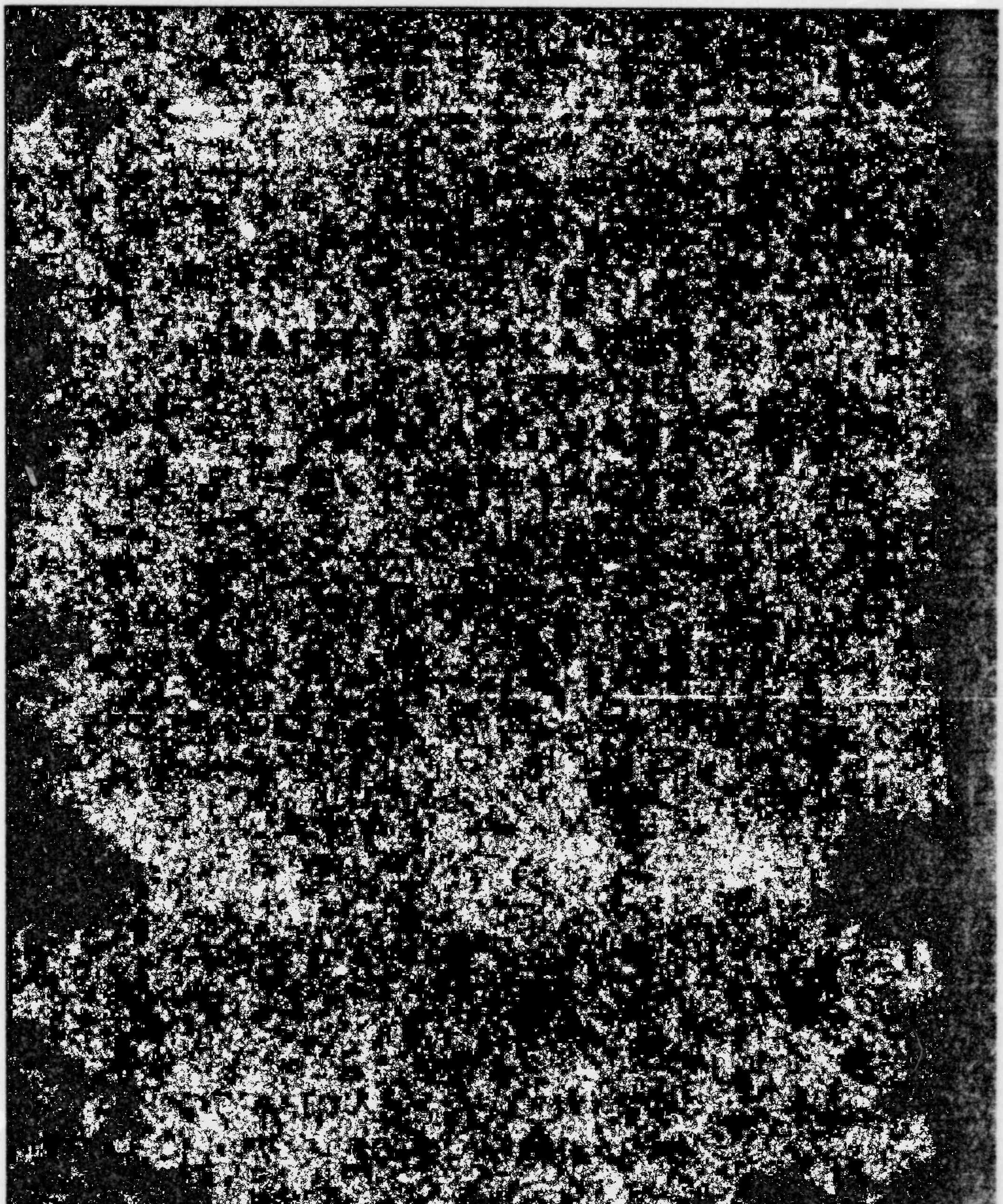

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SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

CONSUMERS POWER COMPANY

MIDLAND PLANT UNITS 1 AND 2

DOCKET NOS. 50-329 AND 50-330

U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

WASHINGTON, D. C. 20555

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1.0 INTRODUCTION

The Midland Plant Units 1 and 2 Safety Evaluation Report was issued on November 12, 1970. On January 14, 1972 a Supplemental Safety Evaluation Report was issued providing the staff's review and conclusions regarding the Midland Plant Units 1 and 2 emergency core cooling system, radioactive waste treatment system and Consumers Power Company financial qualifications.

