSR 9A, dated October 1, 1966.
3. Letter from National Bureau of Standards, dated December 21, 1966, with Supplement B of the Final Safety Analysis Report on the National Bureau of Standards Reactor, NBSR 9B, dated December 16, 1966.
4. Letter from National Bureau of Standards, dated January 23, 1967, with National Bureau of Standards Reactor Proposed Technical Specifications, NBSR-9C, dated January 19, 1967.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

March 12, 1970

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

Subject: REPORT ON NATIONAL BUREAU OF STANDARDS REACTOR

Dear Dr. Seaborg:

At its 119th meeting, on March 5-7, 1970, the Advisory Committee on Reactor Safeguards reviewed the proposal of the National Bureau of Standards to have their provisional operating license No. TR-5 converted to a full-term, 15-year operating license. An ACRS Subcommittee met at the reactor site on February 25, 1970 to discuss this matter. The Committee had the benefit of discussions with representatives of the National Bureau of Standards and the AEC Regulatory Staff, and of the documents listed below.

The applicant reviewed the startup and operating history of the reactor and discussed results of his environmental monitoring program. The startup program was carried out successfully and since June, 1969 the heavy water research reactor has operated at the design power of ten megawatts essentially without incident.

Based on its present review, the ACRS reaffirms its previous conclusion, that the National Bureau of Standards reactor can be operated without undue risk to the health and safety of the public, and recommends conversion of the current provisional operating license to a full-term operating license.

Mr. Harold Etherington did not participate in the review of this project.

Sincerely yours,

/s/
Joseph M. Hendrie
Chairman

References attached.

References - National Bureau of Standards Reactor

1. Letter from National Bureau of Standards, dated April 10, 1969; Request for Conversion of POL No. TR-5 to full-term operating license

2. Letter from National Bureau of Standards, dated January 15, 1970; Answers to AEC Questions re: Application for full-term operating license

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

January 17, 1984

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

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON LICENSE RENEWAL AND POWER INCREASE FOR THE NATIONAL BUREAU OF STANDARDS REACTOR

During its 285th meeting, January 12-14, 1984, the Advisory Committee on Reactor Safeguards reviewed the application for a license renewal and power increase for the National Bureau of Standards (NBS) Reactor. A Subcommittee meeting and tour of the facility were held in Gaithersburg, Maryland on December 21, 1983 to consider this matter. During its review, the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff and its consultants. The Committee also had the benefit of the documents referenced.

Initial planning for the NBS Reactor was begun in 1958. An application for a construction permit (CP) was filed in February 1961, reviewed by the Atomic Energy Commission Staff and the ACRS, and a CP was issued in April 1963. Application for an operating license (OL) (at 10 MW power) was reviewed by the ACRS in February 1967, and a provisional OL issued in November 1967. The reactor was first critical in December 1967 and achieved 10 MW power in February 1969. Routine operation at 10 MW began in September 1969. The ACRS reviewed and reaffirmed its approval of operation at 10 MW in March 1970, and a permanent facility license was issued in June 1970 for a period of 15 years. This license expires on June 30, 1985.

On December 2, 1980 the NBS submitted an application for license renewal for a period of 20 years and for a power increase from 10 to 20 MW. A final Environmental Statement (NUREG-0877) was issued in August 1982. In September 1983, the NRC Staff issued its Safety Evaluation Report (NUREG-1007). The only outstanding issue at the time of the SER was that of emergency plans. That matter is in the course of resolution.

The primary use of the NBS Reactor is to provide a very high flux of thermal neutrons for tests and experiments. This accounts for the choice of heavy water as the moderator, and also for the choice of the reactor fuel. The fuel consists of aluminum-clad plates containing highly enriched uranium.

In the NBS Reactor, heavy water is used to cool the core as well as to moderate and reflect the neutrons. The heavy water in the reactor vessel is circulated through a closed primary system including a heat exchanger in which the heat is transferred to ordinary water. The heat is then dispersed to the atmosphere in a cooling tower. The coolant enters the core at a temperature of about 100°F, and leaves at about 112°F in its present (10 MW) mode of operation. The maximum pressure in the primary coolant circuit is about 80 psig. There is thus very little stored energy and little likelihood of pipe rupture in the cooling system. There are no means of uncovering the core very rapidly, and at the same time reliable alternate means of maintaining core cooling are available. Nevertheless, a number of accident analyses including transients, design basis accidents, and a maximum credible accident have been carried out, assuming operation at 20 MW. In the worst case considered, the radiation exposure of a person remaining at the site boundary (approx. 400 m) for a period of 30 days would be about 1/3 of the maximum whole-body annual dose limit of 10 CFR 20 for a member of the public (approx. 0.17 vs. 0.5 rem) and the dose to the thyroid would be about 1/4 of the minimum of the thyroid dose range recommended by the Environmental Protection Agency for the establishment of an emergency planning zone (EPZ) (approx. 1.2 vs. 5 rems). Thus, the required EPZ falls entirely within the site boundary.

The operating experience since the beginning of routine full power operation approximately 14 years ago has been free of any serious problems. By the end of 1982 the reactor had produced about 800,000 MW-hrs. of energy which, at a power of 10 MW, would have required 80,000 hours of full power operation. The elapsed time, from the start of full power operation until the end of 1982, was approximately 120,000 hours. A continuing environmental surveillance program has been conducted in the neighborhood of the plant that began some years before full power operation without finding any radioactive material in vegetation, soil, or water which was attributable to reactor operation. The main radioactive species released to the atmosphere in normal operation have been argon-41 and tritium. The source of the tritium is neutron capture in the deuterium of the moderator, and a small amount of the heavy water is lost as water vapor. The radioactive argon results from neutron capture in the argon in normal air. Early in the operation, the space between the reactor vessel and the shield was filled with air, but a modification has been made to replace the air in this region by carbon dioxide with the result that the source of argon-41 has been reduced by a factor of about ten. With the present mode of operation the concentrations of these radioactive species at the site boundary are less than 1 percent of the maximum permissible concentrations stipulated in 10 CFR 20.

The changes involved in connection with the proposed increase in power are minimal. With the main exception of the cooling tower, the plant was designed for 20 MW operation. A new and more efficient cooling tower, adequate for 20 MW, has been installed. The original aluminum heat exchanger has been replaced by two 10 MW stainless steel heat exchangers. An additional pump has been installed to provide some increase in the coolant circulation rate. The fuel configuration and amount will not be changed.

The temperature increase in the coolant passing the core will rise from its present 12°F, but to only about 15°F. The release of radioactive argon-41 and tritium will increase, but will still result in off-site exposures no larger than 1 percent of those stipulated in 10 CFR 20.

In its present mode of operation, the facility has been fully occupied with a wide variety of measurements and experiments, and there is usually a backlog of experiment requests for several months. Most of the work done is in response to NBS needs for materials research, measurements of trace elements, neutron radiography, neutron flux standardization and so forth, but about 25 percent is done on behalf of other agencies such as the National Institutes of Health. All of the space available for experimental ports is already in use. By increasing the power (and doubling the flux) the time to conduct many of the present irradiations can be cut in half; and, although set-up times will not be changed, it is estimated that actual utilization will increase by a factor of 1.5 or so. It is also expected that the demand will immediately saturate the increased capacity. In addition, some new and complex procedures which are not presently feasible will become possible. For part of the work done at the NBS Reactor this facility is unique in the U. S., and there is no option to transfer to other facilities in the country. Extension of the operating license and an increase in the power of the NBS Reactor would clearly serve a useful purpose.

The ACRS believes that there is reasonable assurance that the renewal of the license for this reactor at the requested power level of 20 MW may be granted without involving any undue risk to the health and safety of the public.

Mr. H. Etherington, Member Emeritus, did not participate in the Committee's consideration of this matter.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

References:

1. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal and Power Increase for the National Bureau of Standards Reactor," USNRC Report NUREG-1007, dated September 1983
2. U. S. Nuclear Regulatory Commission, "Final Environmental Statement Related to the License Renewal and Power Increase for the National Bureau of Standards Reactor," USNRC Report NUREG-0877, dated August 1982

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON 25, D. C.

April 11, 1966

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

Subject: REPORT ON USE OF THE NATIONAL REACTOR TESTING STATION

Dear Dr. Seaborg:

Recent reviews of reactor projects at the National Reactor Testing Station (NRTS) have indicated a growing problem in connection with off-site dosage levels in the unlikely event of a severe reactor accident. Some of the newer uncontained reactors have power levels considerably higher than previous NRTS reactors and are potentially capable of generating off-site doses significantly in excess of the 10 CFR Part 100 guidelines.

As detailed in its individual reports to you on these reactors, the Advisory Committee on Reactor Safeguards has assured itself in each case that, despite the absence of conventional containment, adequate protection of the health and safety of the public against the consequences of a severe reactor accident exists, in the form of engineered safeguards, remoteness, and planned accident protection capability, including emergency evacuation procedures. With respect to the last point, on February 9, 1966, the Committee's Environmental Subcommittee reviewed the NRTS emergency and disaster plans with representatives of the AEC Idaho Operations Office (ID) Health and Safety Division, who are responsible for the formulation and execution of these procedures. The Subcommittee was in general favorably impressed with the scope and detail of the plans.

The Committee wishes to reaffirm its long held opinion of the suitability of the remote NRTS site for experimental reactor projects. However, it is the Committee's opinion that the presence of several reactors of the power levels now contemplated at NRTS makes it timely to re-evaluate and increase emphasis on accident protection and evacuation procedures for off-site populations. The Committee suggests that a coordinated review should be made by the various NRTS reactor operating groups, AEC-ID, and the AEC Regulatory Staff, of the possible need to extend the areas for which evacuation plans are maintained in a state of readiness and to assure that these plans would be implemented with dispatch in the unlikely event of a serious reactor accident. In addition, the Committee suggests that, if reactors of significantly higher power levels are to be operated at NRTS, additional protective measures, such as containment, should be considered.

Sincerely yours,

/s/
David Okrent
Chairman

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

July 13, 1978

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

SUBJECT: REPORT ON NEW ENGLAND POWER COMPANY NUCLEAR UNITS, NEP 1 AND 2

Dear Dr. Hendrie:

During its 219th meeting, July 6-8, 1978, the Advisory Committee on Reactor Safeguards reviewed the application of the New England Power Company and eight other utilities (Applicants) for a permit to construct the New England Power Company Nuclear Units, NEP 1 and 2.

The proposed site for the plant was visited by members of a Subcommittee on June 28, 1978, and a Subcommittee meeting was held in Warwick, Rhode Island on June 28 and June 29, 1978. During its review, the Committee had the benefit of discussions with the Nuclear Regulatory Commission (NRC) Staff, representatives of and consultants to the Applicants, the Yankee Atomic Electric Company, and the Westinghouse Electric Corporation, as well as comments from members of the public. The Committee also had the benefit of the documents listed.

The NEP units will be replicates of the Seabrook Station Units 1 and 2 on which the Committee reported in its letter of December 10, 1974. These units will utilize two four-loop Westinghouse pressurized water reactor nuclear steam supply systems each having a power level of 3411 MWt. Each unit utilizes the RESAR-3 Consolidated Version and is similar to the Marble Hill Station on which the Committee reported in its letter of October 22, 1976 and the Tyrone Energy Park on which the Committee reported in its letter of December 11, 1975.

The proposed plant will be located on a 549 acre site in the southern part of Washington County, Rhode Island adjacent to Block Island Sound. The proposed site is the location of the abandoned Charlestown Naval Auxiliary Landing Field and is presently owned by the General Services Administration. The site is approximately 35 miles south of Providence, Rhode Island and 18 miles west-southwest of Newport, Rhode Island. The contiguous communities of Westerly, Rhode Island and Pawcatuck, Connecticut (approximately 7.5 miles west of the site) have been designated as the nearest population center (1970 population 19,000; projected 1990 population 25,000). The minimum exclusion area boundary distance is 2130 feet from the center of either containment building and the low population zone radius is 1.5 miles. Land uses in the vicinity of the proposed plant site are primarily for residential and recreational activities.

The Applicants have proposed that horizontal ground accelerations of 0.15g and 0.075g, at the foundation level, are appropriate reference acceleration values for the safe shutdown earthquake and operating basis earthquake, respectively. However, since the NEP design is a replicate of the Seabrook design, the units will be designed for a safe shutdown earthquake acceleration of 0.25g at the foundation level and for an operating basis earthquake acceleration of 0.13g.

The Committee recommends that the Applicants and the NRC Staff evaluate the transient resulting from a loss of the combined offsite and onsite AC power systems as a function of time of system loss, as well as the capability of the plant to tolerate a loss of AC power for extended periods. If appropriate, design modifications to improve reactor capability in this regard should be developed. The Committee wishes to be kept informed.

Although committed to compliance with Regulatory Guide 1.97, Revision 1, the Applicants have taken exception to Position C.3 requiring extended range instrumentation. The intent of this position is to provide the facility operator with a capability for following the course of postulated accidents beyond the design basis accident. To assist in this matter, the Committee recommends that the NRC Staff provide the Applicants with an illustrative model showing an appropriate response to this position. The Committee wishes to be kept informed.

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 NEP Units are II-2, II-3, II-4, II-5B, II-6, II-7, II-9, II-10, II-A2, II-A3, II-A4, II-B2, II-C1, II-C2, II-C3A, II-C3B, II-C4, II-C5, II-C6, II-D2. 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, contingent upon the acquisition of the site by the Applicants, if due consideration is given to the foregoing, the New England Power Company Nuclear Units, NEP 1 & 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,

Stephen Lawroski

Stephen Lawroski
Chairman

References:

1. New England Power Company Preliminary Safety Analysis Report, NEP 1 and 2. Volumes 1 through 9 with Amendments N1 through N11.

2. Safety Evaluation Report related to construction of New England Power Project, Units 1 and 2, NUREG 0424, June 1978.

3. Westinghouse Electric Corporation Reference Safety Analysis Report, RESAR-3 Consolidated Version, Volumes I through VIII with Amendments 1 through 6.

4. Written statement from Dr. Clement A. Griscom, Division of Marine Resources, University of Rhode Island.

5. Written statement from Mr. James E. Hickey. Division of Occupational Health and Radiation Control, Department of Health, State of Rhode Island and Providence Plantations.

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

September 10, 1969

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

Subject: PUBLIC SERVICE ELECTRIC AND GAS COMPANY - NEWBOLD ISLAND SITE

Dear Dr. Seaborg:

At its 112th meeting, August 7-9, 1969, and its 113th meeting, September 4-6, 1969, the Advisory Committee on Reactor Safeguards considered the Newbold Island site, which the Public Service Electric and Gas Company proposes as the location for a nuclear power plant including two boiling water reactors of approximately 3400 MW(t) each. The site consists of approximately 500 acres located on Newbold Island in the Delaware River. A relatively high population density is associated with this site; it is 4-1/2 miles south of Trenton, New Jersey (1960 population - 114,000) and 11 miles northeast of Philadelphia, Pennsylvania (1960 population- 2,000,000). The nearest population center is a grouping of suburbs in Bucks County, Pennsylvania, known collectively as Levittown (1960 population - 70,000), with its nearest boundary 3.4 miles from the site. An ACRS subcommittee visited the site on July 1, 1969. During its review, the Committee had the benefit of discussions with representatives of Public Service Electric and Gas Company and their consultants, and the AEC Regulatory Staff, and of the documents listed below.

Preliminary studies of the geology, seismology, hydrology, and meteorology of the site have been made and have revealed no significant problems. Natural draft cooling towers will be used in the plant.

The conventional dry-well and suppression-chamber containment system will be enclosed in a low-leakage reactor building with air recirculation and filtration to reduce further the releases of radioactivity in the unlikely event of an accident. The Committee believes that the proposed containment system is a useful approach, but cannot comment at this time on its adequacy.

Special attention will be required with regard to the integrity of any portions of the primary system outside the containment and to the steam-line isolation valves. Appropriate additional means for coping with possible valve leakage or a loss of integrity outside the containment should be provided.

Public Service Electric and Gas Company described procedures involving additional hold-up of off-gas releases during routine plant operation. The Committee believes that special attention should be given to the control of liquid waste releases and to the prevention of radwaste accidents, as additional means of keeping radiological releases at a very low level.

The Committee believes that, for this site, additional study of the problems related to possible degradation of reactor vessel integrity, such as leaks and vessel wall ruptures, is needed. Measures that will ameliorate these problems should be implemented to the extent that they are practical and significant to public safety. The features provided should be of such design as to prevent their interference with other engineered safety features.

Other matters noted in previous ACRS letters pertaining to large water reactors should receive appropriately greater attention in the design of the plant. The Committee believes a more conservative approach is appropriate in the design of a plant at this site, with regard to the margins in the engineered safety systems, protection against possible internally-generated missiles, and the number of items to be resolved after the construction permit review.

The Committee emphasizes again the vital importance of quality assurance, and the necessity for adequate consideration of diverse and independent means of protection against common failure modes in safety systems.

The conclusion reached by the Committee regarding this site has been influenced in part by its expectation that some satisfactory experience will have been obtained with reactors of this general type by the time a construction permit is issued, and some satisfactory experience will have been obtained with reactors of this type having the same power and power density as those proposed for this site by the time an operating license is issued.

The Advisory Committee on Reactor Safeguards believes that, subject to the above comments, the Newbold Island site is not unacceptable with respect to the health and safety of the public, for a plant having the general characteristics described above and designed with due attention to the other matters discussed.

Sincerely yours,

/s/

Stephen H. Hanauer
Chairman

References attached.

References - Newbold Island Nuclear Generating Station

1. Newbold Island Nuclear Generating Station Site, Preliminary Site and Environment Description, April 1969.

2. Newbold Island Nuclear Generating Station Site, Preliminary Site and Reference Design Evaluation, April 1969.

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

June 14, 1971

H. L. Price, Director of Regulation

NEWBOLD ISLAND NUCLEAR GENERATING STATION

The ACRS believes that the biological shield for the Newbold Island Station should be strengthened in the manner described by the applicant in recent oral presentations, to provide additional resistance to internal jet forces.

The Committee has requested that this conclusion be transmitted to the applicant.

Original Signed by
R. F. Fraley

R. F. Fraley
Executive Secretary

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

July 13, 1971

H. L. Price, Director of Regulation

ACRS COMMENTS ON NEWBOLD ISLAND NUCLEAR GENERATING STATION

Based on discussion during the 135th ACRS meeting the Committee has reached the following conclusions with respect to the Newbold Island Station:

1) The changes in the ECCS proposed for this project meet Committee comments in its report of September 10, 1969, as they apply to emergency core cooling.

2) It is desirable to obtain as much flow as can be obtained from the HPCI system without making major system changes such as increasing the pump-turbine size.

3) The proposed containment design is considered acceptable.

Original Signed by
R. F. Fraley

R. F. Fraley
Executive Secretary

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

August 10, 1971

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

Subject: REPORT ON NEWBOLD ISLAND NUCLEAR GENERATING STATION UNITS NOS. 1 AND 2

Dear Dr. Seaborg:

At its 136th meeting, August 5-7, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application by the Public Service Electric and Gas Company for a permit to construct the duel-unit Newbold Island Nuclear Generating Station. This project was also considered at the 130th, 133rd, 134th, and 135th meetings of the Committee on February 4-6, May 6-8, June 10-12, and July 8-10, 1971, respectively; and at Subcommittee meetings on June 3, 1970 at Argonne National Laboratory, and on February 3, March 29, April 26, June 3, July 7, and August 4, 1971 in Washington, D. C. During its review the Committee had the benefit of discussions with representatives and consultants of the applicant, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below. The Committee reported the results of its pre-application site review to you in a letter dated September 10, 1969.

The station will be located in New Jersey on 530-acre Newbold Island which is near the east bank of the Delaware River about 4-1/2 miles south of Trenton, New Jersey (1970 population - 105,000) and 11 miles northeast of Philadelphia, Pennsylvania (1970 population - 2,000,000). The nearest population center is a grouping of suburbs in Bucks County, Pennsylvania, known collectively as Levittown (1970 population - 72,000), with its nearest boundary 3.4 miles from the site. The applicant has specified a radius of one mile for the low population zone, which had in 1969 a transient population associated with industry of approximately 1200, and a small resident population which is expected to be about 100 by 1985. The minimum exclusion distance is 700 meters, which extends to the west bank of the Delaware River. As pointed out in the Committee's report of September 10, 1969, a relatively high population density is associated with this site.

Each unit includes a boiling water reactor to be operated at 3293 MWt. With respect to core design, power level, and other features of the nuclear steam supply system, the Newbold Island units are essentially duplicates of the Browns Ferry Units 1, 2 and 3, and Peach Bottom Units 2 and 3. Waste heat from the station will be rejected to the atmosphere by natural draft cooling towers.

In its report of September 10, 1969, the Committee listed several matters which it believed warranted special attention in the design of a plant for the Newbold Island site. In response to these recommendations, the applicant has included in the Newbold Island design several features, in addition to those normally provided for boiling water reactor units, to reduce still further the potential for release of radioactivity to the environment. The principal additional features are described below:

Reactor Building. For each unit, the conventional steel drywell and suppression chamber primary containment, the fuel handling area and spent fuel pool, and the principal components of the engineered safety features are contained in an unlined reinforced concrete building of cylindrical shape with a domed roof. This building is designed to Class I seismic standards and to resist the standard tornado, and missiles from this or other sources. The building can resist an internal pressure of 2 psig, and inleakage at a differential pressure of 1/4-inch of water will be limited to 10 percent of the building volume per day. A filtration, recirculation, and ventilation system (FRVS) is provided to recirculate and filter the reactor building atmosphere and maintain the building at a negative pressure relative to the outside environment.

Main Steam Lines. A low-leakage, slow-acting, stop valve has been added downstream of the two fast-acting valves in each main steam line, and a seal air system has been provided to further reduce leakage of radioactivity after main steam line isolation. The portion of the main steam lines containing the isolation valves is enclosed in a Seismic Class I tunnel chamber connected to the reactor building so that any out-leakage following the unlikely event of a design basis loss-of-coolant accident will be treated by the reactor building FRVS before release to the atmosphere. The entire length of the main steam lines up to and including the turbine stop valve will be designed to Class I seismic standards. The main steam lines from the third isolation valve to the turbine stop valve will be designed and fabricated in substantial accordance with the requirements for AEC quality assurance Classification Group B. In addition, selective inspection of critical areas of this piping will be performed during refueling outages.

Radioactive Waste Disposal. The radioactive waste disposal systems include several features beyond those normally provided in boiling water reactor plants. The liquid waste system permits the recycling of equipment and floor drain wastes and the evaporation of chemical and laundry wastes before discharge to the environment. The gaseous waste system provides for the recombining of hydrogen and oxygen, condensing the vapor, hold-up for decay of short-leved isotopes, and cryogenic separation of the noble gases. Krypton and xenon may be stored for periods sufficiently long that krypton-85 becomes the only significant remaining radioisotope. Provisions will be made to utilize non-radioactive steam in the turbine gland seals and to process containment purge gases when deinerting. The Committee believes that these waste management systems are capable of limiting releases of radioactivity to the environment to levels that are as low as practicable.

Reactor Vessel Integrity. The applicant has described improvements in the design and fabrication of the reactor vessel. These include redesign of the large nozzles to reduce stress concentrations; redesign of the bottom head to reduce the number of welds and improve the capability for in-service inspection; and improved procedures and standards for inspection during fabrication. The applicant has studied the problems related to possible degradation of reactor vessel integrity and has concluded that a nozzle failure or a small break would not impair the integrity of the biological shield, the primary containment, or the reactor internals, and would not affect the ability to cool the core. In addition, the biological shield has been redesigned to increase substantially its ability to withstand internal pressures, jet forces, or missiles.

Emergency Core Cooling System. The emergency core cooling system (ECCS) has been modified in two ways. The high-pressure coolant injection (HPCI) system has been changed to inject water directly to the core through the core spray sparger rather than into the downcomer region via the feedwater sparger. In addition, the applicant has stated that the steam-turbine driven HPCI pump will be modified to the extent feasible to increase the volume of water delivered to the core. The low-pressure coolant injection (LPCI) system has been changed to inject water inside the core shroud through four separate vessel penetrations, rather than through the recirculation lines. The applicant has stated that these changes provide increased reliability of these systems and reductions in the peak clad temperatures attained in the unlikely event of a loss-of-coolant accident.

The Committee believes that the design changes described above are suitably responsive to the concerns stated in its letter of September 10, 1969 regarding additional matters which should be considered for a plant at the Newbold Island site.

In the event of an unisolable break of an instrument line or a process line, reactor coolant will be discharged to the reactor building. Since the instrument lines will contain a 3/8-inch flow-restricting orifice inside the primary containment, failure of as many as eight such lines will not lead to pressures inside the reactor building greater than the 2 psig at which it relieves to the environment. However, failure of a process line, if not isolated in a very short time, could lead to pressures in excess of this relief pressure and significant amounts of reactor coolant would be discharged to the environment. Although the off-site doses from such an accident would be well within the 10 CFR Part 100 guidelines, they would be comparable to or greater than the doses calculated for other less probable accidents. The Committee believes, therefore, that the applicant should make design provisions for reducing the quantity of reactor coolant discharged to the reactor building in the event of a process line break.

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 Newbold Island reactors. 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 AEC Regulatory Staff and the ACRS during construction of the plant.

The applicant has stated that 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 proposed system is not capable of coping with hydrogen generation rates in accordance with current AEC criteria unless the primary containment is inerted. Therefore, 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 AEC Safety Guide 7, "Control of Combustible Gas Concentrations in 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 Newbold Island 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 Newbold Island Nuclear Generating Station Units Nos. 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 comments by Dr. H. O. Monson, Dr. D. Okrent and Dean N. J. Palladino are attached.

Sincerely yours,

/s/

Spencer H. Bush
Chairman

References - Newbold Island Nuclear Generating Station Units Nos. 1 and 2

1. Public Service Electric and Gas Company letter dated February 27, 1970; License Application; Preliminary Safety Analysis Report (PSAR), Volumes 1 through 5

2. Amendments Nos. 1 through 5 and Nos. 7 through 9 to PSAR

ADDITIONAL COMMENTS BY DR. H. O. MONSON,
DR. D. OKRENT AND DEAN N. J. PALLADINO

Although the large, low pressure, high in-leakage secondary reactor building proposed by the applicant for Newbold Island Units 1 and 2 represents an improvement over reactor buildings currently employed for BWRs at sites with lower surrounding population densities, we believe that further improvement is appropriate. The relatively small volume of the steel pressure-suppression type primary containment introduces some crowding of equipment and some attendant problems in the simultaneous accomplishment of full protection against violation of primary containment by possible missiles, jet forces, and pipe whip, and accomplishment of full access for in-service inspection. Some further protection would be provided against extremely low-probability accidents involving a concurrent loss of primary system integrity and a limited violation of primary containment by the use of a large, relatively high-pressure (of the order of 10 psi, as has been proposed for a BWR at another site having a comparable surrounding population density), low-leakage, secondary containment building. Such a high-pressure, secondary containment, coupled with a pressure-suppression primary containment, provides a combination which can tolerate a fairly substantial violation of primary containment arising from the same event which caused a loss of coolant, as well as further protection against unforeseen events. We believe that this improvement in safety capability is warranted for a more densely populated site like Newbold Island, and recommend that the issuance of a construction permit be contingent on the use of a high-pressure, low-leakage secondary containment.

For postulated loss-of-coolant accidents involving small break sizes, the high-pressure coolant injection system (HPCI) arranged so as to inject into one of the core spray loops is predicted by the applicant to be highly effective in limiting peak clad temperatures to moderate levels. We believe that for a high power, high-power-density reactor at a site as densely populated as Newbold Island, the applicant should give further consideration to the use of an HPCI system on the second core spray loop. The purpose would be to provide redundancy of this means of protection in the event that the single HPCI system became ineffective because of failure of an HPCI component or because the accident arose from rupture of the core spray line into which the HPCI injects. The automatic depressurization system which together with the low-pressure emergency cooling systems constitutes an alternate means for coping with small breaks, albeit by introducing a larger opening, would continue to serve as a backup.

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

July 17, 1973

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

Subject: SUPPLEMENTAL REPORT ON NEWBOLD ISLAND NUCLEAR GENERATING STATION UNITS NOS. 1 AND 2

Dear Dr. Ray:

At its 159th meeting, July 12-14, 1973, the Advisory Committee on Reactor Safeguards reviewed additional information on population distribution and preliminary emergency plans submitted by the Public Service Electric and Gas Company in connection with its application for a permit to construct the Newbold Island Nuclear Generating Station. These matters were considered also at Subcommittee meetings on October 24, 1972 and June 6, 1973 in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives and consultants of the Public Service Electric and Gas Company, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee has previously reported the results of its pre-application site review and its construction permit review in letters dated September 10, 1969 and August 10, 1971.

In accordance with an order of the Atomic Safety and Licensing Board, the applicant has made a new and more detailed study of the present and projected population distributions in the area surrounding the site. This study includes, but has not been limited to, the proposed real estate developments by the Warner Realty Investment Company (WRIC). The Committee believes that the results of this revised population study do not differ from those previously considered by such an amount or in such a manner as to change its previous opinion that the site is acceptable, subject to the other considerations noted in its letter of August 10, 1971.

The applicant has prepared a preliminary emergency plan which considers, among other things, the feasibility of evacuating the population within the Low Population Zone (LPZ) in the unlikely event of a major accidental release of radioactivity from the plant. The applicant has also described studies of the feasibility of evacuating an area extending as much as three miles from the plant, assuming the projected population that would result from the full development envisioned by the WRIC, and has concluded that such evacuation is feasible. Detailed emergency plans, to be developed by the State of New Jersey and the Commonwealth of Pennsylvania, have not yet been completed.

The Committee concludes that a suitable emergency plan can be developed for the Newbold Island site. The Committee believes also that plans for appropriate protective measures should extend several miles beyond the proposed LPZ radius of one mile. It is essential also that plant personnel be provided with those instruments, indicators, and measurements that will define clearly the nature and course of an accident so that off-site emergency plans can be initiated at a level and on a time scale consistent with the severity or potential severity of the accident.

The Committee believes that the on-site power system for this plant should be capable of coping with a postulated event involving a LOCA in one unit and a concurrent spurious ECCS actuation signal from the other unit, assuming the failure of a single active component, other than the spurious signal.

The applicant stated that it is a design criterion that failure of components not designed to seismic Category I requirements will not prevent safe shut down or the proper function of any engineered safety features, and that the necessary attention will be given to this aspect of design.

The Committee recommends that further studies be made of methods to enhance the reliability of isolation of low-pressure systems, such as the residual heat removal system, from the primary system while the latter is pressurized, and that such methods as are practical be implemented.

The applicant is giving careful attention to the control of access to the plant and its vital components. The Committee recommends that deliberate consideration be given to other aspects of design and layout that could improve plant security.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items as well as those mentioned in previous reports, the Newbold Island Nuclear Generating Station Units Nos. 1 and 2 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Dr. H. O. Monson, Dr. D. Okrent, and Dean N. J. Palladino, whose additional comments were appended to the Committee's letter of August 10, 1971, believe that those additional comments are still applicable to the Newbold Island Station.

Additional comments by Dean N. J. Palladino are presented below.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

Additional Comments by Dean N. J. Palladino

"In the belief that a more conservative approach is appropriate in the design of a nuclear power plant at this site, I recommend that separate and independent on-site a.c. emergency power systems be provided for each unit of this plant."

References attached.

<u>References</u>

1. Preliminary Safety Analysis Report (PSAR), Volumes 1 through 5.

2. Amendments 1 through 5, 7 through 9, and 11.

3. Directorate of Licensing Safety Evaluation Report dated December 17, 1971.

4. Reid and Priest letter dated August 18, 1972 to the Atomic Safety & Licensing Board submitting a document entitled "Population Estimates and Projections for Newbold Island Region" dated July 11, 1972.

5. Reid and Priest letter dated September 20, 1972 to the Atomic Safety & Licensing Board submitting a document entitled "Preliminary Emergency Plan," dated September 15, 1972, which completes the applicant's response to order, dated July 14, 1972.

6. Public Service Electric and Gas Company letters dated January 10, 1973 and May 1, 1973.

7. Directorate of Licensing Supplement No. 1 to the Safety Evaluation dated May 17, 1973.

8. State of New Jersey, Department of Environmental Protection letter dated June 29, 1973 regarding "Procedures for Implementing Protective Action Guides".

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 NINE MILE POINT NUCLEAR STATION

Dear Dr. Seaborg:

At its fifty-eighth meeting held on October 7-10, 1964, the Advisory Committee on Reactor Safeguards considered the application of the Niagara Mohawk Power Corporation to construct a nuclear power plant to be located on Lake Ontario about seven miles from the City of Oswego, New York. This nuclear plant is to be a direct cycle boiling water reactor capable of generating 1538 MW(t) and approximately 525 MW(e). Pressure absorption containment will be used. The Committee had the benefit of oral presentations by representatives of the Niagara Mohawk Power Corporation, its contractors and consultants, the AEC Regulatory Staff and of the documents listed herewith. In addition, a subcommittee meeting was held at the site on June 10, 1964.

Proposed seismic design criteria are considered adequate. Calculated off-site exposures are based on limited data on halogen retention, and the consequences in the unlikely event of major accidents may be more severe than estimated. However, the Committee believes that more conservative assumptions would not make the proposal unacceptable. The general design and site appear to be satisfactory.

While all design details are not complete, there appear to be no insurmountable safety problems. However, technical analyses with emphasis on certain areas should be developed further as design and construction progress. These areas include: burn-out ratio as affected by recirculation control; limitations on the maximum reactivity of individual control rods; the poison injection system; the core spray system; adequacy of dry-well and suppression-pool heat removal systems; control and safety instrumentation; turbine by-pass action and its effect on core power; possibility of zirconium-water reaction with resulting hydrogen generation in the unlikely event of a major accident.

With due regard to the above comments, the Advisory Committee on Reactor Safeguards believes that the proposed reactor can be constructed at the site selected with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
Herbert Kouts
Chairman

References:

1. Exhibit D, Preliminary Hazards Summary Report, Nine Mile Point Nuclear Station, Volumes I and II, Niagara Mohawk Power Corporation, dated April 1964.

2. Exhibit D (Supp. 1), First Supplement to Preliminary Hazards Summary Report, Nine Mile Point Nuclear Station, Niagara Mohawk Power Corporation, dated August 1964.

April 17, 1969

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

Subject: REPORT ON NINE MILE POINT NUCLEAR STATION

Dear Dr. Seaborg:

During its 108th meeting, April 10-12, 1969, the Advisory Committee on Reactor Safeguards completed its review of the application by the Niagara Mohawk Power Corporation for a license to operate the Nine Mile Point Nuclear Station at power levels up to 1538 MW(t). During this review, the project was considered at Subcommittee meetings held on February 27, 1969 (at the site), and on April 8, 1969. In the course of these meetings, the Committee had the benefit of discussions with representatives and consultants of Niagara Mohawk Power Corporation, General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee previously discussed this project in a construction permit report dated October 15, 1964.

The Nine Mile Point Nuclear Station employs a boiling water reactor. Power level, core design, and other principal features of the nuclear steam supply system are generally similar to those for the Oyster Creek Nuclear Power Plant Unit No. 1, previously discussed in the Committee's report to you dated December 12, 1968.

As in Oyster Creek Unit No. 1, type 304 stainless steel utilized at a number of places in the reactor vessel was furnace-sensitized during fabrication. Careful examination of these parts for evidence of corrosion has been made by the applicant, and none has been found. Although the likelihood of occurrence of significant corrosion (intergranular attack) during the service life of the plant appears small, the applicant plans to install appropriate corrosion test specimens within the vessel for future examination. The Committee believes that the applicant should resolve with the AEC Regulatory Staff, prior to the start of operation, a satisfactory schedule and inspection procedure for at least the initial portion of this corrosion surveillance program.

The Committee wishes to emphasize the importance of periodic inspection of the high pressure coolant system in this and other reactors. The in-service inspection requirements for this reactor as described, and to be stated in the Technical Specifications, appear adequate for initial operation. The Committee agrees with the applicant's intention to review his inspection program after about five years of operation. Because of the difficulties inherent in direct inspection of the bulk of the welds in the reactor pressure vessel after the reactor is in service, it is strongly recommended that alternative means for assuring continued pressure vessel integrity be studied and implemented to the degree practical. In addition, the applicant should develop more specific plans for in-service inspection of the main steam lines beyond the second isolation valve.

The applicant plans to study supplemental and potentially more sensitive methods of primary system leak detection and to implement methods which provide significant improvements in measurement of leak rate, in the time needed to measure leak rate, or in distinguishing the nature of the leak. The applicant should report to the Regulatory Staff his progress in this area within a year after start of power operation.

Studies are continuing on the possible effects of radiolysis of water in the unlikely event of a loss-of-coolant accident. These studies should be evaluated by the Regulatory Staff and appropriate measures taken as deemed necessary. Such measures should make allowance for effects of hydrogen generated by metal-water reactions if the effectiveness of the emergency core cooling system should be less than that predicted by the applicant.

The applicant has stated that he plans to study possible means of instrumenting and 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, by the time of the first refueling outage, to implement such means as are found practical and appropriate.

The safety review and audit function proposed by the applicant appears to be satisfactory. However, the Committee recommends that membership of the Safety Review and Audit Board include one or more experts from outside the applicant's organization, at least for the first few years of operation, to aid in effecting sufficiently independent review.

The applicant indicates that instrumentation which senses radioactivity from the steam system can be used to provide early signs of gross failure of fuel elements. As operating experience is gained, he intends to improve the utilization of this type of instrumentation for this purpose. The Committee strongly endorses this effort.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, the Nine Mile Point Nuclear Station can be operated at power levels up to 1538 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Joseph M. Hendrie

Joseph M. Hendrie
Acting Chairman

References - Nine Mile Point Nuclear Station

1. Volumes I - IV, Final Safety Analysis Report.

2. First - Seventh Supplement to Final Safety Analysis Report.

3. Amendments 2 - 13, to Application for Licenses.

4. Final Safety Analysis Report - Nine Mile Point Nuclear Station - Technical Specifications (Revised), Draft - dated April 1969.

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

June 16, 1970

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

Subject: REPORT ON NINE MILE POINT NUCLEAR STATION

Dear Dr. Seaborg:

At its 122nd meeting, June 11-13, 1970, the Advisory Committee on Reactor Safeguards reviewed the program proposed by the Niagara Mohawk Power Corporation for restoration to service of the Nine Mile Point Station following the discovery during March, 1970, of cracks and leakage in a "safe end" (stainless steel extension of the reactor vessel nozzle). The program was also considered at Subcommittee meetings on May 5, 1970, and June 1 and 2, 1970. During its review, the Committee had the benefit of discussions with representatives of the applicant, the General Electric Company, the AEC Regulatory Staff, and their consultants, and of the documents listed. The Committee previously reported to you on this project on April 17, 1969.

Normal procedures for most reactor pressure vessels have been to join the austenitic stainless steel safe ends to the nozzles prior to the stress relieving heat treatment. This heat treatment sensitizes the safe ends, which makes the steel less resistant to certain types of corrosion. Sensitized austenitic stainless steels in this condition have given reasonably satisfactory service over many reactor years of operation.

The applicant and the General Electric Company have conducted an extensive investigation of the cracking and its causes. An independent stress analysis of the as-built piping has revealed that stresses in the cracked safe end, and one other safe end, were excessive. It is believed that this excessive stress, possibly in combination with a high concentration of oxygen in the non-flowing fluid in the pipe concerned, caused the intergranular cracking of the furnace-sensitized stainless steel safe end. Both of the overstressed safe ends have been removed and replaced with new ones made of unsensitized material. The thermal sleeves have been slotted to avoid the possibility of gas bubbles at the high points. The piping supports have been rearranged, and the entire primary system re-analyzed, for both hot and cold conditions, to give assurance that stresses will remain within allowable limits.

One other safe end made of sensitized material has been removed and examined, found not to contain cracks, and has been replaced with a new one of unsensitized material. All other safe ends made of sensitized material have been non-destructively tested. The minor defects found will be ground out before the reactor is operated again.

The applicant stated that expansions of primary piping will be measured during a hot functional test to be conducted prior to restarting the reactor, to check the pipe supports and the seismic restraints.

The applicant has proposed an augmented surveillance program for the sensitized safe ends remaining in the primary system, including non-destructive testing at least once a year and re-checking piping expansions for several full thermal cycles. The Regulatory Staff should assure itself that the details of the proposed program are appropriate.

The applicant is studying improved leak-detection methods. The Committee believes that detection and location of small leaks is an essential part of the surveillance program. The applicant should expeditiously install such leak-detection devices as seem likely to give improved sensitivity or speed of leak detection. The Committee recommends that at least one leak-detection system in addition to the proposed sump accumulation rate and dew point systems be installed within a few months and wishes to be kept informed of progress in this regard.

The Committee believes that the Regulatory Staff should assure itself that the biological shield surrounding the reactor vessel can withstand the pressure that could be developed by loss of integrity of a safe end or nozzle, or that failure of the shield would have no intolerable consequences.

The ACRS believes that, if due regard is given to the recommendations above and in its previous report to you of April 17, 1969, there is reasonable assurance that the Nine Mile Point Nuclear Station can be operated at power levels up to 1538 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References

1. Niagara Mohawk Power Corporation report, "Reactor Primary System Investigation at Nine Mile Point Nuclear Station," dated May 1, 1970.

2. Niagara Mohawk Power Corporation report, "Reactor Primary System Investigation at Nine Mile Point Nuclear Station, Report No. 2," dated May 11, 1970.

3. Niagara Mohawk Power Corporation report, "Program for Restoration to Service Based on Reports of Primary System Investigation Nine Mile Point Nuclear Station," dated May 11, 1970.

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

February 6, 1971

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

Subject: REPORT ON NINE MILE POINT NUCLEAR STATION

Dear Dr. Seaborg:

During its 128th meeting, December 10-12, 1970, and its 130th meeting, February 4-6, 1971, the Advisory Committee on Reactor Safeguards reviewed the application of the Niagara Mohawk Power Corporation for an increase in the licensed power level of the Nine Mile Point Nuclear Station from 1538 MW(t) to 1850 MW(t). The application was also considered at subcommittee meetings held in Washington, D. C. on December 9, 1970, and February 2, 1971. During its review, the Committee had the benefit of discussions with representatives of Niagara Mohawk Power Corporation, the General Electric Company, the AEC Regulatory Staff, and their consultants, and of the documents listed. The Committee previously reported to you on this project on June 16, 1970.

The proposed increase in power level is based in part on favorable preoperational test results and initial operating experience, and on use of an improved heat transfer correlation for evaluation of core thermal performance. Also, the normal reactor operating pressure will be increased from 1000 to 1030 psig, and a number of minor modifications to the plant will be made.

The applicant intends to install one additional safety valve (for a total of 16) on the reactor coolant system so as to meet at 1850 MW(t) the same design criterion for pressure relief as was met at the original power level.

Two new reactor scram trips will be added, one based on turbine stop valve closure and the other based on turbine control valve high rate of closure. Both trips will be operative at all power levels above 45 percent of full power, and are provided to assure that safety limits within the core are not exceeded during a transient resulting from turbine trip with assumed failure of the steam bypass valves to open.

Performance of the emergency core spray cooling system has been re-evaluated for 1850 MW(t) operation. The applicant proposes to revise time settings on the emergency power system so as to reduce core spray initiation time from 60 seconds to 35 seconds. With this change, and in light of results from the Commission's FLECHT Program, the core spray system appears acceptable for the proposed higher power operation. However, the Committee believes the applicant should continue to seek refinement in the models for evaluation of peak clad temperatures reached during postulated loss of coolant accidents. Also, confirmatory analyses currently underway by the Regulatory Staff should continue to be pursued.

