ACRS R-0889

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

July 16, 1980

The Honorable Victor Gilinsky
Commissioner
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Gilinsky:

Your letter of June 4, 1980 requested the Committee's views on the utility of in-core thermocouples as an aid in the detection of off-normal conditions and for use in accident analysis. Your request was in the context of instrumentation to follow the course of an accident.

As you noted, the ACRS has long been concerned with assuring that qualified instrumentation should be available to follow the course of a serious accident. This concern has extended to the use of data from core outlet thermocouples in PWRs. For example, the ACRS has commented on the desirability of thermocouple availability and use in its reviews of the Davis-Besse, Oconee, and Indian Point Unit 3 plants. More recently, the Committee commented on the use of thermocouples during a discussion with the Commission immediately following the accident at TMI (see the attached correspondence in which appropriate paragraphs are underlined).

The Committee believed then, as it does now, that core outlet thermocouples can provide a readily available check on certain types of core conditions and serve as an additional means for detection of core behavior anomalies. The Committee also believes that, as practical, such information should be made readily available to the operator over the full temperature range of the installed detectors. The Committee recommends that, as practical, faulty thermocouples should be replaced during refueling intervals. New PWRs should be designed for thermocouple replacement. Core subassembly exit thermocouples or other core outlet thermocouples have not generally been used in BWRs; however, the ACRS favors a careful examination of the feasibility of their use in BWRs and the pros and cons of such use.

The ACRS notes that the NRC Staff has developed proposed criteria for installation and use of core outlet thermocouples as part of the Lessons Learned item addressing inadequate core cooling. NRC Staff efforts in this area include the requirement specified in proposed Regulatory Guide 1.97 for core exit temperature measurements. The ACRS supports this Regulatory Guide requirement but suggests that the NRC Staff consider PWR and BWR design differences in its implementation.

The Committee believes that instruments displaying thermocouple readings should be readily available in plant control rooms, consistent with the philosophy underlying ACRS Generic Item 43: "Instrumentation to Follow the

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Course of an Accident". The ACRS notes that in the past the availability of core temperature readings has varied from plant to plant, and offers as an example the fact that had the accident at Three Mile Island Unit 2 occurred at Unit 1, it would have been necessary to enter the containment to obtain core temperature information.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Attachments:

1. Memo dtd April 18, 1979 transmitting, "Recommendations of the Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards Regarding the March 28, 1979 Accident at the Three Mile Island Nuclear Station Unit 2" - Presented Orally to, and Discussed with, the NRC Commissioners during the ACRS - Commissioners Meeting on April 17, 1979
2. Ltr dtd November 14, 1973 from H. G. Mangelsdorf, Chairman, Advisory Committee on Reactor Safeguards to D. L. Ray, Chairman, U.S. Atomic Energy Commission, Subject: "Interim Report on Indian Point Nuclear Generating Station Unit No. 3"
3. Ltr dtd September 23, 1970 from J. M. Hendrie, Chairman, Advisory Committee on Reactor Safeguards to G. T. Seaborg, Chairman, U.S. Atomic Energy Commission, Subject: "Report on Oconee Nuclear Station Unit No. 1"
4. Ltr dtd August 20, 1970 from J. M. Hendrie, Chairman, Advisory Committee on Reactor Safeguards to G. T. Seaborg, Chairman, U.S. Atomic Energy Commission, Subject: "Report on Davis-Besse Nuclear Power Station"

References:

1. U.S. Nuclear Regulatory Commission: Draft 2 of Revision 2 to Regulatory Guide 1.97: "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" June 4, 1980.
2. U.S. Nuclear Regulatory Commission: "TMI-1 Restart Evaluation of Licensee's Compliance With the Short and Long Term Items of Section II of NRC Order dtd August 9, 1979 - Metropolitan Edison Company, et al. Three Mile Island Nuclear Station, Unit 1 - Docket No. 50-289". U.S. NRC Report NUREG-0680, June 1980.
3. Letter dtd May 5, 1980 from Bill Lowery, Lawrence Livermore Laboratory, to Gary Holahan, USNRC, NRR, transmitting, "A Summary of Utility Responses to NUREG-0578, Section 2.1.3.b: 'Instrumentation for Detection of Inadequate Core Cooling'" University of California, Livermore, California, April 28, 1980.
4. Memorandum for V. Gilinsky from H. Denton, Subject: Use of In-Core Thermocouples, dtd November 1, 1979.