Construction permits CPPR-81 and CPPR-82 authorizing construction of the Midland Plant Units 1 and 2 were issued to the Consumers Power Company (Licensee) on December 15, 1972.

On July 21, 1976, the United States Court of Appeals for the District of Columbia Circuit in Nelson Aeschliman, et al. v. U.S. Nuclear Regulatory Commission, 547 F2d 622, among other matters remanded for clarification the June 18, 1970 report issued by the Advisory Committee on Reactor Safeguards (ACRS) for the Midland Plant Units 1 and 2.

On August 16, 1976, the Commission reconvened an Atomic Safety and Licensing Board (Board) to consider whether the construction permits for the Midland Units 1 and 2 should be continued, modified or suspended in light of the issues remanded by the D.C. Circuit.

In a letter dated October 14, 1976, the licensing Board returned the original Advisory Committee on Reactor Safeguards (ACRS) report to the ACRS for clarification. In response to the Board's request, the ACRS issued a "Supplemental Report on Midland Plant Units 1 and 2" dated November 18, 1976. This ACRS report is attached as Appendix B. Therein the ACRS identified eleven items which were:

> "...those items referred to in its paragraph on 'other problems · lated 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." (Page 1 of the November 18, 1976 letter.)

The purpose of this Supplement No. 2 to the Safety Evaluation Report for the Midland Plant Units 1 and 2 is to provide an updated status and identify resolutions of the eleven identified ACRS items for the Midland Plant.

Appendix A to this supplement provides an update of major milestones that have occurred for this facility since the issuance of the Safety Evaluation Report on November 12, 1970.

2.0 DISCUSSION OF THE 11 ITEMS IN ADVISORY COMMITTEE ON REACTOR SAFEGUARDS LETTER REPORT DATED NOVEMBER 18, 1976

2.1 Separation of Protection and Control Instrumentation

"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)."

The physical independence design of the circuits and electric equipment comprising or associated with the Class IE power system, the protection system, systems actuated or controlled by the protection system and auxiliary or supporting systems must assure that operability of the protection system and the systems it actuates to perform their safety-related functions are not compromised by any failure in the control system or other nonprotection systems.

In order to show the resolution of the above Advisory Committee on Reactor Safeguards (ACRS) item, it is necessary to establish the chronology of events. The ACRS concern was first indicated in the Three Mile Island ACRS letter dated January 17, 1968, (See Appendix B to this report) in the third paragraph of the second page. On August 30, 1968, the Institute of Electrical and Electronics Engineers (IEEE-279) standard was issued and accepted for purposes of regulations applicable to nuclear power plants. This standard sets forth the criteria and requirements for separation of control and protection instrumentation used in our evaluation of the Midland Plant design. At the 122nd ACRS meeting held on June 11-13, 1970, the ACRS Committee completed its review of the application by the Consumers Power Company for a permit to construct the Midland Plant, Units 1 and 2. On June 18, 1970, the Report on the Midland Plant, Units 1 and 2, was issued by the ACRS wherein the concern regarding separation of protection and control instrumentation identified by the Three Mile Island ACRS letter was referenced.

In the Safety Evaluation Report by the Atomic Energy Commission (now the Nuclear Regulatory Commission) for the Midland Plant issued on November 12, 1970 in Section 8.1 "Instrumentation and Control" the staff indicates that the applicant (Consumers Power Company) will design the instrumentation for protection and control systems to conform to IEEE-279 dated August 1968, and we concluded this was acceptable.