Doses calculated for design basis accidents have also been reexamined for 1850 MW(t) operation. The applicant proposes to reduce the allowable containment leak rate from 1.6 to 1.5 percent per day (at 22 psig test pressure) and to maintain unchanged the existing primary coolant activity limits. With these provisions, the calculated doses based on the higher power level are no higher than those originally calculated for the stretch power rating of 1779 MW(t), and are within the 10 CFR 100 guidelines.

Further study by the applicant has indicated that adequate integrity of the spent fuel pool may not be assured in the postulated event of dropping of a fuel cask into the pool. Some possible corrective measures have been identified, and the applicant states that appropriate modifications to the plant will be made. The Regulatory Staff should follow this matter and assure implementation on an appropriate time scale.

The applicant has developed improved plans for in-service inspection of the main steam lines both inside and outside of containment. For piping beyond the second isolation valve, two welds in each pipe will be completely inspected by ultrasonic testing each year, with every such weld being so inspected at least once per eight years. This program will be initiated at the next plant outage.

Analyses by the applicant indicate that the biological shield surrounding the reactor can withstand satisfactorily the effects of failure of a reactor vessel safe end. The Regulatory Staff agrees with this conclusion.

The applicant has studied improved leak detection methods for use within the containment, and plans to supplement the existing systems. In addition to the sump accumulation rate and dew point measurement systems already in operation, he will install an atmospheric radioactivity monitoring system. This system will recirculate a portion of the containment atmosphere through an external loop and an air monitor. Installation is expected to be completed within a few months.

The Committee wishes to re-emphasize its belief that additional means for assuring continued reactor pressure vessel integrity, including possible improvement in access to the vessel surfaces for augmentation of in-service inspection, should be actively studied and implemented to the degree practical.

The applicant is actively studying means for control of buildup of hydrogen in the containment which might follow in the unlikely event of a loss of coolant accident. The Committee wishes to be kept informed of the resolution of this matter.

The applicant is continuing to study further means of preventing common failure modes from negating reactor scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee wishes to be kept informed of the resolution of this matter.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above and in its reports of April 17, 1969 and June 16, 1970, there is reasonable assurance that the Nine Mile Point Nuclear Station can be operated at power levels up to 1850 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H. Bush

Spencer H. Bush
Chairman

References - Nine Mile Point Nuclear Station

1. Niagara Mohawk Power Corporation Petition Requesting Amendment of License dated April 20, 1970, with Technical Supplement to Increase Power Level.

2. First through Fifth Addenda to Technical Supplement to Increase Power Level.

3. Niagara Mohawk Power Corporation letter dated November 23, 1970, forwarding corrections to Second Addendum to Technical Supplement to Petition to Increase Power Level.

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

July 17, 1973

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

Subject: REPORT ON NINE MILE POINT NUCLEAR STATION UNIT 2

Dear Dr. Ray:

At its 159th meeting, July 12-14, 1973, the Advisory Committee on Reactor Safeguards completed its review of the application by the Niagara Mohawk Power Corporation for a permit to construct Unit 2 of its Nine Mile Point Nuclear Station. A Subcommittee made a tour of the site of the proposed unit on June 15, 1973. The project was considered during a Subcommittee meeting in Washington, D. C., on June 26, 1973. During its review, the Committee had the benefit of discussions with representatives and consultants of the Niagara Mohawk Power Corporation, the General Electric Company, the Stone & Webster Engineering Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

Nine Mile Point Unit 2 will be located on the south shore of Lake Ontario in Oswego County, New York, about seven miles northeast of the city of Oswego. The reactor will be approximately 900 feet east of the Unit 1 reactor, which is in operation, and about 2300 feet west of the James A. FitzPatrick Nuclear Power Plant which is scheduled to start operation in the near future. The Nine Mile Point site consists of about 900 acres and is contiguous with the 700 acre FitzPatrick site. The minimum exclusion distance for Unit 2 is approximately 5,100 feet; the low population zone extends out to 3.8 miles. The applicant stated that Unit 2 will not share any safety-related facilities with either of the existing units.

Unit 2 will utilize a General Electric boiling water reactor of the 1969 product line (BWR/5) design, to be operated at power levels up to 3323 MWt (1086 MWe). Design of the nuclear steam supply system is similar to that of the Hanford No. 2 Nuclear Power Plant and the La Salle County Station units, previously approved for construction. An over-under type of pressure suppression system, with reinforced concrete primary containment, is used. The design, leak testing capability, and continued integrity of the seal for the peripheral joint between the drywell floor and wall is of particular importance. Excessive bypass leakage could interfere with the effectiveness of the pressure suppression function. The final design should be reviewed in detail by the Regulatory Staff.

It is anticipated by the applicant that, as a result of postulated storm conditions, the water level in Lake Ontario could rise to a point where flooding might be a problem on portions of the reactor site. To protect the site, the applicant proposes to expand the existing dike along the lake shore and to modify the drainage ditch behind it. Calculations by the Regulatory Staff and the applicant presently differ by three feet in the estimated maximum lake surge level under storm conditions. Studies are continuing in order to resolve this difference. The applicant has agreed that the final design and construction of the dike and drainage ditch will be accomplished in a manner acceptable to the Regulatory Staff.

The applicant proposes to install a water seal system, designed as an engineered safety feature, to minimize leakage through the main steam line isolation valves following a postulated loss-of-coolant accident. The Committee favors incorporation of a seal system for this purpose and recommends that the Regulatory Staff review carefully the final design to assure its adequacy.

The Committee recommends that further studies be made of methods to enhance the reliability of isolation of low pressure systems, such as the residual heat removal system, from the primary system while the latter is pressurized, and that such methods as are practical be implemented.

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 Nine Mile Point Nuclear Station Unit 2.

As indicated in its report on ECCS Interim Acceptance Criteria of January 7, 1972, the Committee believes that for plants for which construction permits are requested thereafter, design changes to improve ECCS capability should be sought and, to the extent practical, employed, irrespective of whether the plant design without such changes appears to meet the provisions of the Interim Acceptance Criteria. The Committee recognizes that there may be practical difficulties in fully implementing this recommendation for the Nine Mile Point Unit 2 plant, but believes the applicant should make a serious study of means to satisfy the desired objective. The Committee wishes to be kept informed concerning the applicant's efforts to improve emergency core cooling capability during the construction phase of this project.

The ACRS believes that the above items can be resolved during construction and that, if due consideration is given to these items, Nine Mile Point Unit 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,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References

1. Preliminary Safety Analysis Report for the Nine Mile Point Nuclear Station, Unit 2 - Volumes I through VIII.

2. Amendments 1 through 9 and 11 through 12.

3. Niagara Mohawk Power Corporation letter dated December 13, 1972 regarding stack height and turbine building ventilation.

4. Niagara Mohawk Power Corporation letter dated January 3, 1973 regarding fuel densification.

5. Niagara Mohawk Power Corporation letter dated May 21, 1973 submitting plant design modifications and supplemental information to the PSAR.

6. Directorate of Licensing Safety Evaluation Report dated June 15, 1973.

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

September 10, 1974

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

Subject: REPORT ON NINE MILE POINT NUCLEAR STATION UNIT 1

Dear Dr. Ray:

At its 173rd meeting, September 5-7, 1974, the Advisory Committee on Reactor Safeguards completed a review of the application by the Niagara Mohawk Power Corporation for conversion of its Nine Mile Point Nuclear Station Unit 1 provisional operating license to a full-term operating license. The application also was considered at a Subcommittee meeting in Washington, D. C. on July 29, 1974. During its review, the Committee had the benefit of discussions with representatives of the Niagara Mohawk Power Corporation, General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee previously discussed this project in an operating license report of April 17, 1969 and in subsequent reports dated June 16, 1970 and February 6, 1971.

In its review, the Committee evaluated the operation and performance of this unit with particular emphasis on the response of the applicant to past recommendations for improvements in safety related systems.

Unit 1 is a non-jet pump boiling water reactor of 1850 MW(t) rated power level. Commercial power operation of the plant was begun in December, 1969. The operating history of the unit has been generally satisfactory. However, a number of operating problems or design deficiencies have been encountered during the approximately five year period of power operation. Included among these are: cracking of a core spray nozzle safe end; development of cracks in the steam dryer assembly; control rod scram sluggishness; failure of some control rods to remain fully inserted after scram; increased control rod operating restrictions found necessary to assure protection for a postulated rod drop accident; feedwater control deficiency, with resultant flooding of steam lines; torus baffle dislocation by relief valve steam discharge into the torus; and, failure of a relief valve to reclose. All of these deficiencies appear to have been satisfactorily corrected. Reactor availability has averaged approximately 66%.

Difficulty also has been experienced in respect to repeated occurrences of excessive leakage rates of the main steam isolation valves under test conditions. The applicant now proposes to remachine the valve seats and plugs to an improved configuration and believes that this, together with the probable low levels of residual stresses now existing in these valves, will enable maintenance of acceptable leakage rates in the future. This matter should be followed closely by the Regulatory Staff.

A number of design improvements have been accomplished or committed to since operation began. Among the most significant of these from the point of view of safety are the following. The feedwater system has been modified also to serve as an additional emergency core cooling system for small breaks; emergency power for this system is supplied by an offsite source of hydroelectric power. A fuel cask drop protection system has been designed and approved, and installation will be completed before shipment of spent fuel is undertaken. A containment atmosphere dilution (CAD) system for combustible gas control will be installed and available for operation in 1976. An additional primary pressure boundary leak detection system has been added, and position indication in the control room for the containment vacuum breaker valves has been provided.

Approximately one-fifth of the reactor 7x7 fuel bundles have been replaced with 8x8 fuel; through additional reloads, the core eventually is to consist entirely of 8x8 fuel.

Because of the relatively limited accessibility for in-service inspection of the reactor pressure vessel, the Committee wishes to emphasize again its belief that additional means for assuring continued vessel integrity, including possible improvement in accessibility, should continue to be actively studied and implemented to the degree practical.

The Committee recommends that the Regulatory Staff and the applicant give further consideration to the possible advisability of additional backfitting of Unit 1 where significant and practical safety improvements can be made.

The Committee believes that, in view of the generally satisfactory operating experience to date and the improvements made in the plant as noted herein, and subject to the above comments and those in previous ACRS reports on this plant, there exists reasonable assurance that the Nine Mile Point Nuclear Station Unit 1, can continue to be operated at power levels up to 1850 MW(t) without undue risk to the health and safety of the public. The Committee concurs in conversion of the present provisional operating license to a full-term operating license.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References: See Page 3

References:

1. Niagara Mohawk Power Company Technical Supplement to Petition for Conversion from Provisional Operating License to Full-Term Operating License dated July 1972.

2. Applicant's Environmental Report, Operating License Stage, Conversion to Full-Term Operating License, June 1972.

3. Amendments 1 through 3 to Application for Full-Term Operating License.

4. Directorate of Licensing Safety Evaluation Report dated July 3, 1974.

5. Directorate of Licensing letter dated July 3, 1974 concerning list of outstanding items in connection with their review of application for Full-Term Operating License.

6. Niagara Mohawk Power Corporation letter dated November 20, 1972 concerning Fuel Densification and its Effect on Reactor Operation Including Transients and Postulated Loss-of-Coolant Accident.

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

August 20, 1970

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

Subject: REPORT ON NORTH ANNA POWER STATION UNITS 1 AND 2

Dear Dr. Seaborg:

At its 124th meeting, August 13-15, 1970 the Advisory Committee on Reactor Safegurads completed its review of the application of the Virginia Electric and Power Company for authorization to construct two nuclear units at its North Anna Power Station in Louisa County, Virginia. This project was considered at a Subcommittee meeting in Fredericksburg, Virginia on July 30, 1970 which included an inspection of the site. During its review, the Committee had the benefit of discussions with representatives of the Virginia Electric and Power Company, the Westinghouse Electric Corporation, the Stone and Webster Engineering Corporation, the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents listed.

The North Anna Power Station site comprises approximately 1075 acres located in the northeastern corner of Louisa County, Virginia adjacent to and south of the North Anna River. The nearest population center is Fredericksburg, Virginia, about 24 miles northeast of the site, with a population of approximately 15,000, in 1968. The low population zone, extending six miles from the site had a 1968 population of about 2,000. The minimum exclusion distance is about 0.84 miles. The region surrounding the site is rural and sparsely populated.

Each of the North Anna nuclear units will include a three-loop pressurized water reactor designed for an initial core power level of 2652 MWt. The nuclear steam supply systems and the emergency core cooling systems are essentially identical with those for the previously reviewed Surry Power Station Units 1 and 2 and Beaver Valley Power Station Unit 1 (ACRS reports of April 29, 1968 and March 12, 1970 respectively). The proposed power level and average power density are essentially the same as for Beaver Valley Unit 1. If measurements to be made in Beaver Valley or similar operating cores should not adequately confirm the basis for estimates of hot channel conditions used in the North Anna design, system modifications or restrictions on operations may be appropriate.

The subatmospheric containment systems are similar to those approved for the Surry Power Station Units 1 and 2. In the unlikely event of a loss-of-coolant accident, the pressure in the containment is quickly reduced to below atmospheric by operation of redundant containment spray systems which initially introduce chilled water and then cooled, recirculated water from the containment sump into the containment atmosphere. The spray systems thus provide the heat sink for steam condensation and pressure reduction in the containment. The applicant proposes to flow-test this system only once (during pre-operational testing), to maintain the system in a dry condition thereafter, and to perform periodic rotational tests and cable insulation tests to determine that the powered pumps will rotate. The Committee believes that these tests provide insufficient assurance that the vital containment spray systems will perform as designed. Provisions for appropriate periodic flow-testing of the containment spray systems should be incorporated into the design. This matter should be resolved during construction in a manner satisfactory to the Regulatory Staff. The Regulatory Staff should also review the containment design pressure to assure that an adequate margin of conservatism exists.

Cooling water for the North Anna reactors is supplied from a 13,000 acre reservoir formed by the construction of a dam across the North Anna River about five miles below the station site. Cooling water for both normal and emergency shutdown conditions is supplied by a separate seismic Class I Service Water Reservoir with makeup supplied from the North Anna Reservoir. The Committee believes that a second Class I source of emergency cooling water or its equivalent should be provided.

Cooling water and liquid wastes will be discharged into a series of three lagoons which flow into the North Anna Reservoir. Because of extended periods of low flow of the North Anna River, cooling water will be recirculated with a resulting potential buildup of long-lived radioisotopes in the reservoir. The concentrations are estimated to reach about 40 percent of those in the discharge canal. While the radioactivity concentrations expected in the canal are estimated by the applicant to be a small fraction of the 10 CFR 20 limits, limited dilution and the seasonally exposed near-shore lake bottom may tend to create an unsatisfactory external radiation exposure situation. Reconcentration factors and radiation exposure rates should be estimated for critical radioisotopes, such as cesium, and this information used in the design of the waste treatment system.

The applicant has described his procedures for changeover from normal operation to operation with one circulating loop out of service. The procedures involve reducing power to 50%, manually adjusting several set points on the control room instrumentation and checking the instruments to confirm the proper setting. Power is then raised to 60% of full power for continued operation. The required manual adjustments should be minimal, made in accordance with explicit procedures by approved personnel, on a deliberate time scale, and with final settings calibrated and tested. It is expected that this mode of operation will be infrequent. The Committee believes that these conditions are essential if manual rather than automatic adjustment of set points is to be used for removing a loop from service at power.

The applicant stated that he will install equipment to control the buildup of hydrogen in the containment which might follow in the unlikely event of a loss-of-coolant accident. Consideration is being given to a catalytic recombiner presently under development for limiting hydrogrn concentration. The hydrogen control system and provisions for containment atmosphere mixing and sampling should have redundancy and instrumentation suitable for an engineered safety feature. The capability for controlled purging should also be provided. The Committee wishes to be kept informed of the resolution of this matter.

The applicant should study design changes to improve the capability for testing the actuating circuits for the engineered safety features during reactor operation.

The Committee recommends that the applicant accelerate the study of means to prevent common mode failures 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 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.

The applicant's criteria for design of the irradiated-fuel-storage pool include the provision that adequate cooling water be available in the event of postulated accidents involving large missiles or a dropped fuel cask. The specific design approach adopted and related analyses should be reviewed by the Regulatory Staff.

The Committee reiterates its interest in active participation by applicants in overall quality assurance programs to better assure the construction of safe plants. In this regard, an increased level of direct participation by the applicant in the quality assurance program of the North Anna Station would be desirable

The applicant has stated that he will provide additional evidence obtained by improved multi-node analytical techniques to assure that the ECCS is capable of limiting core temperatures to acceptably conservative values. He will also make appropriate plant changes if further analyses 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 line connecting the refueling water storage tank to the low-pressure safety injection system and the lines connecting each accumulator to the remainder of its piping system contain a normally open motor-operated valve. Since inadvertent closing of these valves would seriously degrade emergency core cooling capability, the Committee believes that more positive assurance of proper valve function should be provided.

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 North Anna Power Station.

The Committee believes that the above items can be resolved during construction and that, if due consideration is given to these items, the North Anna Power 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/
Joseph M. Hendrie
Chairman

<u>References</u>

1) Amendments 1 - 11 to the License Application

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

June 16, 1971

H. L. Price, Director of Regulation

PROPOSED STAFF POSITION REGARDING THE ULTIMATE HEAT SINK FOR THE NORTH ANNA POWER STATION, UNITS 1 AND 2

Reference: Memorandum from P. A. Morris to Dr S. H. Bush, Dated June 7, 1971, with Report to the ACRS, Attached.

Based on discussion during the 134th ACRS meeting, the Committee concurred with the proposed Staff position regarding the seismic classification for the ultimate heat sink for the North Anna Station and application of this criteria to other nuclear plants as proposed.

/s/ R. F. Fraley

R. F. Fraley
Executive Secretary

cc: P. A. Morris, DRL
E. G. Case, DRS

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

March 13, 1973

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

Subject: REPORT ON NORTH ANNA POWER STATION, UNITS NO. 3 AND NO. 4

Dear Dr. Ray:

At its 155th meeting, March 8-10, 1973, the Advisory Committee on Reactor Safeguards completed its review of the application of the Virginia Electric and Power Company for authorization to construct two nuclear units, identified as the North Anna Power Station, Units No. 3 and No. 4, in Louisa County, Virginia. This project was considered at a Subcommittee meeting at the site on December 22, 1972, at a Subcommittee meeting on January 3, 1973, in Washington, D. C., at the 153rd meeting of the Committee, January 11-13, 1973, in Washington, D. C., and at a Subcommittee meeting on February 23, 1973, in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives and consultants of the Virginia Electric and Power Company, the Babcock & Wilcox Company, the Stone and Webster Engineering Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The North Anna Power Station site consists of 1,075 acres of land in the northeastern corner of Louisa County, Virginia, on the south shore of Lake Anna, a 17 mile long, 13,000 acre man-made lake created by impoundment of water in the North Anna River. Units No. 3 and No. 4 share the site with Nuclear Units No. 1 and No. 2, which are now under construction. The nearest population center is Fredericksburg, about 24 miles northeast of the site, with a population of approximately 15,000 in 1970. The minimum exclusion distance is 5,000 feet. The low population zone, extending six miles from the site, had a 1970 population of approximately 2000. The site is in a rural, extensively wooded, area which is interspersed with farms.

Each of the two proposed nuclear steam supply systems will utilize a two-loop pressurized water reactor supplied by the Babcock & Wilcox Company and designed to operate at an initial power of 2,631 MWt. The thermal power level and the design of the nuclear steam supply systems for the North Anna Units No. 3 and No. 4 are generally similar to those of the Davis-Besse Nuclear Power Station. However, the average and maximum linear power ratings are, respectively, about 22 percent and 8 percent higher for North Anna Units No. 3 and No. 4. The Committee reiterates its previous statements with respect to reactors designed for high linear power ratings that, if experience does not confirm the predicted performance, system modifications or restrictions on operations may be appropriate.

Because of the importance of the incore instrumentation for operation of this plant, the Committee urges that careful attention be given to ensuring reliability and adequacy of the incore system. Experience with performance of similar systems in the Oconee Nuclear Station should be thoroughly examined to make certain that the information needed can be obtained with the accuracy required. The applicant stated that the reactor design does not preclude the capability of installing traveling incore instrumentation. The Committee recommends that this flexibility be retained.

The applicant is reviewing the calculated performance of the emergency core cooling system following a postulated loss-of-coolant accident resulting from a break in a core flooding tank line. He has agreed to make such system changes as are determined to be necessary.

The potential effects of fuel performance and LOCA-related phenomena for the possible spectrum of break sizes on acceptable linear power ratings for the North Anna Units No. 3 and No. 4 require further study. In addition, the Committee believes it important that improvements in ECCS effectiveness be investigated and included, as practical. The Committee 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 applicant has under study means to mitigate the consequences of possible rupture of the main steam lines and feedwater lines outside the containment building. This matter should be resolved to the satisfaction of the Regulatory Staff; the Committee wishes to be kept informed.

The applicant has indicated that the design of the reactor vessel and internals will have the benefit of experience obtained in the Oconee Nuclear Power Station Unit No. 1 as well as other Babcock & Wilcox data. The applicant will perform a prototype-plant vibration test program on the North Anna Unit No. 3. The Committee wishes to emphasize the desirability of using the available technology to monitor for excessive vibrations, loose parts, or other anomalous effects in the primary system during operation.

The Committee believes it desirable for the applicant and the Regulatory Staff to review further North Anna Power Station Units No. 3 and No. 4 for design features that should reduce the possibility and consequences of sabotage, in accordance with Safety Guide No. 17, "Protection Against Industrial Sabotage."

The applicant is reviewing the quality group classification of portions of the component cooling system relative to Safety Guide 26, "Quality Group Classifications and Standards." This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The Regulatory Staff is reviewing the pressure margins used by the applicant in the design of containment sub-compartments to withstand the effects of pressure transients. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

A review of the proposed environmental surveillance program revealed several unresolved questions with respect to sample collection and analysis, particularly as to the usefulness of the resulting data for estimating population dose. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

Studies are in progress relating to the effects of a failure to scram during 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.

The Committee believes that, unless the applicant can demonstrate that the probability of a serious accident arising from turbine missile generation is acceptably low, further measures both to reduce the probability and the potential consequences of turbine missile generation, including considerations of overspeed, be studied and implemented. Analytical and experimental work on the penetration of reinforced concrete by missiles of the type of interest is an example of the kinds of data important to evaluation of this problem.

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 North Anna Power Station, Units No. 3 and No. 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,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References:

1) Virginia Electric and Power Company Application for License to Construct and Operate North Anna Power Station, Units No. 3 and No. 4, with Volumes 1 through 4, Preliminary Safety Analysis Report, and Volumes I and II, Supplementary Preliminary Safety Analysis Report

2) Amendments 1 through 15 to the Application

3) Virginia Electric and Power Company letter to DL dated December 27, 1972 re: General Information Required for Consideration of the Effects of a Piping-System Break Outside Containment

4) Virginia Electric and Power Company letter to DL dated December 27, 1972 re: construction permits for North Anna Units No. 3 and No. 4

5) Virginia Electric and Power Company letter dated December 29, 1972 furnishing information regarding the Safety Evaluation Position on the service water reservoir for North Anna Units No. 3 and No. 4

6) DL Safety Evaluation, received December 29, 1972

7) Virginia Electric and Power Company letter to DL dated January 2, 1973 re: analysis of the effects of fuel densification

8) Virginia Electric and Power Company letter to DL dated January 10, 1973 re: Safety Evaluation Position concerning application for a construction permit

9) Virginia Electric and Power Company letter dated January 16, 1973 re: exemption request

10) Virginia Electric and Power Company letter dated January 23, 1973 re: report entitled, "Mixing of Combustible Gases in the Containment Subcompartments Following a LOCA for North Anna Units No. 3 and No. 4

References (Cont'd):

11) Virginia Electric and Power Company letter dated January 25, 1973 to DL re: function of active valves under normal system operation

12) Virginia Electric And Power Company letter to DL dated February 2, 1973 re: report entitled "Radioiodine Releases"

13) Virginia Electric and Power Company letter dated February 9, 1973 to DL furnishing information regarding turbine rotor and disc inspection at the North Anna No. 3 and No. 4

14) Virginia Electric and Power Company letter to DL dated February 16, 1973 re: application for construction permits for North Anna No. 3 and No. 4

15) Virginia Electric and Power Company letter dated February 16, 1973 to DL re: additional information on the construction permits

16) Virginia Electric and Power Company letter dated February 16, 1973 to DL re: additional information on the construction permits

17) Virginia Electric and Power Company letter dated February 21, 1973 to DL re: intentions and conditions which must be met in the application for construction permits

18) DL Supplement No. 1 to the Safety Evaluation dated February 21, 1973

19) Virginia Electric and Power Company letter dated March 7, 1973 re: radioiodine releases and the service water system at North Anna No. 3 and No. 4

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

April 15, 1974

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

Subject: REPORT ON NORTH ANNA POWER STATION, UNITS 1, 2, 3 AND 4

Dear Dr. Ray:

At its 168th meeting, April 11-13, 1974, the Advisory Committee on Reactor Safeguards completed its review of safety matters related to the previous discovery of fault zones under or adjacent to the foundation locations of North Anna Power Station, Units 1, 2, 3, and 4 in Louisa County, Virginia. Members of the ACRS or its consultants visited the site on five occasions between September 29, 1973 and February 6, 1974. The matter was considered further at a Subcommittee meeting on March 6, 1974 and at a meeting of the full Committee, March 7-9, 1974. During its review, the Committee had the benefit of discussions with representatives and consultants of the Virginia Electric and Power Company 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 permit applications for North Anna Units 1 and 2, on August 20, 1970 and on North Anna Units 3 and 4, on March 13, 1973.

On May 17, 1973 the Applicant notified the Regulatory Staff of the existence of a chlorite seam in the containment excavations for Units 3 and 4. Subsequent investigation has led to the conclusion that several minor fault zones are in the immediate vicinity and that the excavation for each of the four units is transected by one or more faults.

Extensive study of the matter has led to general agreement by the Regulatory Staff and its consultants, by the Applicant's consultants, and by consultants to the Committee that all the faults under consideration are very old (perhaps hundreds of millions of years) and that in the four fault zones which transect the excavations there is direct geologic evidence that no significant displacement has occurred in more than 500,000 years. These are therefore not capable faults as defined by Appendix A of 10 CFR Part 100.

Studies of the possibility of Lake Anna inducing seismic activity and displacement along the faults in question lead to the conclusion that such a consequence is highly unlikely. The program of monitoring microearthquakes in the immediate vicinity of the site should provide further substantiation of this conclusion or alert the Applicant to the possible onset of changed conditions.

Adequate provisions with regard to foundation stability are included in the Applicant's design approach.

The Advisory Committee on Reactor Safeguards concludes that provisions to accommodate surface displacement of faults need not be made in the design of the North Anna Power Station, Units 1, 2, 3, and 4.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References: Listed on Page 3

List of References:

1. Amendments 17-20 to the Preliminary Safety Analysis Report of the Virginia Electric and Power Company Application for License to Construct Units 3 and 4 of the North Anna Power Station

2. Letter dated February 25, 1974 from Virginia Electric and Power Company to Directorate of Licensing, U. S. AEC

3. Letters from the North Anna Environmental Coalition dated: October 22, 1973; January 2, 1974; February 25, 1974; and April 9, 1974

4. Questions presented April 11, 1974, from the North Anna Environmental Coalition with attached chronology

5. Press releases (2) dated April 11, 1974, from the North Anna Environmental Coalition with attachments

6. Supplement No. 3, dated February 28, 1974, to the Safety Evaluation by the Directorate of Licensing, U. S. AEC in the Matter of Virginia Electric and Power Company North Anna Power Station, Units 3 and 4

7. U. S. Geological Survey Reports dated: October 4, 1973; December 4, 1973; February 28, 1974; March 19, 1974; and, April 10, 1974

8. Testimony prepared for presentation before the Atomic Safety and Licensing Board in the Matter of Virginia Electric and Power Company (North Anna Power Station, Units 1, 2, 3, and 4) from:

   Bernard Archer
   Dr. Carlos G. Bell, Jr.
   John Briedis
   James L. Calver
   Dr. Lowell A. Douglas
   Dr. Robert Brian Ellwood
   Joseph A. Fischer
   Dr. Todd M. Gates
   John F. Gibbons, II
   Tidu Maini
   Joel Marks
   Roland C. McEldowney
   James Grey McWhorter
   Dr. Robert F. Mueller
   Dr. Antonio V. Segovia
   Dr. David T. Snow
   Donald V. Wise

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

October 26, 1976

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON PARTIAL REVIEW OF NORTH ANNA POWER STATION UNITS 1 AND 2

Dear Mr. Rowden:

At its 198th meeting, October 14-16, 1976, the Advisory Committee on Reactor Safeguards completed a partial review of the application of the Virginia Electric and Power Company for authorization to operate the North Anna Power Station, Units 1 and 2. The project was previously considered at Subcommittee meetings in Washington, D.C., on July 7, 1976, August 11, 1976, and October 13, 1976, and at the 196th meeting of the Committee on August 12-14, 1976. Tours of the facility were made by Subcommittee members on February 3, 1976 and May 27, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Virginia Electric and Power Company, the Westinghouse Electric Corporation, the Stone and Webster Engineering Corporation, the Sun Shipbuilding and Dry Dock Company, the North Anna Environmental Coalition, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed. The Committee discussed the application for a construction permit for the North Anna Power Station, Units 1 and 2, in its report of August 20, 1970. The Committee also discussed matters related to fault zones under or adjacent to the foundations of North Anna Power Station, Units 1, 2, 3, and 4 in its report of April 15, 1974.

The site is located on 1,075 acres on the shores of Lake Anna in Louisa County, Virginia, about 24 miles southwest of Fredericksburg, Virginia, and 40 miles north-northwest of Richmond, Virginia.

The Committee has not completed its review of North Anna Units 1 and 2 with regard to the following matters: adequacy of seismic design basis and seismic design; loss-of-coolant accidents and emergency core cooling; quality assurance and control in on-site fabrication and installation; asymmetric loads on pressure vessel structures arising from certain postulated pipe breaks; and plans for upgrading protection against fires.

Also, in Supplement No. 3 to the Safety Evaluation Report, the NRC Staff has identified several items to be resolved, and the Committee has a few remaining items relating to systems interactions on which it wishes further information.

An unexpected amount of settlement has been experienced by the service water pump house for the North Anna Units 1 and 2. Some cracking of the pump house walls has resulted. The Applicant has examined the causes of the settlement and has made design changes, including the provision of flexible expansion coupling between the piping and the pump house to accommodate additional settlement. The NRC Staff is satisfied with the re-analysis of stresses and, except for review of the design of a system of well points for ground water control, believes the situation is currently acceptable. Future settlement, which should be modest, will be monitored carefully in accordance with technical specifications to be prepared. The Committee concurs with the NRC Staff.

The Applicant has submitted a revised probable maximum flood analysis. The NRC Staff has reviewed the analysis and found it acceptable with the inclusion of a technical specification to restrict facility operation when the water level in Lake Anna exceeds an elevation of 256 feet above mean sea level. The Committee concurs.

The North Anna Power Station, Units 1 and 2 will employ a 17x17 fuel assembly similar to that employed in Beaver Valley Unit 1. A considerable portion of the Westinghouse research and development program on these assemblies has been completed, and has been evaluated and accepted by the NRC Staff. The Committee wishes to be kept informed on those matters still under review.

The steam-generator and reactor-coolant-pump supports are constructed of heavy rolled steel shapes and thick plate. After delivery of these structures at the site, the Applicant found many weld defects and proceeded to remove all welds and to reweld the supports. The Unit 1 steam-generator supports had been installed and were rewelded in place, which made it necessary to substitute peening for thermal stress relieving. The Committee finds this procedure acceptable. The Unit 2 supports were rewelded in the shop and thermally stress relieved. The NRC Staff has not completed its review of this unit. Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for the supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was good, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F. The Applicant, therefore, proposes to operate so that all A572 material is at 180°F or above. He also plans periodic inspection of the A572 members to the extent that they are accessible. The Committee believes that increasing the operating temperature is an acceptable solution, but recommends that the operating temperature of the A572 material be substantially above the proposed temperature. The Committee believes also that it would be prudent not to permit pressurization of the primary system to substantial levels while temperatures of the supports might be well below the operating temperature.

The NRC Staff is satisfied with regard to the Emergency Plan, and the Applicant has made considerable progress in providing instrumentation to follow the course of an accident.

The Committee recommends an early resolution of the matter of anticipated transients without scram for North Anna Units 1 and 2. The Committee wishes to be kept informed.

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 North Anna, Units 1 and 2, should be dealt with appropriately by the NRC Staff and Applicant 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; IIB-2; IIC-1, 2, 3, 4, 5, 6, 7.

The ACRS believes that, if due regard is given to the items mentioned and subject to satisfactory resolution of those matters still under review and to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the North Anna Power Station, Units 1 and 2 can be operated at power levels up to 2775 MW(t) without undue risk to the health and safety of the public. The Committee will report in the future on those matters for which its review is not yet complete.

Additional comments by Dr. Spencer H. Bush are presented on the following page.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Additional Comments by Member Spencer H. Bush

These additional comments are directed to what appears to be the NRC Staff's position regarding acceptance of operation with the North Anna, Units 1 and 2 steam-generator and reactor-coolant-pump supports at or below temperatures of 180°F. I find it difficult to accept system pressurization to substantial levels while temperatures of the supports might be well below those suggested as "equilibrium", e.g., <180°F temperature. I do not consider it unreasonable to require that the minimum temperatures of the supports be at a level of 225-250°F, obtainable by methods such as electric "trace" heating. The combined benefits of operation in the elastic-plastic fracture mechanics regime, major increase in critical flaw size and minimization of fast fracture propagation, admittedly represent conservatisms, but these conservatisms can be achieved relatively easily with no apparent adverse degradation mechanisms. Since we do not have complete impact or fracture mechanics data, equilibrating at 225-250°F prior to pressurizing fully is recognized as conservative, but is considered desirable.

REFERENCES

1. Final Safety Analysis Report, North Anna Power Station, Units 1 and 2, with Amendments 1 through 56.

2. Safety Evaluation Report related to the operation of North Anna Power Station, Units 1 and 2, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, with Supplements 1, 2, and 3. (NUREG-0053)

3. Letter dated October 8, 1976, from Ernst Volgenau, Director, Office of Inspection and Enforcement, USNRC, to R.F. Fraley, Executive Director, ACRS, Subject: "Comments Regarding North Anna Nuclear Plant".

4. North Anna Environmental Coalition (NAEC) letters dated August 17, 1976 and September 1, 1976 and NAEC Statement of August 11, 1976 continued on October 13, 1976.

5. "Interim Report on the Examination of Core Samples from Reworked Steam Generator Supports of VEPCO, North Anna", William S. Pellini, April 8, 1976.

6. "The Safety of Steam Generator Support Structures for North Anna, Units 1 and 2", J.D. Harrison and R.E. Dolby, for Sun Shipbuilding and Dry Dock Company, May 1976.

7. "Additional Information found in VEPCO and Stone and Webster files", 3 pp., Sun Shipbuilding and Dry Dock Company.

8. "The Safety of Steam Generator Support Structures for North Anna, Units 1 and 2", Sun Shipbuilding and Dry Dock Company, May 20, 1976, with Appendix 1, plus a one-page "Final Note".

9. "Book 1, Summary of Information on Core Samples Including Source, Dimensions", (with 30 pages of photographs), Sun Shipbuilding and Dry Dock Company, May 20, 1976.

10. "Book 2, Photographic Documentation of Defects in Core Samples", (with 30 pages of photographs) Sun Shipbuilding and Dry Dock Company, May 20, 1976

REFERENCES (con't)

11. "The Safety of Steam Generator Support Structures for North Anna, Units 1 and 2" by Sun Shipbuilding and Dry Dock Company, July 7, 1976.

12. "The Safety of Steam Generator Support Structures for North Anna, Units 1 and 2" Statement before the ACRS by Sun Shipbuilding and Dry Dock Company, October 13, 1976.

13. "Further Comments on the Safety of the North Anna Support Structures", LD 22955/2, June 1976, J.D. Harrison and R.E. Dolby, the Welding Institute, (for Sun Shipbuilding and Dry Dock Ltd.).

14. "Catalog of Brittle Failures of Bridges and Other Related Structures, and Brittle Failures of Other Items Recorded at Higher Temperatures", VEPCO report to ACRS North Anna Subcommittee, October 13, 1976.

15. "Test Data for Materials in North Anna Units 1 and 2 Steam Generator and Reactor Coolant Pump Supports", VEPCO report to ACRS North Anna Subcommittee, October 13, 1976.

16. "VEPCO North Anna Units 1 and 2 Support Structures, Discussion of Fracture Mechanics Studies Presented by Various Parties", H.T. Corten, October 1976.

17. "Repairs, Inspection and Quality Assurance, Steam Generator and Reactor Coolant Pump Repair Program", VEPCO report to ACRS North Anna Subcommittee, October 13, 1976.

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

January 17, 1977

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON NORTH ANNA POWER STATION, UNITS 1 AND 2

Dear Mr. Rowden:

At its 201st meeting, January 6-8, 1977, the Advisory Committee on Reactor Safeguards completed its review of the application of the Virginia Electric and Power Company for a license to operate North Anna Power Station, Units 1 & 2. This project was also considered during a Subcommittee meeting held in Washington, D.C., on January 5, 1977. The Committee previously completed a partial review of this project at its 198th meeting, October 14-16, 1976, as discussed in its report to you, dated October 26, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Virginia Electric and Power Company, the Westinghouse Electric Corporation, the Stone and Webster Engineering Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

In its report of October 26, 1976, on North Anna, Units 1 & 2, the ACRS had not completed its review of the adequacy of seismic design bases and seismic design; loss-of-coolant accidents and emergency core cooling; quality assurance and control of on-site fabrication and installation; asymmetric loads on pressure vessel structures arising from certain postulated pipe breaks; and plans for upgrading protection against fires.

The NRC Staff has now completed its review of the Stafford fault zone and concluded that the available geological and seismological information supports the conclusion that the Stafford fault zone is not capable within the meaning of Appendix A to 10 CFR Part 100, and that the available information does not warrant any change in the previously approved seismic design bases for North Anna 1 and 2. Representatives of the U.S. Geological Survey concurred that there exists no definitive information showing significant movement during the last million years and that the fault is not capable. Consultants to the ACRS concur with this interpretation. While they generally find the current design bases acceptable for the already constructed North Anna plants, they have recommended that, in view of the uncertainties of knowledge concerning the sources of earthquakes in the Eastern United States, a minimum safe shutdown earthquake (SSE) of 0.2g acceleration should be utilized for new plants for which construction permit applications are submitted in the future.

The Applicant presented partial information concerning the calculated safety factors during safe shutdown earthquake conditions for some of the engineered safety features. The Committee recommends that the NRC Staff review this aspect of the design in detail and assure itself that significant margins exist in all systems required to accomplish safe shutdown of the reactors and continued shutdown heat removal, given an SSE. The Committee believes that such an evaluation need not delay the start of operation of North Anna 1 and 2. The Committee wishes to be kept informed.

The NRC Staff has now completed its review of emergency core cooling system performance and found it to be acceptable. The Committee concurs.

The NRC Staff has conducted and is continuing extensive investigation of construction activities of North Anna Units 1 and 2. These investigations have been separated into four phases:

1. investigation of specific allegations made by three individuals of faulty construction practices;

2. a detailed inspection of certain safety-related piping not directly implicated in the original allegations but which was potentially subject to similar problems;

3. detailed monitoring of the nondestructive preservice baseline examination of selected welds in safety-related piping by the Licensee and his contractors; and

4. inspections of the performance of selected components in specific piping systems during the preoperational testing program.

The NRC Staff has concluded that various items of non-compliance with NRC requirements have occurred and has defined a program to remedy the matter.

The Committee has had the benefit of a review and evaluation of this matter by its own consultant, who supports the adequacy of the NRC investigations and has made several recommendations, including one related to a program to ascertain that significant deficiencies do not exist in safety related piping systems. The ACRS concurs. The Committee wishes to be kept informed regarding resolution of these recommendations.

The NRC Staff has reported that the matter of asymmetric loads on pressure vessel structures is essentially resolved. The ACRS has had the benefit of meetings of an Ad Hoc Working Group on this general subject, in Toronto on August 5, 1976, and in Los Angeles on December 1, 1976. The Committee agrees that, subject to final evaluation by the NRC Staff, this matter is in an acceptable status for North Anna 1 and 2.

The Applicant is in the process of studying fire protection measures at the plant in accordance with the guidelines of Appendix A to Auxiliary and Power Conversion Systems Branch Technical Position 9.5-1. The NRC Staff has stated that, as a plant about to come into operation, North Anna 1 and 2 will be given priority in the evaluation of fire protection matters, and that most, if not all improvements will be implemented prior to the start of operation on the second fuel cycle. The Committee finds this approach to be acceptable.

The Committee notes that post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptible 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 and maintenance procedures to assure continuous long-term seal capability should be developed.

The ACRS believes that, if due regard is given to the items mentioned above and in its report of October 26, 1976, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the North Anna Power Station, Units 1 and 2, can be operated at power levels up to 2775 MWt without undue risk to the health and safety of the public.

Sincerely yours,

M. Bender

M. Bender
Chairman

Attachment:
Report of W.R. Gall, ACRS Consultant, dated January 3, 1977, Subject: Review of Allegations and Inspectors Findings as Reported in NRC Investigation Report #50-338/76-28, 50-339/76-16 North Anna, Units 1 and 2.

REFERENCES:

1. North Anna Power Station, Units 1 & 2 Final Safety Analysis Report, with Amendments 1 through 60.
2. Safety Evaluation Report (NUREG-0053) related to operation of North Anna Power Station, Units 1 and 2, with Supplements 1 through 5.
3. Virginia Electric and Power Company (VEPCO) letter Serial No. 338 to Mr. Benard C. Rusche, ONRR, NRC, dated November 24, 1976, on environmental testing of safety related instrumentation.
4. VEPCO letter Serial No. 350 to Mr. Benard C. Rusche, ONRR, NRC, dated November 30, 1976, forwarding a document entitled, "Safety Related Equipment Temperature Transients During the Limiting Main Steam Line Break."
5. VEPCO letter Serial No. 346 to Mr. Benard C. Rusche, ONRR, NRC, dated November 30, 1976, on measures considered for use at North Anna re overpressurization events.
6. VEPCO letter Serial No. 316A, dated December 3, 1976, re model testing of LHSI pumps.
7. VEPCO letter Serial No. 298/102276, dated December 16, 1976, containing information on LOCA effects on reactor fuel. (Westinghouse PROPRIETARY).
8. NRC letter of December 14, 1976, from D.B. Vassallo to Dr. Dade W. Moeller, Chairman, ACRS, subject "Staff Report - Assessment of the Stafford Fault Zone."
9. NRC memo dated December 2, 1976, from Dudley Thompson and Boyce H. Grier to Ernst Volgenau, I&E, subject, "Transmittal and Evaluation of Investigation Report, No. 50-338/76-28, 50-339/76-16 - North Anna Units 1 and 2."
10. VEPCO letter Serial No. 371, dated December 9, 1976, forwarding a copy of VEPCO's reply to E. Volgenau re I&E Investigation Report Number 50-338/76-28 and 50-339/76-16.
11. NRC letter dated December 6, 1976 from E. Volgenau , I&E, to VEPCO Attn: Mr. T. Justin Moore, President referring to the I&E investigation of construction activities at North Anna 1 and 2 forwarding a "Notice of Violation", and a "Notice of Proposed Imposition of Civil Penalities."