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

April 18, 1979

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

FROM: R. F. Fraley, Executive Director
Advisory Committee on Reactor Safeguards

Attached for your information and use is a copy of the recommendations of the Advisory Committee on Reactor Safeguards which were orally presented to and discussed with you on April 17, 1979 regarding the recent accident at the Three Mile Island Nuclear Station Unit 2.

R F Fraley
R. F. Fraley
Executive Director

Attachment:
Recommendations of the NRC Advisory Committee on Reactor Safeguards Re. the 3/28/79 Accident at The Three Mile Island Nuclear Station Unit 2

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ATTACHMENT 1

April 17, 1979

RECOMMENDATIONS OF THE NUCLEAR REGULATORY COMMISSION ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REGARDING THE MARCH 28, 1979 ACCIDENT AT THE THREE MILE ISLAND NUCLEAR STATION UNIT 2

Presented orally to, and discussed with, the NRC Commissioners during the ACRS-Commissioners Meeting on April 17, 1979 - Washington, D. C.

Natural circulation is an important mode of reactor cooling, both as a planned process and as a process that may be used under abnormal circumstances. The Committee believes that greater understanding of this mode of cooling is required and that detailed analyses should be developed by licensees or their suppliers. The analyses should be supported, as necessary, by experiment. Procedures should be developed for initiating natural circulation in a safe manner and for providing the operator with assurance that circulation has, in fact, been established. This may require installation of instrumentation to measure or indicate flow at low water velocity.

The use of natural circulation for decay heat removal following a loss of offsite power sources requires the maintenance of a suitable overpressure on the reactor coolant system. This overpressure may be assured by placing the pressurizer heaters on a qualified onsite power source with a suitable arrangement of heaters and power distribution to provide redundant capability. Presently operating PWR plants should be surveyed expeditiously to determine whether such arrangements can be provided to assure this aspect of natural circulation capability.

The plant operator should be adequately informed at all times concerning the conditions of reactor coolant system operation which might affect the capability to place the system in the natural circulation mode of operation or to sustain such a mode. Of particular importance is that information which might indicate that the reactor coolant system is approaching the saturation pressure corresponding to the core exit temperature. This impending loss of system overpressure will signal to the operator a possible loss of natural circulation capability. Such a warning may be derived from pressurizer pressure instruments and hot leg temperatures in conjunction with conventional steam tables. A suitable display of this information should be provided to the plant operator at all times. In addition, consideration should be given to the use of the flow exit temperatures from the fuel subassemblies, where available, as an additional indication of natural circulation.

The exit temperature of coolant from the core is currently measured by thermocouples in many PWRs to determine core performance. The Committee recommends that these temperature measurements, as currently available, be used to guide the operator concerning core status. The range of the information displayed and recorded should include the full capability of the thermocouples. It is also recommended that other existing instrumentation be examined for its possible use in assisting operating action during a transient.

The ACRS recommends that operating power reactors be given priority with regard to the definition and implementation of instrumentation which provides additional information to help diagnose and follow the course of a serious accident. This should include improved sampling procedures under accident conditions and techniques to help provide improved guidance to offsite authorities, should this be needed. The Committee recommends that a phased implementation approach be employed so that techniques can be adopted shortly after they are judged to be appropriate.

The ACRS recommends that a high priority be placed on the development and implementation of safety research on the behavior of light water reactors during anomalous transients. The NRC may find it appropriate to develop a capability to simulate a wide range of postulated transient and accident conditions in order to gain increased insight into measures which can be taken to improve reactor safety. The ACRS wishes to reiterate its previous recommendations that a high priority be given to research to improve reactor safety.

Consideration should be given to the desirability of additional equipment status monitoring on various engineered safeguards features and their supporting services to help assure their availability at all times.

The ACRS is continuing its review of the implications of this accident and hope to provide further advice as it is developed.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

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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-

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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 ssurance that Indian Point Nuclear Generating Station U it 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

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ADVISORY COMMITTEE ON REACTOR SAFEGUA:RDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

Septem r 23, 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Com ssion
Washington, D. C. 2054:

Subjec: REPORT ON OCO E 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 stud 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.

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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

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 DAVIS-BESSE NUCLEAR POWER STATION

Dear Dr. Seaborg:

At its 124th meeting, August 13-15, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by the Toledo Edison Company and The Cleveland Electric Illuminating Company for a permit to construct the Davis-Besse Nuclear Power Station. A Subcommittee met to review the project on May 26, 1970, at the site and in Toledo, Ohio, and on August 4, 1970, in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives and consultants of the applicants, the Babcock and Wilcox Company, the Bechtel Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The plant will be located on the southwestern shore of Lake Erie approximately 21 miles east of Toledo, Ohio. The nearest population centers are Toledo and Sandusky, Ohio, each about 20 miles from the site, with populations in 1960 of 379,000 and 32,000, respectively. The city of Fremont, Ohio, with a 1960 population of about 18,000, is located 17 miles from the site. The minimum exclusion distance is 2400 feet and the low population zone distance is two miles. Approximately 3200 people live within five miles of the site.