Therefore the separation of protection and control instrumentation design for the Midland Plant was resolved in the Staff's Safety Evaluation Report issued November 12, 1970 by the applicant's commitment to meet the requirements of the IEEE-279 Standard dated August 1968. In Section 8.4, "Installation Criteria," of our Safety Evaluation Report dated November 12, 1970, we indicated the applicant will develop more detailed criteria and procedures for installation of protection and emergency power systems as recommended by the ACRS. In Amendment 32 to the Preliminary Safety Analysis Report, the applicant, to comply with this recommendation, submitted the detail criteria and procedures for installation of the protection and emergency power system for the balance-of-plant. These detailed criteria and procedures are currently being reviewed by the staff and will be completed prior to installation of these systems. The detail criteria and procedures for installation of the protection and emergency power systems for the nuclear steam supply system will be submitted for staff review and approval prior to installation of these systems.

Acceptability of the final implementation of these requirements will be determined during the operating license review stage for the Midland facility.

2.2 Vibration and Loose Parts Monitoring

"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)."

The loose part monitoring system is designed to provide early warning of reactor components which have failed or are vibrating to such an extent that failure may be imminent.

The staff Safety Evaluation Report on the Midland Plant, issued on November 12, 1970, and the Supplemental Safety Evaluation issued on January 14, 1972, make no mention of a requirement for a loose parts and vibration monitor. However, in a response to the staff question 4.5 in Amendment No. 5 to the PSAR, the licensee committed to install a loose parts monitor if a practical and reliable system were available. Such equipment as might ultimately be required can be in the nature of add-on equipment which could be added to a plant at any time. Loose part monitoring systems are currently available and we will resolve this matter during the operating license review stage for the Midland Plant Units 1 and 2.

Acceptability of the final implementation of this requirement will be determined during the operating license review for the Midland Plant, Units 1 and 2.

2.3 Potential for Axial Xenon Oscillations

"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)."

This subject is addressed in the staff's Safety Evaluation Report in the Midland Plant issued on November 12, 1970. The staff noted that analyses at that time indicated that the core would be stable to potential radial or azimuthal power oscillations due to xenon, and that potential axial oscillations could be controlled by use of part-length control rods.

Tests of core stability were performed during start-up tests for the Oconee Unit 1 reactor, a sister or similar type unit to the Midland Plant reactors. A diagonal (combination of axial and azimuthal) oscillation was induced at 75 percent full power and the reactor response was monitored for 72 hours. The azimuthal component of the oscillation was damped, but the axial component was divergent. At 70 hours into the transient, the part-length rods were used to suppress the axial imbalance which was reduced to near zero where it was maintained.

On the basis of this demonstration of azimuthal stability of the Oconee Unit 1 reactor (essentially identical to the Midland Plant reactors) and the ability of the control system to suppress axial oscillations, we conclude that this concern is resolved for the Midland Plant.

The use of solid poison shim rods in the fuel element provides a means to assure that the moderator temperature coefficient is only slightly positive or negative throughout the core life. Where some Technical Specifications allow a slightly positive coefficient, the accident and stability analyses take this into account. Burnable poison provisions have been designed into the Midland Plant fuel to reduce excessive positive coefficients to allowable values.

Acceptability of the final implementation of these requirements will be determined during the operating license review for the Midlant Plant.

2.4 The Behavior of Core-Barrel Check Valves in Normal Operation

"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)."

This matter is of generic concern to nuclear steam supply systems designed by Babcock and Wilcox. Other reactor vendors do not use core-barrel vent valves. The concern of the Committee was that there was a potential for the core-barrel check valves to open during normal operation allowing excessive core by-pass flow.

For the Oconee units, which are of Babcock and Wilcox design and sister or similar type units to the Midland design, the staff initially imposed a 4.6 percent reactor coolant flow penalty in the thermal-hydraulic design analysis to provide conservatism due to the possibility of leakage through the vent valves during normal operation. By letter to the licensee of the Oconee Nuclear Station (Duke Power Company) dated January 30, 1976, the staff advised the licensee that it had concluded that sufficient evidence had been provided by Babcock and Wilcox to assure that the core-barrel vent valves would remain closed during normal operation. Accordingly, we advised the licensee that the vent valve flow penalty could be eliminated provided the licensee established appropriate surveillance requirements to demonstrate, at each refueling outage, that the vent valves are not stuck open and that they operate freely.