REFERENCES (con't)

12. USNRC, IE Investigation Report 50-338/76-28, 50-339/76-16, Subject: "Investigation of alleged discrepancies in the construction and quality control program for piping installation at the North Anna Power Station."
13. VEPCO letter serial 390 to Dr. Dade W. Moeller, Chairman, ACRS, forwarding a copy of Mr. T. Justin Moore's letter of December 23, 1976 to Dr. Ernst Volgenau re the North Anna investigation.
14. VEPCO letter Serial No. 391, dated January 4, 1977, providing information re concerns related to auxiliary power and containment systems.
15. North Anna Environmental Coalition (NAEC) letter dated January 5, 1977, to Dr. Dade W. Moeller and Dr. David Okrent, ACRS, requesting that certain items be made a part of the record of the January 6-8, 1977, ACRS meeting.
16. NAEC letter dated January 7, 1977, to Dr. Dade W. Moeller and Dr. David Okrent, ACRS, adding two additional items to the list submitted in the NAEC letter of January 5, 1977.

ATTACHMENT TO CTE. LTR. DTD. 1/17/77
on NORTH ANNA POWER STATION, UNITS 1 & 2

Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, Tennessee 37830
January 3, 1977

Dr. David Okrent
Energy and Kinetics Department
5532 Boetler Hall
School of Engineering and Applied Science
University of California
Los Angeles, CA 90024

Dear Dr. Okrent:

Subject: Review of Allegations and Inspectors Findings as Reported in NRC Investigation Report #50-338/76-28, 50-339/76-16 North Anna, Units 1 and 2

The purpose of this memo is to transmit my conclusions and recommendations regarding the reported allegations and the inspectors' findings as reported in the NRC Investigation Report on North Anna Units 1 and 2.

It is my opinion that the investigation of specific allegations as covered in the report has been sufficiently thorough to provide an evaluation of probable validity of the allegations and their possible effect on the integrity of the system. My comments are based on the study of their report supplemented by two visits to the plant site, discussions with Stone & Webster, and Vepco staff members and with NRC Inspection and Enforcement staff in Bethesda, Maryland.

The report presents the inspectors' findings and explains the method of investigation upon which their conclusions are based. It does not in all cases cover corrective actions that have been taken or that may be proposed as a result of the findings. In some cases, an evaluation of the integrity of the affected systems will depend upon the corrective action that is proposed or taken.

I have the following specific items of concern:

1. Cutting of Rebar

Apparently the cutting of rebar became so prevalent that Stone & Webster themselves became concerned about it and initiated actions to curtail or control such cutting. But prior to initiation of those actions, various methods of cutting rebar were used, some of which may be detrimental to the properties of the concrete and particularly the use of carbon-arc, oxygen-flame cutting and welding rod processes which could provide high levels of heat input to the concrete. The proposed analysis described in the licensee's response may be sufficient to establish the adequacy of the rebar but further evaluation may be necessary to determine if the concrete was damaged.

2. Allegations Concerning Fake Anchor Bolts

The interference between anchor bolts and rebar may be responsible for the faking of two anchor bolts which were reported in allegations B-7 and P-1. It is also possible that some anchor bolts were cut to avoid cutting rebar which would result in the length of bolts being shorter than specified, thus affecting the strength of the anchor. It is my understanding that ultrasonic measurements will be made to detect those bolts which were shortened. I recommend that an evaluation be made by the licensee to determine the adequacy of any bolts which are found to be short.

3. Welders Performing Welds Outside the Range of Their Qualifications

It was established by the inspector that 30 Class 1 type welds in Units 1 and 2 were performed by welders qualified for thinner sections. I believe all of these welds are identified in QC records and that all of the welds have been examined by radiography and found to be acceptable. The welders were qualified on thinner sections than those cited. The acceptance of radiographed production welds as qualification welds may be a valid procedure provided the initial weld performed outside the previously qualified thickness range meets the requirements of QW-301.4, QW-302.2, and QW-305.2 of Section IX of the ASME Code,is acceptable without weld repairs, and also provided the complete weld was performed by the same welder. One instance was observed during a visit to North Anna site in which a single welded joint in a primary coolant loop had the weld identification numbers of twelve different welders. I believe this would not be a satisfactory way to qualify any of the welders.

4. 32-Inch Main Steam Riser in Safety Valve Station

The circumferential joints performed in the modification to the 32-inch main steam riser have been evaluated and seem to be satisfactory except in the matter of mismatch. Permissible mismatch allowed by the Code is 3/32 of an inch. The inspector determined in at least one case a maximum of 5/16 inch mismatch. This is a factor of 3 over that permitted by the Code and assuming that the stress in the longitudinal direction was judged satisfactory with the permissible misalignment, the affect of mismatch would increase that stress by a factor of 3 in the case of 5/16 inch mismatch. This pipe is probably subject to extreme axial compressive loads when the safety valves operate.

I have not found evidence that a failure of this pipe could not cause a pipe whip in the main steam valve house which would react on the penetration of the containment wall sufficiently to breach the containment.

5. Welding Electrodes

There were two items concerning improper storage of welding electrodes and one concerning use of welding electrodes prior to receipt of material certifications for them. It would be difficult if not impossible to determine whether welding electrodes which had been stored overnight or over a shift outside of the required drying ovens have been used in welds or in which welds they may have been used. Furthermore, detection of the effect of excessive moisture or other contaminates principally hydrogen embrittlement, would be difficult to detect by means of radiography. It would be desirable to establish that all electrodes held over were being kept for personal use. The use of welding materials prior to receipt of proper documentation requires verification after the weld material has been used and could result in a determination that incorrect materials were used. This verification should be made in all cases where this was done and corrective action taken where necessary.

6. Defective Shop Welds

Two instances are reported in the inspector's findings - Items 2-C and 2-K in which noncorming shop welds in pipes performed by others were discovered by Stone & Webster quality control. Corrective action is not indicated. Of particular interest is the disposition of those Southwest Fabricating and Welding Company's pipe welds which were not included in the 1.5% sample examined by Stone & Webster QC and the applications in which they were used. Approximately half of the 1.5% sample were found to be nonconforming and presumably were repaired. Corrective action should be applied to all other welds represented by those samples to assure conformance with quality requirements in the Class 2 system.

7. Improper Identification of Materials and Parts

In the four reported incidents of improper identification of materials, it was possible to establish acceptability for the materials affected. Can it be established with a reasonable degree of confidence that these reported instances are the only ones in which materials were improperly identified, or that all materials installed are in compliance with requirements?

8. Conclusions

A. I agree in general with the Evaluation of Findings enclosed with the transmittal of the report of the investigation. Corrective action to correct the deficiencies in the quality assurance program must be augmented by actions to verify quality of construction already completed. Phases 2, 3 and 4 of the investigation, I believe, were conceived for this purpose. Phase 2 has been completed with some deficiencies yet to be resolved. Phases 3 and 4 should form a basis for establishing the integrity of the system.

An effort should be made to establish that welding electrodes which were improperly stored were not used in welding of safety related systems, or that if used the effects will not compromise safety.

B. It is my opinion that all of the identified "quality of work" non-conformities can be corrected by corrective action. Some of the quality control non-conformities affecting work that is already completed cannot be corrected now, but the quality of the work affected may be verified by preoperational testing, and examination, and if deficient it can be corrected.

C. The "unresolved items" listed in Part E of the report can also be resolved by appropriate corrective actions.

D. The licensee's quality assurance program has not functioned in accordance with established procedures and requirements in some cases. This leads to concern about possible undetected non-conformances. Phases 1 and 2 of the investigation constitute a thorough study of these possible deficiencies in the important safety related systems and it resulted in disclosure of some additional deficiencies which should be corrected.

E. In my study of the report and my discussions with persons involved at the site, I have developed a number of detailed questions related to the allegations and the findings which are given in Attachment I.

F. The allegations which were concluded to be unsubstantiated are reviewed in Attachment II to this letter and the substantiated allegations are reviewed in Attachment III.

Very truly yours,

W. R. Gall

W. R. Gall

WRG:mb

cc: S. H. Bush
J. C. Ebersole
H. Etherington
M. S. Plesset
File - RC

Attachment I

Questions

1. What degree of conservatism is used in design of the supports which depend on anchor bolts?

2. What action will be taken to establish that the length of anchor bolts is adequate?

3. What action will be taken to establish the integrity of concrete affected by arc, or flame cutting of rebar?

4. What action is proposed to verify adequacy of cadwelding performed in non-conformance with requirements?

5. What corrective action will be taken on welds performed by welders outside of their qualified thickness range?

6. If the main steam pipe fails outside of the containment, between the penetration and the stop valve, will containment be breached?

7. What action will be taken to correct misalignment at welded joint in main steam riser?

8. In determining acceptable thinning of pipe walls during grinding, is $0.875 \times t_n$ (t_n = nominal thickness) used as an acceptable thickness?

9. What defects could be incurred as a result of lack of QC in-process surveillance?

10. Will the welds in the reactor coolant loops be examined by the ultrasonic method during pre-service testing?

11. What action will be taken to ascertain whether improperly stored weld rods were used in production and may have affected quality of welds - especially welds in the reactor coolant system?

12. What action will be taken on the Southwest Fabricating Company's welds which were not examined by S&W during their audits.

Attachment II

Unsubstantiated Allegations

A total of 58 allegations were made by the allegors A, B, and C including the additional allegations. Of this total, 45 were found not to be substantiated for various reasons. In the following list allegations are grouped according to the reasons for which they were not substantiated.

Reason for Not Substantiating	No. of Allegations
1. Allegations that were not substantiated because the investigators examined the affected part or the records and found them to be in conformance with the requirements. A-3, 8, 9, 11, 12, 17, 21, B-1, 12, 13, 14, 16, 18, 20, 21, 22, 24, 25, 27, 31, and P-6	21
In the report on Allegations A-8 and B-24, the inspector reported that quality control Nonconformance and Disposition reports showed that quality control had identified and documented instances of welders welding beyond limit qualifications. The report does not indicate what disposition was made of these occurrences. Inspector-identified Item 2b (Appendix 3) cites as an infraction the welding of more than 30 welds in Class 1 piping in Unit 2 by welders who were not qualified for the thickness of the pipe which they were welding.	
2. Allegations which are shown by quality control records to have been identified and corrected in accordance with procedures. A-6, 7, 13, 15, 18, B-6, 26, P-2, 4, and 5	10
3. Allegations that were true but either were not related to quality or were in accordance with procedures and requirements. A-14, 16, 19, B-4, 17, 30, and P-3	7
4. Allegations that were found to be a mistake on the part of the allegor. A-20	1
5. Allegations which were not related to quality whether true or not. B-9, C-1	2
6. Allegations that could not be verified by interviews with personnel, review of records, or other means and were concluded to be not substantiated. A-2, 10, B-10 and 15	4
These allegations having to do with falsification of records are the type which would be difficult to verify or disprove. The investigators' conclusions on these items were based on review of available records and interviews with persons on the job and, though not considered substantiated, some of these violations could have occurred either without the knowledge of those interviewed or without being recorded in the documents.	

(continuation of Attachment II)

Allegations A-2, A-10, and B-15 were that a welding inspector and a QC inspector signed off papers without fully reviewing the work, that no one cares about quality or checks the work being done by welders, and QC inspectors had craftsmen perform fit-up inspections for them. Allegation B-10 was that a particular field weld was performed by a different welder than the one whose number was recorded as having performed the weld. It is my opinion that the conclusions drawn by the inspector are correct, but I believe it is possible for violations of this type to occur in such a way that substantiation is almost impossible. However, examination of the completed work by nondestructive methods can be performed to show that the work is satisfactory.

Attachment III

Substantiated Allegations

Of the 13 substantiated allegations, four allegations (B-3, B-8, B-9, and B-11) dealt with incorrect identification numbers on materials for Class 3 systems. However, traceability was established through the heat numbers and the materials were found to be acceptable.

Allegation A-1, cutting of concrete reinforcement steel ("rebar"), was substantiated and it was established that rebar was cut during the drilling of about 25% of the anchor bolt holes for anchor bolts for supports. The Licensee's response to this finding indicates that an engineering analysis will be made to establish the adequacy of the concrete structures. It is my opinion that corrective action can be taken to assure the adequacy of these structures.

Allegations A-4 and B-2 dealt with the 32-inch main steam risers to the safety valve headers. A serious problem with this incident is the verification that the joint, in one case at least, had a mis-match of 5/16 inch as compared to the Code maximum of 3/32 inch. Bending stresses in the pipe wall as a result of such misalignment would be increased by a factor of approximately three due to this effect. Corrective action should be taken in regard to this misalignment.

Allegation A-5 concerns lack of in-process surveillance of piping work. The Licensee's response indicates that the procedures called for this surveillance and that it was carried out in part. It is likely that an increase in quality control personnel would be required if this is implemented as it is supposed to be, which would tend to substantiate allegations A-12 and B-31 that there are too few QC personnel.

Allegations B-7 and P-1 concerning fake anchor bolts in pipe supports are related to the cutting of rebar. The difficulty in installing anchor bolts without interfering with rebar apparently has caused some people to subvert the requirements by faking the anchor bolt installation. It is my opinion that action should be taken to check the length of all the anchor bolts used for supports of this type to establish that the lengths are in accordance with the requirements or that deviations are permissible as established by engineering verification.

Allegation B-5 refers to unrecorded welds which were made in 2-inch pipe in a Class 3 system for which corrective action has not been reported. It is my opinion that if the procedures described in the response from the Licensee are followed, any additional unrecorded welds of this type will be discovered and they should be examined to establish their acceptability.

Allegation B-23, holding over welding rod. This problem seems to be very difficult to control but it is important that uncontrolled electrodes not be used in pipe welds. The Licensee's response to this allegation does not indicate that they plan to take any corrective action on this item.

(continuation of Attachment III)

Allegation B-32, improper storage of stainless steel and carbon steel pipe and valves. It was evident during my visit to the site that many items are stored throughout the plant awaiting installation. Although this is probably only temporary storage it appears that damage could occur and dirt could be accumulated in some of the valve operators and controls which could effect their performance. I believe corrective action is required on this item.

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

February 17, 1977

Benard C. Rusche, Director
Office of Nuclear Reactor Regulation

SUBJECT: ACRS REPORT ON THE NORTH ANNA POWER STATION, UNITS 1 AND 2, DATED JANUARY 17, 1977

This memorandum is in response to your letter of January 31, 1977 concerning interpretation of the ACRS report of January 17, 1977 on the North Anna Power Station, Units 1 and 2. The Committee considered your request for clarification during the 202nd ACRS meeting. The members discussed the bases for the Committee's report on the North Anna Station and the comments noted below are reflected in the meeting minutes.

(1) The Committee concurs with its consultants in the matter of the Stafford fault zone.

(2) The Committee concurs in general with the recommendation of its consultants that a minimum safe shutdown earthquake (SSE) of 0.2g should ordinarily be utilized for new plants for which construction permit applications are submitted in the future, although the Committee believes that flexibility in this nominal floor is appropriate to allow for special site conditions and specific aspects of plant design for which site dependent spectra may be important or for situations where a sound and non-controversial basis exists for setting lesser criteria.

(3) The systems to be investigated are those required to accomplish safe shutdown of the reactors and continued shutdown heat removal. The Committee has recommended that such systems have significant margins in the event of the SSE, so that safe shutdown has a high probability of accomplishment, should a lower probability earthquake having a response spectrum somewhat larger than that of the usual broad band spectrum over part of the frequency range occur. Instances in which "current acceptance limits" may be exceeded in such an evaluation may be considered acceptable on a judgment basis.

R. F. Fraley
Executive Director

cc: L. Gossick, EDO
S. Chilk, SECY

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

July 20, 1977

Lee V. Gossick
Executive Director for Operations

REVIEW OF THE NORTH ANNA POWER STATION, UNITS 1 AND 2

During its 207th meeting, July 14-15, 1977, the Advisory Committee on Reactor Safeguards considered whether, on the basis of information presented to the full Committee and to the North Anna Subcommittee at a meeting on July 6, 1977, the ACRS should reopen its review of the application of the Virginia Electric and Power Company for a license to operate the North Anna Power Station, Units 1 and 2. The Committee concluded, as noted in the minutes of the meeting, that, on the basis of this information, there was no reason to alter its report of January 17, 1977 on the North Anna Station.

Mr. Bender informed the NRC Staff of this conclusion and suggested that the Staff affirm that the hydrology of the site is under control.

Raymond F. Fraley
Raymond F. Fraley
Executive Director

References

1. Letter dated 1/31/75 from N. C. Mosely, NRC, to S. Ragone, VEPCO, reporting the results of a NRC insepction conducted on January 7-10, 1975 at the Surry Power Station.
2. Letter dated 11/1/76 from S. C. Brown, Jr., VEPCO, to B. C. Rusche, NRC, containing information on the groundwater control beneath the service water pumphouse.
3. Letter dated 12/4/76 from S. C. Brown, Jr., VEPCO, to B. C. Rusche, NRC, re: Responses to Comments 2.19, 2.20 and 2.21 forwarded in the NRC letter of November 24, 1976.
4. Letter dated 1/14/77 from S. C. Brown, Jr., VEPCO, to N. C. Mosely, NRC re: repair of overstressed service water piping at North Anna Unit 2.
5. Letter dated 3/1/77 from S. C. Brown, Jr., VEPCO, to N. C. Mosely, NRC, reporting the findings of the insepction during the January 11-14, 1977 visit to North Anna Station.
6. Letter dated 3/4/77 from C. M. Stallings, VEPCO, to B. C. Rusche, NRC, furnishing information requested by NRC relating to specifications for the measurement of the suspended solids and turbidity in the effluent from the horizontal drains beneath the service water pumphouse.

7. Letter dated 3/8/77 from F. R. Brown, Army Corps of Engineers, to W. P. Gammill, NRC, transmitting a report entitled "The Undrained Cyclic Triaxial Response to A Saprolitic Soil" and the subject report.
8. Letter dated 4/20/77 from J. Allen, NAEC, to E. Volgenau, NRC, re: North Anna Station.
9. Letter dated 5/2/77 from J. Allen, NAEC, to F. Coufal, ASLB, re: North Anna Station.
10. Letter dated 5/5/77 from J. Allen, NAEC, to D. Okrent, ACRS, re: North Anna Station.
11. Letter dated 5/16/77 from O. D. Parr, NRC, to W. L. Proffitt, VEPCO, re: Request for Additional Information.
12. Report to the House Committee on Interstate and Foreign Commerce, Allegations of Poor Construction Practices on The North Anna Nuclear Power Plants, dated June 2, 1977.
13. Letter dated 6/8/77 from J. Allen, NAEC, to D. Okrent, ACRS re: North Anna Station.
14. Letter dated 6/14/77 from J. Allen, NAEC, to M. Bender, ACRS, re: North Anna Station.
15. Letter dated 7/12/77 from J. Allen, NAEC, to M. Bender, ACRS, re: North Anna Station.

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

November 10, 1977

Mr. Lee V. Gossick
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: NORTH ANNA POWER STATION, UNIT 1, MODIFICATION OF RECIRCULATION AND QUENCH SPRAY SYSTEMS

Dear Mr. Gossick:

Reference: (a) Vepco Report, "Analysis and System Modification for Recirculation Spray Pumps Net Positive Suction Head," North Anna Power Station Unit 1, September 16, 1977

*(b) VEPCO letter Serial No. 362, dated, August 20, 1977 (Docket Nos. 50-280 and 50-281, License Nos. DPR-32 and DPR-37) from C. M. Stallings to E. G. Case, NRC, (Attn: Mr. Karl R. Goller)

In Reference (a) (Section 1.2) the Applicant has indicated that the Recirculation and Quench Spray systems would be modified to ameliorate the calculated Net Positive Suction Head (NPSH) deficiency by the installation of flow restricting orifices on the discharge sides of the outside Recirculation Spray pumps and by diverting part of the Quench Spray flow to cool the containment sump.

The Committee suggests that such modifications may be unnecessary and undesirable in view of the assurance given in Reference (b) that the recirculation spray pumps, as installed, will function satisfactorily in the event that their NPSH is somewhat reduced for a period of time.

Sincerely,

M. Bender

M. Bender
Chairman

* Ref. (b) revised per R. F. Fraley's memo to Mr. L. V. Gossick dated November 15, 1977

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

March 16, 1978

Mr. L. V. Gossick
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Gossick:

Subject: MONITORING OF MICROEARTHQUAKES IN THE VICINITY OF THE NORTH ANNA POWER STATION

During its discussion of microseismicity near the site of the North Anna Nuclear Station, the members of the ACRS concluded that it would be prudent to continue operation of the seismic monitoring network established by VEPCO to monitor microearthquakes in this area.

Sincerely,

Stephen Lawroski

Stephen Lawroski
Chairman

cc:
E. G. Case, NRR
L. Crocker, NRR
O. Parr, NRR
V. Stello, DOR
R. Boyd, DPM

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

May 12, 1981

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

SUBJECT: PROPOSED MODIFICATIONS TO THE NORTH ANNA STATION UNIT 2 LOW PRESSURE INJECTION AND RESIDUAL HEAT REMOVAL SYSTEMS

Dear Dr. Hendrie:

Commissioner Gilinsky, in a separate statement dated August 20, 1980 on the full power authorization for the North Anna Nuclear Plant Unit 2, noted that it appeared that the low pressure injection (LPI) and residual heat removal (RHR) systems could be improved substantially in terms of their ability to deal with accidents. The specific changes suggested were the environmental qualification of the RHR system and the addition of a heat exchanger to the LPI system. Commissioner Gilinsky requested that the desirability of these modifications be examined by the Advisory Committee on Reactor Safeguards (ACRS) and the NRC Staff.

At a meeting of an Ad Hoc ACRS Subcommittee on Decay Heat Removal Systems held on May 5, 1981 and again at a meeting of the ACRS on May 7, 1981, the NRC Staff and the North Anna licensee discussed the North Anna decay heat removal systems and the modifications proposed by Commissioner Gilinsky. The Staff has concluded that the existing North Anna design conforms with all regulatory requirements and that, within the framework of current NRC review criteria, the proposed modifications would not improve these systems significantly. The licensee described the design bases for the LPI and RHR systems and the ability of the plant to cope with a spectrum of abnormal situations involving accidental degradation of the normal systems' capacity. Further, results were presented of a study conducted for the NRC Staff by Brookhaven National Laboratory which indicated there would be little or no safety improvement with addition of a heat exchanger in the LPI system. The study pointed out that the North Anna LPI system already includes in its design a significant improvement over that of the Surry system analyzed in the Reactor Safety Study, WASH-1400. This improvement reduces potential for one accident sequence (S_2C) which was a major contributor to the core melt probability calculated for Surry.

Also presented to the Subcommittee by the Staff was an updated description of preliminary Task Action Plan A-45, on the Unresolved Safety Issue of decay heat removal requirements, which had been presented to the ACRS on April 10, 1981.

We concur with Commissioner Gilinsky's desire to improve the reliability of decay heat removal systems. Probabilistic and deterministic assessments have concluded that potential failures in decay heat removal systems are major contributors to risk in operation of light water reactors. We conclude that aggressive development of Task A-45 and rapid implementation of any new decay heat removal requirements deriving from the Task Action Plan will best serve to improve the reliability of such systems for all LWRs, including North Anna. We have listed below several suggestions which we believe will improve the Task Action Plan, and we urge the Commission to take these into account while assigning a high priority to Task A-45.

- The plan should give careful consideration to alternative decay heat removal systems, such as those used in some foreign LWRs and reviewed in the current Sandia study being conducted for RES.

- Probabilistic studies such as RSSMAP, RSS, and IREP should furnish valuable insights in assessing system improvements, but engineering evaluation should not rely solely on these studies.

- The estimated completion date, 1984, is not likely to be realized for the ambitious program outlined unless the Commission assigns high priority to the work and allocates Staff resources to assure its timely completion.

We understand that a draft Task Action Plan for Task A-45 will be available in the very near future. We request the opportunity to comment on the plan in more detail.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

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

October 11, 1962

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

Subject: REPORT ON NUCLEAR FUEL SERVICES, INC.

Dear Dr. Seaborg:

At its forty-fourth meeting, October 4-6, 1962, the Advisory Committee on Reactor Safeguards considered the site proposed by Nuclear Fuel Services, Inc., for a plant to process reactor fuels. The Committee had the benefit of discussion with representatives of the applicant and members of the AEC staff, of the reports listed below, and of a visit by a subcommittee to the site.

The site is located in a sparsely populated area in Cattaraugus County, New York, about thirty miles southeast of Buffalo. The proposed plant has a nominal capacity of 1000 kg. per day of nuclear fuels of various types. It will employ a number of chemical processes, most of which have already been used on a substantial scale at other locations. The Committee has not reviewed these processes or the equipment design in detail.

The applicant estimates that for typical operating conditions none of the general public will be exposed to an integrated radiation dose in excess of the limits of 10 CFR Part 20. The most severe accident postulated would not cause exposures in excess of the guides suggested in 10 CFR Part 100. While these limits would not be exceeded, even if releases several times those estimated by the applicant were to occur, such increase in load is not visualized.

It is the opinion of the Advisory Committee on Reactor Safeguards that the site selected may be considered as suitable for a fuel reprocessing plant of the type of capacity proposed with reasonable assurance that it may be operated without undue hazard to the health and safety of the public.

Mr. K. R. Osborn did not participate in the discussions of this project.

Sincerely yours,

/s/

F. A. Gifford, Jr.
Chairman

References:

1. Application for Construction Permit & License for a Spent Fuel Processing Plant, Part A - General Information, dated July 25, 1962.
2. Safety Analysis, Spent Fuel Processing Plant, Part B - Vol. I and Vol. II, dated July 1962.
3. Letter from Scharfeld, Bechhoefer, Baron & Stambler, dated September 7, 1962, transmitting additional information to Part B.

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

December 26, 1962

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

Subject: REPORT ON NUCLEAR FUEL SERVICES, INC.

Dear Dr. Seaborg:

At its forty-fifth meeting on December 13-15, 1962 at Oak Ridge, Tennessee, the Advisory Committee on Reactor Safeguards considered the nuclear fuel processing plant proposed by Nuclear Fuel Services, Inc. to be constructed at the state-owned, Springville site located southwest of Buffalo, New York. In its letter of October 11, 1962, the Committee commented on the suitability of the site for the proposed operations. The Committee had the benefit of oral presentations by representatives of Nuclear Fuel Services, Inc., Bechtel Corp., the AEC Regulatory Staff and its consultant, and of the reports listed.

The process to be used is a batch dissolution of fuel elements which are usually chopped into small pieces. The plant will be designed to handle a nominal throughput of 1000 kilograms of uranium per day. A variety of types of fuel with varying exposure time histories are to be processed, the upper limit of which is approximately represented by the following parameters:

Burnup	20,000	mwd/ton
Specific power	27.5	mw/ton
Irradiation time	2	years
Load factor	85	percent
Cooling time	150	days

Similar chemical processing operations have been conducted on a production basis at various Commission-owned plants for several years. A prototype fuel element chopping operation has been carried on at Oak Ridge for about three years. This experience furnishes an adequate basis for plant design.

The Committee believes that this plant can be designed and constructed with reasonable assurance that it may be operated without undue hazard to the health and safety of the public.

Mr. K. R. Osborn did not participate in the discussions of this project.

Sincerely yours,

/s/

F. A. Gifford, Jr.
Chairman

References:

1. Application for Construction Permit & License for a Spent Fuel Processing Plant, Part A - General Information, dated July 25, 1962.
2. Safety Analysis, Spent Fuel Processing Plant, Part B - Vol. I and Vol. II, dated July 1962.
3. Letter from Scharfeld, Bechhoefer, Baron & Stambler, dated September 7, 1962, transmitting additional information to Part B.
4. Application for Construction Permit & License for a Spent Fuel Processing Plant, Part B -- Safety Analysis, Amendment No. 1, dated October 12, 1962.
5. Letter from Scharfeld, Bechhoefer, Baron & Stambler, dated October 24, 1962, transmitting seven (7) drawings referred to in Amendment No. 1.
6. Amendment No. 2 dated November 23, 1962 to Application for Construction Permit & License.

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

July 19, 1965

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

Subject: REPORT ON NUCLEAR FUEL SERVICES, INC.

Dear Dr. Seaborg:

At its sixty-fourth meeting, July 8-10, 1965, the Advisory Committee on Reactor Safeguards considered the application of Nuclear Fuel Services, Inc., for a provisional operating license for its Springville, New York irradiated nuclear fuel processing plant. The Committee commented on the suitability of this site for the proposed operation in its October 11, 1962 report and provided a brief description of the proposed plant design and operation in its report of December 26, 1962.

Subcommittee meetings were held at the Springville site on June 1, 1965 and in Washington on July 7, 1965. During its present review, the Committee had the benefit of discussion with representatives of Nuclear Fuel Services, Inc., Bechtel Corporation, and the New York State Atomic and Space Development Authority, and considered the reports listed.

Construction of the NFS irradiated fuel processing plant is nearly complete. Spent fuel is now being received and stored under license, and chemical processing operations are expected to begin near the end of this year. The chemical processing operations involved are basically the same as those that have been conducted on a production basis at Commission-owned plants for many years. Fuel elements will be mechanically chopped into small pieces prior to dissolution, an operation that has been studied for several years on a prototype basis at Oak Ridge.

The NFS processing operations are organized in such a way that major adjustments and shifts in the process streams will be made between processing of different types of fuel batches. This will require considerable reliance to be placed on administrative control to achieve safe plant operation, and the Committee has accordingly reviewed examples of administrative procedures in some detail. The applicant has placed limitations on the kinds and enrichment of fuel elements that will be processed.

It is the opinion of the Committee that this facility can be operated as proposed without undue hazard to health and safety of the public.

Mr. D. A. Rogers did not participate in the review of this project.

Sincerely yours,

/s/

W. D. Manly
Chairman

References:

1. Amendment #3 dated December 10, 1962.
2. Amendment #1 to Part A, dated February 12, 1963.
3. Letter dated July 1, 1963 from Walton A. Rodger, Nuclear Fuel Services, Inc. with attached Submission No. 1 Final Safety Analysis Report, dated July 1, 1963.
4. Letter dated October 10, 1963 from Scharfeld, Bechhoefer, Baron & Stambler with attached Submission No. 2 Final Safety Analysis Report, dated October 10, 1963.
5. Letter dated December 6, 1963 from Walton A. Rodger, NFS with attached Submission No. 3 Final Safety Analysis Report, dated December 9, 1963.
6. Letter dated February 25, 1964 from Walton A. Rodger, NFS with attached Submission No. 4 Final Safety Analysis Report, dated February 25, 1964.
7. Submission No. 5 - Final Safety Analysis Report, dated March 23, 1964.
8. Submission No. 6 - Final Safety Analysis Report, dated April 20, 1964.
9. Submission No. 7 - Final Safety Analysis Report, dated April 29, 1964.
10. Submission No. 8 - Final Safety Analysis Report, dated June 12, 1964.
11. Letter dated July 3, 1964 from W. A. Rodger, NFS to Mr. Alexander E. Aikens, Jr., AEC.
12. Submission No. 9 - Final Safety Analysis Report, dated June 30, 1964.
13. Submission No. 10 - Final Safety Analysis Report, dated June 30, 1964.
14. Letter dated July 20, 1964 from W. A. Rodger, NFS to Mr. Alexander E. Aikens, Jr., AEC.
15. Submission No. 11 - Final Safety Analysis Report, dated June 30, 1964.
16. Submission No. 12 - Final Safety Analysis Report, Revision 2, dated August 1, 1964.
17. Submission No. 13 - Final Safety Analysis Report, Revision 1, dated May 30, 1964.

References - Nuclear Fuel Services, Inc.

18. Submission No. 14 - Final Safety Analysis Report, Revision 2, dated August 15, 1964.
19. "Table 6-36a - Accountability Sample Summary", single page, Revision 1, dated May 30, 1964.
20. Submission No. 15 - Final Safety Analysis Report, Revision 2, dated August 20, 1964.
21. Submission No. 16 - Final Safety Analysis Report, Revision 2, dated August 20, 1964.
22. Submission No. 17 - Final Safety Analysis Report, Revision 2, dated September 23, 1964.
23. Submission No. 18 - Final Safety Analysis Report, undated, received October 14, 1964.
24. Submission No. 19 - Final Safety Analysis Report, Revision 3, dated October 2, 1964.
25. Letter dated October 19, 1964 from S. L. Reese, NFS with attached Submission No. 20 - Final Safety Analysis Report, dated October 20, 1964.
26. Submission No. 21 - Final Safety Analysis Report, dated October 26, 1964.
27. Submission No. 22 - Final Safety Analysis Report, dated October 31, 1964.
28. Submission No. 23 - Final Safety Analysis Report, Revision 1, dated October 31, 1964.
29. Letter dated January 15, 1965 from Walton A. Rodger, NFS; letter dated January 11, 1965 from Scharfeld, Bechhoefer, Baron & Stambler with attached "Part A, General Corporate Financial and Technical Information - Information Subsequent to Construction Permit Submission No. 1."
30. Submission No. 24 - Final Safety Analysis Report, dated April 9, 1965.
31. Letter dated January 25, 1965 from Walton A. Rodger, NFS to Mr. A. E. Aikens, Jr., AEC
32. Letter dated February 17, 1965 from Walton A. Rodger, NFS to Mr. A. E. Aikens, Jr., AEC.
33. Letter dated February 15, 1965 from W. A. Rodger, NFS to Alexander E. Aikens, Jr., AEC
34. Letter dated March 31, 1965 from Walton A. Rodger, NFS to Mr. A. E. Aikens, Jr., AEC.
35. Letter dated May 3, 1965 from W. A. Rodger, NFS to Mr. Alexander E. Aikens, Jr., AEC, with attachment.
36. Letter dated May 24, 1965 from W. A. Rodger, NFS to Mr. Alexander E. Aikens, Jr., AEC, with attachment.

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

March 6, 1960

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

Subject: NUCLEAR POWER PLANTS IN CALIFORNIA

Dear Mr. McCone:

This is in reply to the letter of February 27, 1960, addressed to the Chairman of the Advisory Committee on Reactor Safeguards signed by A. R. Luedecke, General Manager. This letter and the accompanying memorandum acquainted the Committee for the first time with the proposals of the Westinghouse Electric Corporation and the General Electric Company for pressurized water and boiling water nuclear power plants, respectively, with capacities of approximately 300 MWE. The letter informed us that these companies had been negotiating with the Southern California Edison Electric Company for a reactor in the Los Angeles area and with the Pacific Gas and Electric Company for a reactor in the San Francisco area. In addition the letter informed us of the interest of the City of Pasadena and the Bureau of Power and Light of the City of Los Angeles in a 50-100 MWE boiling water reactor under the Second Round cooperative arrangement.

Because of the preliminary stage of planning for these cases, the Committee has not been furnished with preliminary hazards summary reports or staff analyses. Accordingly the views expressed in this letter must necessarily be of a general nature and may be subject to revision upon receipt of further information.

In reply to the request for an advisory report from the Advisory Committee on Reactor Safeguards on the feasibility and acceptability of locating the proposed reactors in the Los Angeles area and in an area within a fifty-mile radius of San Francisco in terms of the possible hazards associated with inversion and earthquake conditions, the following advice is given.

With respect to seismic considerations, we understand that it is present utility industry practice in California to locate generating stations at least one mile from known surface faults; and to design and construct these stations using local codes supplemented by special analyses and increased seismic design factors for those critical plant components necessary to maintain the station on the line. In addition, in the case of a nuclear reactor facility, special analyses and increased seismic design factors are needed for those reactor plant systems whose failure could result in a release of radioactive material. With these precautions, the Committee believes the reactor facility would be adequately protected against seismic disturbance.

With respect to the question, raised on page 2 of the subject letter, concerning the specific consideration given to the inversion question in connection with various reactor projects, inversion frequency information is invariably included in hazards summary reports by the applicant and considered by the ACRS in addition to other pertinent factors affecting site selection and safety. The attached appendix is a tabulation of inversion frequencies for a number of sites, culled from these reports and United States Weather Bureau sources.

Referring to the frequency of inversion conditions, the situation of the Southern California coastal strip (south of San Francisco) is essentially unique in the United States. The semipermanent Pacific high pressure area induces a slow, large-scale, persistent subsiding motion in the atmosphere there. Air, warmed by this descent, contacts the coastal water surface which is cold as a result of upwelling. By this mechanism an inversion is formed; and the air layer extending up to a few thousand feet above the surface becomes a trap for air pollution.

Whereas persistent poor dispersion (stagnation) conditions of meteorology, lasting several days, may be expected on the average once per year anywhere east of the Rockies, the frequency of such episodes in the Southern California coastal strip is of the order of several per month. For example, during a two-year period, from July 1956 through June 1958, the Los Angeles weather was of the "smog warning" type 164 days.

For the Southern California Edison Electric Company reactor, the three locations given in the letter of February 27, 1960, cover a very considerable area. These locations have meteorological conditions varying from those approaching the area east of the Rockies to those characteristic of the Southern California coastal strip. On the basis of rather meager information it appears that the reactors proposed in the letter can be so designed and constructed that suitable sites can be found within the locations given. A specific reactor and its site should be given a detailed review at the earliest opportunity.

As to the expected joint proposal of the City of Pasadena and the Bureau of Power and Light of the City of Los Angeles regarding a 50-100 MWE boiling water reactor, no specific site has been identified. It should be mentioned that there are, within Los Angeles County, few if any sites on which a power reactor, at the present state of technology, could be built and operated with the degree of assurance of protecting the health and safety of the public that the Commission has previously afforded. This is due to the combination of high population density and the factors described above for the Southern California coastal strip. For these reasons some other location, as referred to in General Luedecke's letter, is preferable.

The Pacific Gas and Electric Company site approximately fifty miles north of San Francisco was identified to the ACRS. We presently do not have any reason to expect that detailed information which will be compiled and evaluated later will reveal significant factors weighing seriously against this location as a power reactor site. It has not yet been demonstrated to the ACRS that the vapor suppression system can be relied upon to protect the health and safety of the public.

In selecting a site for a high power reactor, consideration should be given to an adequate exclusion radius and the population density, not only in the immediate vicinity, five to ten miles, but also for greater distances. Obviously the lower the population density the better. The meteorology of the Southern California coastal strip is so unfavorable for dissipating pollutants that this area should be avoided if it is coupled with a high population density. In theory a reactor can be so designed, constructed, and operated that it will offset the unfavorable meteorology and high population density. Because of the present limited experience with the operation of power reactors and the large power level of the proposed reactors, the provision of an adequate degree of safety in practice may require an extreme of conservative design and containment.

The opportunity to render advice upon a reactor proposal at an early stage is valuable. This is especially true in cases whose significance is of this magnitude.

Sincerely yours,
/s/ Leslie Silverman

Leslie Silverman
Chairman

cc: A.R.Luedecke

Attachment: Appendix consisting
of four pages.

FREQUENCY OF SURFACE-BASED INVERSIONS
(Percent of Total Seasonal Hours)

The following are estimates based on limited data, length of day-night periods and synoptic features pertinent to areas.

REGION		Winter	Spring	Summer	Fall
San Diego	- coastal	60%	15	5	40
	- inland	75	40	30	60
Los Angeles	- coastal	60	15	5	45
	- inland	80	40	30	60
San Francisco	- coastal	65	20	5	45
	- inland	75	35	30	50

(Appendix to letter of 3/6/60
to AEC Chairman re Nuclear
Power Plants in California)

FREQUENCY OF SURFACE-BASED INVERSIONS

(Percent of total seasonal hours)

Reactor Site	Winter	Spring	Summer	Fall
Idaho Falls	47%	42	50	52
Oak Ridge	55	57	51	59
Dresden	37		53	
Indian Point	26		38	
Shippingport	12	19	23	27
Yankee	55		49	
CANEL	33	23	26	32
Enrico Fermi	26		22	
Argonne	37	38	53	53
Hanford	57	45	38	57

Above statistics based on information obtained from Reactor Hazards Reports.

(Appendix to letter of 3/6/60
to AEC Chairman re Nuclear
Power Plants in California)

FREQUENCY OF SURFACE-BASED INVERSIONS

5 years data, USWB

STATION	LST	Winter	Spring	Summer	Fall
San Diego	0700	61%	12	3*	36
Oakland	0700	66	21	2*	46
Tatoosh Is, Wash.	0700	14	14	21	38
*Reflects prevailing sea-breeze effect.					
Nashville, Tenn.	0900	21		1	
Boise, Idaho	0800	65		9	
Portland, Me.	1000	23		1	
St. Paul, Minn.	0900	35		3	
Tampa, Fla.	1000	6		2	
Phoenix, Ariz.	0800	93		14	

The above indicate the frequency of observations, taken at a specified time each day for a five-year period, showing a surface-based inversion. It would be expected that inland desert areas of California would show inversion frequencies similar to those of Boise and Phoenix.

(Appendix to letter of 3/6/60
to AEC Chairman re Nuclear
Power Plants in California)

FREQUENCY OF INVERSIONS BASED ABOVE SURFACE

BUT BELOW 1800 FEET

5 years data, USWB

STATION	Winter	Spring	Summer	Fall
San Diego	25%	25	49	41
Oakland	29	40	63	43
Brownsville, Texas	19	25	12	10
Burwood, Ia.	10	7	3	8
Jacksonville, Fla.	21	23	20	16
Norfolk, Va.	27	31	25	20

San Diego and Oakland frequencies of 0700 LST.

Other stations frequencies for 2100 or 2200 LST.

Inversions based below 1800 ft. would be expected to occur more frequently during nighttime and early morning hours.

(Appendix to letter of 3/6/60
to AEC Chairman re Nuclear
Power Plants in California)

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

May 12, 1958

Honorable Lewis L. Strauss
Chairman, U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: OAK RIDGE NATIONAL LABORATORY RESEARCH REACTOR (ORR)

Dear Mr. Strauss:

The Oak Ridge National Laboratory Research Reactor was reviewed at the Sixth Meeting of the Advisory Committee on Reactor Safeguards, May 9, 1958, at the request of the Commission. The Committee, before it received statutory status, had previously submitted its recommendations on location and design of this reactor 1/ to the Commission.

For the present review, the Committee had access to information referenced below, 2/-9/ inclusive. In addition, representatives from Oak Ridge presented amplifying comments orally.

The physics and engineering of reactors of the Oak Ridge Research type are well understood. Considerable operating experience has been accumulated. Moreover, the Oak Ridge staff has demonstrated its ability to operate research and testing reactors.

The Advisory Committee on Reactor Safeguards concludes, in agreement with its initial recommendation and with the Hazards Evaluation Branch, that there is reasonable assurance that the Oak Ridge National Laboratory Research Reactor can be operated without endangering the health and safety of the public.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman
Advisory Committee on
Reactor Safeguards

Herzel Plaine

cc: K. E. Fields, GM
H.L. Price, DL&R
ACRS Members and RHG

References - See Page 2

1/ ACRS Fifth Meeting, April 21-23, 1954; Eighth Meeting, October 21-24, 1954; Tenth Meeting, February 3-4, 1955; Eleventh Meeting, March 3, 1955. See Letter C. Rogers McCullough to C. K. Beck, January 29, 1958.

2/ The Oak Ridge National Laboratory Research Reactor Safeguard Report, by F. T. Binford, T. E. Cole, and J. P. Gill, October 7, 1954; Vol. I, ORNL 1794; Vol. II, TID-10083.

3/ A Method of Disposal of Volatile Fission Products from an Accident in the Oak Ridge Research Reactor, by F. T. Binford and T. H. J. Burnett, August 2, 1956, ORNL-2086.