Camp Perry, an Ohio National Guard facility, is located on Lake Erie about five miles east of the site. This installation is used during a short period each year for target practice with small arms and with 40-mm. anti-aircraft shells armed only with a small destruct charge. At the Erie Industrial Park, about three to four miles east of the site, Cadillac Gage Company is engaged in testing ordnance equipment firing 120-mm. mortar shells with a maximum range of about two miles. All firing from both locations is directed into restricted areas in Lake Erie. The applicants have provided studies which demonstrate that none of the projectiles now being fired from these installations could

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penetrate the heavy reinforced concrete structures provided to protect the essential portions of the plant. The Committee recommends, however, that the applicants and the Regulatory Staff make suitable arrangements to be informed of any changes in these activities so that their possible effect on the safety of the plant may be evaluated.

An area in Lake Erie about ten miles north of the site is used by aircraft from the Selfridge Air Force Base in Michigan as an Anti-Submarine Warfare practice area and by the Lockbourne Air Force Base at Columbus, Ohio, as an impact area for automatic weapon firing from aircraft. The applicants have been given assurance by officials of the Department of Defense that military aircraft enroute to or from this area will not be routed closer than ten miles from the site. The Committee believes that this arrangement reduces, to acceptably low levels, the probability of an aircraft striking the plant, but recommends that formal arrangements be made to enable the applicants and the Regulatory Staff to maintain continuing awareness of the operational patterns of military aircraft in this area.

The Davis-Besse plant will include a two-loop pressurized water reactor similar to those for the Midland units except that the internal vent valves have been eliminated by changes in the elevations of the steam generators to obviate their need. Since the proposed arrangement eliminates the possibility of coolant flow bypass through an open vent valve, the Davis-Besse reactor is designed for an initial core power level of 2633 MWt as compared to 2452 MWt for the Midland units.

The applicants stated that it will be possible to anneal the pressure vessel if this should become necessary at some time after operation is begun.

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.

The containment consists of a steel vessel surrounded by a reinforced concrete shield building, with the annular space maintained at a slightly negative pressure and the air from this space exhausted through filters. This design is similar to that for the Prairie Island, Kewaunee, and Hutchinson Island plants, except that the free volume of the steel containment is much greater, nearly three million cubic feet. The Regulatory Staff should review the containment design pressure to assure that an adequate margin of conservatism exists.

Detailed criteria remain to be formulated by the applicants for the design of the penetrations for the hot process pipes which traverse the annulus between the two containment barriers. In view of the importance of these penetrations, criteria should be reviewed by the Regulatory Staff to assure adequate conservatism, and the applicants should arrange for an independent review of the actual design.

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 applicants are studying various methods of coping with this problem, including purging and the use of catalytic recombiners. 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 applicants have stated that they 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 acceptably conservative values. They will also make appropriate plant changes if further analysis demonstrates that such changes are required. This matter should be resolved during construction in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The Committee recommends that the applicants accelerate the study of means to prevent common failure modes from negating scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The applicants 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.

The Committee believes that consideration should be given to the utilization of instrumentation for prompt detection of gross failure of a fuel element. Consideration should be given also to the use of core exit thermocouples as an aid to reliable operation and as an additional method of detecting behavior anomalies.

The applicants propose batch discharge of liquid wastes following treatment. Concentrations of radionuclides in the discharge will be kept well below 10 CFR 20 limits with positive dilution being provided from several equipment cooling water streams. Plans for operation of waste treatment equipment should be such as to minimize the quantities of radioactivity discharged, and provisions should be made to achieve rapid dispersion in the lake.

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 Davis-Besse plant.

The Committee believes that the above items can be resolved during construction and that, if due consideration is given to these items, the Davis-Besse Nuclear Power Station can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Joseph M. Hendrie

Joseph M. Hendrie
Chairman

<u>References</u>:

1. Letter from Toledo Edison Company, dated August 1, 1969; License Application, Volumes 1, 2 and 3 of the Preliminary Safety Analysis Report (PSAR)

2. Volume 4 of the PSAR, dated April 16, 1970

3. Amendments 1 through 9 to License Application