The Oconee resolution of this matter is directly applicable to the Midland Plant design since the designs are identical and the matter is therefore satisfactorily resolved for the Midland Plant.

Acceptability of the final implementation of the core-barrel check valves requirements will be determined during the operating license review for the Midland Plant.

2.5 The Potential Consequences of Fuel Handling Accidents

"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)."

This concern is resolved by General Design Criterion 61 of Appendix A to 10 CFR Part 50 which requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions. Regulatory Guide 1.13 "Spent Fuel Storage Facility Design Basis" describes a method acceptable to the staff for implementing General Design Criterion 61.

By letter to the Licensee dated September 29, 1976, the staff noted that the initial design of the Midland Pland did not include charcoal filters in the exhaust system for the spent fuel storage facility. However, the staff also noted that during discussions with the Licensee, the Licensee had agreed to install charcoal filters in conformance with Regulatory Guide 1.13. On the basis of this commitment by the Licensee, the staff concluded that the design of the Midland Plant is in conformance with Regulatory Guide 1.13 and is acceptable.

Acceptability of the final implementation of the Regulatory Guide 1.13 requirements will be determined during the operating licensee review for the Midland Plant.

2.6 The Effects of Blowdown Forces on Core Internals

"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)."

In the Safety Evaluation Report on the Midland Plant dated November 12, 1970 the above ACRS item was resolved to the staff's satisfaction as discussed in Section 5.4, "Reactor Vessel Internals," and Section 15.10, "Blowdown Forces on Core Internals."

This matter is partially covered by Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing." By letter dated September 29, 1976, the staff informed the Licensee of our conclusion that the Midland design was in full conformance to Regulatory Guide 1.20.

There have been recent additional concerns raised about the loads on reactors internals during a loss-of-coolant accident. The staff now is working with all the reactor vendors on this matter. The vendors, including Babcock and Wilcox, are developing thermal- hydraulic codes that properly handle the loadings on the core internals during subcooled blowdown. We expect that versions of these codes acceptable to the staff will be available within about one year. To date, the preliminary indications are that the internals design of the Babcock and Wilcox reactors will withstand the blowdown loads and is acceptable. In the event analyses indicate that the internals design is not acceptable design modifications may be required. Resolution will be made during the staff's operating license review for the Midland Plant Units 1 and 2.

2.7 Assurance That Loss-of-Coolant-Accident-Related Fuel Rod Failures Will Not Interfere With Emergency Core Cooling System

"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)."

This matter is considered resolved on the basis of the generic rulemaking hearing on Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, RMS-50-1, which resulted in promulgation of regulations, specifically 10 CFR 50.46 and Appendix K to 10 CFR Part 50 to which all nuclear plants must comply. The Midland Plant will be required to conform to these requirements which will assure that fuel rod failure will not interfere with the emergency core cooling system function. Operating Plants of the Midland type are now meeting the requirements of Appendix K to 10 CFR Part 50 and 10 CFR 50.46. As indicated in our Supplemental Safety Evaluation Report dated January 14, 1972, the Midland Plant Unit 1 and 2 Emergency Core Cooling Systems met the Commission's Interim Policy Statement and Acceptance Criteria.

During the operating license review the staff will require that the Midland Plant Units 1 and 2 emergency core cooling system meet the requirements set forth in Appendix K to 10 CFR Part 50 and 10 CFR 50.46. The matter is therefore satisfactorily resolved for the Midland Plant.

2.8 The Effect On Pressure Vessel Integrity of Emergency Core Cooling System Induced Thermal Shock

"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)."

Regulatory Guide 1.2, "Thermal Shock to Reactor Pressure Vessels," covers current information on this subject. The ultimate position as to the significance of thermal shock requires input of fracture mechanics data on irradiated steels from the Heavy Section Steel Technology (HSST) program.