4/ The Oak Ridge National Laboratory Research Reactor (ORR), A General Description, by T. E. Cole, J. P. Gill, January 17, 1958, ORNL-2240.

5/ Letter J. A. Swartout to H. M. Roth, ORR Test Operation, February 28, 1958, and attachments.

6/ Dispersion of Airborne Activity from a Cold Cloud Accident to the ORR, by U. S. Weather Bureau Office, Oak Ridge, Tenn., April 9, 1958.

7/ Report to ACRS by Division of Licensing & Regulation on Oak Ridge Research Reactor (ORR), April 15, 1958.

8/ Meteorological Aspects of the Oak Ridge Research Reactor, by Special Projects Section, Office of Meteorological Research, Weather Bureau (Donald H. Pack), April 22, 1958. (Presented at the ACRS Meeting.)

9/ Comments on ORR by John F. Newell, April 25, 1958 (presented at the ACRS Meeting).

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: X-10 ANNEALING

Dear Mr. McCone:

The method proposed by Oak Ridge National Laboratory for the release of stored energy in the graphite of the X-10 reactor was reviewed by the Advisory Committee on Reactor Safeguards at its Seventeenth Meeting on July 23-25, 1959. Details of the planned energy release are contained in report ORNL-2725.

The Committee concurs with the Oak Ridge National Laboratory that the X-10 stored energy can be released safely by the proposed method which involves allowing the temperature of the graphite to rise slowly under carefully controlled conditions to a predetermined value higher than normal operating temperature. The energy release program will require approximately three days to complete and will be carried out at a time to be determined by the Oak Ridge National Laboratory management.

Dr. Henry W. Newson did not participate in these reviews and discussions.

Sincerely yours,

/S/
C. Rogers McCullough
Chairman

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

References:

ORNL-2725 - Safeguard Report on the Proposed Method of Annealing Graphite in the X-10 Reactor, 4/30/59.

Division of Licensing and Regulation Report to the ACRS on the Proposed Method of Annealing Graphite in the X-10 Reactor, 7/1/59.

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

July 11, 1967

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

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

Dear Dr. Seaborg:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved by the applicant and the Regulatory Staff during construction of the reactors. On the basis of the foregoing comments, the Committee believes that the proposed Oconee Nuclear Station can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/
N. J. Palladino
Chairman

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

References:

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

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: REPORT ON OCONEE NUCLEAR STATION UNIT NO. 1

Dear Dr. Seaborg:

During its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application of the Duke Power Company for a license to operate Unit 1 of the Oconee Nuclear Station at power levels up to 2568 MW(t). The Committee met with the applicant during its 124th meeting, August 13-15, 1970 and Subcommittee meetings were held on June 23, 1970, at the site and on July 31, 1970 and September 9, 1970, in Washington, D. C. In the course of the review, the Committee had the benefit of discussions with representatives and consultants of the applicant, the Babcock and Wilcox Company, the Bechtel Corporation, and the AEC Regulatory Staff, and of study of the documents listed.

The Oconee Station is located in a rural area of Oconee County, South Carolina. The nearest population center is Anderson, 21 miles south, with a population of about 41,000. The minimum exclusion distance for the completed three-unit power station will be one mile and the Low Population Zone radius will be six miles containing about 3,400 people. The water supply for the plant is taken from Lake Keowee which was created by the applicant. The lake and associated recreational facilities are expected to attract a transient population to the area.

The application covers Oconee Units 1, 2, and 3, but this report applies only to Unit 1, which will employ the first of the Babcock and Wilcox two-loop, four-pump, pressurized water reactor, nuclear steam supply systems. The three units are designed to be nearly identical, but some facilities and services are shared in various arrangements. The Committee has reviewed the temporary arrangements necessitated by operation of Unit 1 while Units 2 and 3 are still under construction. It is believed that the proposed physical measures and administrative procedures to isolate the operating unit from construction activities are adequate.

The Committee reported to you on the construction permit application for this power station on July 11, 1967. At that time the proposed operating power was to have been 2452 MW(t); the current proposal for operating at powers as high as 2568 MW(t) is justified by the applicant, primarily on the basis of a flatter power distribution. Prior to operation at the higher power level, reactor operation should be reviewed by the Regulatory Staff.

The prestressed concrete containment building is similar to those for the Palisades and Point Beach plants which have been reviewed recently for operation.

The Committee recommends that the applicant accelerate his studies of means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram when required during anticipated transients. As solutions develop and are evaluated by the Regulatory Staff, appropriate action should be proposed and taken by the applicant on a reasonable time scale. The Committee wishes to be kept informed.

The applicant has proposed using a power-to-flow ratio signal as a diverse means to cause shutdown of the reactor if emergency core cooling action should be initiated. The Committee believes it is necessary that either the equipment associated with this signal be demonstrated to be able to survive the accident environment for an adequate time or a different, diverse trip signal be employed. This matter should be resolved to the satisfaction of the Regulatory Staff.

The Committee suggests that developmental techniques, such as neutron noise analysis and use of accelerometers, be considered as an aid in ascertaining displacements, changes in vibration characteristics, and the presence of loose parts in the primary systems. The Committee notes the desirability of the continuing use of some thermocouples in the core.

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 purging technique after a suitable time delay subsequent to the accident. Relatively high off-site doses possibly could result following purging of the containment. The Committee recommends that purging systems be incorporated in the plant but that the primary protection in this regard should utilize a hydrogen control method which keeps the hydrogen concentration within safe limits by means other than purging. The hydrogen control system and provisions for containment atmosphere mixing and sampling should have redundancy and instrumentation suitable for an engineered safety feature; these should be made available within the first two years of power operation. The Committee wishes to be kept informed of the resolution of this matter.

The applicant stated that the amount of radioactivity in liquid wastes normally will not be greater than one percent of 10 CFR Part 20 limiting concentrations after dilution with the minimum flow (30 cfs) below the Keowee dam. Larger flows will have proportionately smaller limiting concentrations. The mean annual discharge from the Keowee dam is expected to be 1,100 cu. ft./sec. The off-gas system has holding tank and filtering capability and gas release rates are not expected to exceed a few percent of 10 CFR Part 20 limits.

In order to protect against the postulated consequences of the accidental dropping of a fuel element, the applicant has stated that either, he will install filters in the fuel pool building exhaust system, or the equivalent control and protection will be assured by another method. This matter should be resolved to the satisfaction of the Regulatory Staff within the first year of power operation.

Improved calculational techniques are being applied to the analysis of the efficacy of the emergency core cooling system in the unlikely event of a loss-of-coolant accident. Interim results appear to be acceptable, but further calculations are needed and some phenomena important to the course of the accident require further study. This matter should be resolved in a manner satisfactory to the Regulatory Staff prior to operation at power. The Committee wishes to be kept informed.

The reactor is calculated to have a positive moderator coefficient of reactivity at power which will become negative as boron is removed from the coolant concurrent with build-up of fission products and fuel burnup. The applicant plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses related to these tests.

A conservative method of defining pressure vessel fracture toughness should be employed that is satisfactory to the Regulatory Staff.

Other problems relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous reports to you should be dealt with appropriately by the Staff and applicant in the Oconee Unit 1 power plant 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 the Oconee Nuclear Plant Unit 1 can be operated at power levels up to 2568 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/
Joseph M. Hendrie
Chairman

Additional comments by Dr. W. R. Stratton are presented below:

"The high off-site doses which are stated to accompany the proposed purging operation are based on calculations which include a number of assumptions which I believe to be overly conservative. It is my opinion that the situation, should it ever arise, would be much less severe and that the proposed purge system would provide adequate protection for the health and safety of the public in this regard and therefore the additional hydrogen control equipment required by this letter is not necessary."

Attachment: List of References

References:

1. Amendment No. 7 to Duke Power Company Application for Oconee Nuclear Station, Units 1, 2, and 3, consisting of Final Safety Analysis Report, Volumes I and II, received June 4, 1969

2. Amendments Nos. 8 through 21 and Revised Amendment No. 13 to the License Application.

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 OCONEE NUCLEAR STATION UNITS 2 AND 3

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 Duke Power Company for a license to operate Units 2 and 3 of the Oconee Nuclear Station at power levels up to 2568 MW(t). This project was considered during a Subcommittee meeting near the site at Clemson, South Carolina, on July 23 and 24, 1973, subsequent to a tour of the plant. In the course of the review, the Committee had the benefit of discussions with representatives and consultants of the Duke Power Company, 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 11, 1967, and on operation of Unit 1 of the Oconee Nuclear Station on September 23, 1970.

The Oconee Nuclear Station is located in Oconee County, South Carolina. The nearest population center is Anderson, 21 miles southeast with a population of about 28,000. The water supply for the plant is taken from Lake Keowee.

The application for a construction permit for Units 1, 2 and 3 proposed initial operation of each unit at power levels up to 2452 MW(t) although the safety studies had been made for a power level of 2568 MW(t). The application for an operating license included a request for the higher power and the Committee agreed to this value for Unit 1, but recommended that the Regulatory Staff review operation of Unit 1 prior to allowing the full requested power for this first of a type. The Committee believes that this review should be completed and satisfactory performance of Unit 1 demonstrated before Units 2 and 3 operate at full licensed power.

The hot functional testing of Oconee Nuclear Station Unit 1 which was conducted in 1972 caused damage to some components, including reactor vessel internals. The design changes which were required for Unit 1 have been applied to Units 2 and 3. The Committee believes that these changes are acceptable and notes, in addition, that a loose parts monitoring system has been installed in each unit and that a vibration monitoring system is being tested in Unit 1.

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 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 Units 2 and 3 of the Oconee Nuclear Station can be operated at power levels up to 2568 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/

H. G. Mangelsdorf
Chairman

References Attached

References

1. Final Safety Analysis Report, Volumes I through IV

2. Amendments 22 through 42 to Application

3. DL Safety Evaluation for Oconee Nuclear Station, Unit 1, dated December 29, 1970, with Supplements 1, 2, and 3, dated March 24 and December 20, 1972, and July 10, 1973, respectively.

4. DL Safety Evaluation for Oconee Nuclear Station, Units 2 and 3, dated July 6, 1973, and Supplement 1, dated August 2, 1973

5. Duke Power Company letter dated July 27, 1972, transmitting a list of B&W Topical Reports

6. Duke Power Company letter dated November 20, 1972, furnishing information on auxiliary service water system for Oconee Units 2 & 3

7. Duke Power Company letter dated December 29, 1972 transmitting their analysis regarding the consequences of main steam and feedwater piping ruptures at the Oconee Station, Units 1, 2, and 3

8. Duke Power Company letter dated January 12, 1973, regarding the installation of flow restrictors in core flooding nozzles entering the reactor vessel at Oconee Units 1, 2, and 3

9. Duke Power Company letter dated March 2, 1973, concerning the analysis of reactor cavity and steam generator subcompartment pressure response

10. Duke Power Company letter dated April 4, 1973, furnishing comments of items under the heading "Units 1 & 2 and Units 1, 2, & 3 Operations"

11. Duke Power Company letter dated April 27, 1973 regarding quality assurance program for operation of Oconee Nuclear Station

12. B&W Interim Report on Fuel Densification for the Oconee 2 and 3 Reactors, May 1973

13. Duke Power Company Report No. 08-73.2, dated April 25, 1973, "Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Units 1, 2, and 3"

14. Duke Power Company letter dated May 1, 1973, on Oconee Units 2 and 3 Active Valve Operability

15. Duke Power Company letter dated May 3, 1973, regarding control circuits and safety related equipment

16. Duke Power Company letter, dated May 4, 1973, transmitting three reports:

References Continued

1) Failure of the operating Mechanism to Fully Open the Core Flood Line Isolation Valve CF-1
2) Failure of Reactor Building Spray Valves to Open During ES System Testing
3) March 6, 1973, 1A1 Reactor Coolant Pump Oil Fire Incident Report

17. BAW-1395 (Proprietary) "Oconee 2 Fuel Densification Report"

18. DL Technical Report on Densification of B&W Reactor Fuel, July 6, 1973.

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

August 28, 1964

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

Subject: REPORT ON OYSTER CREEK NUCLEAR POWER PLANT OF JERSEY CENTRAL POWER AND LIGHT COMPANY

Dear Dr. Seaborg:

At its fifty-seventh meeting, on August 24-26, 1964, the Advisory Committee on Reactor Safeguards considered the proposal of the Jersey Central Power and Light Company to construct and operate a nuclear power plant on Oyster Creek in New Jersey. This will be a 1600 MW(t) boiling-water type reactor with pressure absorption containment.

The Committee had the benefit of an oral presentation by representatives of the applicant and consultants and contractors, advice by the AEC Staff, and the reports cited. A Subcommittee meeting was held at the site on May 1, 1964, and a further Subcommittee meeting was held in Washington, D. C. on August 7, 1964.

Many details of the proposed design have not yet been completed. The applicant is continuing to study the limitation of maximum reactivity of individual control rods and the design of the reactor protection system. The following additional points should be given examination and consideration:

(1) Under some credible accident conditions, the dry well and absorption pool may require provisions for additional heat removal.

(2) In the unlikely event of a melt-down accident, a zirconium-water reaction may produce hydrogen. Provision should be made to prevent any hydrogen-oxygen reaction that would disrupt the containment.

(3) The adequacy of the reactor protection system when operating at partial recirculation flow rates should be established.

Estimates made by the applicant on halogen retention by absorption in water and by plate-out are based on limited data, and the consequences of the unlikely accident may be more severe than estimated. However, the Committee believes that more conservative assumptions would not make the proposal unacceptable.

With due regard to the above comments, the ACRS believes that the proposed reactor can be constructed at the proposed location with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References:
1. Part B, Preliminary Safeguards Summary Report, Application to the United States Atomic Energy Commission for Construction Permit and Operating License, Oyster Creek Nuclear Power Plant Unit No. 1, Jersey Central Power and Light Company, undated, received April 2, 1964.
2. Amendment No. 2, Application Reactor Construction Permit and Operating License, Oyster Creek Nuclear Power Plant Unit No. 1, Jersey Central Power and Light Company, dated June 26, 1964, with enclosures.

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

December 12, 1968

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

Subject: REPORT ON OYSTER CREEK NUCLEAR POWER PLANT UNIT NO. 1

Dear Dr. Seaborg:

During its 104th meeting, December 5-7, 1968, the Advisory Committee on Reactor Safeguards completed its review of the application by the Jersey Central Power and Light Company for a license to operate the Oyster Creek Nuclear Power Plant Unit No. 1 at power levels up to 1600 MW(t). During this review, the project has been considered at eight Subcommittee meetings (including one at the site) and four full Committee meetings. In the course of these discussions, the Committee has had the benefit of discussions with representatives of the Jersey Central Power and Light Company, the General Electric Company, the AEC Regulatory Staff and with consultants of these organizations. The Committee also had the benefit of the documents listed. The Committee previously discussed this project in a construction permit report dated August 28, 1964.

The Oyster Creek plant is the first of a new generation of boiling water reactors to be reviewed for an operating license; the increase of power level over that of previously licensed boiling water reactors is more than a factor of two. The time for construction of this plant was extended because of defective welds and stress-corrosion cracking in stainless steel portions of the pressure vessel envelope and internals. Items such as control rod stub tubes, nozzle safe-ends, and the core support ring were involved. These cracks were discovered during and after the system hydrostatic test. The causes of the stress-corrosion have not been definitely determined; however, studies to establish the effects of various contaminants are continuing. The Committee is satisfied that the repair procedures should prevent or minimize recurrence of stress-corrosion cracking.

The Committee wishes to emphasize the importance of periodic inspection of the high pressure coolant system in this and other reactors. The in-service inspection requirements for this reactor are stated in the Technical Specifications, and the Committee finds these adequate for initial operation. It is expected that experience with this first large BWR will give useful information regarding the practicality of inspection methods. The Committee endorses the applicant's proposal to review his in-service inspection program with the Regulatory Staff after four years of reactor operation. In view of the difficulties inherent in direct inspection of the bulk of the welds in the Oyster Creek pressure vessel after the reactor is in service, it is recommended that alternative means for assuring continued pressure vessel integrity be studied, and implemented to the degree practical.

It is recommended that supplemental and potentially more sensitive methods of primary system leak detection be studied, evaluated, and implemented if they provide significant improvements in measurement of leak rate, in the time needed to measure leak rate, or in distinguishing the nature of the leak. The study and evaluation should be completed within a year.

The emergency core cooling system will be supplemented in about a year by the addition of a third diesel generator. This extra source of power will allow the use of one feedwater pump (as well as one core spray system) in the case of the loss of off-site power. The Committee has reviewed the design criteria for this emergency Feedwater Coolant Injection System and recommends that the applicant submit the design for review by the Regulatory Staff prior to installation. In this regard, the Committee urges caution to avoid the overloading of cable trays.

The applicant has recently reviewed design and construction criteria in regard to the separation of redundant protection components and circuits. An audit of the Oyster Creek plant revealed some deficiencies in this respect, and the applicant is proceeding with a remedial program.

Studies are continuing on the possible effects of radiolysis of water in the unlikely event of a loss-of-coolant accident. These studies should be evaluated by the Regulatory Staff and appropriate measures taken as deemed necessary.

The applicant stated that instrumentation which senses radioactivity from the steam system can be used to provide early signs of gross failure of fuel elements. The Committee believes that, as operating experience is gained with the facility, the applicant should improve the utilization of this type of instrumentation for this purpose, particularly to provide the reactor operators with direct, early indication.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, the Oyster Creek Unit No. 1 can be operated at power levels up to 1600 MW(t) without undue hazard to the health and safety of the public.

Sincerely yours,

/s/
Carroll W. Zabel
Chairman

References:
1. Jersey Central Power and Light Company Application for Reactor Construction Permit and Operating License for Oyster Creek Unit No. 1, Amendments No. 3 through 5 and 7 through 48.

2. Jersey Central Power and Light Company telegram, dated October 11, 1967, regarding Request for Permit for Fuel Loading and Testing of Oyster Creek Reactor Prior to Completion of Review of Application for Provisional Operating License.

3. Jersey Central Power and Light Company letter, dated February 9, 1968, transmitting General Electric Summary Report, dated February 2, 1968, regarding Reactor Vessel Problems.

4. Jersey Central Power and Light Company letter, dated April 9, 1968, regarding Oyster Creek Pressure Vessel Repair Program.

5. Jersey Central Power and Light Company telegram, dated July 3, 1968, regarding Oyster Creek Reactor Vessel Repair.

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

November 17, 1970

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

Subject: REPORT ON OYSTER CREEK NUCLEAR POWER PLANT UNIT NO. 1

Dear Dr. Seaborg:

During its 126th meeting, October 15-17, 1970, and its 127th meeting, November 12-14, 1970, the Advisory Committee on Reactor Safeguards reviewed amendments to the application by Jersey Central Power and Light Company for a modification to the license to operate the Oyster Creek Nuclear Power Plant Unit No. 1 at increased power levels up to 1690 Mw(t). The operation of this plant was the subject of a report to you dated December 12, 1968. This matter was considered during Subcommittee meetings held in Washington, D. C. on October 14, 1970 and November 11, 1970. The Committee had the benefit of discussion with representatives and consultants of the Jersey Central Power and Light Company, the General Electric Company and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The applicant has requested this increase in power on the basis of favorable preoperational test results and operating experience. He has found that the full power recirculation flow rate capability is about 15% greater than that used in the provisional operating license application analysis. Because of an increased flow, the same thermal margins can be maintained at 1690 Mw(t) as were calculated for 1600 Mw(t). To further protect the reactor system the applicant stated that he will install two anticipatory scrams -- one actuated by electrical load rejection and the other by turbine trip.

During this review the Committee examined the efficacy of the Oyster Creek Emergency Core Cooling System in the light of results from the Commission's BWR FLECHT program. These experiments and associated analyses provide reasonable assurance as to the adequacy of the core sprays for the proposed power level.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned in its report of December 12, 1968, there is reasonable assurance that the Oyster Creek Nuclear Power Plant Unit 1 can be operated at power levels up to 1690 Mw(t) without undue risk to the health and safety of the public.

Additional remarks by Dr. David Okrent are attached.

Sincerely yours,

/s/

Spencer H. Bush
Acting Chairman

<u>References</u>

1) Amendments 55, 57, 58 and 63 to the License Application

2) Semi-Annual Report No. 1 (May 3, 1969-December 31, 1969)

3) Supplement No. 1 to Semi-Annual Report No. 1

4) Semi-Annual Report No. 2 (January 1, 1970-June 30, 1970)

ADDITIONAL REMARKS BY DR. DAVID OKRENT

I can concur with the Committee's conclusion on the basis of explicit statement of the following qualifying remarks:

1. That Oyster Creek Unit No. 1, which has only core sprays rather than both the core spray and core flooding systems included in Dresden Nuclear Power Station Unit 3, be provided on a reasonable time scale with both a controlled purging system and an independent means for coping with the buildup of hydrogen and oxygen by radiolysis and other mechanisms, as was recommended by the ACRS for Dresden Station Unit 3 in its report to you of July 17, 1970.

2. That, as recommended by the ACRS for Dresden Station Unit 3, studies be made for Oyster Creek Unit No. 1 of anticipated transients in the presence of a failure to scram, and that, as appropriate, design changes be made as are found to improve significantly the safety of the plant in this regard.

3. That, as recommended in several recent ACRS reports, special attention be given to assuring the continued integrity and isolability of the large number of small diameter instrument lines which penetrate the containment.

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 OYSTER CREEK NUCLEAR POWER PLANT NO. 1

Dear Dr. Seaborg:

During its 134th meeting, June 10-12, 1971, the Advisory Committee on Reactor Safeguards reviewed the application by Jersey Central Power and Light Company for a modification to the license to operate the Oyster Creek Nuclear Power Plant Unit No. 1 at increased power levels up to 1930 MW(t). The operation of this plant was the subject of a report to you dated December 12, 1968, and an increase in power from 1600 to 1690 MW(t) was discussed in a letter to you dated November 17, 1970. The current increase in power was considered during a Subcommittee meeting at the site on May 26, 1971. The Committee had the benefit of discussions with representatives and consultants of the Jersey Central Power and Light Company, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The proposed increase in maximum power is based on favorable operating experience, and on the application of the heat transfer correlation currently in use to evaluate core thermal-hydraulic performance in boiling water reactors.

A fifth relief valve will be added to the primary coolant system to prevent opening of the safety valves in the event of a severe pressure transient. This additional pressure relief is required only for power levels greater than 1865 MW(t), and the applicant proposes to install the valve before exceeding this value. The installation and testing of this valve should be completed to the satisfaction of the Regulatory Staff.

Performance of the emergency core spray cooling system has been reevaluated for 1930 MW(t) operation. The core spray system is acceptable for the proposed higher power operation in view 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 emergency core cooling systems (ECCS).

Off-site doses calculated for design basis accidents have been reexamined for 1930 MW(t) operation. The applicant proposes to reduce the allowable containment leak rate from 1.25 percent per day to 1.0 percent per day and to reduce the allowable primary coolant activity limit from 20 μCi/ml to 8 μCi/ml. With these provisions, the calculated doses based on the higher power level are no higher than those calculated previously for the current power level.

The applicant has examined methods for the control of the buildup of hydrogen in the containment which might follow in the unlikely event of a loss-of-coolant accident. A submittal is to be made which will describe the proposed approach, necessary hardware, procedures and calculational assumptions. The Committee wishes to be kept informed of the resolution of this matter.

Studies by the applicant have indicated that integrity of the spent fuel pool may not be maintained if a fuel cask is dropped into the pool. Some corrective measures have been identified, and the applicant stated that appropriate modifications will be made. The Regulatory Staff should follow this matter and assure implementation on an appropriate time scale.

The applicant has developed improved plans for in-service inspection of the main steam lines both inside and outside of containment. The Committee re-emphasizes its belief that continued attention be given to possible improvements in access to the reactor pressure vessel surfaces for augmentation of in-service inspection. The Committee recommends that inspection plans be implemented to the satisfaction of the Regulatory Staff.

The applicant has installed an atmospheric radioactivity monitoring system for leak detection in the containment drywell. The Committee recommends continued use and testing of this system and urges the applicant to search for means to improve leak detection sensitivity.

The applicant has proposed a testing procedure for the instrument lines that penetrate the containment. Further analyses of the effect of an instrument line failure on the integrity of the secondary building and of the off-site doses are being made. These studies will be reviewed by the Regulatory Staff.

The Committee feels that as means become available on a practical basis to provide information concerning the possibility of excessive vibration, structural damage, or loose parts within the reactor vessel, consideration should be given to their use in the Oyster Creek Plant.

The applicant is continuing 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 wishes to be kept informed of the resolution of this matter.

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 Oyster Creek Nuclear Power Plant can be operated at a power as high as 1930 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. 65 to the License Application, dated December 31, 1970
2. Semi-Annual Report No. 3 (July 1, 1970 - December 31, 1970)
3. Supplement No. 1 to Amendment No. 65, dated January 26, 1971
4. Supplement No. 2 to Amendment No. 65, dated June 4, 1971

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

November 16, 1976

Dr. G. Paulson
Assistant Commissioner for Science
Department of Environmental Protection
State of New Jersey
Trenton, NJ 08625

Dear Dr. Paulson:

The Advisory Committee on Reactor Safeguards has considered your request for comments on the preliminary analysis by Mr. Peter Davis of the probability of a catastrophic accident during the remaining lifetime of the Oyster Creek plant. It is unfortunately not possible for the ACRS to provide a detailed review of this study. We note, however, that Mr. Davis suggests that the incorporation of the automatic recirculation pump trip would lead to a substantial reduction in the probability of a core melt following a failure to scram on an anticipated transient.

The ACRS has been following closely the continuing evaluation of the matter of anticipated transients without scram (ATWS) since the question was raised by this Committee several years ago. The recirculation pump trip feature is included in most, if not all, of the newer boiling water reactors in operation or under construction, and in March 1976 the ACRS recommended that this feature be implemented promptly on all operating BWR's unless analyses have shown that such a trip is not required to mitigate the consequences of ATWS.

We understand that the Jersey Central Power and Light Company has made a commitment to install such a pump trip unless they are able to provide a convincing demonstration that other features incorporated in this plant make this trip unnecessary (see attachments).

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Attachments:
1) ACRS letter to L. V. Gossick
March 12, 1976
2) Jersey Central Power & Light
Company letter October 4, 1976

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

March 12, 1976

Lee V. Gossick
Executive Director
for Operations

Subject: RECIRCULATION PUMP TRIP TO HELP LIMIT THE CONSEQUENCES OF AN ATWS EVENT IN BWRs

It has recently come to the attention of the ACRS that not all operating boiling water reactors have installed a recirculation pump trip to help limit the consequences of an ATWS event. Such a trip was proposed by applicants for construction permits in 1971 and for operating licenses in 1972.

The Committee believes that the recirculation pump trip represents a substantial improvement in protection for BWRs and should be implemented promptly on all operating BWRs unless analyses have shown that such a trip is not required to mitigate the consequences of ATWS.

Please provide the Committee with a list of those reactors in operation, or nearing operation, which do not have such a trip and a schedule for installation of this trip on a timely basis.

Sincerely,

Dade W. Moeller

Dade W. Moeller
Chairman

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111

MEMBER OF THE General Public Utilities Corporation SYSTEM

October 4, 1976
EATJM-66

RECEIVED
OCT 8 1976
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Mr. Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Stello:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Anticipated Transient Without Scram-
Recirculation Pump Trip

Your letter dated September 1, 1976, stated that you have found that a recirculation pump trip similar in design to the one described in your Anticipated Transient Without Scram (ATWS) status report dated December 9, 1975, and the General Electric Report NEDO-20626 will provide substantial additional protection for the mitigation of an ATWS event. You further requested that we inform you of our commitment to modify our plant accordingly and our schedule for doing so.

We do plan to modify the Oyster Creek Nuclear Generating Station to incorporate the recirculation pump trip, unless additional information is supplied by the industry during the finalization of your review that demonstrates that this modification will not significantly limit the consequences of an ATWS event for the Oyster Creek Station. At present we plan to implement the modification during the spring refueling outage of 1978.

We cannot, at this time, provide our schedule for submitting the detailed modification design to you for approval, nor can we now provide our procurement schedule. We will provide these by November 17, 1976.

Very truly yours,

Ivan R. Finfrock, Jr.

Ivan R. Finfrock, Jr.
Vice President

pk

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

November 9, 1982

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

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE NRC SYSTEMATIC EVALUATION PROGRAM REVIEW OF THE OYSTER CREEK NUCLEAR GENERATING STATION

During its 271st meeting, November 4-5, 1982, the ACRS reviewed the results of the Systematic Evaluation Program (SEP), Phase II, as it has been applied to the Oyster Creek Nuclear Generating Station. These matters were discussed also during a Subcommittee meeting in Washington, D.C. on October 26, 1982. During our review, we had the benefit of discussions with representatives of the General Public Utilities Nuclear Corporation, the Jersey Central Power & Light Company (Licensee), and the NRC Staff. We also had the benefit of the documents listed below.

This is our third review of the application of Phase II of the SEP. We reported to you on our reviews of the Palisades and R. E. Ginna plants in letters dated May 11, 1982 and August 18, 1982, respectively. The first report included comments also on the objectives of the SEP and the extent to which they have been achieved. Our review of the SEP in relation to the Oyster Creek plant has led to no changes in our previous findings regarding the program as reported in our letter on the Palisades plant.

The remainder of this letter relates specifically to the SEP review of the Oyster Creek plant.

Although the Oyster Creek plant is the first boiling water reactor (BWR) to be reviewed under the SEP, the findings by the NRC Staff regarding the number and nature of topics for which the plant did not meet current criteria were not markedly different from those for the Palisades and Ginna plants. A large number of these topics related to the adequacy of the design to resist extreme external phenomena (flooding, tornado, earthquake), and most of the remaining topics related to balance-of-plant items, or items of a generic nature not specific to BWRs.

Of the 137 topics to be addressed by the SEP, 30 were not applicable to the Oyster Creek plant, and 24 were deleted because they were being reviewed generically under either the Unresolved Safety Issues (USI) program or the TMI Action Plan. Of the 83 topics addressed in the Oyster Creek review, 38 were found to meet current NRC criteria, and 5 were found 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.

For all or parts of the remaining 38 SEP topics, the Oyster Creek plant was found not to meet current criteria. These topics were addressed by the Integrated Plant Safety Assessment, and various resolutions have been proposed.

The Integrated Assessment has not yet been completed for all or parts of 13 topics, 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 topics are of such a nature that hardware backfits may be required by the NRC Staff for their resolution. The Staff's assessments will be provided in a supplemental report that will be available for review in connection with the application for a full-term operating license (FTOL) for the Oyster Creek plant.

For all or parts of 10 topics included in the Integrated Assessment, the NRC Staff concluded that no backfit is required. We concur.

For the remaining topics for which the assessment has been completed, the NRC Staff requires the addition or modification of structures or equipment in about half of the cases, and the development or modification of procedures or Technical Specifications in the other half. The Licensee does not agree with the NRC Staff's requirements for three of the hardware backfits, two of which relate to leakage detection systems, and for five of the required procedural backfits, all of which relate to the Technical Specifications. Our comments on these areas of disagreement are given below.

In connection with Topic III-4.A, Tornado Missiles, the NRC Staff's concern is that all of the components that could be used for shutdown heat removal could be disabled by multiple missiles transported by a single tornado. The NRC Staff requirement is that at least one system capable of shutdown heat removal should be protected against tornado missiles. The Licensee believes that the total loss of shutdown heat removal capability as a result of multiple missile strikes is of such low probability that no protection is needed. We agree that this is a very low probability event, but we do not believe that the probability has been quantified with any significant degree of certainty. Further, we recognize the importance of having at least one shutdown heat removal system available following a tornado, or other extreme environmental event. We recommend therefore that one such system be protected against tornado missiles (and other possible effects of high winds, such as sandstorms) unless the cost of such protection clearly outweighs the reduction in risk.

For Topic III-5.B, Pipe Break Outside Containment, the NRC Staff requires an automatic local leakage detection system for the isolation condenser piping, which is lagged and is outside of containment. The system should be capable of detecting leaks from stable cracks before they grow to be too large. The detectable leak rate is based on an analysis of tight cracks whose length is two to four times the wall thickness. The Licensee contends that the leak rate corresponding to such a crack will be large enough that it can be detected by visual inspection. If they can show this to the NRC Staff's satisfaction, we feel such an approach is simple and reliable. If they cannot, an automatic leak detection system would be a more delicate but acceptable approach.

Topic V-5, Reactor Coolant Pressure Boundary Leakage Detection, relates to the requirement for a reliable system to detect leakage inside the containment with a sensitivity adequate to provide early warning so that timely actions can be taken to preclude a pipe break. The Licensee believes that the existing system, utilizing the containment sump, is satisfactory. We believe that this matter should be resolved in a manner satisfactory to the NRC Staff.

In connection with Topics V-5, VI-7.A.3 and VI-10.A, the NRC Staff requires that certain limiting conditions of operation, and surveillance or test requirements, be added to the Technical Specifications for the Oyster Creek plant. We concur.

Topics XV-16 and XV-18 relate to the calculated radiological consequences for certain design basis accidents; thyroid doses calculated in accordance with current criteria are considerably in excess of the siting criteria. To correct this situation, the NRC Staff requires that the iodine concentration in the reactor coolant be limited by appropriate changes to the Technical Specifications. We believe that this proposal is acceptable.

As was the case for the Palisades and Ginna plants, a plant-specific probabilistic risk assessment (PRA) was not available for the Oyster Creek plant. Because a plant-specific PRA was not available, the NRC Staff utilized in its Integrated Assessment the results of the Millstone Unit 1 PRA developed as part of the Interim Reliability Evaluation Program (IREP), suitably modified and interpreted to reflect the differences between the two plants. The PRA study for Oyster Creek addressed 20 of the topics included in the Integrated Assessment, a somewhat greater number than for either Palisades or Ginna. However, because the Millstone IREP did not include extreme external events, topics relating to design criteria for such events could not benefit from the use of PRA in the Integrated Assessment.

Our conclusions regarding the Oyster Creek SEP review are similar to those for the Palisades and Ginna plants:

1. The SEP has been carried out in such a manner that the stated objectives have been achieved for the most part for the Oyster Creek plant and should be achieved for the remaining plants in Phase II of the Program.

2. The actions taken thus far by the NRC Staff in its SEP assessment of the Oyster Creek plant are acceptable.

3. The ACRS will defer its review of the FTOL for the Oyster Creek Nuclear Generating Station until the NRC Staff has completed its actions on the remaining SEP topics and the USI and TMI Action Plan items.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. U. S. Nuclear Regulatory Commission Draft Report, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Oyster Creek Nuclear Generating Station," NUREG-0822, September 1982.
2. NRC Staff consultants' reviews of the Oyster Creek Integrated Plant Safety Assessment Report consisting of consultant reports from H. S. Isbin, Z. Zudans, J. M. Hendrie, and S. H. Bush, dated October 22, October 25, October 21, and October 20, 1982, respectively.
3. U.S. Nuclear Regulatory Commission Safety Evaluation Reports, Oyster Creek Systematic Evaluation Program Topics, Volumes 1 through 3, dated October 1982.

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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. 20545

Subject: REPORT ON PALISADES PLANT

Dear Dr. Seaborg:

At its seventy-ninth meeting, November 10-12, 1966, eightieth meeting, December 8-10, 1966, and eighty-first meeting, January 12-14, 1967, the Advisory Committee on Reactor Safeguards considered the application by Consumers Power Company for a construction permit for the Palisades Plant, which is to be located on the eastern shore of Lake Michigan, on a site sixteen miles north of Benton Harbor. The proposed Palisades Plant contains a pressurized water reactor having an initial maximum operating power level of 2200 MWt, a design power level of 2450 MWt, and an ultimate stretch capability of 2640 MWt. During its review, the Committee had the benefit of discussions with representatives of the applicant, Combustion Engineering Inc., Bechtel Corporation, the AEC Regulatory Staff and its consultants, and the documents listed below. Subcommittee meetings were held at the site on September 15, 1966 and in Washington, D. C. on November 9, 1966, December 7, 1966 and January 6 and 7, 1967.

The Palisades Plant containment structure is a steel-lined concrete shell provided with steel prestressing tendons that carry the principal loads. The design makes provision for inspection and corrosion control of the prestressing tendons. The effect on containment design of damping of earthquake notions by the soil will be resolved between the applicant and the Regulatory Staff, based on the outcome of proposed analyses, and of experiments at the site.

Emergency core cooling systems (ECCS) are proposed, consisting of a high pressure system with three 200 gpm pumps, a low pressure system with two 3000 gpm pumps, and four accumulators. The applicant has stated that the ECCS will be designed to prevent fuel and cladding damage that would interfere with adequate emergency core cooling, and to limit the cladding-water reaction to less than approximately 1%, for all break sizes in the primary system piping up to the double ended rupture of the largest primary coolant pipe, for any break location, and for the applicable break time. An analysis will be performed to show the expected margin in the design to prevent melting of the cladding.

The reactor has a negative power coefficient of reactivity. The magnitude and sign of the moderator coefficient are subject to details of the final design. The applicant is making detailed studies of this question and in addition will carefully study the effect of possible positive moderator coefficients on the potential xenon oscillations, on postulated rod-ejection accidents, and on the unlikely large scale loss-of-coolant accident. The applicant reported that, if necessary, the moderator coefficient will be made more negative by the addition of solid burnable poisons.

The Committee believes that the question of moderator coefficient and the detailed design of the emergency core cooling systems, including the effects of blowdown forces, should be reviewed carefully by the Regulatory Staff as soon as sufficient details are available. The Committee would like to be kept informed of the results.

The applicant is basing the requirement for and design of a containment iodine removal system on the outcome of a meteorological observation program to be conducted at the site over the next two years, and is providing at this time only the capability for adding this equipment if it is required. The applicant suggests that observations at this site will justify the use in calculations of more rapid atmospheric diffusion than given in the AEC siting guides (TID-14844). This matter can be resolved at the time of review for the operating license.

The applicant described the general arrangement of plant protection instrumentation and gave design criteria for this equipment. The AEC Regulatory Staff should review the detailed design of this instrumentation before its fabrication and be assured that its performance and reliability are commensurate with the importance of its function. Also, the applicant stated that he would review the need for redundancy in rod-position instrumentation.

The Committee continues to emphasize the importance of quality assurance during fabrication of the primary system, and inspectability during its service life, and recommends that the applicant give further attention to the possibility of improvements in these areas.

The Committee believes that the items mentioned above can be resolved by the applicant and the Regulatory Staff during construction and that the proposed reactor can be built at the Palisades 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. Palisades Plant Facility Description and Safety Analysis Report, General Information and Volumes I, II, and III.
2. Amendment No. 1 to Consumers Power Company Application for Reactor Construction Permit and Operating License, dated September 2, 1966.
3. Amendment No. 2 to Consumers Power Company Application for Reactor Construction Permit and Operating Oicense, dated September 12, 1966.
4. Amendment No. 3 to Consumers Power Company Application for Reactor Construction Permit and Operating License, dated October 31, 1966.
5. Amendment No. 4 to Consumers Power Company Application for Reactor Construction Permit and Operating License, dated November 29, 1966.
6. Amendment No. 5 to Consumers Power Company Application for Reactor Construction Permit and Operating License, dated December 2, 1966.
7. Amendment No. 6 to Consumers Power Company Application for Reactor Construction Permit and Operating License, dated December 30, 1966.
8. Amendment No. 7 to Consumers Power Company Application for Reactor Construction Permit and Operating License, dated January 10, 1967.
9. Amendment No. 8 to Consumers Power Company Application for Reactor Construction Permit and Operating License, dated January 13, 1967.

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

January 27, 1970

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

Subject: REPORT ON PALISADES PLANT

Dear Dr. Seaborg:

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

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

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

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

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

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

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

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

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

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

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

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Palisades Plant can be operated at power levels up to 2200 MWt without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References:

1. Final Safety Analysis Report for the Palisades Plant

2. Amendments No. 9-19 to license application

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 SYSTEMATIC EVALUATION PROGRAM, PHASE II, AND ITS APPLICATION TO THE PALISADES PLANT

During its 265th meeting, May 6-8, 1982, the ACRS reviewed the results of the Systematic Evaluation Program, Phase II, as it has been applied to the Palisades Plant. These matters were discussed also at a subcommittee meeting in Washington, D.C. on April 15, 1982. During our review we had the benefit of discussions with representatives of the Consumers Power Company (Licensee) and the NRC Staff. We also had the benefit of the documents listed below.

The Systematic Evaluation Program (SEP) was initiated in 1977 to review the designs of older operating nuclear power plants in order to provide:

a. an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed,

b. a basis for deciding how these differences should be resolved in an integrated plant review, and

c. a documented evaluation of plant safety.

The original SEP objectives were:

1. The program should establish documentation that shows how the criteria for each operating plant reviewed compare with current criteria on significant safety issues, and should provide a rationale for acceptable departures from these criteria.

2. The program should provide the capability to make integrated and balanced decisions with respect to any required backfitting.

3. The program should be structured for early identification and resolution of any significant deficiencies.

4. The program should assess the safety adequacy of the design and operation of currently licensed nuclear power plants.

5. The program should efficiently use available resources and minimize requirements for additional resources by NRC or industry.

The program objectives were later interpreted to ensure that the SEP also provide safety assessments adequate for conversion of provisional operating licenses (POLs) to full-term operating licenses (FTOLs).

Ten plants are now included in Phase II of the SEP. The Palisades Plant is the first for which the safety reviews and the Integrated Plant Safety Assessment have been completed.

We believe that the program itself, its scope, and its methodology have been appropriate for providing the information listed in Items a. through c., above, and in meeting the objectives listed as Items 1. through 3., above. As is discussed below, the SEP can only meet objective 4. in part. With regard to objective 5., there has been a learning period. It is our understanding that the interaction between the NRC Staff and licensees is becoming more efficient.

Of the 137 topics to be addressed by the SEP, 23 were not applicable to the Palisades Plant. Twenty-four topics were found to be identical with one or more matters being reviewed by the NRC Staff in connection with the resolution of Unresolved Safety Issues (USI) or TMI Action Plan requirements. The evaluation and resolution of these topics are not included as a part of the SEP for the Palisades Plant. We believe that this was appropriate from a procedural standpoint; any other approach would have required duplication of effort within the NRC Staff or would have extended considerably the completion of Phase II of the SEP. It must be recognized, however, that because of this separation of topics, all of the SEP objectives, as listed above, have not been achieved completely at this stage of the program. For example, the documentation of objective 1 is not yet complete, the integrated and balanced decisions on backfitting did not involve all of the omitted topics (objective 2), and the assessment of safety adequacy (objective 4) is not complete.

Of the 90 topics addressed in the SEP for the Palisades Plant, 57 were found to meet current criteria or were found to be acceptable on other defined bases. In addition, as a result of modifications made by the Licensee during the review, two additional topics and parts of three others were found to meet current criteria. We have reviewed the assessments and conclusions of the NRC Staff in relation to these topics and have found them appropriate.

For all or parts of 31 SEP topics, the Palisades Plant was found not to meet current criteria. These topics were addressed by the Integrated Assessment and have been resolved in various ways: For five topics, addition or modification of equipment was required for resolution; for 12 topics, resolution required only the development or modification of procedures or Technical Specifications; and for five topics, a decision was reached that no backfit was required.

We have reviewed the treatment of these topics, and have found no reason to disagree substantially with the NRC Staff's approach, assessments, and recommended actions for resolution.

There remain nine topics for which the Integrated Assessment has not been completed, chiefly because additional information is to be provided by the Licensee. This information consists of calculations, evaluations, and various other submittals that are required by the NRC Staff as bases for its assessments and decisions. None of these topics is minor in importance to safety and most will not be easier to resolve than topics already considered. The NRC Staff expects to report the resolution of these topics in a supplemental report in the near future. Until this is done, the Integrated Assessment is incomplete by a further increment beyond that resulting from deletion of the USI and TMI topics from the SEP. As a result our endorsement and acceptance of the SEP and its application to the Palisades Plant is limited to what we have learned of the treatment of a representative group of the SEP topics. If the remaining topics are treated in a comparable manner, the objectives of the SEP will have been achieved.

The question of management performance and capability has been considered in relation to the operational history and record of regulatory compliance of the Palisades Plant. This is important because the NRC Staff has recommended changes in procedures as remedial measures for several of the SEP topics. We have noted reports of relatively recent changes in management organization, intentions, and performance. The results are encouraging but not conclusive in view of the limited length of time during which they have been observed. Nevertheless, we are satisfied with those resolutions involving procedural changes, chiefly because we are satisfied that the NRC Staff has exhibited a suitable level of concern about their effective implementation, and we are satisfied that they will continue to monitor management performance at the Palisades Plant.

A plant-specific Probabilistic Risk Assessment (PRA) was not available for the Palisades Plant. The NRC Staff utilized a limited risk assessment in portions of the Integrated Assessment, in a qualitative and subjective manner. We believe that this was done with appropriate caution and with adequate appreciation of the limitations of the analysis and the data as they applied to the Palisades Plant. We note, however, that the draft Calvert Cliffs PRA, which was utilized in the limited risk assessment, has not been available to us for use in connection with our review.

For some plants in Phase II of the SEP, and for additional plants in Phase III, it is expected that more complete plant-specific PRAs will be available. We believe that these will be useful and highly desirable as inputs to the Integrated Assessment portion of the SEP.

The Integrated Plant Safety Assessment portion of the SEP for the Palisades Plant will be documented in NUREG-0820 and its Supplements. However, the safety evaluation reports for each of the 90 topics are included only by reference. Since these reports are an essential and important part of the SEP and constitute the only documentation of why 57 topics were found to meet current criteria or were acceptable on other defined bases, we believe that these reports should be published or otherwise made more generally available than simply by putting them in the Public Document Room.

It is expected that the results of the SEP evaluations will be among the bases used in considering the conversion of the provisional operating license for the Palisades Plant to a FTOL. We believe that these results will be very useful for this purpose. However, we defer our review of an FTOL for the Palisades Plant until such time as the remaining SEP topics have been assessed and disposed of and the topics related to the USI and TMI items have been addressed appropriately, at least in a manner similar to that being used for new operating licenses.

Our conclusions can be summarized 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 Palisades Plant and should be achieved for the remaining plants in Phase II of the program.

2. The actions taken thus far by the NRC Staff in its SEP assessment of the Palisades Plant are acceptable.

3. The ACRS will defer its review of the FTOL for the Palisades Plant until the NRC Staff has completed its actions on the remaining SEP topics and the USI and TMI items.

Dr. William Kerr did not participate in consideration of this matter.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. U.S. NRC Draft Report, "Integrated Plant Safety Assessment, Systematic Evaluation Program" - Palisades Plant, NUREG-0820 dated April 1982.
2. Letter from G. C. Lainas, Division of Licensing, USNRC, to P. G. Shewmon, Chairman, ACRS, dated 4/30/82, Subject: NRC Staff Consultants' Review of Palisades Draft Integrated Plant Safety Assessment Report transmitting Consultant Reports from R. J. Budnitz, S. H. Bush, J. M. Hendrie, H. S. Isbin, and Z. Zudans

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

November 12, 1975

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

Subject: REPORT ON THE PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, & 3

During its 187th meeting, November 6-8, 1975, the Advisory Committee on Reactor Safeguards reviewed the application of the Arizona Public Service Company, the Salt River Project Agricultural Improvement and Power District, the El Paso Electric Company, the Public Service Company of New Mexico, the Arizona Electric Power Cooperative, Inc., and the Southern California Edison Company (Applicants) for a permit to construct the Palo Verde Nuclear Generating Station Units 1, 2, and 3. The members of this joint application have designated the Arizona Public Service Company as the Project Manager and Operating Agent with full authority to construct and operate the power station. The site of the proposed power station was visited on October 20, 1975, and a Subcommittee meeting was held the same day in Phoenix, Arizona. During its review the Committee had the benefit of discussions with the Nuclear Regulatory Commission (NRC) Staff, representatives and consultants of the Applicants, Combustion Engineering, Inc., and the Bechtel Corporation. The Committee also had the benefit of the documents listed.

The Palo Verde 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. This policy allows for a reference system that involves an entire facility design or major fraction of a design outside the context of a license application. 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."

This power station will be located in a sparsely populated section of Maricopa County, Arizona, about 36 miles west of the nearest boundary of Phoenix, Arizona, which is the designated population center. The exclusion area will be within the boundaries of the 3800 acre site.

The Palo Verde Nuclear Generating Station will utilize three two-loop pressurized water nuclear steam supply systems, each having a power of 3817 MW(t). The turbine generators will be supplied by the General Electric Company and will be oriented so as to minimize damage should turbine failure occur.

The ultimate heat sink for each reactor of this station will consist of two seismic Category I essential spray ponds, which will be of a size to provide sufficient cooling water for 30 days with the reactor shutdown. These ponds will be filled initially from an on-site storage reservoir which in turn will receive water from the water reclamation plant of the city of Phoenix.

The Applicants described their investigations of the geologic and seismic characteristics of the site and the surrounding region. The Committee recognizes the extensiveness of these studies but recommends that approval of the site and the plant seismic design bases be subject to receipt of a favorable report from the U.S. Geological Survey. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

Each of the Palo Verde units employs a cylindrical, steel-lined, reinforced, post-tensioned concrete containment structure with a free volume of about 2.7×10^6 ft^3. The design pressure and temperature are 60 psi gauge and 300 degrees Fahrenheit respectively. The Committee believes that this containment design is satisfactory for the Palo Verde Station. The Applicants compared the parameters of this containment with the containment model assumed for the Reference System-80 to show that, in regard to calculated cladding temperatures following a loss-of-coolant accident, the Palo Verde design is conservative. The Committee believes that this analysis is adequate for this phase of the project.

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 Palo Verde design is in this category. These units will use the 16X16 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 are less for 16X16 assemblies than for the 14X14 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 these units.

In conjunction with their presentation of results of analyses of events subsequent to a postulated loss-of-coolant accident, the Applicants discussed development of best judgment calculations for the same class of accidents. Preliminary results indicated that a considerable margin of safety may exist; however, the methodology used has not been subjected to critical evaluation. The Committee recognizes the potential importance of studies of this type in the improvement and optimization of design of safety features and encourges the Applicants and the NRC Staff to accelerate their efforts to this end.

The Palo Verde Station will be the first commercial nuclear power plant in the state of Arizona. For this reason, the Committee recommends that the Applicants 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 Palo Verde Station design for features that could reduce the possibility and consequences of sabotage.

The Committee recommends that the NRC Staff and the Applicants review further the design features that are intended to prevent the occurrence of 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 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 Palo Verde Nuclear Generating Station Units 1, 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.

Additional comments by Mr. Myer Bender, Mr. William Stratton, and Mr. Milton Plesset are attached.

Sincerely,

WKerr

W. Kerr
Chairman

References Attached.

Additional Comments by Mr. Myer Bender, Mr. William Stratton, and Mr. Milton Plesset

Questions related to soil liquefaction are still being evaluated. The Applicants have indicated their willingness to commit very substantial funds for special features to avoid the consequences of such an event. In view of the soil conditions, the climatology, the meteorology, and the population distribution at the Palo Verde Site, the off-site consequences of an accident resulting from soil liquefaction in the absence of the proposed features would probably not cause intolerable damage to public health and safety. The likelihood of such an event is sufficiently low to make the cost effectiveness of the soil liquefaction provisions doubtful from a cost-risk-benefit standpoint.

References:

1. Palo Verde Nuclear Generating Station Preliminary Safety Analysis Report (PSAR) Volumes I-XV

2. Amendments 1-13 to the PSAR

3. Division of Reactor Licensing Safety Evaluation Report, dated October 10, 1975

4. Letter, dated August 22, 1975, Arizona Public Service Company to the Nuclear Regulatory Commission (NRC), Division of Reactor Licensing (DRL), concerning geology and tectonics in southeastern Arizona

5. Letter, dated August 4, 1975, Arizona Public Service Company to the DRL, concerning the Quality Assurance Program and control of mineral rights within the exclusion area

6. Letter, dated January 10, 1975, Arizona Public Service Company to the DRL, concerning the calculated thyroid dose at the exclusion zone boundary

7. Letter, dated September 27, 1974, Arizona Public Service Company to the DRL, concerning Anticipated Transients Without Scram

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

December 15, 1981

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

SUBJECT: ACRS REPORT ON THE PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3

Dear Dr. Palladino:

During its 260th meeting, December 10-12, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of the Arizona Public Service Company, the Salt River Project Agricultural Improvement and Power District, the El Paso Electric Company, the Public Service Company of New Mexico, and the Southern California Edison Company (Applicants) for a license to operate the Palo Verde Nuclear Generating Station Units 1, 2, and 3. The joint applicants have designated the Arizona Public Service Company as the Project Manager and Operating Agent with full authority to construct and operate the power station. The project was considered at a Subcommittee meeting in Phoenix, Arizona on November 23-24, 1981, and members of the Committee toured the facility on November 23, 1981. In its review the Committee had the benefit of discussions with representatives of the Arizona Public Service Company, Combustion Engineering, Inc., Bechtel Power Corporation, 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 the Palo Verde Nuclear Generating Station Units 1, 2, and 3 in a report dated November 12, 1975 to the NRC Chairman.

The Palo Verde application is submitted in accordance with the Commission's regulations 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. NRC policy stated in the Federal Register (42 FR 34395 and 43 FR 38954) allows for a reference system that involves an entire facility design or major fraction of a design outside the context of a license application. For this application the reference system is the Combustion Engineering standard nuclear steam supply system known as its Standard Reference System 80. This design has been reviewed by the ACRS and discussed in its report dated December 15, 1981, "Final Design Approval for Combustion Engineering, Inc. Standard Nuclear Steam Supply System (Standard Reference System 80)".

This power station is located in a sparsely populated section of Maricopa County, Arizona, about 36 miles west of the nearest boundary of Phoenix, Arizona. The nearest densely populated center is Sun City, Arizona, about 35 miles east-northeast of the site, which had a 1980 population of about 57,800 persons. Palo Verde is the first commercial nuclear power station to be operated by Arizona Public Service Company and the first in the state of Arizona.

The Palo Verde Nuclear Generating Station uses three System 80 pressurized water nuclear steam supply systems designed by Combustion Engineering, Inc. Each of these has a design core power output of 3800 MWt. The turbine generators are oriented so as to minimize plant damage should turbine failure occur. The containment is a steel-lined, prestressed concrete cylindrical structure with a hemispherical dome and a design pressure of 60 psig. The cooling tower makeup is supplied from treated sewage effluent from the city of Phoenix.

The Committee's review included consideration of the management organization and capability, and the operator training program. The organizational plan for technical support of the operating plant is still being formulated. The Committee notes that the Arizona Public Service Company management personnel have extensive experience in both commercial and other nuclear plant operation and construction. The utility anticipates using many of its installation surveillance staff members as part of the technical support team. The ACRS encourages this organizational arrangement, but believes the Applicant should promptly analyze the skill requirements needed to support operations and make certain that the necessary capabilities will be available when needed. In order that the Committee be kept informed, we request an update on the organizational arrangement in about one year from this date.

The Committee notes that Arizona Public Service Company has a training simulator in operation at the Palo Verde site. The Committee's review indicated that the training program is being developed and that use of the plant simulator is still in the process of being integrated into the program. The Committee recommends that Arizona Public Service Company examine industry-sponsored programs concerning effective use of simulators for training and make certain that its approach takes account of current understanding of simulator training limitations.

Discussion with the Arizona Public Service Company staff indicated that emergency operating procedures for dealing with off-normal plant behavior are incomplete. Development of such procedures should be expedited to provide maximum time to make use of them in the operational training program.

In the Palo Verde design the primary system does not include capability for rapid, direct depressurization when the plant has been shut down. This places extra importance on the reliability of the auxiliary feedwater system and makes it necessary that the NRC Staff and the Applicant assure the availability and dependability of this system for a wide variety of transients. It also places extra requirements on the continued integrity of the two steam generators as the only method of heat removal immediately after shutdown. The ACRS recommends that the NRC Staff and the Arizona Public Service Company give additional attention to the matter of shutdown heat removal for Palo Verde and develop a detailed evaluation and justification for the position judged to be acceptable. The Committee wishes to be kept informed.

Arizona Public Service Company should expand its studies on systems interactions and systems reliability.

A number of items have been identified as Outstanding Issues, Confirmatory Issues, and proposed License Conditions in the NRC Staff's Safety Evaluation Report dated November 1981. The ACRS is satisfied with the progress on these topics and believes that they should be resolved in a manner satisfactory to the NRC Staff.

Our approval of the operation of this plant is contingent upon the satisfactory completion of construction and preoperational testing. For this reason, we request that, prior to fuel loading on Unit 1, a report be provided to the Committee describing significant construction deficiencies and their disposition, effectiveness of the quality assurance program, and results of the preoperational test program. In addition, a review of the startup experience on Unit 1 should be made prior to fuel loading on Unit 2 and the Committee kept informed.

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 Palo Verde Nuclear Generating Station Units 1, 2, and 3 can each be operated at power levels up to the design core power output of 3800 MWt without undue risk to the health and safety of the public.

Additional comments by ACRS member M. Bender and ACRS members H. W. Lewis and M. S. Plesset are presented below.

Sincerely yours,

J. Carson Mark

J. Carson Mark
Chairman

Additional Comments by ACRS Member M. Bender

The NRC requirements for instrumentation to follow the course of an accident have been generally outlined in Regulatory Guide 1.97. The ACRS has concentrated most of its attention on instrumentation to detect inadequate core cooling, sometimes called pressure vessel coolant level measuring instrumentation. The Regulatory Guide 1.97 requirements and the emphasis on measurement of vessel coolant levels both seem to have confused the real accident diagnosis requirements.

The proposed coolant level indicators could only have value under quiescent conditions. The proposed devices, differential pressure indicators and heated junction thermocouples, require considerable information about hydraulic conditions, pressure distribution, and density variations in the primary coolant circuit to be useful for unambiguous interpretation of changing coolant inventory in the reactor core. A full understanding of mass and energy distribution and related physical behavior of the nuclear system would be needed to make such information diagnostically useful under most accident conditions. The main value would appear to be for conditions where the system has been depressurized and the coolant state is known, for example, prior to refueling. Such knowledge does not appear relevant to the circumstances of primary concern such as accident conditions comparable to the TMI-2 event.

Regulatory Guide 1.97 has a mixture of requirements, some directed to pre-accident symptom identification, some to actual surveillance of rapidly changing transients, and some to surveillance of accident recuperation conditions. Although all of these requirements could be justified under some circumstances, it is likely that, if everything listed in the guide were provided, the operators could be overwhelmed by the informational detail and their diagnostic capability actually impaired.

At a time when unambiguous accident diagnostic information is urgently needed, a maze of indicating and analytical devices that might confuse the operators hardly makes sense. I propose the following criteria as a basis for determining accident diagnostics adequacy.

1. Does the operator have a well-defined set of signals to guide his emergency response to important accidents?

2. Do the emergency procedures enable the operator to avoid misinterpretation of those signals under circumstances where accident diagnosis is needed in conjunction with emergency actions?

3. In accident recovery is the sensor capability adequate to enable the operators to establish whether a stable and safe operating condition is being maintained until the system can be brought to cold shutdown and reliable decay heat removal functions assured?

4. If fuel failures occur, is there capability to determine whether the failures are of minor or major significance (clad reaction

with water and fuel melting); whether bulk quantities of radioactive nuclides have been released to the primary coolant circuitry, the containment interior, or are leaking from containment; and whether the containment boundary is jeopardized by overpressure or overtemperature?

Only a few additions to the pre-TMI-accident instrumentation appear necessary to address these considerations. However, to be certain that necessary information is available, the actions required of operators during accidents must be thoroughly examined. Emergency procedure guidance is now being developed by the nuclear steam supply equipment vendors. This guidance must be converted into usable procedures that may be testable on nuclear plant simulators. Palo Verde and a few other installations have simulators that might be used for this purpose. Those operating organizations having appropriate simulation equipment should give priority attention to proving the effectiveness of the diagnostic equipment in conjunction with proposed emergency procedures in order to verify diagnostic adequacy. No serious effort in this direction appears to have been initiated up to this time.

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

We do not wish to belabor the points we made in our addendum to the ACRS letter dated November 17, 1981 on the St. Lucie Plant Unit 2, but they are as relevant here as there. The Staff continues to accept instruments that do not provide an unambiguous measure of liquid level in the pressure vessel, and continues to lack an adequate rationale therefor. We do not find fault with the Applicants for their efforts to be responsive to the Staff, but are concerned about the proliferation of inadequately considered requirements, of which this is only one example. To sanctify an ambiguous indication of core water level is to play with fire. In this particular case (heated thermocouples in a separator tube), not only dynamic effects, but a pressure vessel full of high-void-fraction water will spoof the instrument, and tend to lull the operator into a false sense of security about the coolant inventory. In that specific case, the instrument will indicate that the vessel is nearly full.

None of the above is meant to suggest that we oppose the provision of instrumentation to follow the course of an accident or to detect the onset of inadequate core cooling - unambiguous diagnosis of accident conditions through improved instrumentation and training is a high priority. Our concern is a piecemeal and incoherent approach to the problem, as exemplified here.

References:

1. Arizona Public Service Company, "Palo Verde Nuclear Generating Station, Final Safety Analysis Report," with Amendments 1 through 6.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Palo Verde Nuclear Generating Station, Units 1, 2, and 3," NUREG-0850, dated November 1981.

3. Combustion Engineering, Inc., "System 80 CESSAR FSAR," with Amendments 1 through 5.
4. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80," NUREG-0852, dated November 1981.

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

December 16, 1981

MEMORANDUM FOR: Mr. William J. Dircks
Executive Director for Operations

FROM: Mr. Raymond F. Fraley
Executive Director
Advisory Committee on Reactor Safeguards

Subject: OCCUPATIONAL EXPOSURES AT PALO VERDE NUCLEAR GENERATING STATION AND OTHER SYSTEM 80 PLANTS

During its review of the CESSAR-80/Palo Verde Nuclear Generating Station, Units 1, 2, and 3, the ACRS was provided estimates of the annual collective occupational dose associated with the operation of each unit at the Palo Verde Station which may average well over one-thousand person rem. In view of the fact that these units are based on a standard design which supposedly incorporates application of the ALARA principle, the members expected somewhat lower dose estimates.

In this connection, it should be noted that the occupational dose estimates may have been unduly conservative and therefore misleading. The Committee's review did not provide an opportunity to examine the basis for these dose estimates in detail and this should be done to determine if they result from the CESSAR-80 design, the balance of plant design, the proposed method of operation, or other factors.

The Committee urges attention to this matter regarding the Palo Verde Station and the CESSAR-80 standardized plant design. The ACRS Subcommittee on Reactor Radiological Effects would be pleased to discuss this matter further with the NRC Staff.

Raymond F. Fraley

Raymond F. Fraley
Executive Director
ACRS

cc: C. Mark, ACRS
H. Denton, NRR
E. Goodwin, NRR
D. G. Eisenhut, NRR
W. E. Kreger, NRR

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: NORTHERN STATES POWER COMPANY - PATHFINDER ATOMIC POWER PLANT

Dear Mr. McCone:

The Pathfinder Reactor proposal was described to the Advisory Committee on Reactor Safeguards by the applicant for a license, the Northern States Power Company and his contractor, Allis-Chalmers Manufacturing Company, on August 14, 1959, at its eighteenth meeting. Subsequently discussions were held among the applicant, his contractor, members of the ACRS Subcommittee and the Hazards Evaluation Branch.

At its nineteenth meeting on September 10-12, 1959, the Advisory Committee on Reactor Safeguards, upon request of the Director of the Division of Licensing and Regulation, gave further consideration to the Pathfinder proposal and held a discussion with representatives of the Division of Licensing and Regulation.

The Advisory Committee on Reactor Safeguards has concluded that the proposed site near Sioux Falls, South Dakota, is a suitable location for a power reactor of this rating. However, there are a number of special design features in the proposal on which the applicant is conducting further theoretical analysis and for which a significant research and development effort is planned. These include an in-core nuclear superheater, aluminum alloy fuel elements and a circulation control system.

The applicant has not yet proposed alternate solutions using presently known technology which may be substituted in case his research and development program does not confirm his expectations. The Committee therefore cannot now conclude that the proposed facility can be constructed and operated at this site without undue hazard to the health and safety of the public.

Sincerely yours,

C. Rogers McCullough
Chairman

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

References

1) ACNP-5905 - Pathfinder Atomic Power Plant Safeguards Report, March 10, 1959.

2) Supplement No. 1 to the Safeguards Report (ACNP-5905), June 26, 1959.

3) Division of Licensing and Regulation report to the ACRS, April 28, 1959.

4) Division of Licensing and Regulation report to the ACRS, August 11, 1959.

5) U. S. Weather Bureau Comments on ACNP-5905 "Supplement No. 1 to Pathfinder Atomic Power Plant Safeguards Report, dated June 26, 1959," August 3, 1959

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C.

December 14, 1959

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

Subject: NORTHERN STATES POWER COMPANY - PATHFINDER ATOMIC POWER PLANT

Dear Mr. McCone:

The Northern States Power Company proposal to build a power development reactor near Sioux Falls, South Dakota, was considered by the Advisory Committee on Reactor Safeguards at its twenty-second meeting, December 10-11, 1959. The proposal had previously been considered at the sixteenth, eighteenth, and nineteenth meetings and in two Subcommittee meetings with the Hazards Evaluation Branch staff and the applicant. The necessary information is contained in the reports listed below. The Hazards Evaluation Branch and representatives of the applicant participated in the discussion.

This reactor is a 204 thermal megawatt boiling water unit with internal superheater, which new developmental feature is one of its important objectives. Design to incorporate the superheater, controls necessary for the unique fuel loading, and a new optional power control introduce special problems. Use of aluminum-clad uranium oxide fuel, previously untried under these plant operating conditions, is proposed.

The applicant has research and design work in progress on these problems which must be resolved before operation as proposed. Part of this development includes partial power range operation and testing in the reactor before full-scale operation.

The applicant proposes stepwise approach to full power and to designed superheat. Operation with reduced superheat or without superheator fuel, with Zircaloy fuel cladding instead of aluminum or without using the special control features has been suggested as feasible by the applicant, and proposed in case of need.

The Committee concludes that the approach suggested as above by the applicant will enable construction of the reactor proposed at its site with reasonable assurance that it can be operated without undue hazard to the health and safety of the 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 - 12/16/59
bc: L. K. Olson, GC - 12/16/59
H. H. Plaine, OGC - 12/16/59

References

1) ACNP-5905 - Pathfinder Atomic Power Plant Safeguards Report, March 10, 1959.

2) Supplement No. 1 to the Safeguards Report (ACNP-5905), June 26, 1959.

3) Amendment No. 3 to the Safeguards Report (ACNP-5905), November 2, 1959.

4) Division of Licensing and Regulation report to the ACRS, November 25, 1959.

5) Comments by the Office of Health and Safety, December 2, 1959.

6) Comments on "Amendment No. 3" by the U. S. Weather Bureau, December 4, 1959.

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

February 1, 1960

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

Subject: NORTHERN STATES POWER COMPANY - PATHFINDER ATOMIC POWER PLANT

Dear Mr. McCone:

At its twenty-second meeting, December 10-11, 1959, and as stated in its letter of December 14, 1959, to you, the Advisory Committee on Reactor Safeguards considered that "the approach suggested as above by the applicant will enable construction of the reactor proposed at its site with reasonable assurance that it can be operated without undue hazard to the health and safety of the public."

Amendments No. 4 and No. 5 to the application have since been received and were considered at the twenty-third meeting of the Committee on January 28-30, 1960. The opinion of the Committee is not changed due to these amendments.

Sincerely yours,

/s/ Leslie Silverman

Leslie Silverman
Chairman

cc: A.R.Ludecke, GM
W.F.Finna, CGM
H.L.Price, DL&R
ACRS Members & Dr. Duffey

<u>References</u>:

1) Amendment No. 4 to the Safeguards Report (ACNP-5905), and Supplement No. 3, November 20, 1959.

2) Amendment No. 5 to the Safeguards Report (ACNP-5905), December 18, 1959.

3) U. S. Weather Bureau Comments on Amendment No. 4 and Supplement No. 3 to Safeguards Report ACNP-5905, December 22, 1959.

4) Division of Licensing and Regulation Report to the ACRS on Pathfinder Amendments No. 4 and No. 5, January 12, 1960.

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

November 5, 1960

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

Subject: REPORT ON NORTHERN STATES POWER COMPANY - (PATHFINDER)

Dear Mr. McCone:

At its twenty-ninth meeting November 3-5, 1960, the Advisory Committee on Reactor Safeguards considered the hazards associated with the fuel elements of the Pathfinder reactor.

The Committee believes that the proposed substitution of Zircaloy II for aluminum in the fuel elements of the boiling region of the core constitutes an acceptable change in the design of these elements.

The Committee notes that there is no experience with the superheater fuel elements under the proposed operating conditions. Although we doubt that this feature of the reactor poses an undue hazard, this uncertainty should be resolved by adequate prior testing of the fuel elements. The Committee believes that operation of the superheater elements can be carried out without undue hazard to the public if increases in power level are made stepwise and are supplemented by a concurrent fuel element testing program.

The Committee urges that the significant problems of the superheater design be resolved by the results of the Division of Reactor Development program on the superheater concept, (such as BORAX V and loop tests) before a high power load is placed on the superheater of this reactor.

Sincerely yours,

Sgd/ LESLIE SILVERMAN

Leslie Silverman
Chairman

Reference:
Supplement No. 4 to Preliminary Safeguards Report, Pathfinder Atomic Power Plant dated August 15, 1960

cc: A. R. Luedecke, GM
W. F. Finan, OGM
H. L. Price, 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 NORTHERN STATES POWER COMPANY - PATHFINDER ATOMIC POWER PLANT

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 application of Northern States Power Company for a permit to operate the Pathfinder Atomic Power Plant. A similar meeting on the same subject was held at the 45th meeting in Oak Ridge in December 1962. Subcommittee meetings were held in October 1962, and June 1963. At the present review, the Committee had the benefit of references listed below, and discussions with representatives of the Northern States Power Company, Allis-Chalmers Manufacturing Company, Bendix Corporation, and the AEC Regulatory staff. Because of novel features of this reactor, there are a number of unresolved problems which still need attention. These may be classified into problems which should be resolved before fuel loading begins, and those which become important after there is an appreciable inventory of fission products.

In the first category, the Committee must consider the safety design philosophy, particularly that of the instrumentation and controls. The Committee is concerned by an increasing tendency among designers to provide instrumentation which eliminates spurious scrams at the possible expense of safety. For instance, in the Pathfinder reactor, the Committee has not found adequate independence of safety channels. A detailed review of these instruments and controls is now in progress.

Problems of the second category approaching resolution include the following: nuclear superheater corrosion and other tests, containment ventilation system isolation valves, containment spray system, remote control of emergency make-up water valves, control rod embrittlement, recirculation pump cavitation, main steam line isolation valving, the location of the readout of in-core monitors, containment leak testing, and the proposed operating limits. The Committee urges conservative solutions for these problems.

The following points are in particular need of scrutiny:

(1) The Committee believes that discharge of the reactor system pressure relief valves into the turbine condenser does not provide adequate containment.

(2) The Committee believes that present plans do not provide adequate protection so that the operators can carry out their functions in event of a serious incident. The Committee recommends that further study be made to determine whether it is better to provide adequate shielding in the present control room or to provide a secondary emergency control center. The Committee suggests that alarms be installed which will signal a command for immediate evacuation by the operators at either a specified radiation rate in the corridor or a specified integrated dose in the control room.

(3) The Committee suggests greater use of experienced and technically qualified engineers for the independent safety committee and for the reactor startup, and points out the advisability of incorporating professionally trained nuclear engineers into the operating organization.

In conclusion, the Committee believes that the important questions indicated above must be resolved before the applicant is permitted to operate at a power greater than one megawatt. However, if independent safety channels suitable for low power reactors are provided, the Committee believes that this reactor may be loaded, brought to critical, and operated at powers up to one megawatt (thermal) without undue risk to the health and safety of the public.

Sincerely yours,

/s/
D. B. Hall
Chairman

References Attached.

References:
1. ACNP-5905, Pathfinder Atomic Power Plant Safeguards Report, dated January 15, 1962, transmitted by Amendment 10 dated June 12, 1962.

2. ACNP-6112, Program and Organization for Preoperational and Nuclear Testing, transmitted by Amendment 10 dated June 12, 1962.

3. ACNP-6121, Components of the Pathfinder Reactor to be Fabricated of Precipitation Hardened Stainless Steels, transmitted by Amendment 10 dated June 12, 1962.

4. Amendment 11, Additional Technical Information, dated February 22, 1963.

5. Amendment 12, dated April 24, 1963.

6. ACNP-62-25, Reactor Vessel Materials, Fabrication and Inspection, dated December 1, 1962, transmitted by Amendment 12 dated April 24, 1963.

7. Amendment 13, dated May 11, 1963.

8. ACNP-6112, Rev. 1, dated May 15, 1963, Program and Organization for Preoperational and Nuclear Testing, transmitted by Amendment 13, dated May 11, 1963.

9. Amendment 14, dated May 29, 1963.

10. Technical Specifications for Pathfinder Atomic Power Plant, dated May 28, 1963, transmitted by Amendment 14, dated May 29, 1963.

11. Amendment 15, Additional Technical Information, dated June 11, 1963.

12. PSE-6301, Pathfinder Atomic Power Plant, - Steam Line Report, dated June 14, 1963, transmitted by Amendment 16, dated June 17, 1963.

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 NORTHERN STATES POWER COMPANY - PATHFINDER ATOMIC POWER PLANT

Dear Dr. Seaborg:

At its sixty-third meeting in Washington, D. C. on May 13-15, 1965, the Advisory Committee on Reactor Safeguards considered the application of Northern States Power Company to operate the Pathfinder Atomic Power Plant up to the design power, 190 MW(t) with full superheat. Previous discussions were held at the forty-fifth meeting in December 1962, the forty-eighth meeting in July 1963, and the fifty-first meeting in November 1963. Subcommittee meetings were held in October 1962, June 1963, November 1963, and May 1965. At the present review the Committee had the benefit of the references listed, and discussions with representatives of the Northern States Power Company, Allis-Chalmers Manufacturing Company, and the AEC Regulatory Staff.

The Committee believes that adequate consideration has now been given to the following: nuclear superheater corrosion, containment ventilation system isolation valves, containment spray system, remote control of emergency make-up water valves, control rod embrittlement, recirculation pump cavitation, main steam line isolation valving, the location of the readout of in-core monitors, containment leak testing, and shielding of the control room. Implementation of these considerations is nearly complete.

The Committee believes that discharge of the reactor pressure relief valves into the condenser or through a rupture disc into the containment building is acceptable.

The Committee recognizes the inclusion of experienced and technically qualified nuclear engineers on the independent safety committee and in the operating organization.

The Committee is pleased to note that a sound criterion for safety design of instrumentation and controls has been adopted and has been implemented rigorously.

The applicant reported orally on some preliminary studies of the effects of a control rod ejection accident; the consequences appear to be acceptable if rod worths are appropriately limited. It is assumed that such limits will be defined during discussions between the applicant and the Regulatory Staff.

In conclusion, the Committee believes that, with the implementation of the above mentioned considerations, the Pathfinder Plant may be operated at its designed power of 190 MW(t) without undue risk to the health and safety of the public.

Mr. Harold Etherington did not participate in the deliberations on this project.

Sincerely yours,

/s/

W. D. Manly
Chairman

References:

1. Amendment No. 17, dated August 14, 1963, with attachments.
2. Amendment No. 18, dated August 28, 1963, with attachment.
3. Amendment No. 20, dated October 24, 1963, with attachment.
4. Amendment No. 21, dated October 29, 1963, with attachment.
5. Amendment No. 27, dated February 12, 1965, with attachments.
6. Amendment No. 28, Attachment 1, dated March 5, 1965.
7. Amendment No. 28, Attachment 2, dated March 5, 1965.
8. Amendment No. 29, Attachment 1, dated April 28, 1965.
9. Amendment No. 29, Attachment 2, dated April 28, 1965.
10. Amendment No. 29, Attachment 3, dated April 28, 1965.

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

March 14, 1960

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

Subject: PEACH BOTTOM ATOMIC POWER STATION - PHILADELPHIA ELECTRIC COMPANY

Dear Mr. McCone:

At its twenty-fourth meeting the Advisory Committee on Reactor Safeguards considered the proposal of the Philadelphia Electric Company to construct a 115 MW (thermal), helium cooled, graphite moderated, high temperature reactor at its Peach Bottom site, a location on the west shore of the Susquehanna River, nine miles upstream from the Conowingo Dam. In addition to the applicant's presentation and the Site Evaluation Report, the ACRS had the benefit of comments from the Staff of the AEC and others. A subcommittee meeting was held with the applicant, his contractors and consultants, and members of the AEC Staff, on February 17, 1960.

The location of this reactor on Conowingo Pond, which is a potential supply of potable water to the City of Baltimore, and serves several smaller cities, makes mandatory an especially careful consideration of factors which might lead to pond contamination. The applicant has presented preliminaty evidence, in the form of the results of preliminary analyses, indicating that in fact pond contamination will not present an undue hazard.

The design of this reactor, although not yet fixed, will necessarily be such that routine reactor operation may be accompanied by considerable fission product contamination of the coolant gas stream. This places particular emphasis on the need for reliability of the helium coolant system, the associated fission product traps, and the outer containment shell.

The Advisory Committee on Reactor Safeguards believes that the Peach Bottom site provides a generally acceptable degree of isolation when considered in relation to the proposed high integrity containment, and concludes the site is suitable for a reactor of the general design and power level proposed.

Sincerely yours,

/s/

Leslie Silverman
Chairman

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

References

1) Site Evaluation Report for the Peach Bottom Atomic Power Station, Part I, January 1960.

2) Site Evaluation Report - Part II - Analysis of Physical Data Pertaining to Environmental Survey for the Peach Bottom Atomic Power Station, January 1960.

3) Supplement to Site Evaluation Report, Peach Bottom Atomic Power Station, and letter of transmittal dated March 4, 1960.

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: PEACH BOTTOM ATOMIC POWER STATION - PHILADELPHIA ELECTRIC COMPANY

Dear Mr. McCone:

At its thirtieth meeting on December 7-10, 1960, the Advisory Committee on Reactor Safeguards reviewed the status of the research and development program on the advanced gas-cooled reactor as related to current design and construction plans for the Peach Bottom Reactor. Present at the discussions on December 8 were representatives of General Atomics Corporation, Bechtel Company, Philadelphia Electric Company and the AEC Division of Licensing and Regulation. The reference material listed herein had previously been made available to the Committee.

The Committee continues to be optimistic that a reactor of the general type proposed in the advanced gas-cooled concept can be constructed and operated at the Peach Bottom site without undue risk to the health and safety of the public. General Atomics has embarked on an extensive research and development program related to this reactor. The initial results from this program appear to be favorable.

However, there are many questions that remain to be answered by the research and development program which is still in its early stages. Several of these questions are in areas which could require major changes in the present design concepts and could conceivably change our early optimism. For example, the long-term integrity of the graphite under the proposed design conditions, which lie far outside past experience with graphite, has yet to be established. Also, the experimental data required to determine that the fission products emitted continuously from the homogeneous fuel components can be collected, stored and disposed of safely have yet to be developed. A novel design is proposed for the control rod system, but the development is not yet sufficiently advanced to permit adequate evaluation of its reliability. The applicant has not provided either a secondary back-up safety system or an emergency coolant system but has not yet established to the satisfaction of the Committee that these systems are unnecessary. The relatively large assumed Doppler coefficient has yet to be confirmed.

In view of the foregoing, the Advisory Committee on Reactor Safeguards is not now prepared to go beyond its original conclusion reached in our letter to you under date of March 14, 1960, that the Peach Bottom site is suitable for a reactor of this general design and power level.

Sincerely yours,

/s/

Leslie Silverman
Chairman

References:

Application of Philadelphia Electric Company, Part A General Information, Peach Bottom Atomic Power Station, undated, received July 28, 1960.

Application of Philadelphia Electric Company, Part B Preliminary Hazards Summary Report, Peach Bottom Atomic Power Station, Vol. I - Plant Description and Safeguards Analysis, and Vol. II - Site and Environmental Information, undated, received July 28, 1960.

Amendment #1 to the application of Philadelphia Electric Company, dated September 27, 1960.

Supplement to Site Evaluation Report - Peach Bottom Atomic Power Station, undated, received March 8, 1960.

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. 20545

November 1, 1961

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

Subject: REPORT ON THE PEACH BOTTOM ATOMIC POWER STATION - PHILADELPHIA ELECTRIC COMPANY

Dear Dr. Seaborg:

At its thirty-seventh meeting, October 26-28, 1961, the Advisory Committee on Reactor Safeguards reviewed the 115 MW (thermal) helium cooled, graphite moderated, high temperature reactor to be constructed at the Philadelphia Electric Company Peach Bottom site in southeastern Pennsylvania. This reactor was considered previously at the Committee's twenty-fourth and thirtieth meetings. In addition, a subcommittee has met with the applicant, contractor, and the staff on February 17, 1960, March 15-16, 1961 (at La Jolla, California) June 2, 1961 and October 3, 1961.

At the discussion on October 27, 1961, representatives of General Atomic Division of General Dynamics Corporation, Bechtel Company, and members of the AEC staff were present. The Committee has also had the benefit of reports from its subcommittee and the documents referenced below.

In its reports dated March 14, 1960 and Deeember 10, 1960, the Committee expressed the opinion that the site is suitable for a reactor of this general design and power level. In those reports several questions were raised relative to problems requiring investigation because of novel design features.

The extensive research and development program which is being carried on by General Atomic Division has resulted in the development of pertinent information. Design modifications have been made which appear to resolve the safety questions that have been raised.

While the hydraulic control rod system remains basically the same, the added rod separation detection system, electrically driven emergency shutdown rods, fusible-link poison rods, and installation of a finger-type holding lock on control rods provide a satisfactory control and backup scheme. A testing program which is underway on a prototype hydraulic control rod system involving starts and stops, a large number of scrams, and a series of malfunction tests appears to indicate its reliability.

The questions raised concerning the inherent shutdown characteristics appear to have been resolved by changes in thorium concentration and addition of rhodium to the core, and recalculation and measurements on the Doppler contribution. It has been stated by the applicant that, as a result of these changes, the temperature coefficient is negative throughout core life and at all temperatures up to 4000°F.

In order to prevent reaction between core graphite and moisture, provision has been made for rapid moisture detection, loop isolation, and scramming the reactor if excessive moisture is detected in the primary system. Further protection of the graphite is provided by maintaining the oxygen content of the containment vessel at a level below 5%. An emergency cooling system has been provided around the reactor cavity to remove decay heat after shutdown in the event of loss of coolant circulation. Design specifications, including inspection procedures provide a basis for assuring the integrity of the containment shell. In addition, the research and development program gives reasonable assurance as to the long term integrity of the graphite.

Considerable information has been developed on barriers against fission product release. Pyrolytic coating of fuel particles, the use of an impervious graphite sleeve around the fuel compacts, internal fission product traps on fuel elements, and external fission product traps are proposed as the means of controlling fission product concentration in the coolant. The current results of the fission product research program appear to be favorable. However, should later results indicate that a reliable system can not be obtained by the present approach, alternate methods appear to be available to insure that the fission product concentration in the helium coolant will be kept low.

Since the continuing research program gives reasonable assurance that all health and safety problems can be satisfactorily resolved, the ACRS believes that the proposed reactor can be constructed at the Peach Bottom site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Dr. John C. Geyer did not participate in these reviews or discussions.

Sincerely yours,

/s/

T. J. Thompson
Chairman

References:

1. Amendment #2 to Application of Philadelphia Electric Company, Part B, dated August 4, 1961.
2. Amendment #3 to Application of Philadelphia Electric Company, dated October 17, 1961.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON 25, D.C.

November 18, 1964

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

Subject: REPORT ON THE PEACH BOTTOM ATOMIC POWER STATION - PHILADELPHIA ELECTRIC COMPANY

Dear Dr. Seaborg:

At its fifty-ninth meeting, held on November 12-14, 1964, the Advisory Committee on Reactor Safeguards considered the proposal of the Philadelphia Electric Company to operate the Peach Bottom Atomic Power Station at power levels up to one megawatt (thermal). This station incorporates a helium-cooled, graphite-moderated, high-temperature reactor designed to produce 115 MW(th), with a net electrical output of 40 MW. The proposal to construct this plant was reported on by the Committee in its letter dated November 1, 1961. At the present review, the Committee had the benefit of oral presentations by representatives of the Philadelphia Electric Company and its consultants, the General Atomic Division of General Dynamics Corporation, the Bechtel Corporation, the AEC Staff, and of the documents listed below. In addition, a Subcommittee meeting was held on October 1, 1964.

The Peach Bottom reactor is designed to operate with a core-exit coolant temperature of approximately 1350° F, and it incorporates a number of novel features. These features include an essentially all-graphite core, pyrolytically-coated thorium-uranium-carbide fuel, rhodium-103 in the fuel to provide a negative moderator temperature coefficient, and a fission-product trapping system. These features have evolved from a research and development program carried on by General Atomic. The applicant plans to obtain further confirmation of the characteristics of many of these features by testing during initial operations up to 1 MW(th).

It is the opinion of the Advisory Committee on Reactor Safeguards that the Peach Bottom Atomic Power Station reactor can be operated at power levels of up to 1 MW(th) without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References Attached.

References:

1. "Duties, Qualifications and Training Program for Operating personnel, Peach Bottom Atomic Power Station", Philadelphia Electric Company, dated February 15, 1963.

2. "Design of Gas and Liquid Waste Disposal Systems, Peach Bottom Atomic Power Station", Philadelphia Electric Company, dated February 15, 1963.

3. Amendment No. 4 to Application of Philadelphia Electric Company for Construction Permit and Class 104 License, dated February 25, 1964, transmitting "Part C, Final Hazards Summary Report, Peach Bottom Atomic Power Station, Volumes I-V."

4. Amendment No. 5, dated July 7, 1964, transmitting Appendix A, "Proposed Technical Specifications, Peach Bottom Atomic Power Station."

5. Amendment No. 6, dated August 7, 1964, transmitting "Part C, Final Hazards Summary Report, Peach Bottom Atomic Power Station, Volume V (A)".

6. Semi-Annual Reports of Philadelphia Electric Company on the Peach Bottom Atomic Power Station:

   a. First, dated August 23, 1962.
   b. Second, dated February 23, 1963.
   c. Third, dated August 23, 1963.
   d. Fourth, dated February 23, 1964.
   e. Fifth, dated August 24, 1964.

7. Replacement pages for Section V, Plant Operation, dated October, 1964.

8. "Part C, Final Hazards Summary Report, Peach Bottom Atomic Power Station, Supplement," undated, received October 15, 1964.

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

October 12, 1966

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

Subject: REPORT ON PEACH BOTTOM ATOMIC POWER STATION

Dear Dr. Seaborg:

At its seventy-eighth meeting on October 6-8, 1966, the Advisory Committee on Reactor Safeguards considered the proposal of the Philadelphia Electric Company to operate the Peach Bottom Atomic Power Station at power levels up to 115 MW(t). The Committee had the benefit of discussion with representatives of the Philadelphia Electric Company, Bechtel Corporation, the General Atomic Division of General Dynamics Corporation, and the AEC Staff, and of the documents listed. A Subcommittee of the ACRS met at the plant site on September 23, 1966.

Proposed operation and testing of this plant at power levels up to 1 MW(t) had been reviewed previously by the Committee in November 1964. Completion of the plant in preparation for operation and testing up to 1 MW(t) was delayed by a cable fire in the containment during construction; the fire damage has been repaired and steps taken to reduce the possibility of recurrence.

The applicant reported that the planned program of tests at power levels up to 1 MW(t) was completed in May 1966 and that the measured nuclear characteristics of the reactor were in reasonable agreement with predicted values. Pre-operational shakedown tests of other components, however, disclosed leaks in the superheater section of the steam generators and some problems with the control rod drive mechanisms and the fuel transfer machine.

The leaks in the steam generators were reported to have been caused by stress corrosion cracking in the stainless steel superheater tubes, the superheater outlet piping, and the expansion bellows in the domes of the generators. These components were removed from each of the generators and are being replaced with Inconel and Incoloy components to reduce the possibility of recurrence of stress corrosion cracking. The expansion bellows is being replaced by an expansion loop. Testing after repairs are completed is to be performed in accordance with the requirements of

ASME Code Section VIII and will include radiographing of welds, helium mass spectrometer leakage testing of the tube-to-tubesheet welds and pneumatic pressure testing at 1.25 times design pressure.

Four of the control rod drive mechanisms exhibited erratic sticking in the regulating mode. The problem was traced to fractured balls in one of the three races of the ball nut assemblies of the linear actuators of these rods. It was reported, however, that these mechanisms were able to scram in every case despite the fractured balls. All drive mechanisms are being modified and tested to confirm operability.

The fuel transfer-machine performed well during early pre-operational testing, but gave problems with binding of the telescoping shaft during tests in hot helium. Modifications were made to eliminate interferences and the machine was used successfully in loading the core at room temperature. Subsequent to core loading, the machine was shipped to the General Atomic facilities for further modification and proof testing to confirm its operability in hot helium.

The Philadelphia Electric Company proposes a stepwise approach to power and has outlined a program of tests to be conducted during this period. The applicant proposes a detailed program of core surveillance and plant performance evaluation during the first year of power operation.

The Committee believes that the proposed testing during the ascent to full power and during subsequent operation can be conducted safely. After reaching full power, the applicant should develop, in co-operation with the AEC Staff, appropriate limits on reactivity and power anomalies. The Committee also believes that the applicant should develop and implement a program of periodic inspection of accessible primary system components during service life; this program should be developed in co-operation with the AEC Staff.

Some questions arose with regard to operating procedures involving the isolation valves and the prevention of negative pressure in the containment. These questions should be resolved in co-operation with the AEC Staff before the ascent to power.

The ACRS believes that, with satisfactory completion of the repair and testing of the steam generators and with successful completion of an adequate pre-operational testing program for the modified control rod drive mechanisms, the Peach Bottom reactor can be operated at power levels up to 115 MW(t) without undue risk to the health and safety of the public.

Dr. S. H. Bush did not participate in the Committee's review of this project.

Sincerely yours,

/s/
David Okrent
Chairman

References:

1. Philadelphia Electric Company letter dated November 3, 1964 to AEC Division of Reactor Licensing transmitting Amendment No. 8.
2. Sixth Semi-Annual Report, dated February 23, 1965.
3. Seventh Semi-Annual Report, dated August 23, 1965.
4. Final Semi-Annual Report, dated January 11, 1966.
5. Monthly Operations Report No. 1, March 1966.
6. "Steam Generator Superheater Section Repairs", dated April 1966.
7. Monthly Operations Report No. 2, April 1966.
8. Monthly Operations Report No. 3, May 1966.
9. Monthly Operations Report No. 4, June 1966.
10. Amendment No. 13 containing "Supplemental Technical Information for Operation at 115 MW(t)", transmitted by Philadelphia Electric Company letter dated August 18, 1966 to AEC Division of Reactor Licensing.
11. Monthly Operations Report No. 5, July 1966.
12. Philadelphia Electric Company letter dated September 22, 1966 to AEC Division of Reactor Licensing transmitting Amendment No. 14.

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 PEACH BOTTOM ATOMIC POWER STATION UNITS NO. 2 AND 3

Dear Dr. Seaborg:

At its ninetieth meeting, on October 5-7, 1967, the Advisory Committee on Reactor Safeguards completed its review of the application by Philadelphia Electric Co. for authorization to construct the Peach Bottom Atomic Power Station Units No. 2 and 3. This project was previously considered at ACRS Subcommittee meetings held at the Peach Bottom Atomic Power Station site on August 25, 1967, and in Washington, D. C. on September 20, 1967. During its review, the Committee had the benefit of discussions with representatives of Philadelphia Electric Co., General Electric Co., Bechtel Corporation, and the AEC Regulatory Staff, as well as the documents listed below.

The two units are to be located adjacent to the existing high-temperature, gas-cooled nuclear power plant (Unit No. 1) on a 600-acre site in Peach Bottom Township, York County, Pennsylvania. The site, located approximately 38 miles north-northeast of Baltimore, Maryland and 63 miles west-southwest of Philadelphia, Pennsylvania, is on the west bank of Conowingo Reservoir, formed by the Conowingo Dam on the Susquehanna River.

Each unit includes a boiling water reactor to be operated at a maximum power level of 3295 MWt. With respect to core design, power level, and other features of the nuclear steam supply system, Peach Bottom Units 2 and 3 are essentially duplicates of the Browns Ferry units of the Tennessee Valley Authority, previously discussed in the Committee's letter to you dated March 14, 1967.

In the unlikely event of failure of Conowingo Dam, the normal source of cooling water for the two units would no longer be available. The applicant described several possible schemes for removing shutdown heat from the plant in the event that the reservoir level should fall below the normal cooling water inlet. Such a system should be designed and constructed to the same criteria as applied to other Class I structures in the plant. The design of this system should be reviewed by the Regulatory Staff.

The present design of the units includes a ring header to supply water from the torus to the emergency core cooling systems. The applicant discussed a possible modification intended to simplify the piping and reduce susceptability to single point failure. The Committee believes that this matter should be resolved between the applicant and the Regulatory Staff.

To meet water temperature criteria of the Commonwealth of Pennsylvania, the use of cooling towers may be required for plant cooling water. A hydraulic model of the Conowingo Reservoir has been built and is being tested to determine how the criteria will be met. The Committee believes that one or more of the possible arrangements of cooling towers could be installed without adverse effects on the health and safety of the public, and that this matter can be resolved between the applicant and the Regulatory Staff.

The film condensation coefficient used to predict the depressurization performance of the High Pressure Coolant Injection (HPCI) system is based on extrapolation of available heat transfer data. Additional experiments or other supporting studies are needed to confirm the effectiveness of the HPCI system, and the results should be reviewed by the Regulatory Staff.

The Committee, in its letter to you of March 14, 1967, called attention to a number of matters that warrant careful consideration with regard to reactors of the Browns Ferry design, and other matters of significance for all large water-cooled power reactors. These matters apply similarly to Peach Bottom Units No. 2 and 3.

As in the case of the Browns Ferry units, a careful startup program will be required. If the startup program or additional information on fuel behavior fail to confirm adequately the design basis, system 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 proposed reactors. On the basis of the foregoing comments and in view of the favorable characteristics of the site, the Committee believes that the proposed Peach Bottom Atomic Power Station Units No. 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,

/s/

N. J. Palladino
Chairman

References:

1. Philadelphia Electric Company letter dated February 10, 1967; License Application, Peach Bottom Atomic Power Station Units No. 2 & 3, dated February 6, 1967; Preliminary Safety Analysis Report, Volumes 1 and 2.
2. Philadelphia Electric Company letter dated July 12, 1967; Amendment No. 1 to License Application, dated July 11, 1967; Supplement No. 1 to Preliminary Safety Analysis Report.
3. Philadelphia Electric Company letter dated September 8, 1967; Amendment No. 2 to License Application, dated September 7, 1967; Supplement No. 2 to Preliminary Safety Analysis Report.
4. Philadelphia Electric Company letter dated September 26, 1967; Amendment No. 3 to License Application, dated September 25, 1967.
5. Amendment No. 4 to License Application of Philadelphia Electric Company, Peach Bottom Atomic Power Station, Units No. 2 and 3, dated October 6, 1967.

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 PEACH BOTTOM ATOMIC POWER STATION UNITS NO. 2 AND 3

Dear Dr. Schlesinger:

At its 149th meeting, on September 14-16, 1972, the Advisory Committee on Reactor Safeguards reviewed the application by the Philadelphia Electric Company for authorization to operate the Peach Bottom Atomic Power Station Units No. 2 and 3 at power levels up to 3293 MW(t). The application was also considered at Subcommittee meetings held at the site on August 18, 1972, and in Washington, D. C., on August 31, 1972. During its review, the Committee had the benefit of discussions with representatives of the Philadelphia Electric Company, the General Electric Company, the Bechtel Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed below. The Committee reported on the construction permit application for these units in its report of October 12, 1967.

The two units are located adjacent to the existing high-temperature gas-cooled reactor (Unit No. 1) on a 620-acre site on the west shore of Conowingo Pond, formed by the Conowingo Dam on the Susquehanna River. The site is approximately 38 miles north-northeast of Baltimore, Maryland and 63 miles west-southwest of Philadelphia, Pennsylvania.

The Peach Bottom Units No. 2 and 3 reactors have essentially the same power density and linear heat generation rate as the Vermont Yankee reactor (the Committee reported on operation of this reactor in its letter of March 9, 1971), but have the highest power level of any boiling water reactor reviewed for operation to date. The reactor core design has been substantially modified from that proposed at the construction permit stage, and employs five different fuel enrichments as well as gadolinia bearing rods for reactivity control augmentation (instead of boron-steel control curtains). Some of the gadolinia rods are uniformly axially loaded and are similar to those used in the Quad Cities reactors (ACRS operating license report of March 9, 1971). Others, however, are non-uniformly loaded (part-length) and their use in Peach Bottom Units No. 2 and 3 will represent the first application in a commercial reactor.

Analyses of postulated control rod drop accidents recently have been revised by the applicant to employ a more realistic rate of reactivity insertion than formerly assumed, and to account for the changes made in the core design, in particular the use of a number of fuel enrichments and employment of full and part-length gadolinia bearing fuel rods. These analyses indicate that, for accidents occurring during certain portions of the fuel cycle, the results are unacceptable. The applicant has proposed possible changes in plant design or operating procedures which he believes would render the probability of occurrence of such an accident negligibly low. The general approach appears feasible; however, details of the proposal are not yet available and will require thorough evaluation after submittal. This matter should be resolved in a manner satisfactory to the Regulatory Staff and the Committee. Approved measures should be placed into effect prior to operation above 1% of rated power.

Cooling towers have been installed in order that the water temperature criteria of the Commonwealth of Pennsylvania may be met. The applicant has also provided an emergency cooling tower to remove shutdown heat from the reactors in the unlikely event of a failure of the Conowingo Dam. This emergency heat sink and its associated systems are designed to seismic Class I requirements and are to be operable during the design basis flood and during a loss of off-site power.

For control of combustible gas concentrations in the containment following a postulated loss-of-coolant accident, the applicant proposes use of a containment atmospheric dilution (CAD) system. With this system the desired dilution is accomplished by controlled addition of nitrogen, and results in the maintenance of higher containment pressure during a portion of the post-LOCA period than would otherwise exist. The Committee believes that, in general, use of such dilution schemes, which involve repressurization of the containment, is not desirable. However, as a backfitted provision on a plant well along in construction, use of this approach is believed by the Committee to be acceptable. The Committee nevertheless recommends that the applicant study means to assure that the peak repressurization pressure will be limited to a value substantially below the containment design pressure.

The inservice inspection program proposed for the reactor coolant system complies with Section XI of the ASME Boiler and Pressure Vessel Code to the extent permitted by the existing design. The Committee believes the program is acceptable, but recommends that the applicant continue to study means of assuring reactor vessel integrity in regions currently inaccessible for inspection.

The applicant proposes to employ recirculation pump trip as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The Committee believes that this recirculation pump trip represents a substantial improvement and should be provided for Units 2 and 3 prior to the start of commercial power operation. The specific means employed for implementing the pump trip should be resolved in a manner satisfactory to the Regulatory Staff

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 preoperational testing, there is reasonable assurance that the Peach Bottom Atomic Power Station, Units No. 2 and 3 can be operated at power levels up to 3293 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

References:

1. Philadelphia Electric Company letter dated August 31, 1970, transmitting Final Safety Analysis Report (FSAR), Volumes 1 through 7 to Peach Bottom Power Station Units 2 and 3

2. Amendments 8, 9, 11 through 16, and 18 through 20 to the License Application for Peach Bottom Power Station Units 2 and 3

3. Philadelphia Electric Company letter dated June 12, 1972, transmitting "Additional Information Required by the AEC Regarding Control of Combustible Gas Concentration in the Peach Bottom Units 2 and 3 Containments", dated June 5, 1972

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

February 11, 1976

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

SUBJECT: INTERIM REPORT ON PEBBLE SPRINGS NUCLEAR PLANT, UNITS 1 & 2

Dear Mr. Anders:

At its 190th Meeting, February 5-7, 1976, the Advisory Committee on Reactor Safeguards completed an interim review of the application of the Portland General Electric Company for permission to construct the Pebble Springs Nuclear Plant, Units 1 and 2. This Plant was previously considered at a Subcommittee meeting on January 30, 1976, at Portland, Oregon, and the site for the proposed plant was visited on January 29, 1976. During its review the Committee had the benefit of discussions with representatives of the Portland General Electric Company (PGE) and consultants, the Babcock and Wilcox Company (B&W), the Bechtel Power Corporation, and the NRC Staff. The Committee also had the benefit of the documents listed.

The proposed Pebble Springs Plant will be located on an 8650-acre tract of land near Arlington, Oregon, approximately 55 miles west-southwest of the Tri-Cities (Kennewick, Pasco, and Richland, Washington) area, the nearest population center (1970 population - 55,000). The exclusion radius is 800 meters; the low population zone radius is 2 miles. In 1970 there were 6 residents within the low population zone.

The seismic design basis and matters related to the deposition of volcanic ash arising from major volcanic eruptions at Mount Hood and Mount Saint Helens are still under review by the NRC Staff, the United States Geological Survey and the Applicant.

The ultimate heat sink will include a seismic Category 1 spray pond with a seismic Category 1 intake structure housing the two backup service water pumps. The system also includes the Pebble Springs reservoir, which is nonseismic Category 1 but is protected against tornado damage. Makeup to the reservoir will be from the Columbia River and makeup to the spray pond will be from the reservoir.

The Pebble Springs Nuclear Plant is currently planned to consist of two identical nuclear generating units. The nuclear steam supply systems (NSSS's) will be supplied by B&W and will be identical to other B&W 205 Mark C fuel assembly NSSS's, including Bellefonte Nuclear Plant, Units 1 and 2, on which the ACRS reported on July 16, 1974.

The NRC Staff and the Applicant report that, employing the currently accepted LOCA-ECCS B&W evaluation model, peak clad temperatures have a margin to the limiting condition of 2200° F.

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 inasmuch as they will be operated at lower linear heat generation rates and are expected to yield greater thermal margins to fuel design limits. 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 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 in situ 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 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 Pebble Springs Nuclear Plant in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

Specific consideration of the question of anticipated transients without scram is now under way by the NRC Staff. The Committee recommends that the design of Pebble Springs Nuclear Plant, Units 1 and 2, be such that potential design changes to minimize serious consequences from ATWS can be readily incorporated, should they be deemed necessary. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

The Committee believes that the Applicant and the NRC Staff should continue to review the Pebble Springs 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

The Applicant has calculated that the probability of adverse effects on the ability to shut the plant down safely due to turbine-generated missiles is acceptably low. The ACRS believes this analysis requires further evaluation, particularly with regard to the assumptions concerning missile energy and penetration capability. The Applicant has stated that he has backup positions including a steel turbine-missile shield which can be implemented late in the construction phase. The Committee recommends that this matter be resolved in a timely fashion, during construction, in a manner satisfactory to the NRC Staff and the ACRS.

The exact schedule for construction of Pebble Springs Nuclear Plant, Units 1 and 2 remains to be determined. The Committee recommends that if appreciable delay arises in the initiation of construction of Unit 1 from the originally planned schedule, or if delays lead to a completion date for Unit 2 significantly more than 10 years from now, the Plant should be reevaluated in terms of new regulatory requirements which may have significant effects in further protecting the health and safety of the public.

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 ACRS will review the site-related aspects of the application for a construction permit when the appropriate information has been developed, and evaluation has been completed by the NRC Staff.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

REFERENCES

1. Pebble Springs Nuclear Plant, Units 1 and 2, Preliminary Safety Analysis Report (October 29, 1974) with Amendments 1 through 8.

2. Safety Evaluation Report NUREG-0013 related to construction of Pebble Springs Nuclear Plant, Units 1 and 2, January 1976, with Supplements Nos. 1 and 2.

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

January 12, 1978

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

Subject: REPORT ON PEBBLE SPRINGS 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 Portland General Electric Company for a permit to construct the Pebble Springs Nuclear Plant, Units 1 and 2. This project was also considered during a Subcommittee meeting held in Portland, Oregon, on October 28, 1977. The Committee previously completed a partial review of this project at its 190th meeting, as discussed in its interim report to the Nuclear Regulatory Commission (NRC) dated February 11, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Portland General Electric Company, the Babcock and Wilcox Company, the Bechtel Power Corporation, and the NRC Staff. The Committee also had the benefit of presentations on the regional tectonics of the Pacific Northwest by representatives of the NRC Staff, the U. S. Geological Survey (USGS), Puget Sound Power and Light Company, Portland General Electric Company, Washington Public Power Supply System, their consultants, and members of the public at Subcommittee meetings held on September 1-2, 1977 in San Francisco, California, and on October 27-28, 1977 in Portland, Oregon. Matters related to the regional tectonics of the Pacific Northwest were considered at the 209th and 211th Committee meetings as reported in the Committee's letter dated November 15, 1977 to the NRC Executive Director for Operations. The Committee also had the benefit of the documents listed.

At the time of the Committee's interim report, February 11, 1976, the NRC Staff, the USGS, and the Applicant had not yet completed their reviews of the seismic design basis and of matters related to the possible deposition of volcanic ash arising from major volcanic eruptions of Mount Hood or Mount Saint Helens. These reviews have now been completed and the ACRS finds the Staff positions on these matters acceptable.

The Committee believes that the Applicant and the NRC Staff should review the Pebble Springs Nuclear Plant for design features that could further reduce the possibility or consequences of sabotage. (Generic Item IIC-2 in ACRS Report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977).

Since the Committee's earlier partial review, the Staff has identified 13 additional issues, 11 of which require resolution prior to construction. These matters should be resolved in a manner satisfactory to the NRC Staff.

With regard to the 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 Pebble Springs Nuclear Plant, Units 1 and 2 are: II-3, 4, 5B, 6, 7, 9, 10; IIA-2, 3, 4; IIB-1, 2; IIC-1, 3A, 3B, 4, 5, 6; IID-2; IIE-1. These problems should be dealt with by the NRC Staff and Applicant as solutions are found.

The ACRS believes that, if due regard is given to the items mentioned above and in its report of February 11, 1976, the Pebble Springs 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. Pebble Springs Preliminary Safety Analysis Report, Volumes 1-9 and Amendments 1 through 10.

2. Safety Evaluation Report, NUREG 0013, related to construction of Pebble Springs Nuclear Plant, Docket Nos. 50-514 and 50-515, January 1976, with Supplements 1 through 4.

3. USGS letter, dated January 3, 1978 from Henry W. Coulter, to Mr. Edson G. Case, ONRR, USNRC, re review of geologic and seismologic data relevant to Pebble Springs Nuclear Plant, Units 1 & 2.

References (con't)

4. Shannon & Wilson report to Portland General Electric Company entitled "Volcanic Hazard Study - Potential for Volcanic Ash Fall, Pebble Springs Nuclear Plant Site, Gilliam County, Oregon," dated January 1976.

5. Pebble Springs Nuclear Plant Fire Protection Review, PGE 2013, March 1977 with Amendment 1 dated November 1977.

6. Portland General Electric Company letter dated September 7, 1977, from J.W. Lindblad to Director of Nuclear Reactor Regulation, USNRC, re identification of significant items not formally documented with NRC.

7. Portland General Electric Company letter dated November 17, 1977, from W.J. Lindblad to Director of Nuclear Reactor Regulation, USNRC, re forwarding detailed description of Solid State Interposing Logic System.

8. Portland General Electric Company letter dated November 29, 1977, from W.J. Lindblad to Director of Nuclear Reactor Regulation, USNRC, re evaluation of geological and seismological aspects and outstanding issues.

9. Portland General Electric Company letter dated November 30, 1977, from Joseph L. Williams, Executive Vice President to Director of Nuclear Reactor Regulation, USNRC, re response to questions raised by ACRS.

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 PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2

Dear Dr. Ray:

At its 176th meeting, on December 5-7, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application of the Cleveland Electric Illuminating Company, the Duquesne Light Company, the Ohio Edison Company, the Pennsylvania Power Company, and the Toledo Edison Company (the applicants), for a permit to construct the Perry Nuclear Power Plant, Units 1 and 2. The Committee also considered this application during its 172nd meeting on August 8-10, 1974. The site for the proposed plant was visited by Committee members on June 28, 1974. Subcommittee meetings were held on this project in Painesville, Ohio, on June 28, 1974, and in Washington, D. C., on July 23 and November 23, 1974. In its review, the Committee had the benefit of discussions with representatives of the applicants, their consultants and contractors, and representatives of the Regulatory Staff and of the documents listed.

The Perry Nuclear Power Plant will be located on the southern shore of Lake Erie in Lake County, Ohio, approximately 35 miles northeast of Cleveland and seven miles northeast of Painesville, Ohio, which has been identified as the nearest population center since its population is expected to exceed 25,000 by 1980.

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

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

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

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

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

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

The applicants propose to use two hyperbolic, natural draft cooling towers in a closed cycle cooling system for the normal mode of thermal energy rejection. Lake Erie will be utilized as the Ultimate Heat Sink. The applicants are reviewing possible localized meteorological effects of the natural draft cooling towers on structural loads in the safety-related structures and on onsite meteorological measurements. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The applicants are arranging to control the mineral rights within 1800 feet, and the underground storage rights for propane within two miles, of all safety-related structures, systems, and components. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

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

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

The ACRS believes that the above items can be resolved during construction and that, if due consideration is given to these items, the Perry Nuclear Power 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.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

PERRY PLANT REFERENCES

1. The Cleveland Electric Illuminating Company (CEI) Preliminary Safety Analysis Report (PSAR), dated June 22, 1974, Volumes 1-10, for the Perry Nuclear Power Plant, Units 1 and 2.

2. Amendments 1-6, 8-13, and 15-21 to PSAR including Volumes 11 and 12.

3. CEI letter dated March 1, 1974, concerning new design items.

4. CEI letter dated April 6, 1974, concerning additional commitments and clarifications.

5. CEI letter dated August 12, 1974, concerning 8X8 fuel assembly spray cooling test and qualifications of personnel involved in quality assurance and control.

6. CEI letter dated September 20, 1974, concerning commitments involving salt rights.

7. CEI letter dated November 7, 1974, concerning clarification of information submitted with Amendment 21 to PSAR.

8. CEI letter dated November 11, 1974, concerning effect of cooling towers on wind velocities.

9. Directorate of Licensing letter dated July 22, 1974 transmitting "Summary Statement of Outstanding Safety-Related Issues" and "Safety Evaluation Report" issued July 1974.

10. Directorate of Licensing letter dated December 4, 1974 transmitting "Summary Statement of Outstanding Safety-Related Issues" and "Supplement No. 1 to the Safety Evaluation Report".

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

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
May 12, 1975

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

SUBJECT: REPORT ON PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2

Dear Mr. Anders:

At its 181st meeting, May 8-10, 1975, the Advisory Committee on Reactor Safeguards completed its review of safety matters related to a proposal by the Cleveland Electric Illuminating Company, the Duquesne Light Company, the Ohio Edison Company, the Pennsylvania Power Company, and the Toledo Edison Company (the Applicants) to design and install a permanent dewatering system which will lower the existing groundwater level during the construction phase and during the operating lifetime of the Perry Nuclear Power Plant, Units 1 and 2. The system was also considered at a Subcommittee meeting held at Painesville, Ohio, on April 25, 1975. During its review, the Committee had the benefit of discussions with representatives of the Applicants, their consultants and contractors, and representatives of the NRC Staff. The Committee also had the benefit of the documents listed.

The Committee previously reported on the construction permit application for the Perry Nuclear Power Plant, Units 1 and 2, on December 12, 1974.

The Applicants later proposed, in Amendment 22 to the Preliminary Safety Analysis Report (PSAR), to reduce the groundwater level from the maximum natural elevation of 618 ft. mean sea level (MSL) to an elevation of 568.5 ft. MSL because calculations using the 618 ft. elevation indicated that the factors of safety against overturning of structures during an operating basis earthquake (OBE) and a safe shutdown earthquake (SSE) would be inadequate. The design was modified and described in greater detail in Amendment 23 to the PSAR. The proposed system is composed of two separate and redundant subsystems, the principal components of which include a porous blanket, porous concrete piping, pumps, and inspection manholes. One of the subsystems is a pumped-discharge subsystem, not seismically qualified, which would maintain the groundwater at an elevation between about 566 and 568.5 feet MSL during normal operation. The other is a gravity drain subsystem, seismic Category I, which would maintain the groundwater at or below elevation 594 ft. MSL under the design basis accident (DBA) condition; the NRC Staff has defined the DBA for the dewatering system as the sudden release to the underdrain system of all the water stored on the site not contained by seismic Category I structures. The Applicants have committed to design all safety-related structures to withstand the hydrostatic head of the water table at 618 ft. MSL under normal operating conditions. There is agreement by the NRC Staff with the Applicants' estimate that there are adequate factors of safety against overturning of safety-related structures under OBE and SSE conditions with the water table at 594 feet. The Applicants have further committed to various actions including notification, remedial steps, and plant shutdown, depending on specific water levels exceeded.

The Applicants have not yet provided specifications for the design criteria of the porous concrete blanket, nor completed all the necessary physical and chemical tests of the pertinent geological strata on the plant site. These matters should be resolved in a manner satisfactory to the NRC Staff.

Methods of testing, monitoring, and maintaining the underdrain system performance as well as monitoring for and venting of methane gas accumulation (from natural occurrence) in the system should be resolved in a manner satisfactory to the NRC Staff.

The Committee believes that the proposed dewatering system design can provide a drawdown capability with adequate safety margin. To achieve and maintain the required performance capability, the Applicants' quality assurance program for the design, construction, and operation of the dewatering system should include special attention to protecting the porous concrete blanket against clogging and protecting the lower till and Chagrin shale against downgrading.

The Advisory Committee on Reactor Safeguards believes that the proposed permanent dewatering system is acceptable and, if due regard is given to the items mentioned above and in the Committee's letter of December 12, 1974, the Perry Nuclear Power 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.

Sincerely,

WKerr

W. Kerr
Chairman

References:

1. The Cleveland Electric Illuminating Company (CEI), Preliminary Safety Analysis Report (PSAR), Amendments 22 and 23..

2. Supplement No.2 to the Safety Evaluation of the Perry Nuclear Power Plant Units 1 and 2 by the Office of Nuclear Reactor Regulation, USNRC, dated April, 1975..

3. Letter, Evelyn Stebbins (Coalition for Safe Electric Power) to Executive Secretary (ACRS), commenting on information not included in the Applicant's description of the underdrain system March 1, 1975.

4. Letter, Cleveland Electric Illuminating Co., reiterating and replying to questions from the NRC Staff on the proposed underdrain system, March 13, 1975.

5. Letter, Cleveland Electric Illuminating Co., reiterating and replying to questions from the NRC Staff on the proposed underdrain system, April 3, 1975.

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

July 13, 1982

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

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE PERRY NUCLEAR POWER PLANT, UNIT 1

During its 267th meeting, July 8-10, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of the Cleveland Electric Illuminating Company (Applicant), acting on behalf of itself and as agent for Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and the Toledo Edison Company, for a license to operate the Perry Nuclear Power Plant, Units 1 and 2. The plant is to be operated by the Cleveland Electric Illuminating Company. A tour of the facilities was made by members of the Subcommittee on the morning of June 28, 1982, and a Subcommittee meeting was held in Cleveland, Ohio on June 28 and 29, 1982 to consider the application. During its review the Committee had the benefit of discussion with representatives of the Applicant, the NRC Staff, and members of the public. The Committee also had the benefit of the documents listed. The Committee commented on the application for a permit to construct this plant in its reports dated December 12, 1974 and May 12, 1975.

The Perry Nuclear Power Plant is located in Lake County, Ohio near Lake Erie approximately 35 miles northeast of Cleveland, Ohio and 21 miles southwest of Ashtabula, Ohio. Units 1 and 2 use General Electric BWR-6 nuclear steam supply systems with a rated power of 3579 MWt and a Mark III pressure suppression containment system with a design pressure of 15 psig. Construction of Unit 1 is about 83% complete and Unit 2 is about 43% complete.

Because loading of fuel for Unit 2 is scheduled for May 1987, the Committee does not believe it appropriate to report at this time on the operation of Unit 2.

Our review included the management organization, technical support staff, status of operational staffing, and the training program. This is the first nuclear power plant to be operated by the Applicant. The plant staff has a minimum amount of boiling water reactor (BWR) nuclear background. We agree with the NRC Staff on the urgent need for additional personnel with BWR experience within the operating management. The Applicant should fill the position of Superintendent of Plant Operations in the near future. Experienced senior technical support personnel should be included in the staffing plans of the Applicant. This matter should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed.

As a result of adverse experience on the Perry project several years ago, the Applicant restructured its quality assurance procedures and its quality control and assurance organization. The revised organization has been reviewed and audited by the NRC Staff. We wish to receive a report from the NRC Staff which discusses design and construction problems, their disposition, and the overall effectiveness of the effort to assure appropriate quality.

The Applicant has committed several technical staff members to matters related to probabilistic analysis and studies of systems interactions. We believe that efforts of this sort by the operating utilities are to be encouraged.

The Mark III suppression pool dynamic loads have been identified as an Outstanding Issue in the NRC Staff's review. The NRC Staff has provided the Applicant with a proposal for the appropriate design basis loads, and it appears that the Perry design will be able to accommodate these loads. Additional concerns with the design of the Mark III containment have been recently brought to our attention. The NRC Staff is currently assessing these issues for impact on the Mark III design. We will continue to discuss with the NRC Staff, on a generic basis, Mark III suppression pool dynamic loads and other additional Mark III issues.

Hydrogen control systems for Mark III containments are being developed by the Mark III Owners Group. Efforts by this Owners Group are being directed toward the development of a hydrogen ignition system which makes use of distributed ignition sources. The NRC Staff has indicated that they will be able to meet with the Committee on this matter in the near future. We expect to review this system on a generic basis. Acceptability of this system is designated as a License Condition.

We recommend that the Applicant 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. We believe it is important that there should 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. We recommend that any needed modifications be made before the plant resumes operation following the second refueling. We wish to be kept informed on the progress and results of these studies.

During our review, the NRC Staff identified a number of other License Conditions, Confirmatory Matters, and Outstanding Issues which remain to be resolved. Except for the issue of turbine missiles, we are satisfied with the progress on these topics, and we believe that they should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed concerning resolution of the turbine missile issue, and wish to receive a technical report which discusses and evaluates the problems involved.

If due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, the ACRS believes there is reasonable assurance that the Perry Nuclear Power Plant, Unit 1 can be operated at power levels up to 3579 MWt without undue risk to the health and safety of the public.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References

1. Cleveland Electric Illuminating Company, "Final Safety Analysis Report, Perry Nuclear Power Plant, Units 1 and 2," with Amendments 1-6
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report, Perry Nuclear Power Plant, Units 1 and 2," USNRC Report NUREG-0887, dated May 1982
3. Memorandum from D. Houston/J. Kudrick, NRC, to A. Schwencer/W. Butler, NRC, Subject: Summary of May 13, 1982 telecon with John Humphrey - Concerns about Grand Gulf Mark III Containment, dated May 18, 1982
4. Letter from John M. Humphrey, Humphrey Engineering, Inc., to L. F. Dale, Mississippi Power and Light, Subject: BWR-6/Mark III Containment Design Issues, dated May 8, 1982

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

May 11, 1977

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON PHIPPS BEND NUCLEAR PLANT, UNITS 1 AND 2

Dear Mr. Rowden:

During its 205th meeting, May 5-6, 1977, the Advisory Committee on Reactor Safeguards reviewed the application of the Tennessee Valley Authority (TVA) for a license to construct the Phipps Bend Nuclear Plant, Units 1 and 2. This application was previously reviewed at a Subcommittee meeting in Kingsport, Tennessee on April 15, 1977 following a visit to the site by Committee members on the same day. During its review the Committee had the benefit of discussions with representatives of the TVA, the Nuclear Regulatory Commission Staff, and the General Electric Company. The Committee also had the benefit of the documents listed.

The Phipps Bend plant consists of two 3579 MWt reactors of the GESSAR-238 design which uses a BWR/6 boiling water reactor with a Mark III containment. The design of the Phipps Bend nuclear units is identical to that of the Hartsville units on which the Committee reported May 13, 1976. The Staff has issued a Safety Evaluation Report dated April 1977 for the Phipps Bend plant and a Preliminary Design Approval No. PDA-1 dated December 22, 1975, for the GESSAR-238 plant. PDA-1 covers the nuclear island which consists of the nuclear steam supply system, the reactor building, and associated facilities. The TVA will design the turbine island portion and other installations external to the nuclear island.

The plant will be located in Hawkins County in eastern Tennessee, approximately 15 miles southwest of Kingsport. The site consists of approximately 1,270 acres bordering on the Holston River. The minimum exclusion area distance measured from the center of the Unit 1 containment building is approximately 2,490 feet. The low population zone has a radius of 3 miles and included a population of 2,090 in 1970. The nearest population center is Kingsport in combination with the suburb of Kingsport North (1970 population 45,056).

An acceleration of 0.25g has been selected for the safe shutdown earthquake and 0.09g for the operating basis earthquake. The Committee considers these values acceptable for this plant.

The Applicant stated that all matters of concern expressed by the ACRS relating to the Hartsville plant and its use of the GESSAR-238 nuclear island design will be resolved in connection with the Hartsville plant, which will be licensed, constructed, and operated prior to the time similar stages are reached for the Phipps Bend units.

The ACRS recommends that the Staff and the Applicant review and evaluate the probability of loss of all AC power as a function of the duration of such power loss and develop criteria and a specific approach to assure that the plant can withstand such an event with acceptable reliability.

The proposed temperature limits on suppression pool water during an ATWS and structural problems associated with steam condensation require timely evaluation by the Staff. These matters are related to generic Item IIB-3. The Committee wishes to be kept informed.

Various generic problems are discussed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 5," dated February 24, 1977 (Attached). Those problems relevant to the Phipps Bend plant should be dealt with by the Staff and the Applicant as solutions are found. The relevant items are: II-4, 6, 7, 9, 10; IIA-1, 7; IIB-2; IIC-1, 2, 3, 5, IID-2.

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 Phipps Bend 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.

Sincerely yours,

M. Bender

M. Bender
Chairman

Attachment:
[*]Ltr. to Honorable M. A. Rowden, "Status of Generic Items Relating to Light-Water Reactors: Rpt. No. 5," dated 2/24/77

[*] See pages 2287-2330, Volume IV

REFERENCES

1. Phipps Bend Nuclear Plant, Units 1 and 2, Preliminary Safety Analysis Report, Volumes 1 - 4
2. Amendments 1 - 13 to the Preliminary Safety Analysis Report
3. Safety Evaluation Report, NUREG 0101, related to the construction of the Phipps Bend Nuclear Plant, Units 1 and 2, April 1977
4. Letter from Tennessee Valley Authority to Nuclear Reactor Regulation, NRC, concerning information regarding the still water flood level, dated April 22, 1977
5. Letter from Tennessee Valley Authority to Nuclear Reactor Regulation, NRC, concerning commitments to design to the new load profile or relocate structures to elevations greater than 19.5 feet above the suppression pool, dated March 31, 1977
6. Letter from Tennessee Valley Authority to Nuclear Reactor Regulation, NRC, concerning commitments regarding resolution of issues on the probable maximum flood, dated March 15, 1977
7. Letter from Tennessee Valley Authority to Nuclear Reactor Regulation, NRC, concerning design criteria for the construction of the Central Service Facility substructure wall, dated March 4, 1977
8. Letter from Tennessee Valley Authority to Nuclear Reactor Regulation, NRC, concerning the potential impacts of movement of the reactor pressure vessels and heads to the Phipps Bend site, dated February 25, 1977

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

July 8, 1961

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

Subject: REPORT ON ORDNANCE CORPS RESEARCH REACTOR AT PICATINNY ARSENAL

Dear Dr. Seaborg:

At its thirty-fifth meeting the Advisory Committee on Reactor Safeguards reviewed the site proposed by the U. S. Army Ordnance Corps for a 20 to 30 MW(t) research reactor and associated research facilities. The Committee had access to the reports referenced below and the benefit of discussions with the applicant and the AEC staff. A Subcommittee had previously visited the site on June 12, 1961. The proposed site is a remote section of Picatinny Arsenal in Green Pond valley. The Ordnance Corps Research Reactor, OCRR, will be patterned after the Oak Ridge Research Reactor except that the OCRR will be completely enclosed in a high integrity containment shell having a designed leak rate of 0.2% per day or less.

This site has adequate isolation with respect to accidental release of air-borne radioactivity so long as the Arsenal retains its restrictive casement in the direction of Green Pond. The largest exposed group is that composed of the Arsenal employees -- about 6,000 people. The normal evacuation plan of the Arsenal, modified to reflect the hazards of radioactivity, can be made adequate to protect this body of essentially mobile people.

The major problem at the site is whether or not a severe accident in the reactor can seriously contaminate the Boonton Reservoir, which is the primary source of water for Jersey City, population roughly 300,000. Because of the relatively close coupling between liquid releases and the Boonton Reservoir, it will be essential to limit the release of highly contaminated water, generated in the event of a severe accident, to very small volumes. It is the opinion of the ACRS that adequate limitation can be achieved by proper design and construction of the containment vessel and associated storage facilities for liquid wastes.

With these provisions, the ACRS believes that the Picatinny site is suitable for a research reactor of the type and power level proposed.

Sincerely yours,

/s/
T. J. Thompson
Chairman

References:

1. Study of Hydro-Geology of Picatinny Arsenal, N. J., dated July 1960.
2. Preliminary Site Survey for OCRR, KLX-1820, dated August 11, 1960.
3. Preliminary Scope Report - Research Reactor Facility, KLX-1822, dated October 21, 1960.
4. Preliminary Hazards Report - Research Reactor Facility, KLX-1825, dated February 10, 1961.
5. Amendment #1, dated June 1961, to Preliminary Hazards Report.
6. Amendment #2, dated June 1961, to Preliminary Hazards Report.
7. Letter from H. J. Matsugama (Picatinny Arsenal) to J. Newell (DL&R) re Page 19 of Amendment #2, dated June 22, 1961.

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

April 12, 1968

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

Subject: REPORT ON PILGRIM NUCLEAR POWER STATION

Dear Dr. Seaborg:

At its ninety-sixth meeting, on April 4-6, 1968, the Advisory Committee on Reactor Safeguards reviewed the application by the Boston Edison Company for authorization to construct its Pilgrim Nuclear Power Station. An ACRS Subcommittee had previously reviewed the project and visited the site during a meeting with the applicant in Boston, Massachusetts, on March 26-27, 1968. During its review, the Committee had the benefit of discussions with representatives and consultants of Boston Edison Company, General Electric Company, Bechtel Corporation, and the AEC Regulatory Staff. The Committee had the benefit of the documents listed.

The plant will be located on the west shore of Cape Cod Bay approximately 3-1/2 miles south of Plymouth, Massachusetts. The city of Boston is 36 miles to the northwest and the city of Providence, Rhode Island, 44 miles approximately to the west. The plant includes a boiling water reactor designed for 1912 MWt, at a lower power density than employed in the Browns Ferry reactors.

The applicant is continuing his studies of water rise and runup during severe coastal storms. The design of the structures is stated to be sufficiently flexible to permit adjustment for an unexpectedly high calculated flood level.

The Committee has, in the past, called attention to several problem areas pertaining to large, water-cooled, power reactors - these apply also to the Pilgrim plant. The applicant and the Staff should resolve the manner in which the intent of General Design Criterion Number 35 (10 CFR 50.34 proposed) will be met for the Pilgrim plant.

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.

The applicant stated that he would give further consideration to a suitable interlock to ensure that low-pressure cooling capability would be available before the auto-relief depressurization could be initiated.

The Committee recommends that the Boston Edison Company assume an active role in quality assurance at all stages of fabrication and construction.

The Committee was informed that the Commonwealth of Massachusetts is responsible for preparing off-site emergency plans, with inputs provided by the applicant. The Committee believes that the applicant should assure himself of the adequacy of all emergency plans.

The Advisory Committee on Reactor Safeguards believes that the items mentioned can be resolved during construction. The Committee believes the proposed plant can be constructed at the Pilgrim 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 - Pilgrim</u>

1. Letter from Boston Edison Company, dated June 23, 1967; Application for License; Volumes I, II and III of Design and Analysis Report for Pilgrim Nuclear Power Station

2. Letter from Boston Edison Company, dated July 21, 1967; Amendment No. 1 to License Application

3. Letter from Boston Edison Company, dated October 11, 1967; Amendment No. 2 to License Application

4. Letter from Boston Edison Company, dated December 15, 1967; Amendment No. 3 to License Application

5. Letter from Boston Edison Company, dated December 28, 1967; Amendment No. 4 to License Application

6. Letter from Boston Edison Company, dated February 6, 1968; Amendment No. 5 to License Application

7. Letter from Boston Edison Company, dated March 5, 1968; Amendment No. 6 to License Application

8. Letter from Boston Edison Company, dated March 11, 1968; Amendment No. 7 to License Application

9. Letter from Boston Edison Company, dated March 11, 1968; Amendment No. 8 to License Application

10. Letter from Boston Edison Company, dated March 11, 1968; Amendment No. 9 to License Application

11. Letter from Boston Edison Company, dated March 26, 1968; Amendment No. 10 to License Application

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

April 7, 1971

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

Subject: PILGRIM NUCLEAR POWER STATION

Dear Dr. Seaborg:

At its one-hundred thirty-second meeting, April 1-3, 1971, the Advisory Committee on Reactor Safeguards reviewed the application by Boston Edison Company for authorization to operate the Pilgrim Nuclear Power Station, comprising a single nuclear power generating unit, at power levels up to 1998 MW(t). The application was previously considered at a Subcommittee meeting held at the site on March 22, 1971. During its review the Committee had the benefit of discussions with representatives and consultants of Boston Edison Company, General Electric Company, Bechtel Corporation, and the AEC Regulatory Staff, and of the documents listed below. The Committee reported to you at the construction permit stage for this station on April 12, 1968.

The Pilgrim Nuclear Station is on the west shore of Cape Cod Bay, approximately five miles from the center of Plymouth, Massachusetts (population about 11,000). Boston is 36 miles northwest of the site, and Providence is 44 miles west. The Pilgrim reactor is a boiling water reactor generally similar to Millstone Unit 1 and other boiling water reactors recently reviewed by the Committee for operation.

The applicant has not provided equipment for concentrating and separating radioactivity from liquid wastes, and he states that the radioactivity concentration in the condenser circulating water discharge will not exceed that permitted by 10 CFR 20. During the first reactor shutdown for refueling, the applicant will install an evaporator designed to permit the holdup of liquid wastes and thereby reduce the gross radioactivity discharged. The Committee believes that the design and operation of this evaporator system should be such as to reduce to levels as low as practicable the amount of long-lived radioisotopes discharged. The Regulatory Staff should review and approve the design and operating mode of this equipment. The Committee also believes that prior to the installation of this equipment, effort should be made to reduce the radioactivity released.

The applicant proposes that the gaseous and particulate radioactivity discharged through the stack will not exceed 10 CFR 20 limits. The Committee believes the applicant should set a much lower operating limit and should make such equipment changes as may be necessary to accomplish this.

In previous reports, the Committee has commented on the following matters common to boiling water reactors recently reviewed for operation; these comments apply also to the Pilgrim Plant. The Committee believes that the reactor containment should be inerted during normal operation, and that the primary control of accident-generated hydrogen should be by some method other than purging; the need for inerting should be reviewed periodically as operating experience and further knowledge from development work are obtained and as other means of coping with hydrogen are found. The applicant proposes to protect the containment against breaching that may be caused by whipping of unrestrained piping in the event of a pipe rupture, and also to guard against missiles that could be generated from the biological shield by rupture of pipes, including safe-ends, within the shield.

The applicant proposes to assure that accidental dropping of the spent-fuel cask into the fuel storage pool will not cause leakage in excess of the make-up capacity, and will make such modifications as may be necessary. The applicant said he would make tests adequate to confirm the predicted vibrational characteristics of the vessel internals. The Committee believes the applicant should make timely proposals for resolution of the problem of possible failure to scram on anticipated transients. The applicant should reevaluate, before routine operation at full power, the performance of the emergency core cooling system, using recent heat transfer data and calculational methods. Several items regarding plant instrument systems and electrical systems are under review by the Regulatory Staff. All these matters should be resolved to the satisfaction of the Regulatory Staff; the Committee wishes to be kept informed.

The Committee believes the applicant should continue to explore means of improving access to vessel welds for inservice inspection. The Committee also believes that the reactor vessel pressure should be limited in accordance with current AEC bases when the vessel temperature is below $180^{\circ}F$.

The site is served by two 345 kv electrical transmission lines on the same towers and a separate 23 kv line. Over a short distance the lines are adjacent and it is physically possible for the fall of a tower to break the 23 kv line. The Committee believes that the applicant should explore the feasibility of using an alternative 23 kv supply or of making local changes to reduce the possibility of losing the 345 and 23 kv lines simultaneously.

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 Pilgrim Nuclear Power Station can be operated at power levels up to 1998 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

/s/

Spencer H. Bush
Chairman

References - Pilgrim Nuclear Power Station

1. Amendment Nos. 12 through 27 to License Application for the Pilgrim Nuclear Power Station and Volumes 1 through 5 of FSAR.

2. Letter from Boston Edison Company, dated September 26, 1968; Report entitled "Installation of Stub Tubes - Pilgrim Station Reactor Vessel - Boston Edison Project".

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 PILGRIM NUCLEAR POWER STATION UNIT 1

Dear Dr. Ray:

At its 171st meeting, the ACRS reviewed the safety considerations related to the 1974 inservice inspection results for the Pilgrim Station Unit 1 reactor pressure vessel. The matter had been considered previously at a Subcommittee meeting on July 9, 1974. During its review the Committee had the benefit of presentations from Boston Edison Company and its consultants, the AEC Heavy Section Steel Technology Program, and the documents listed.

The mechanized ultrasonic examination of the Pilgrim Station Unit 1 reactor vessel during the current shutdown yielded an ultrasonic reflection signal from the weld between the vessel and the N2B recirculation coolant inlet nozzle, which was greater than the signal found during the manual baseline inspection prior to initial operation. Subsequently, the licensee, Boston Edison, verified this increase in signal by repeating the examination using a manual technique. During the inservice inspection, the licensee examined the N4A feedwater nozzle weld, which also had an indication of a subsurface discontinuity when the baseline inspection was performed. This examination yielded a signal similar to that which had been obtained during the baseline inspection, thus confirming that the mechanized examination technique could provide results comparable to those obtained during the manual baseline inspection. Boston Edison has concluded that the signal change in nozzle N2B can probably be attributed to an alteration in the character of the defect, rather than to an extension of the defect size.

The results from both the recirculation inlet nozzle and the feedwater nozzle ultrasonic examinations were evaluated using the criteria set forth in the 1974 ASME Code Section XI. Under this Code, if the ultrasonic signal is larger than a pre-established value dimensional characterization and analysis of the fracture propagation potential of the indicated defect are required. The defect indications from both nozzles have been shown by studies performed independently by the Boston Edison Company and by the Regulatory Staff to be within acceptable size limits for the stress and temperature conditions considered in the Pilgrim reactor safety analyses. The neutron fluence in the nozzle regions is too low to alter the fracture toughness of the affected portions of the vessel.

The licensee has agreed to repeat the inspection of the affected vessel regions at subsequent refueling outages and to add acoustic emission sensors at the next refueling outage as an additional monitoring provision. As part of his evaluation of the defect in accordance with the 1974 Code, the licensee has calculated that there will be an insignificant increase in defect size between now and the next scheduled inspection. The Committee recognizes that ultrasonic examinations by several individuals have validated the defect sizes in the two nozzles. Even so, the Committee recommends independent examinations by at least two qualified organizations during the next inspection period to certify defect dimensions and possible changes.

In view of the above considerations, the ACRS believes that the Pilgrim Station Unit 1 reactor may resume normal operation without undue risk to the health and safety of the public.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References attached

References:

1. AEC letter from Karl R. Goller to Dr. William R. Stratton dated July 1, 1974, transmitting Safety Evaluation

2. AEC letter from John F. O'Leary to Dr. William R. Stratton dated June 19, 1974

3. Boston Edison Company letter to Dennis L. Ziemann dated June 11, 1974

4. Boston Edison Company letter to J. P. O'Reilly dated May 3, 1974, w/6 attachments

5. Boston Edison Company letter to James P. O'Reilly dated April 29, 1974

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

November 14, 1975

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

SUBJECT: INTERIM REPORT ON PILGRIM NUCLEAR GENERATING STATION, UNIT NO. 2

Dear Mr. Anders:

At its 187th meeting, November 6-8, 1975, the Advisory Committee on Reactor Safeguards completed a partial review of the application of Boston Edison Company and joint applicants (Applicants) for a permit to construct the Pilgrim Nuclear Generating Station, Unit No. 2. The site was visited on February 20, 1975, and the project was considered at a Subcommittee meeting at Plymouth, Massachusetts on November 4, 1975. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, Combustion Engineering, Inc., Bechtel Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

The plant will be located in Plymouth County, Massachusetts approximately 38 miles southeast of Boston. The NRC Staff has designated a group of contiguous communities consisting of Plymouth Center, West and North Plymouth, and Kingston Center, some located as near as 2.2 miles from the site, to be the nearest population center (1970 population of 20,000 and the projected 1990 population of 25,000). The minimum exclusion distance is 441 meters and the low population zone radius is 1.5 miles. Major land uses in the vicinity of the plant site are for residential and recreational activities.

The Nuclear Steam Supply System (NSSS) for Pilgrim Unit 2 will be furnished by Combustion Engineering, Inc. It will consist of a pressurized water reactor with a two-loop reactor coolant system and will be rated at a thermal power output of 3473 megawatts. The design of the NSSS is similar to that of San Onofre Units 2 and 3 which was reported on in the Committee's report of July 21, 1972.

The Committee has not completed its review of the seismicity of the site region, the proposed seismic design basis, and the foundation engineering for Category I structures. These matters will be reviewed by the Committee following completion of the NRC Staff review. The Committee will complete its review of LOCA-ECCS at the same time.

The source of normal and emergency cooling water will be Cape Cod Bay. The intake structure and the intake channel will be protected by existing breakwaters constructed for Pilgrim Unit 1.

Pilgrim Unit 2 will employ a containment consisting of a steel-lined, pre-stressed, post-tensioned concrete cylinder and hemispherical dome roof with a total free volume of 2.48×10^6 cu. ft. The design pressure and temperature are 60 psig and 300°F., respectively. The Committee believes that this containment design, with its auxiliary systems, is satisfactory for this plant.

The NRC Staff has identified other outstanding issues 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 recommends that the NRC Staff and the Applicants review further the design features that are intended to prevent the occurrence of 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 ACRS considered the problem of turbine missiles in its report of April 18, 1973, where recommendations were made concerning overspeed protection systems, optimum turbine orientation, and projectile penetration. The Committee recommends that the NRC Staff continue to review the combination of overspeed protection systems and low angle missile barriers to determine if changes would enhance the safety of Pilgrim Unit 2, recognizing that design of this plant, which utilizes a non-optimum turbine orientation was well advanced prior to 1973. For future plants, the ACRS reiterates its recommendation that a peninsular arrangement, optimized to be non-interactive with critical components in both single and multi-unit stations, is preferred.

The Committee believes that the Applicants and the NRC Staff should continue to review the Pilgrim Unit 2 design for features that could reduce the possibility and consequences of sabotage. The Committee recommends that adequate attention be given by the Applicants and the NRC Staff to ensure that satisfactory measures are developed and implemented to assure the protection of Pilgrim Unit 1 during the construction of Unit 2.

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.

With satisfactory conclusions on LOCA-ECCS, the seismic-related items, and the foundation engineering of Category I structures, identified above as matters requiring further Committee review, and with due consideration to the other items mentioned above, the Committee believes that Pilgrim Nuclear Generating Station, 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,

W. Kerr
W. Kerr
Chairman

References

1. Boston Edison Company and joint applicants, "Pilgrim Nuclear Generating Station, Unit No. 2 Preliminary Safety Analysis Report," (PSAR), Vols. I-XI.
2. Amendments 1-21 to PSAR.
3. U.S.N.R.C., Safety Evaluation Report for the Pilgrim Nuclear Generating Station, Unit No. 2, June 1975.
4. Letter, dated January 3, 1975, Boston Edison Company to DRL, concerning Anticipated Transients Without Scram.

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

October 12, 1977

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

SUBJECT: REPORT ON PILGRIM NUCLEAR GENERATING STATION, UNIT NO. 2

Dear Dr. Hendrie:

During its 210th meeting, October 6-8, 1977, the Advisory Committee on Reactor Safeguards completed its review of the application of Boston Edison Company and joint applicants (Applicants) for a permit to construct the Pilgrim Nuclear Generating Station, Unit No. 2. The application was previously reviewed by the Committee in 1975 and the results of its review are covered in its Interim Report of November 14, 1975. Subsequently, the Applicants undertook an extensive investigation to provide additional data on foundation stability and on tectonic interpretations for establishing the seismic design basis for the site. These items, along with LOCA-ECCS, fire protection, and industrial security, were the principal matters of this review. A Subcommittee meeting with the Nuclear Regulatory Commission (NRC) Staff and Applicants was held in Boston, Massachusetts on September 22, 1977. The Committee had the benefit of discussions with representatives and consultants of the Applicants, Combustion Engineering, Inc., Bechtel Corporation, U. S. Geological Survey, and the NRC Staff. The Committee also had the benefit of the documents listed.

As noted in the Committee's Interim Report of November 14, 1975, the Nuclear Steam Supply System (NSSS) for Pilgrim Nuclear Generating Station, Unit No. 2 will be furnished by Combustion Engineering, Inc. It will consist of a pressurized water reactor with a two-loop reactor coolant system and will be rated at a thermal power output of 3473 megawatts. The design of the NSSS is similar to that of San Onofre Units 2 and 3 on which the Committee reported in its letter of July 21, 1972.

The plant will be located in Plymouth County, Massachusetts, approximately 38 miles southeast of Boston. The NRC Staff has designated a group of contiguous communities consisting of Plymouth Center, West and North Plymouth, and Kingston Center as the nearest population center (1970 population of 20,000 and projected 1990 population of 25,000). The minimum exclusion distance is 441 meters and the low population zone radius is 2.3 miles, revised from 1.5 miles since the Committee's Interim Report. Major land uses in the vicinity of the plant site are for residential and recreational activities.

The Applicants have performed extensive investigations to assess site and regional geologic and seismic conditions in accordance with the guidelines of Appendix A to 10 CFR Part 100 and have determined that the controlling earthquake is one of intensity VII (MM) assumed to occur in the vicinity of the site. Using either the Neumann or the Trifunac-Brady relationship between intensity and acceleration, the trend of the means of the peak acceleration values corresponding to intensity VII (MM) is 0.13g. The NRC Staff also has concluded that the controlling earthquake should be intensity VII (MM) but that the appropriate design acceleration for the safe shutdown earthquake (SSE) should be increased from 0.13g to 0.2g at the ground surface due to the possibility of soil amplification at the Pilgrim site. The U. S. Geological Survey's conclusion is consistent with that of the NRC Staff. Based on the above, the Committee agrees that a SSE with 0.2g acceleration at the ground surface is appropriate.

The resistance of the foundation soils to liquefaction has been evaluated using both observational and analytical methods. The lowest determined safety factor was 1.8 allowing for ground acceleration levels as high as 0.25g. Settlement of Category 1 structures is expected to be minimal. Measurements of actual settlement will be evaluated at the operating license stage of review.

Recent calculations show compliance with the criteria of 10 CFR Part 50.46 based on a peak linear heat generation rate of 13.0 kilowatts per foot.

The Committee believes that resolution of the seismic, foundation, and ECCS-LOCA matters noted above are adequate and that the other items mentioned in its November 14, 1975 Interim Report can be satisfactorily resolved during construction.

With regard to generic problems applicable to this plant, refer to "Status of Generic Items Relating to Light-Water Reactors: Report No. 5" dated February 24, 1977. Items considered relevant to the Pilgrim Nuclear Generating Station, Unit No. 2 are: II-1, 2, 3, 4, 5 (loose parts monitoring resolved) 6, 7, 10; II-A 3, 4, 5, 6, 7; II-B 1, 2; II-C 1, 2, 3, 4, 5, 6; and II-D 2. These problems should be dealt with by the NRC Staff and Applicants as solutions are found.

The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the foregoing, the Pilgrim Nuclear Generating Station, 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,

M. Bender

M. Bender
Chairman

REFERENCES:
1. Boston Edison Company and joint applicants, "Pilgrim Nuclear Generating Station, Unit No. 2 Preliminary Safety Analysis Report" (PSAR) Vols. I-IX
2. Amendments 1-36 to the PSAR
3. Safety Evaluation Report, NUREG-75/054, related to the construction of the Pilgrim Nuclear Generating Station, Unit No. 2, June 1975
4. Supplement Nos. 1-3 to NUREG-75/054

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

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

August 5, 1958

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

Subject: PROPOSED NUCLEAR POWER REACTOR FOR THE CITY OF PIQUA, OHIO

Dear Mr. McCone:

At its Ninth Meeting, August 4, 1958, the Advisory Committee on Reactor Safeguards was given an oral presentation by the Hazards Evaluation Branch of the general characteristics and site of the proposed nuclear power reactor for the City of Piqua, Ohio.

The matter has not been formally submitted to the Committee as yet and no other information has been made available.

The tentative view of the Advisory Committee on Reactor Safeguards is that the site is not a suitable one.

Sincerely yours,

/s/ C. Rogers McCullough

C. Rogers McCullough
Chairman

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

November 12, 1958

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

Subject: PROPOSED NUCLEAR POWER REACTOR FOR THE CITY OF PIQUA, OHIO

Dear Mr. McCone:

At its Tenth Meeting on October 16, 1958, the Advisory Committee on Reactor Safeguards reviewed the Organic Moderated Reactor proposed for installation at the Piqua Municipal Power Plant as a nuclear steam generator. Discussions of the proposal were held with the Division of Licensing and Regulation, representatives of the City of Piqua Municipal Power Commission and Atomics International. In addition, the Committee had available for reference purposes the Preliminary Safeguards Report on the project, NAA-52-3100, and the report of the Hazards Evaluation Branch dated October 14, 1958.

Although the Committee is favorably impressed with the organic moderated reactor concept, and is aware of the generally favorable results of the OMRE experience to date, it wishes to reaffirm its opinion that the Piqua site proposed on October 16 for installation of a nuclear power plant based on this concept is unsuitable. That site is in an urban area and makes no provision for an exclusion zone. Both the meteorological and the hydrological conditions are unfavorable to the safe dispersal of radioactive by-products. The organic moderator presents a local fire hazard which is increased by the inclusion of a decay heat removal system employing a xylene boiler.

Representatives of Atomics International and the City of Piqua appeared before the Committee at its Eleventh Meeting on November 6, 1958, with oral proposals for a new location for the reactor farther removed from the populated area, and for better containment of the facility. Because of the preliminary nature of the information presented, the Committee has no basis for arriving at any firm conclusion with respect to these new proposals. However, it can be said that the proposed new location with its approximately quarter mile removal from immediately populated areas is an improvement over the site previously proposed; and, with adequate containment provisions, may prove to be acceptable for a reactor of the general type proposed.

Chairman C. Rogers McCullough did not participate in these reviews or discussions.

Sincerely yours,

/s/

R. C. Stratton
Acting Chairman

cc: P.F.Foster, GM
H.L.Price, DLR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON

January 12, 1959

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

Subject: PROPOSED NUCLEAR POWER REACTOR FOR THE CITY OF PIQUA, OHIO

Dear Mr. McCone:

At the twelfth meeting of the Advisory Committee on Reactor Safeguards on December 11-13, 1958, representatives of Atomics International, the City of Piqua, and the Division of Reactor Development described a revised plan for construction of an organic moderated nuclear power plant in Piqua, Ohio. Earlier plans for this plant were reviewed by the Committee at its tenth and eleventh meetings, and were the subject of letters to you dated August 5 and November 12, 1958. The revised plan presented at the twelfth meeting is described in NAA-SR-MEMO 3405 entitled, Supplement III to Preliminary Safeguards Report for the Piqua Organic Moderated Reactor (NAA-SR-3100).

A subcommittee reviewed Supplement III prior to the thirteenth ACRS meeting. At the thirteenth meeting, a report by the Hazards Evaluation Branch was reviewed, and oral discussion by representatives of the Division of Reactor Development indicated some changes in containment had been proposed.

The site now proposed appears more suitable than the location originally selected. However, the Committee does not consider the installation at this site of a nuclear power plant of this capacity of a relatively untried type to be without undue public hazard until the present proposed unconventional type of containment is replaced by a more substantial and dependable system.

Chairman C. Rogers McCullough did not participate in these reviews and discussions.

Sincerely yours,

W. P. Conner, Jr.
Acting Chairman

cc: Alvin R. Luedecke, GM
Harold L. Price, DLR

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: PROPOSED NUCLEAR POWER REACTOR FOR THE CITY OF PIQUA, OHIO

Dear Mr. McCone:

At the Sixteenth Meeting of the Advisory Committee on Reactor Safeguards on May 14-15, 1959, representatives of Atomics International, the City of Piqua, and the Division of Reactor Development presented for review a revised plan, described in NAA-SR-3575, April 13, 1959, for construction of an organic moderated nuclear power plant in Piqua, Ohio.

Previous proposals were considered at several earlier meetings of the Advisory Committee on Reactor Safeguards and reports were sent to you following each review.

While in principle the Committee does not look with favor upon the location of power reactors immediately adjacent to populated areas, the site as now proposed may be considered as not creating an undue public risk provided (a) adequate containment is constructed as now described by the applicant, (b) the maximum leakage rate for the containment is reduced to an acceptable low value, and (c) that this relatively new reactor system is adequately designed.

Chairman C. Rogers McCullough did not participate in these reviews and discussions.

Sincerely yours,

/s/ W. P. Conner, Jr.
W. P. Conner, Jr.
Acting Chairman

References:

1. NAA-SR-3575 - Preliminary Safeguards Report for the Piqua Organic Moderated Reactor (Revised), April 13, 1959.

2. Report to ACRS by Division of Licensing and Regulation on the Piqua Organic Moderated Reactor, April 28, 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: PROPOSED NUCLEAR POWER REACTOR FOR THE CITY OF PIQUA, OHIO

Dear Mr. McCone:

At the Seventeenth Meeting of the Advisory Committee on Reactor Safeguards on July 23-25, 1959, the Committee received a report from its Subcommittee and reviewed a memo, NAA-SR-MEMO-4048, July 1, 1959, documenting an earlier meeting between representatives of Atomics International, the Subcommittee and members of the Hazards Evaluation Branch. At the Seventeenth Meeting, representatives of the Division of Reactor Development, Hazards Evaluation Branch, Atomics International and the City of Piqua were present.

The Committee's comments on previous proposals have been reported. In the most recent letter upon such proposals, May 18, 1959, it was stated: "... the site as now proposed may be considered as not creating an undue public risk provided (a) adequate containment is constructed as now described by the applicant, (b) the maximum leakage rate for the containment is reduced to an acceptable low value, and (c) that this relatively new reactor system is adequately designed."

From the information obtained and discussed, it appears requirements (a) and (b) will be satisfied and progress toward (c), adequate design and control of this new reactor system, is satisfactory.

Dependent upon meeting the qualifications above, the Committee concludes that this reactor, as now described, may be constructed and operated at the site selected without undue risk to the health and safety of the public.

Chairman C. Rogers McCullough did not participate in these reviews and discussions.

Sincerely yours,

R. C. Stratton
Acting Chairman

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

References:

NAA-SR-3575 - Preliminary Safeguards Report for the Piqua Organic Moderated Reactor (Revised), April 13, 1959.

NAA-SR-MEMO-4048 - Supplement I to the Preliminary Safeguards Report for the Piqua Organic Moderated Reactor (Revised), July 1, 1959.

Report to ACRS by the Division of Licensing and Regulation on the Piqua Organic Moderated Reactor, April 28, 1959.

Report to ACRS by the Division of Licensing and Regulation on the Piqua Organic Moderated Reactor, July 2, 1959.

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 PIQUA NUCLEAR POWER FACILITY, PIQUA, OHIO

Dear Dr. Seaborg:

At its thirty-fourth meeting on May 18-20, 1961, in Cambridge, Massachusetts, the Advisory Committee on Reactor Safeguards considered the final safeguards report, a supplement and a proposal to operate as submitted by the applicant, Atomics International. Representatives of Atomics International, the City of Piqua, and the AEC staff were present at this meeting and participated in the discussion. On April 20, 1961 a meeting was held at the site between the ACRS subcommittee, representatives of the applicant, and the AEC staff.

The Piqua Nuclear Power Facility is a heterogeneous, organic-cooled and -moderated reactor with adequate containment, designed for a power level of 45.5 MW(t). The steam generated will be utilized in an existing city-owned power station. Previous ACRS letters raised questions which have now been resolved.

During the term of the applicant's control of the operation of this reactor, it is suggested that at each partial fuel reloading cycle (approximately at four-to-six month intervals of power operation) an inspection be made of a typical sample portion of the remaining fuel elements and a report forwarded to the Division of Licensing and Regulation.

Based upon the information presented and discussed, it is the opinion of the ACRS that this reactor can be operated by Atomics International without undue hazard to the health and safety of the public.

Dr. C. Rogers McCullough did not participate in these reviews or discussions.

Sincerely yours,

/s/

T. J. Thompson
Chairman

(References attached)

References

1. NAA-SR-5608 Final Safeguards Summary Report for the Piqua Nuclear Power Facility, dated February 1, 1961.

2. NAA-SR-Memo-5608 (Suppl), Supplement 1, Final Safeguards Summary Report for the Piqua Nuclear Power Facility, dated May 8, 1961.

3. Figure IV-6 to Final Safeguards Summary Report undated, received May 1, 1961.

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 PIQUA NUCLEAR POWER FACILITY

Dear Dr. Seaborg:

At its fifty-third meeting held on February 13-15, 1964, and at the fifty-fourth meeting on April 2-4, 1964, the Advisory Committee on Reactor Safeguards considered the application of the City of Piqua, Ohio, for authorization to assume the operating responsibility for the Piqua Nuclear Power Facility, which is now being operated by Atomics International. The Committee heard at each meeting oral presentations by representatives of the City of Piqua, Atomics International, and the AEC staff, and had the benefit of the listed reports. A subcommittee meeting was held at this facility on March 5, 1964.

This organic-cooled and moderated reactor has been operated at its design full power of 45.5 MW(t) by Atomics International, and it is reported that no operational problems remain unresolved.

It appears to the Committee that the nuclear plant operating organization proposed contains only an acceptable minimum of fully qualified supervisory personnel. A Safety Review Committee has been established, containing both plant and consultant personnel, under a charter clearly defining its functions, procedures, and responsibilities.

The Committee assumes that at least the present minimum number of competent personnel will be retained, and that the Safety Review Committee will function within the requirements of its charter, subject to minor revisions as discussed with the Regulatory Staff, but not yet documented by the applicant. Based on this assumption, it is the opinion of the Advisory Committee on Reactor Safeguards that this reactor can be operated by the City of Piqua without undue hazard to the health and safety of the public.

Sincerely yours,

/s/

Herbert Kouts
Chairman

References - Piqua

1. Letter from City of Piqua to AEC Division of Licensing and Regulation dated February 21, 1963, with enclosure "Application for Operation Authorization".
2. Letter from City of Piqua to Mr. E. R. Price, PNPF-365-63 dated September 9, 1963, with enclosure "Application for Operating Authorization (Revised)".
3. Letter from City of Piqua to Mr. E. R. Price, PNPF-410-63, dated October 14, 1963, with enclosures.
4. Letter from City of Piqua to Mr. Saul Levine, PNPF-26-64, dated January 21, 1964, with enclosures.
5. Letter from Atomics International to Mr. R. Lowenstein, 64AT420, dated January 17, 1964, with enclosure, "Summary Report of Testing and Operation to 50% Rated Power".
6. Letter from Atomics International to Mr. R. Lowenstein, 64AT1150, dated February 7, 1964, with enclosure, "Supplementary Summary Report of Testing and Operation to 100% Rated Power".
7. Letter from Atomics International to Mr. R. Lowenstein, 64AT1635, dated March 12, 1964, with enclosure, "Report of Power Operation".
8. Letter from City of Piqua to Mr. Saul Levine, PNPF-118-64, dated March 23, 1964, with enclosure, "Charter for the Piqua Nuclear Power Facility Safety Review Committee".

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: PLUTONIUM RECYCLE TEST REACTOR

Dear Mr. McCone:

At its eleventh meeting (November 6-8, 1958), the Advisory Committee on Reactor Safeguards reviewed the design (at its present state of development) of the Plutonium Recycle Test Reactor. In addition to the reports referenced below, descriptions of the proposed designs were presented by members of the Hazards Evaluation Branch and representatives of Hanford. Additional clarifying information has been received since that meeting. The views of the Committee are summarized below:

(1) The proposed site appears to be suitable for the proposed facility in view of the power level, the inherent nuclear stability of the active lattice, and the intended containment features.

(2) The proposed scram mechanism is unusual but appears to be adequate.

(3) An untried method for fine control is proposed, but we do not feel that this feature increases the public risk appreciably.

The Committee understands that the design is proceeding towards the following objectives:

(4) The shim rods are to be designed so that the rods will move only when the drive motor is operating. Only one shim rod motor may be run at a time.

(5) Emergency light water cooling for the fuel elements will be applied automatically less than 30 seconds after severe loss of pressure in the coolant system.

(6) To facilitate testing or maintenance, the designers propose that instruments which initiate a scram signal may be by-passed. A by-pass which is to be used only when the pile is shut down must be interlocked so as to make operation or startup impossible

until the by-pass has been removed. In order to by-pass such an instrument during operation, duplication of instrumentation must be provided so that the reactor will not operate without proper protective devices.

Sincerely yours,

/s/
C. Rogers McCullough
Chairman

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

References:

HW-46461 - Plutonium Recycle Program Demonstration Reactor Site Study, November 7, 1956.

HW-48800 - Plutonium Recycle Program Reactor Preliminary Safeguards Analysis, July 12, 1957.

Hw-48800 - (Rev) Plutonium Recycle Test Reactor Preliminary Safeguards Analysis, June 5, 1958.

Hazards Evaluation Branch Report to the Advisory Committee on Reactor Safeguards, October 8, 1958.

U. S. Weather Bureau Comments, September 16, 1958.

February 1, 1960

Mr. A. R. Luedecke
General Manager
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: PLUTONIUM RECYCLE TEST REACTOR (PRTR)

Dear Mr. Luedecke:

We list below a number of technical matters on the Plutonium Recycle Test Reactor (PRTR) which the subcommittee on this reactor wants to pursue with the AEC Staff.

A. Study

1. Possible positive void coefficients. Finally measure using the complete reactor as a critical assembly and include information in accident calculations. These measurements and calculations should be repeated with each substantial change of core loading. Physics studies on the all-plutonium core should precede such critical experiments.

2. Effect of broken pressure tubes. The Committee does not understand the analysis of this effect in the Hazard Report (HW-61236).

B. Interlocks

1. Shim rods and moderator level should not move simultaneously.

2. It should be impossible to increase reactivity at the maximum rate except when the high level trip is set at about one-tenth full power.

3. There should be a warning signal if the voltage across any ionization chambers falls appreciably below the specified value.

4. Standard interlocks should prevent startup without a positive signal from startup or period channel.

C. Procedure

1. Maximum rate of increasing reactivity should not be used during a prolonged shakedown period. The rate should be decreased by about a factor of four during this time.

2. The Committee cannot understand the claimed advantages of automatic control under any other condition than constant power.

3. Manual operation should be utilized during initial startup tests and full power tests. The automatic level control should be used only after initial tests.

Sincerely yours,

/s/
Leslie Silverman
Chairman

References:

HW-61236 - Plutonium Recycle Test Reactor Final Safeguards Analysis, October 1, 1959.

U. S. Weather Bureau Comments on HW-61236, December 16, 1959.

Division of Licensing and Regulation Report to the ACRS on the Plutonium Recycle Test Reactor (PRTR), January 12, 1960.

Comments of the Office of Health and Safety on the Plutonium Recycle Test Reactor Final Safeguards Analysis, January 15, 1960.

cc: H. L. Price, DL&R
W. F. Finan, OGM

February 1, 1960

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

Subject: PLUTONIUM RECYCLE TEST REACTOR (PRTR)

Dear Mr. McCone:

At its twenty-third meeting, January 28-30, 1960, the Advisory Committee on Reactor Safeguards reviewed the Plutonium Recycle Test Reactor. We considered the Final Safeguards Analysis (HW-61236), information presented orally by Hanford personnel, and the views of the Hazards Evaluation Branch.

On December 15, 1958, the ACRS advised the Commission that the proposed site and containment were suitable to the general reactor as conceived at that time. For the most part, the design objectives mentioned in our previous letter have been attained and we see no difficulties which probably cannot be resolved prior to the completion of construction. Operating procedures and certain design details are still under consideration and we must reserve our final judgment until we are informed on the results of these studies.

Sincerely yours,

/s/

Leslie Silverman
Chairman

cc: A.R.Luedecke, GM
W.F.Finan, OGM
H.L.Price, DL&R
ACRS Members & Dr. Duffey
bc: L. K. Olson, gc

References

1) HW-61236 - Plutonium Recycle Test Reactor Final Safeguards Analysis, October 1, 1959.

2) U. S. Weather Bureau Comments on HW-61236, December 16, 1959.

3) Division of Licensing and Regulation Report to the ACRS on the Plutonium Recycle Test Reactor (PRTR), January 12, 1960.

4) Comments of the Office of Health and Safety on the Plutonium Recycle Test Reactor Final Safeguards Analysis, January 15, 1960.

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

May 9, 1960

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

Subject: PLUTONIUM RECYCLE TEST REACTOR (PRTR)

Dear Mr. McCone:

At its twenty-fifth meeting on May 5-7, 1960, the Advisory Committee on Reactor Safeguards heard a report of its PRTR Subcommittee which had conferred extensively with representatives of Hanford, Division of Reactor Development and the Hazards Evaluation Branch. It was concluded that most of the details mentioned in our letters of February 1, 1960, to you and Mr. Luedecke are being satisfactorily settled. However, the following comments seem pertinent:

1) Filters should be installed at the outlet of the containment ventilating system.

2) Experimental information on void coefficients will be developed during the critical experiment program. Efforts to avoid a positive void coefficient should not be relaxed.

3) The Committee is skeptical of the claimed advantages of automatic control under any other condition than constant power.

Sincerely yours,

/s/
Leslie Silverman
Chairman

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

Reference:
HW-61236 SUP 1, 4/15/60

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

July 8, 1961

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

Subject: REPORT ON PM-1 NUCLEAR POWER PLANT

Dear Dr. Seaborg:

At its thirty-fifth meeting, July 6-8, 1961, the Advisory Committee on Reactor Safeguards reviewed the PM-1 reactor which is scheduled to operate at Sundance, Wyoming, reaching full power in September 1961. During the first six months of operation the Martin Company will be in charge. After this time the Air Force will take over the operation. The present review is for the first six months of operation.

The Committee recognizes that this power reactor does not have containment, as the term is usually understood, although it is installed in vertical tanks below grade. In view of its low power (9.37 MW thermal) and its remote location, the Committee agrees this is acceptable.

The Committee is of the opinion that the Martin Company staff, which is assuming full responsibility for the initial operation, is technically competent to operate this reactor. It should be noted that their contract is under the New York Operations Office and there is no evident means of having inspection of this reactor by any group independent of the operator or the contracting office. The Committee believes such an independent inspection to be desirable and recommends it be accomplished in a manner comparable to a licensed reactor installation.

With such independent inspection, the ACRS believes this reactor can be operated as proposed without undue hazard to the health and safety of the public, including site personnel.

Dr. T. J. Thompson did not participate in the reviews or discussions of this project.

Sincerely yours,

/s/

C. Rogers McCullough
Acting Chairman

References:

Hazards Summary Evaluation-MND-M-1853, dated October 15, 1959.

Addendum to Hazards Summary Evaluation MND-1853 (ADD I), dated February 1961.

Use of Precipitation Hardened Stainless Steel in the PM-1 Reactor with attached drawings, undated, received June 28, 1961.

Answers to Question Eight (8) of "January Reactor Facility Questionnaire," undated, received June 28, 1961.

Analysis of Core Thermal Conditions Following a Scram, undated, received June 28, 1961.

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

August 30, 1962

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

SUBJECT: REPORT ON PM-1 NUCLEAR POWER PLANT

Dear Dr. Seaborg:

At its forty-third meeting, August 23-25, 1962, at Idaho Falls, Idaho, the Advisory Committee on Reactor Safeguards reviewed the proposed U. S. Air Force plans for maintenance and operation of the PM-1 reactor now installed at Sundance, Wyoming. The PM-1 reactor became critical February 25, 1962. All testing is now complete. It is planned to transfer responsibility for operation of the plant from the Martin-Marietta Company to the U. S. Air Force on October 31, 1962. The ACRS reviewed the referenced reports and on August 23, 1962, discussed the operation of the PM-1 reactor with U. S. Air Force personnel and with the AEC staff.

In its letter dated July 8, 1961, the ACRS stated its belief that this reactor should periodically be given an independent inspection comparable to the inspections given licensed reactors. U. S. Air Force regulations require that a safety survey be made of all organizations at least yearly. Consultants may be used in specialized areas. A special regulation governs surveys of nuclear reactor installations.

The ACRS believes that consultants should be used in the inspection of PM-1 to provide a degree of outside review of staff competence, operational procedures and maintenance techniques. The ACRS also believes that training and experience requirements for key operating staff should be established at a level so that operation of the PM-1 reactor will continue to be by fully qualified personnel. In particular, the Committee agrees with the AEC staff recommendation that three qualified reactor engineers be members of the PM-1 staff.

With adequate attention to details of inspection and personnel qualifications, the ACRS believes that the operation of this reactor, following transfer of responsibility, can be continued at its present site without undue hazard to the health and safety of the public, and the site personnel.

Dr. T. J. Thompson did not participate in the reviews or discussions of this project.

Sincerely yours,

/s/

F. A. Gifford, Jr.
Chairman

References:

1. U. S. Air Force Report, PM-1 Nuclear Power Plant Capability Report to the AEC, dated June 1962.
2. U. S. Air Force letter to U.S. AEC, subject: PM-1 Presentation to AEC, dated July 27, 1962.

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

December 13, 1961

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

Subject: REPORT ON PM-2A

Dear Dr. Seaborg:

At its thirty-eighth meeting on December 7-9, 1961, the Advisory Committee on Reactor Safeguards considered the Army pre-packaged nuclear power station PM-2A. The Committee listened to representatives from the Corps of Engineers, Department of the Army, as operators of the station and to representatives of ALCO Products, Inc., as designers and fabricators of the reactor. The Committee also had the benefit of analysis by the staff and the documents listed.

The PM-2A is a 10 MW(th) pressurized water reactor located at Camp Century, on the Greenland Ice Cap, 150 miles east of Thule Air Force Base. The plant has been in operation since March 8, 1961 and has accumulated two megawatt years of operation to date. It should be noted that this is the first time that this project has been referred to the Committee for advice. Accordingly, some of the comments which follow concern design features which may not be capable of being changed in the PM-2A but deserve to be considered for incorporation in future reactors of a similar type.

The information which was presented to the Committee was lacking in detail in some areas. However, in view of the remote location and the very low population density near the reactor site, the deficiencies are not considered to be an important safety factor.

There are several points which are worth mentioning as representing departures from accepted good practice.

A generally accepted criterion for the design of reactors provides that criticality should not be achievable on the removal of any one control rod. It is recognized that this feature alone does not make a reactor safe nor guard against a careless operator. It remains, however, a useful bulwark against the malfunctions which are assignable to human errors. This feature was not included in the original design with a clean core but will be present in the next modified replacement core. With 20% of design life expended, the first core now almost meets the criterion and because of this, no issue is made of this exception.

A definitive measurement of the integral leak rate of the vapor containment after installation was not reported. It is suggested that periodic experimental verification of the containment integrity would be desirable. Refueling periods would seem appropriate times to make these measurements.

The vapor container which has been designed to provide containment in the event of primary vessel rupture has an internal obstruction which might, under conceivable circumstances, effectively reduce the available volume for expansion of steam. The resulting high pressure which could be attained in a primary system failure might also breach the secondary containment. Relatively simple design modifications could avoid this situation.

Instrumentation used for low range startup is marginally acceptable in that only one channel of counting equipment has been provided for the lowest and intermediate ranges. It is understood that methods of providing desirable duplication of at least the startup channel are under consideration.

The presentation did not clearly describe the emergency procedures which have been established to cope with disasters. In particular, information was lacking on provisions for control of plant ventilation and of access of contaminated air to the balance of the camp in the event of a radioactive release.

It is unfortunate that this first review of the PM-2A reactor and its installation comes after the unit has been constructed and operated. It is recommended that deficiencies be corrected where practicable but, in view of the remote location, it is the opinion of the Committee that the PM-2A can continue to be operated without undue hazard to the health and safety of the public including camp personnel.

Sincerely yours,

Sgd/T. J. THOMPSON

T. J. Thompson
Chairman

References attached

References:

APAE No. 49, Revision 1 - Hazards Summary Report for a Prepackaged Nuclear Power Plant for an Ice Cap Location (PM-2A), Issued July 14, 1961.

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

September 11, 1961

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

Subject: REPORT ON PM-3A

Dear Dr. Seaborg:

At the thirty-sixth meeting of the Advisory Committee on Reactor Safeguards on September 7-9, 1961, the PM-3A Nuclear Power Plant was reviewed. Representatives of the contractor and AEC Staff participated. Documents as listed were available.

The PM-3A is a low power, 9.36 MW(th), pressurized water reactor designed for operation in isolated areas such as the Antarctic. The plant, including its containment, is designed for air transportation and is readily assembled at site. The interconnected containment vessels do not meet the ASME code but stress analyses reported by the manufacturer have shown the vessels to be satisfactory for the intended use.

Meteorology at the Antarctic site differs markedly from that in the temperate zones. The contractor should re-evaluate a fission product release using meteorological techniques and parameters suitable to the area and more realistic fission product release rates to establish the conditions that might exist at the time of an accident and its effect on evacuation.

The Martin Company will start up the reactor and remain in responsible charge until Naval personnel take over the operation. The initial owner's acceptance of the installation is a responsibility of the New York Operations Office. The ACRS advises that an independent safety inspection should be arranged in a manner comparable to a licensed reactor installation as suggested for the PM-1.

Subject to the above considerations, the ACRS believes this reactor may be operated as proposed without undue hazard to the health and safety of the site personnel.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

References:

1. MND-M3A-2496, Vol. 1 - "PM-3A Nuclear Power Plant Hazards Summary Report Plant Design," dated March 1961.
2. MND-M3A-2496-II, Vol. II - "PM-3A Nuclear Power Plant Hazards Summary Report, Site Description and Safety Evaluation, "dated June 1961.
3. Errata Sheet, PM-3A Hazards Summary Report, Plant Design, Vol. I, undated, received August 2, 1961.
4. Letter from Martin Company to AEC, dated September 5, 1961, Additional Information to Complete PM-3A ACRS Review.
5. Letter from Martin Company to AEC, dated September 1, 1961, Additional Information to Complete PM-3A ACRS Review.

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

February 25, 1964

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

Subject: REPORT ON PM-3A

Dear Dr. Seaborg:

At a special meeting of the Advisory Committee on Reactor Safeguards held on February 24, 1964, the Committee considered the transfer of operating responsibility for the PM-3A nuclear power plant from the Martin Company to the Department of the Navy. Representatives of the U. S. Navy, their consultants, the Martin Company and the AEC staff participated in the discussions. Documents as listed were available.

The PM-3A was considered previously by the Committee at its thirty-sixth meeting on September 7-9, 1961. At that time, the Committee commented on the unusual meteorology of the Antarctic site and expressed the opinion that an evaluation of the consequences of a realistic fission product release should be made. A subsequent analysis has been provided, but this is still based on meteorological assumptions that are not conservative. A more thorough analysis is called for.

Some degree of uncertainty still remains as to the appropriate leakage rate to be assumed in evaluating accident conditions. A separate but related consideration is the need for developing suitable procedures for periodic leak-testing of the containment vessels.

The uncertainty in leakage rate is further complicated by a lack of confidence in the ability of stress-relieved T-1 steel, as used for the containment vessels, to perform adequately at stress levels calculated for the maximum postulated accident. The available data on other heats of T-1 steel suggest that containment may in fact be violated with a correspondingly greater release of fission products.

The Committee notes that criteria for application of emergency procedures have not been formulated and does not believe that presently proposed procedures visualize accidents as severe as implied in the foregoing. The Committee, therefore, suggests that emergency procedures be reviewed to determine if adequate protection can be provided for the site personnel during all conceivable releases. Emergency procedures should be rehearsed at intervals so that all personnel are fully informed.

In the absence of advance testing, the Committee cautions that the joining of the control rod thimble and the coil-can to minimize corrosion of the thimble may cause other unforeseen problems.

The Committee believes that the instrumentation planned for the plant (including modifications now underway) can and should be improved. For example, better monitoring of ionization chamber voltage and much more frequent testing of safety system parts are desirable. The ACRS understands that the Martin Company and the AEC staff are discussing modifications that should improve performance. The Committee believes that the remaining instrumentation problems can be resolved by these discussions.

The concern expressed about the possible environmental releases and instrument inadequacies do not relate directly to the question of operating responsibility. The ACRS was informed of the long-range plans being formulated by the Bureau of Yards and Docks of the U. S. Navy to maintain technical supervision of plant operations from the continental United States based on detailed weekly and special reports from the site and annual inspection trips. The remote area, and associated difficulties of staffing, force the Navy to adopt this procedure. The Committee is concerned, however, over the practical efficacy of this mode of operation and urges vigilant supervision.

In spite of the major reservations that the Committee has expressed above on the PM-3A nuclear power plant, the ACRS believes that its operation by the U. S. Navy will not result in any change in hazard to the health and safety of the site personnel.

The Committee would like to be kept informed of progress made in resolution of the reservations stated above.

Sincerely yours,

/s/
Herbert Kouts
Chairman

References Attached.

References - PM-3A

1. Letter from Department of the Navy, Bureau of Yards and Docks, to Director of Regulation, U. S. Atomic Energy Commission, dated 13 Jan. 1964, with attachment, "The Bureau of Yards and Docks Policies and Instructions for the Operation of Nuclear Shore Power Plants".
2. Memorandum from Director, Division of Reactor Development, to Director, Division of Licensing and Regulation, dated Jan. 23, 1964, subject: PM-3A Safety Review.
3. "PM-3A Nuclear Power Plant - Hazards Summary Report - Corrections for As-Built Conditions", Volume I Addendum, MND-M3A-2496-I Add., dated December 1962.
4. "PM-3A Control Rod Drive Mechanism Armature Housing Metallurgical Examination", dated November 11, 1963; Reprinted Dec. 18, 1963.
5. "PM-3A Operating and Test Report - March 1962 to May 1963", MND-M3A-3068, dated October 1963.
6. "PM-3A Nuclear Power Plant - Hazards Summary Report - Description of Plant Experience and Changes", MND-3315-1, dated January 1964.
7. "Hazards Analysis - Control Rod Actuator Thimble Rupture", MDN-M3A 3108 Part D, dated November 20, 1963; Revised Jan. 28, 1964.
8. "Hydrogen Release Hazards in the PM-3A Containment Vessels", MND-M3A-3108 Part C, dated November 15, 1963; Revised - Jan. 28, 1964.
9. "Radioactive Iodine Release from PM-3A Containment Vessels", MND-M3A 3108 Part B, dated Nov. 15, 1963; Revised Jan. 28, 1964.
10. Letter from C. W. Mallory, Director Nuclear Power Div. Bureau of Yards and Docks, to USAEC, Div. of Licensing & Regulation, dated Feb. 24, 1964.

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

May 16, 1967

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

Subject: REPORT ON POINT BEACH NUCLEAR PLANT, UNIT NO. 1

Dear Dr. Seaborg:

At its eighty-fifth meeting, May 11-13, 1967, the Advisory Committee on Reactor Safeguards completed its review of the application of Wisconsin Michigan Power Company for authorization to construct Point Beach Nuclear Plant Unit No. 1. The project was previously considered at an ACRS Subcommittee meeting on March 24, 1967. During its review, the Committee had the benefit of discussions with representatives of Wisconsin Michigan Power Company, Westinghouse Electric Corporation, Bechtel Corporation, and the AEC Regulatory Staff and its consultants. The Committee also had the benefit of the documents listed.

Point Beach Unit No. 1 is to be located in Manitowoc County, Wisconsin, on the west shore of Lake Michigan approximately 30 miles southeast of Green Bay. It includes a two-loop pressurized water reactor of design similar to the R. E. Ginna Plant Unit No. 1, previously reviewed. The plant is designed for an initial power output of 1396 MWt, and the applicant states that his present application is made with reference to this power level.

The containment structure is similar to those for the Turkey Point Units Nos. 3 and 4, previously reviewed. It consists of a steel-lined concrete shell with shallow spherical dome and flat slab base. The shell and dome are fully pre-stressed, with steel tendon systems carrying the principal loads. Design 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 (ECCS) includes a high head safety injection system, a low head residual heat removal system, and an accumulator system incorporating two accumulator tanks, one on each of the two coolant loops. The applicant states that the plant will not be operated if one of the accumulators is isolated. He also plans to analyze more fully any possible effects upon reactor vessel integrity of thermal shock induced by accumulator discharge in postulated loss-of-coolant accidents. The Committee wishes to note that the possible effects of blowdown on core internals should be conservatively evaluated and that evidence should be obtained to show that the effects of fuel failure in loss-of-coolant accidents will not affect significantly the ability of the ECCS to prevent clad melting. The AEC Regulatory Staff should review carefully the final design of the ECCS, including the analyses of system characteristics. The applicant states that provision will be made so that, in the event of flooding of one of the two residual heat removal pumps, the other will not be flooded.

Calculations by the applicant show that the reactor may have a positive moderator coefficient during some portion of core life. He is continuing his analysis of all consequences of the positive coefficient and, if necessary, will adjust the core composition. The Committee recommends that the Regulatory Staff follow these studies and the conclusions resulting.

The applicant states that detailed analysis will be made of the effects of sudden failure (e.g., by seizure) of a main coolant pump, particularly in respect to the minimum departure-from-nucleate-boiling ratio reached in the core. The Regulatory Staff should follow this work closely.

The applicant stated that, although the general arrangement of the protection instrumentation system is still under study, the system will be designed to meet proposed IEEE Standards. The Committee recommends that the Regulatory Staff review the protection system design thoroughly before its fabrication and installation. The applicant will explore further possibilities for improvement, particularly by diversification, of the instrumentation that initiates emergency core cooling, to provide additional assurance against delay of this vital function.

The Advisory Committee on Reactor Safeguards believes that the various items mentioned can be resolved during construction and that the proposed reactor can be built at the Point Beach 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 dated August 31, 1966 from Wisconsin Michigan Power Company to AEC Division of Reactor Licensing.
2. Volume 1, Point Beach Nuclear Plant, Application for Licenses, dated August 19, 1966.
3. Volumes 2 and 3, Point Beach Nuclear Plant, Preliminary Facility Description and Safety Analysis Report (enclosures to August 31, 1966 letter).
4. First Supplement to Preliminary Facility Description and Safety Analysis Report, dated December 14, 1966.
5. Application for Exemption under Section 50.12 of Regulation of AEC, dated January 3, 1967.
6. Second Supplement to Preliminary Facility Description and Safety Analysis Report, dated February 3, 1967.
7. Supplement No. 3 to Application for Licenses, dated March 15, 1967.
8. Supplement No. 4 to Application for Licenses, dated April 14, 1967.
9. Addendum to Supplement No. 4 to Application for Licenses, dated May 5, 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: REPORT ON POINT BEACH NUCLEAR PLANT, UNIT NO. 2

Dear Dr. Seaborg:

At its ninety-seventh meeting, May 9-11, 1968, the Advisory Committee on Reactor Safeguards completed its review of the application of Wisconsin Electric Power Company and Wisconsin Michigan Power Company for authorization to construct Point Beach Nuclear Plant, Unit No. 2. The project was previously considered at an ACRS Subcommittee meeting on April 29, 1968. During its review, the Committee had the benefit of discussions with representatives of Wisconsin Electric Power Company, Wisconsin Michigan Power Company, Westinghouse Electric Corporation, Bechtel Corporation, and the AEC Regulatory Staff and its consultants. The Committee also had the benefit of the documents listed.

Unit No. 2 is to be located in Manitowoc County, Wisconsin, at the same site as Point Beach Nuclear Plant, Unit No. 1, previously reviewed. Unit No. 2 is virtually identical in design to Unit No. 1. It includes a two-loop pressurized water reactor with initial power output of 1396 MWt, and a steel-lined concrete containment structure with pre-stressed shell and dome.

The complex of emergency core cooling systems includes a high head safety injection system, a low head residual heat removal system, and an accumulator system incorporating two accumulator tanks, one on each of the two coolant loops. The plant is to be shut down if either of the accumulators is isolated. The Committee recommends that the AEC Regulatory Staff carefully review the final design of the entire emergency core cooling system.

Tanks of concentrated boric acid solution are incorporated in the high head safety injection system by manifolding them to the suction of the high head pumps. Initial injection of this solution reduces the time at criticality during a steam line break accident and provides additional assurance of reactor shutdown in a loss-of-coolant accident.

Fixed burnable poison rods are provided in the first core loading to preclude existence of a positive moderator coefficient at any time during core life.

Four part-length rod cluster control assemblies are incorporated in the reactor core. These are expected to aid in control of potential xenon oscillations. The applicant expects to be able to monitor adequately the influence of these assemblies, particularly on axial core power distribution by use of detectors located external to the core. However, he has agreed to provide in-core instrumentation for this purpose if satisfactory determination of axial power peaking with external detectors is not demonstrated by the analytical and experimental study now being pursued.

The applicant is performing a detailed anslysis of the effects of sudden failure (e.g., by seizure) of a main coolant pump, particularly in respect to the minimum departure-from-nucleate-boiling ratio reached in the core. The Regulatory Staff should continue to review this work.

The Committee continues to believe that control and protection instrumentation should be separated to the fullest extent practical. There remain questions in this area on the Point Beach design. The Committee recommends that the Regulatory Staff review the protection system design before its fabrication and installation.

The Committee also calls you attention to those matters previously emphasized which it deems to be important for all large water-cooled power reactors.

The Advisory Committee on Reactor Safeguards believes that the various items mentioned can be resolved during construction and that the proposed Unit No. 2 can be built at the Point Beach 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.

References - Point Beach Unit No. 2

1. Application for License, dated July 17, 1967, from Wisconsin Electric Power Company and Wisconsin Michigan Power Company; Volumes 1, 2 and 3 of Preliminary Facility Description and Safety Analysis Report, Point Beach Nuclear Plant, Unit No. 2

2. Letter from Wisconsin Electric Power Company, dated November 3, 1967; Amendment to Application for License

3. Letter from Wisconsin Electric Power Company, dated December 21, 1967; Amendment No. 2 to Application for License

4. First Supplement to Application for License, dated January 11, 1968

5. Letter from Wisconsin Electric Power Company, dated January 12, 1968

6. Second Supplement to Application for License, dated February 24, 1968

7. Third Supplement to Application for License, dated April 4, 1968

April 16, 1970

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

Subject: POINT BEACH NUCLEAR PLANT UNITS 1 AND 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 Wisconsin Electric Power Company and the Wisconsin Michigan Power Company for a license to operate the Point Beach Nuclear Plant Units 1 and 2 at powers up to 1518 MWt for each unit. Subcommittee meetings with the applicant were held on February 19, 1970, at the site, and on April 8, 1970, in Washington, D. C. In the course of the review, the Committee had the benefit of discussions with representatives of the applicant, the Westinghouse Electric Corporation, Bechtel Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Point Beach Nuclear Plant is located in Manitowoc County, Wisconsin, on the west shore of Lake Michigan approximately 30 miles southeast of Green Bay. The Committee's reports of May 16, 1967, and May 15, 1968, reviewed the applications for the construction of Units 1 and 2, respectively, at the design initial power outputs of 1396 MWt for each unit. Justification for the increase to 1518 MWt has been made by the applicant on the basis of improved flux flattening. Operation of Unit 2 is scheduled to follow Unit 1 by about nine months. The Committee's review is the first for operating licenses for twin, pressurized water reactors at a plant. The Committee's conclusion in this report, with respect to Unit 2, is contingent upon the successful operating experience of Unit 1 and other similar reactors prior to operation of Unit 2.

The nuclear steam supply systems are essentially the same as that used in the Ginna Nuclear Power Plant, previously approved for operation (ACRS Report of May 15, 1969). The prestressed concrete containments are similar in design to the Palisades containment noted in the ACRS Report of January 27, 1970. The items mentioned in the ACRS reports at the construction stage have been duly taken into account by the applicant, and other improvements have been incorporated in the plant.

The applicant has presented a program for preoperational tests of the plant, including proof testing of the containments. The applicant is performing studies to determine the appropriate number of tendons and the interval for tendon inspection. The applicant is following the work of others for inservice vibration monitoring and loose parts detection so as to evaluate the applicability and appropriateness of implementing such means when developed. Neutronic and external accelerometer signature measurements of the reactors during initial operation should be considered in order to provide a basis for comparison with possible future monitoring results. These matters should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant has determined that turbine failure could release missiles that might damage fuel elements in the fuel pool. He has stated that, for each turbine, a second, completely independent speed control system designed to meet nuclear protection system criteria of redundancy, separation, and reliability, will be installed to reduce the probability of an overspeed condition. In a related matter, in the evaluation of refueling accidents, studies pertaining to reduction of fission product releases have not been completed. The Committee recommends that irradiated fuel not be handled outside the containment building until these matters are resolved in a manner satisfactory to the Regulatory Staff.

Some question exists as to the advisability of automatic or manual transfer of the d-c control power to the switchgear for the engineered safety features. This matter should be resolved between the applicant and the Regulatory Staff.

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.

The applicant is using a partial loading of helium "pre-pressurized" fuel rods. The Committee believes that some surveillance of the Point Beach 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 end-of-life.

Studies by the applicant are underway on two problems identified in previous reports of the Committee, as follows:

(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 ttansients.

(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 Staff and applicant in the Point Beach plant 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 of each unit, there is reasonable assurance that the Point Beach Nuclear Plant Units 1 and 2 can be operated at power levels up to 1518 MWt for each unit without undue risk to the health and safety of the public.

Sincerely yours,

/s/
Joseph M. Hendrie
Chairman

References:

1. License Application for Operating Licenses, dated March 12, 1969; Final Safety Analysis Report, Volumes 1, 2, 3, and 4
2. Amendment No. 1 to FSAR, dated May 12, 1969
3. Amendment No. 3 to FSAR, dated January 19, 1970; Volume 5 of FSAR
4. Amendment No. 3 to FSAR, dated February 11, 1970
5. Amendment No. 4 to FSAR, dated March 13, 1970
6. Amendment No. 5 to FSAR, dated April 2, 1970

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: POOL-TYPE REACTORS

Dear Mr. McCone:

In response to its request of November 17, 1960, the Advisory Committee on Reactor Safeguards received a report dated December 6, 1960 from the Division of Licensing and Regulation on pool-type control rod problems.

From the information contained in this report, it is evident that in certain pool-type reactors a control rod jamming in a fuel element could withdraw the element from the core and then subsequently permit it to drop back into the core. This type of malfunction could result in a serious accident.

It is therefore the recommendation of the Advisory Committee on Reactor Safeguards that, in view of the possibility of a serious accident, all operators of pool-type reactors should be notified to take special immediate action to make sure that no fuel elements can be withdrawn with control rods.

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 25, D. C.

November 24, 1965

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

Subject: REPORT ON THE POWER BURST FACILITY

Dear Dr. Seaborg:

At its sixty-eighth meeting, November 10-12, 1965, the Advisory Committee on Reactor Safeguards considered the proposal of Phillips Petroleum Company to construct the Power Burst Facility (PBF) at the National Reactor Testing Station. The Committee had the benefit of discussions with representatives from Phillips Petroleum Company and the AEC Staff, and of the documents referenced below. A Subcommittee reviewed the proposal on November 3, 1965.

The reactor is designed to be capable of generating self-limiting power bursts with peak power levels up to about 350 x 10^9 watts and with initial periods as short as one millisecond. The total core energy release associated with the bursts is expected to be as large as 3000 Mw-sec. The principal purpose of the facility is to provide, without damaging the driver core, large neutron bursts within a central test section in which experimental samples, fuel elements, and clusters of fuel elements can be heated to failure.

The reactor has a steady state heat removal capacity of 20 Mw, but it will be run at such powers only for brief periods in connection with the experimental program.

The core will be about 52 inches in diameter with an active height of 36 inches. It will be supported in a water filled open tank 15 feet in diameter and 28-3/4 feet deep.

Cooling is provided by forced circulation of water through the core. The reactor is regulated by eight control rods. In addition, four pneumatically driven transient rods are provided to initiate self-limiting power bursts. The reactivity insertion rates may be varied up to a maximum rate of $150/sec. The maximum temperature of the core fuel will be limited to temperatures less than the melting point of the fuel pellets (about 2450° C).

The present fuel rod design consists of 304 stainless steel outer clad, an inner zirconia sleeve, and pellets of UO_2 and calcia-stabilized zirconia (ZrO_2 - CaO).

The central experimental region will accommodate a pressurized water loop with a heavy walled in-pile section. The first loop will be designed for operating conditions of 2250 psig and 650° F. An external cooling system will provide for removing 500 kw from the loop experiment. A loop clean-up system will be provided to accommodate fission products which might be released from destructive experiments in the loop. Other coolants have been mentioned for possible use in the central loop, but no designs are available, and the safety of operation with such loops was not reviewed at this time.

Representatives of Phillips Petroleum Company stated that autocatalytic reactivity effects are associated with the loss of water from the loop, and care must be taken to limit the reactivity which could result from voiding the loop.

Core and loop structure will be exposed to peak fluxes of 10^{13} n/cm^2-sec during bursts, and the Phillips Company will look into possible undesirable effects of such fluxes during its development program.

Careful analysis of the possibility of loop failure must be made prior to the acceptance and performance of specific experiments. Representatives of the Phillips Company stated that it will develop an overall group of criteria on which to judge the acceptability of experiments.

It should be noted that the fuel elements proposed for use in the PBF core are still under development. Although tests have been run at the TREAT Reactor and during the KIWI-TNT experiment, the design has not been frozen and further testing programs are planned.

It is planned to use a "lead element" approach in which typical fuel elements are thoroughly tested in the experimental region under more extreme conditions than the elements in the PBF core.

Approach to the maximum burst conditions will be made in a gradual, step-wise fashion. It must be recognized that problems may develop which preclude obtaining the maximum burst condition for which PBF is designed. This could result from uncertainties in the Doppler effect, from limitations imposed on available reactivity due to the consequences of major reactivity accidents, from the possibility that various modes of fuel failure lead to serious core damage or autocatalytic contributions to reactivity excursions.

Since a reactivity excursion considerably beyond that planned will free the bulk of the radioactivity tied up in the core and the experimental section, the Phillips Company proposes to strictly limit the inventory of radioactivity in the core plus experimental loop, and not to perform experiments on plutonium fuel elements.

It is recognized that the potential of an accidental release of fission products is far greater in the PBF than in more conventional test facilities or power reactors. Careful plans for evacuation of the site personnel from the control center must be made and kept in operation.

Because of the low fission product inventory and the care being taken to provide for adequate evacuation of the site personnel, the Advisory Committee on Reactor Safeguards believes that the proposed facility can be constructed and operated without undue risk to the health and safety of the public, in spite of the relatively high likelihood that a fission product release may occur.

Sincerely yours,

/s/
W. D. Manly
Chairman

References:

1. IDO-17060, Preliminary Safety Analysis Report, Power Burst Facility and Revised Pages VI-12, 13 and 26, received August 2, 1965.

2. Additional Information Provided for Safety Review of the PBF dated October 13, 1965.

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

November 12, 1969

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

Subject: ACRS COMMENTS ON THE POWER BURST FACILITY (PBF)

Dear Dr. Seaborg:

In response to a request from the Director, Division of Reactor Development and Technology, the ACRS has reviewed the preliminary draft program outline for the Power Burst Facility and has transmitted comments on this program in a letter to Mr. Robert E. Hollingsworth, General Manager, dated November 12, 1969. A copy of this letter is attached.

Sincerely yours,

/s/
Joseph M. Hendrie
Acting Chairman

Attachment:
Letter from Joseph M. Hendrie,
Acting Chairman, ACRS to
Mr. Robert E. Hollingsworth,
dated November 12, 1969

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

November 12, 1969

Mr. Robert E. Hollingsworth
General Manager
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: ACRS COMMENTS ON THE POWER BURST FACILITY (PBF)

Dear Mr. Hollingsworth:

In response to the letter from Mr. Shaw of June 4, 1969, requesting comments on the preliminary draft of the "PBF Test Program Outline", the Advisory Committee on Reactor Safeguards held a Safety Research Subcommittee meeting at Chicago, Illinois on September 26, 1969, at which time the document was reviewed with the Regulatory Staff and with the Division of Reactor Development and Technology and its contractors. Our comments with regard to the PBF and to the proposed program are herein provided.

1) The Committee believes that PBF is potentially a very valuable facility for reactor safety research; every effort should be made to make it available for experiments as soon as possible.

2) Although the PBF program discussed in the draft document is quite broad, the Committee believes that the effort currently proposed for the first two years is too heavily oriented toward study of the detailed behavior of unirradiated water-cooled, oxide-fuel elements during severe reactivity transients. It is recommended that this portion of the proposed program be reduced considerably and reoriented to emphasize experiments under transient conditions not already studied in other facilities. A considerably greater proportion of the transient experiments should be conducted with pre-irradiated fuel specimens, including a substantial fraction having had high burnup. Less emphasis than now planned should be placed on cases involving transients of very short period. The experiments should be aimed primarily at previously unexplored or poorly explored effects, including fuel-coolant interactions.

3) The water-cooled reactor safety research program in PBF should concurrently investigate, with high priority, the mechanisms and phenomena associated with the initiation, growth, and propagation of fuel pin failure, including the circumstances under which melting of fuel could progress beyond one fuel element. Such a situation could develop in a large power reactor because of a local reduction in heat removal rate (as by flow blockage),

a locally abnormal power density (as by incorrect enrichment of fuel), or a more widespread perturbation in power or flow. These experiments are required in order to ascertain the probability of a local incident progressing into a serious accident and, if possible, the course and consequence of such a sequence of events.

4) The possible early use of PBF for LMFBR research on fuel failure propagation, fuel-failure modes during transients, fuel melting during power to flow mismatches, fuel-coolant interactions, and molten fuel containment should be pursued urgently. While PBF was originally designed primarily for water reactor safety research, some LMFBR safety experiments appear to be practical with the current core configuration. Furthermore, a more realistic simulation of the LMFBR environment may be possible with an altered core configuration (and experimental cavity). This possibility should be pursued expeditiously while preparations proceed for concurrent vigorous programs of water reactor and LMFBR safety research using the first PBF core configuration.

In view of the long lead time required to prepare for experiments in PBF, especially those involving irradiated fuel or a sodium environment, the ACRS recommends prompt implementation of a program revised in accordance with the above comments.

Sincerely yours,

/s/
Joseph M. Hendrie
Acting Chairman

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

December 15, 1972

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

SUBJECT: REPORT ON POWER BURST FACILITY (PBF) TEST PROGRAM

Dear Dr. Schlesinger:

In response to a request from the Director, Division of Reactor Development and Technology (DRDT), the Advisory Committee on Reactor Safeguards has reviewed the PBF Test Program Plan dated March, 1972. The Committee last reported on the proposed PBF Test Program in its letter to Mr. Hollingsworth of November 12, 1969. In this review, the Committee had the benefit of a Subcommittee meeting held at Argonne, Illinois on August 3, 1972 and of discussions with representatives of DRDT, the Aerojet Nuclear Company, the nuclear industry and the Regulatory Staff. The Committee's comments with regard to the proposed PBF Test Program follow:

1) As in November, 1969, the ACRS believes the PBF is potentially a very valuable facility for reactor safety research. Every effort should be made to expedite an aggressive, carefully planned experimental program accompanied by a strong analytical support effort and the necessary, related out-of-reactor studies.

2) The March, 1972, program plan reflects a desirable major change in emphasis which is in general accord with the ACRS recommendations of November, 1969, which were partly reiterated in its report to you on water reactor safety research of February 10, 1972. Highest priority has been placed on potential events involving a serious power-flow mismatch which might lead to departure from nucleate boiling (DNB), fuel element melting and failure, the generation of pressure pulses, or fuel element failure propagation. Priority has also been placed on studying fuel element structural integrity and coolability under postulated loss-of-coolant accident conditions more severe than those which have been studied experimentally out-of-reactor. The Committee concurs in these general priorities.

Some water reactor safety matters which may be amenable to study in PBF have not been specifically identified in the experimental program given in the March, 1972 plan. These include the following: study of the behavior of previously irradiated fuel in anticipated transients, wherein it is intended that a loss of cladding integrity be unlikely; and study of the behavior of fuel, coolant and structural materials under simulated decay-heat conditions to examine phenomena related to the feasibility of retention of molten fuel. The importance of research and development on these matters has been the basis for recommendations by the ACRS in several previous reports.

3) Detailed plans of the first experiments to be run in PBF, including their specific objectives and the mode of accomplishing these objectives, are not yet available. Although the PBF represents a unique facility which will enable tests related to fuel element behavior to be made in-reactor, it will require very considerable pre-planning and analysis of each test to enable the achievement of meaningful and interpretable results. Radial power variations through the test cluster, differences in hydraulic and heat transfer conditions from actual power reactors, and differences in axial power gradient and the design and form of the fuel elements, represent a few of the considerations which will require care in the planning of the actual tests.

A considerable analytical effort in support of the experimental program has been initiated by Aerojet Nuclear Company. However, this analytical effort requires augmentation, and should have the benefit of active participation by other groups in the planning of tests, particularly the water reactor vendors and the Regulatory Staff.

For the future, the Committee recommends timely participation by the Regulatory Staff in the planning and interpretation of each experiment. The Committee also recommends that the nuclear industry accept and pursue vigorously a commensurate share of the efforts required to study these fuel-related safety matters.

4) Partly because of the complexity of in-reactor experiments and partly to provide better understanding of the phenomena involved, the ACRS recommends that an appropriate out-of-reactor experimental program be pursued both in advance of and concurrently with the in-reactor PBF test series. Fuel-coolant interaction represents a potentially vital phenomenon in many of the planned tests; a vigorous program should be initiated promptly to study this effect directly for water power reactors under conditions applicable to both power-flow-mismatch and loss-of-coolant accidents. Also, extensive detailed analyses should be pursued to determine those events in which previous irradiation history, as well as fuel element and sub-assembly design, can be important to the course of the in-reactor experiment.

5) The ACRS recommends that the detailed description and the results of each experiment should be reported as early as possible, and with a wide distribution, so that full advantage can be made of all technical resources in their interpretation and possible beneficial feedback to future tests.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

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

December 18, 1972

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

Subject: REPORT ON THE POWER BURST FACILITY

Dear Dr. Schlesinger:

During its 152nd meeting, December 7-9, 1972, the Advisory Committee on Reactor Safeguards completed its review of the construction of the Power Burst Facility (PBF) and the Commission's plans to operate the reactor. The Committee reported previously on this project on November 24, 1965. The project was also considered at the Committee's 151st meeting, November 9-11, 1972, and at Subcommittee meetings at the site on September 2, 1971, and in Washington, D. C., on October 25, 1972. During its current review, the Committee has had the benefit of discussions with representatives and consultants of the Commission's Division of Reactor Development and Technology, the Aerojet Nuclear Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The PBF is located about 40 miles from Idaho Falls, Idaho, at the National Reactor Testing Station. The nearest boundary of the Testing Station controlled area is about 7 miles to the south; other boundaries are substantially more distant. About 3500 persons normally work in the test area within a radius of 7 miles.

The facility consists of a water-moderated, water-cooled, ambient pressure reactor designed to operate either in a pulsed mode, in a shaped power run (high constant power for a short time), or in a steady state at lower power. The fuel rods are of a unique design. The fuel pellets are composed of a ternary mixture of UO_2, CaO, and ZrO_2. These pellets are separated from the stainless steel cladding by layers of helium gas and ZrO_2. About 2400 such rods make up the core. Reactivity management is accomplished by motion of control rods or special transient rods, or by injection of liquid poison into the system.

An in-pile tube about 8 inches in diameter, designed to accommodate a variety of experiments, will occupy the center of the reactor. An out-of-pile circulation and cooling system is connected to the tube. The purpose of the experiments to be conducted is to test reactor fuels (primarily light water reactor oxide fuels) under simulated accident conditions. These include power-cooling mismatch, loss of coolant, and reactivity-initiated accident conditions.

While the PSAR specified either a pulsed mode or a power-time limit of 20 MW for one hour, the proposal in the FSAR is for a power-time limit of 40 MW for 48 hours. However, the contractor has stated that for the first year or two he expects that 40 MW runs will last no longer than one hour.

Because the Uniform Building Code seismic zone classification for the site was changed from II to III during construction, the confinement building is being strengthened by the addition of a 3/16-inch carbon steel liner to cover the ceiling and walls above the operating floor. Presence of this liner will also reduce leakage from the building and the calculated accident doses.

The Technical Specifications have not yet been issued by the Idaho Operations Office of the AEC. The Committee recommends that these specifications and a safety envelope for the proposed experimental program be reviewed by the Regulatory Staff.

The PBF has specified evacuation procedures, as do all sites at NRTS, and the contractor states that the control center personnel can be evacuated in less than five minutes.

The Advisory Committee on Reactor Safeguards believes that, subject to satisfactory completion of construction and preoperational testing and start-up programs and if due care is taken in the conduct of the experimental program, there is reasonable assurance that the Power Burst Facility can be operated as proposed without undue risk to the health and safety of the public.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

References attached

References

1. DRD&T memo dated August 4, 1971, Final Safety Analysis Report (FSAR) Parts I and II, July 1971
2. DRD&T memo dated August 16, 1971, replacement pages for FSAR
3. DRD&T memo dated January 3, 1972, Section XI of FSAR
4. DRD&T memo dated February 2, 1972, Section XV of FSAR
5. DRD&T memo dated April 24, 1972, answers to questions
6. DRD&T memo dated May 25, 1972, Additional Information
7. DRD&T memo dated August 2, 1972, Additional Information
8. DRD&T memo dated August 11, 1972, Additional Information
9. DRD&T memo dated September 12, 1972, Additional and Confirmatory Information
10. DRD&T memo dated November 7, 1972, Information on PBF Accident Analyses

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

March 12, 1968

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

Subject: REPORT ON PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Dear Dr. Seaborg:

At its ninety-fifth meeting, on March 7-9, 1968, the Advisory Committee on Reactor Safeguards completed a review of the application by the Northern States Power Company for authorization to construct nuclear generating plants Units 1 and 2 at its Prairie Island site, in Goodhue County, Minnesota. This project previously had been considered at Subcommittee meetings at the site on October 27, 1967 and in Washington, D. C. on March 1, 1968. During its review, the Committee had the benefit of discussions with representatives of the Northern States Power Company and their consultants, the Westinghouse Electric Corporation, and the AEC Regulatory Staff and their consultants. The Committee also had the benefit of the documents listed.

The Prairie Island site comprises approximately 560 acres located six miles northwest of the city of Red Wing, Minnesota, on the Mississippi River. Red Wing has a population of 10,500 while the Twin Cities metropolitan area, 28 miles northwest of the site, has a population of 1,700,000. The land surrounding the site is rural and agricultural.

The soils at the site consist of sandy alluvium, ranging in thickness from approximately 160 to 185 feet. Several hundred feet of sound sandstone bedrock underlie the site. The soil will be dewatered, excavated, classified and replaced to a depth of approximately 30 feet and compacted to a relative density of 85%.

The Prairie Island units are to be identical, two-loop, pressurized water reactors operated at maximum power levels of 1650 MWt. With respect to core design and other features of the nuclear steam supply system, the units are essentially duplicates of the Point Beach reactor. The units have a power level and average heat flux about 18% higher than the Point Beach reactor with a power density nearly that of the Diablo Canyon reactor.

Each reactor and its steam generators are enclosed in a structure which consists of a steel primary containment shell and a reinforced concrete vertical shield cylinder with a shallow dome. The applicant stated that the vacuum relief valves for the containment will be sized to accommodate simultaneous operation of the two spray subsystems and the four finned coolers.

The applicant has stated that protection will be afforded against the maximum probable flood.

The applicant has proposed using signals from 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 Prairie Island Units 1 and 2.

The Committee believes that, if due consideration is given to the foregoing items, and in view of the favorable site location, the nuclear units proposed for the Prairie Island 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 Attached.

References - Prairie Island

1. Volumes 1, 2, and 3 - Prairie Island Nuclear Generating Plant, Facility Description and Safety Analysis Report, received September 1, 1967.

2. Supplement No. 1 to Application for Licenses, Prairie Island Nuclear Generating Plant, Facility Description and Safety Analysis Report, received September 1, 1967.

3. Northern States Power Company letter, dated August 24, 1967, transmitting Amendment No. 1 to the Construction Permit and Operating License Application.

4. Northern States Power Company letter, dated December 15, 1967, transmitting Amendment No. 2 to the Construction Permit and Operating License Application.

5. Volume 4, Prairie Island Nuclear Generating Plant, Facility Description and Safety Analysis Report, Appendices J, K and L, received December 28, 1967.

6. Northern States Power Company letter, dated January 12, 1968, transmitting Amendment No. 3 to the Construction Permit and Operating License Application.

7. Northern States Power Company letter, dated February 15, 1968, transmitting Amendment No. 4 to the Construction Permit and Operating License Application.

8. Northern States Power Company letter, dated February 27, 1968, transmitting Amendment No. 5 to the Construction Permit and Operating License Application.

9. Northern States Power Company letter, dated March 6, 1968, transmitting Amendment No. 6 to the Construction Permit and Operating License Application.

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

April 18, 1973

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

Subject: REPORT ON PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Dear Dr. Ray:

At its 156th meeting, April 12-14, 1973, the Advisory Committee on Reactor Safeguards completed its review of the application of the Northern States Power Company for authorization to operate Units 1 and 2 of the Prairie Island Nuclear Generating Plant at power levels up to 1650 MW(t). This project had been considered previously at the Committee's Special Meeting, October 26-28, 1972, at its 151st meeting, November 9-11, 1972, and at Subcommittee meetings at the plant on October 24, 1972, and in Washington, D. C. on March 31, 1973. The applicant expects Unit 2 to be ready for operation approximately a year after Unit 1. During its review, the Committee had the benefit of discussions with the Northern States Power Company, the Westinghouse Electric Corporation, the Pioneer Service and Engineering Company, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee previously discussed this project in a construction permit report dated March 12, 1968.

The Prairie Island Nuclear Generating Plant is located on the west bank of the Mississippi River in Goodhue County, Minnesota, approximately 6 miles northwest of Red Wing, Minnesota, and 30 miles southeast of the Twin Cities metropolitan area.

Each containment consists of a free-standing steel vessel within a reinforced-concrete shield building. An annular space of about five feet separates the two structures. A common auxiliary building serves both units. A region of this building which contains the outer terminations of penetrations from the interior of the steel vessel is designated as a special ventilation zone. Both this zone and the annular spaces are provided with redundant fan-filter systems which can maintain negative pressures relative to the environment and filter any leakage from the steel vessel prior to release. The design and function of this containment system are similar to that of the Kewaunee Plant.

The Committee considered the problem of the unlikely rupture of a high energy piping line outside of containment. The applicant described an evaluation, carried out in accordance with criteria established by the Regulatory Staff. Design modifications including encapsulation sleeves, equipment protection, equipment relocation, equipment qualification and impingement barriers are under consideration. The Committee recommends that the final design and installation meet the requirements of the Regulatory Staff.

The Interim Acceptance Criteria for ECCS effectiveness are under review by the Regulatory Staff. When new criteria have been established, these may influence the allowable peak power or peak linear fuel rating. Also, defects in unpressurized fuel due to pellet densification have been observed in some plants. The Prairie Island fuel is prepressurized and there is reason to expect improved performance with this fuel; however, further information and experience are needed. The applicant will propose limits for linear power and procedures for surveillance of core power distribution and fuel condition. This matter should be resolved in a manner satisfactory to the Regulatory Staff prior to operation at appreciable power. The Committee wishes to be kept informed.

Studies indicate that a false signal from one unit coincident with an actual LOCA in the other unit, coupled with a loss of offsite power, would probably cause tripping of both diesel generators due to overload. The Committee recommends that, prior to operation of both units at power, either analyses should show that the likelihood of the above combination of events is sufficiently low, or changes should be made to avoid such a loss of onsite power. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

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.

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. In particular, 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 Emergency Core Cooling System.

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 Prairie Island Nuclear Generating Plant Units 1 and 2 can be operated at power levels up to 1650 MW(t) without undue risk to the health and safety of the public.

Mr. Hill did not participate in the review of this project.

Sincerely yours,

H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References Attached.

References

1) Final Safety Analysis Report (FSAR), Prairie Island Nuclear Generating Plant

2) Amendments 8-12, 14-20, 22-25, and 27-32 to FSAR

3) Nathan M. Newmark report, August 22, 1972

4) Safety Evaluation by Directorate of Licensing dated September 28, 1972

5) Anonymous letter, undated and unsigned, received October 25, 1972

6) Northern States Power Company (NSP) Letter, November 6, 1972, Response to AEC-DL Staff Oral Questions of November 3, 1972

7) Pioneer Service and Engineering Company letter, Draft Main Steam and Feedwater Line Rupture Study, November 22, 1972

8) Letter, K. Kniel, AEC, to A. V. Dienhart, NSP, January 3, 1973

9) Letter on Fuel Densification, E. C. Ward, NSP, to A. Giambusso, DL, January 8, 1973

10) Letter, R. C. DeYoung, AEC, to Northern States Power Company, January 11, 1973

11) Letter, A. Giambusso, AEC, to Northern States Power Company, February 9, 1973

12) Supplement No. 1 to Safety Evaluation by Directorate of Licensing, March 21, 1973

13) Letter, G. J. Merritt, Minnesota Pollution Control Agency to Executive Secretary, ACRS, March 27, 1973

14) Letter, K. Kniel, AEC, to Northern States Power Company, March 27, 1973

15) Letter, K. Kniel, AEC, to Northern States Power Company, March 30, 1973

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 SITES FOR A 200 MW THERMAL ORGANIC COOLED PROTOTYPE REACTOR

Dear Dr. Seaborg:

The Advisory Committee on Reactor Safeguards was requested by the Director of the Division of Licensing and Regulation in his letter of March 9, 1961, to review site data submitted by four utilities that have shown interest in an invitation to participate in a power demonstration program of the Division of Reactor Development. The project will utilize an organic cooled and moderated reactor of approximately 200 MW thermal which is to generate 50 MW electrical.

The sites are identified in proposals submitted by the following four utilities and referenced below:

1. Dairyland Power Cooperative, Inc., LaCrosse, Wisconsin
2. Plains Electric Generation and Transmission Cooperative, Inc., Albuquerque, New Mexico
3. Grand River Dam Authority, Vinita, Oklahoma (two sites)
4. Burlington Electric Light Department, Burlington, Vermont (two sites)

At its thirty-third meeting on April 6-8, 1961, the Committee reviewed the descriptive site data submitted by the utilities and had the benefit of information contributed by the AEC staff. The Committee concludes that each of the sites proposed is acceptable for the location of a reactor of this general type to operate at a power level not to exceed 200 MW thermal.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

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

References:

1. Invitation for expression of interest, dated December 14, 1960, with supplemental information, dated March 20, 1961.
2. Letter, Board of Electric Light Commissions, Burlington, Vermont, dated January 27, 1961, with appendices.
3. Letter, W. A. Stebbins (Burlington Electric Light Company) to USAEC, dated February 16, 1961.
4. Letter, J. P. Madgett (Dairyland Power Cooperative) to F. K. Pittman (USAEC), dated January 13, 1961, with exhibits.
5. Letter, J. P. Madgett (Dairyland Power Cooperative) to F. K. Pittman (USAEC), dated February 3, 1961, with enclosures.
6. Letter, J. P. Madgett (Dairyland Power Cooperative) to A. Giambusso (USAEC), dated February 14, 1961
7. Letter, J. P. Madgett (Dairyland Power Cooperative) to M. K. Ray, (USAEC), dated March 31, 1961.
8. Letter from C. J. Dugan (Grand River Dam Authority) to USAEC, dated January 27, 1961 with supplemental information.
9. Grand River Dam Industrial Area near Pryor, Oklahoma - Reactor Plant Map B.
10. Grand River Dam Authority - Reactor Plant Map A 2.
11. Telegram, B. T. Ownes (Grand River Dam Authority) to A. Giambusso (USAEC), dated February 16, 1961.
12. Letter, N. Davis (Plains Electric Generation and Transmission Cooperative, Inc.) to F. K. Pittman (USAEC), dated January 26, 1961.
13. Letter from N. Davis (Plains Electric Generation and Transmission Cooperative, Inc.) to USAEC, dated February 10, 1961.
14. Letter, N. Davis (Plains Electric Generation and Transmission Cooperative, Inc.) to USAEC, dated February 21, 1961 with exhibits.

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: PUERTO RICO WATER RESOURCES AUTHORITY - TORTUGUERO SITE

Dear Dr. Seaborg:

At its sixty-fifth meeting, August 5-7, 1965, and its eighty-eighth meeting, August 10-12, 1967, the Advisory Committee on Reactor Safeguards considered the Tortuguero site, at which the Puerto Rico Water Resources Authority (PRWRA) plans eventually to construct a nuclear power station consisting of up to four units. The site consists of approximately 500 acres and is located on the north coast of Puerto Rico, twenty-one miles west of San Juan. Each unit would include a light water moderated and cooled reactor of approximately 500 MWe capacity. Construction of the first unit may begin in time to provide power production by January 1, 1975. An ACRS Subcommittee visited the site and met with representatives of PRWRA on June 20, 1965. A Subcommittee meeting was also held in Washington, D. C., on August 9, 1967. During its review, the Committee had the benefit of discussions with representatives of PRWRA and their consultants, the AEC Regulatory Staff and their consultants, and of the documents listed below.

Studies of the geology, seismology, and meterology of the site have been made and have not revealed significant problems. At the site, Aymamon Limestone is found at a depth of approximately twenty to fifty feet below sea level. The applicant proposes that the facility foundation be supported on piles 70 to 120 feet long, driven into the Aymamon Formation. Further drillings will be made to determine the contour of the load-bearing formation and to confirm the absence of large voids.

Centers of seismic activity in the region of the Puerto Rico Trench as well as to the east and west of the island have been reported, but no evidence of faulting or displacement in the vicinity of the proposed site has been found. The applicant proposes to design the facility using 0.15 g and 0.25 g, respectively, for the design and the maximum potential earthquake-induced accelerations in the Aymamon Formation at the site. The applicant currently plans to design the facility to withstand a 20-foot tsunami. However, further studies of the topography of the north coast and the lagoon are required prior to resolution of the question of appropriate protection against tsunamis and hurricanes.

The present and projected populations in the vicinity of the proposed site are appreciable, and appropriate attention to the engineered safety features and the conservatism of design will be required. The Committee believes that special attention should be given to providing reliable emergency power sources.

The Advisory Committee on Reactor Safeguards believes that, subject to the above comments, the Tortuguero site is acceptable for four reactors of the general type and power level proposed.

Sincerely yours,

/s/
N. J. Palladino
Chairman

<u>References</u>:

1. Letter from Puerto Rico Water Resources Authority, dated April 19, 1965, with attached Request for Preliminary Analysis by the AEC for Licensing the Tortuguero, Palo Seco and South Coast Sites, dated April 21, 1965.
2. Letter from Puerto Rico Water Resources Authority, dated August 2, 1965, with attached Addendum No. 1 to Request for Preliminary Analysis by AEC for Licensing the Tortuguero, Palo Seco and South Coast Sites, dated July 30, 1965.
3. Letter from Puerto Rico Water Resources Authority, dated January 17, 1967, and Geological and Geophysical Investigations of the Proposed Tortuguero Nuclear Power Station for the Puerto Rico Water Resources Authority.
4. Letter from Puerto Rico Water Resources Authority, dated July 12, 1967, with attached Addendum No. 2, Tortuguero Nuclear Plant Site.

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