The Nuclear Regulatory Commission's confirmatory safety assessment of hot reactor pressure vessels subjected to thermal shock effects due to injection of cold emergency core cooling system water following a loss-of-coolant accident is continuing and the third thermal shock experiment has been conducted at the Oak Ridge National Laboratory. In this experiment, a hot 21-inch diameter steel cylinder with a deliberate flaw was quickly cooled with a water-alcohol mixture at -10°F, thereby producing a severe thermal shock. Preliminary studies of the flaw indicate that it grew uniformly in depth, as predicted.

These results, plus those from the first two tests, have provided important assurance of the validity of the thermal shock analysis methodology to predict crack initiation and extension in reactor presure vessels under LOCA-type conditions. The work done to date at ORNL, plus results from the Naval Research Lab on the beneficial effects of warm prestressing, demonstrate that flawed, irradiated reactor vessels subjected to thermal shock from LOCA-ECCS water, will not fail catastrophically.

Two tests on warm prestressing are scheduled for FY 77, to provide the final verification necessary for routine application of the methodology by the licensing staff. Additional testing of 39-inch cylinders may also be conducted provided liquid nitrogen can be used as the coolant.

In a letter to the Applicant dated September 24, 1976, the staff concluded that the Midland design conforms to Regulatory Guide 1.2. Pending results from the HSST program, which is designed to confirm the validity of the analytical design model for irradiated pressure vessels, conformance to Regulatory Guide 1.2 and design of vessels in accordance with the ASME code and subsequent adherence to guidelines for surveillance of radiation damage and nil-ductility transition temperature changes resulting there from are acceptable to the staff as proper assurance against pressure vessel failure.

Should the pressure vessel surveillance program for the Midland Plant Units 1 and 2 indicate that greater than anticipated irradiation damage is occurring to a Midland reactor pressure vessel, the Licensee will be required to anneal the vessel to restore the toughness properties to acceptable values.

2.9 Environmental Qualifications of Vital Equipment in Containment

"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)."

The concern regarding this matter is verification that systems and components located in the containment, and required to function during and following a loss-of-coolant accident, can withstand the temperature, pressure, humidity and radiation conditions which could occur in the containment. The qualification requirements of critical components are now covered by Regulatory Guides 1.40, 1.63, 1.73 and 1.89 and by IEEE Standards 382-1972, 383-1974, 317-1972, and 323-1974 which provide acceptable methods to meet these qualification requirements.

The staff review of the Midland Plant regarding qualification f vital equipment in containment is not complete. Completion most likely will not occur until the staff review of the operating license application for the plant. However, since this matter deals exclusively with components, rather than structures, continued construction of the plant would not preclude possible upgrading of components, if required, during the operating license review.

Due to the advanced state of construction and procurement on the Midland facility, complete compliance with the above guides and standards will not be required on all components and equipment, however, exceptions will be required to be justified during the operating license review.

Regulatory Guide 1.40 "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants" was reviewed by the staff and the Licensee. By letter dated September 29, 1976, we informed the Licensee that the staff had concluded that the Midland design was in full conformance to this Regulatory Guide.

Regulatory Guide 1.63 "Electrical Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants" endorses IEEE Standard 317-1972. By letter to the Licensee dated September 29, 1976, the staff informed the Licensee that additional information would be required regarding the ability of penetrations to withstand, without loss of mechanical integrity, the maximum possible fault current vs time conditions (position C.1 of the Guide).

Regulatory Guide 1.73 "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants" endorses IEEE Standard 382-1972. By letter to the Licensee dated September 29, 1976, the staff informed the Licensee that implementation of this guide is acceptable.

Regulatory Guide 1.89 "Qualification of Class 1E Equipment for Nuclear Power Plants" endorses IEEE Standard 323-1974. This guide was issued in November of 1974 and it notes that the staff may reevaluate the plant design on a case-by-case basis to assure that acceptable methods for qualification of Class 1E equipment have been specified in purchase orders executed after November 15, 1974. The degree of conformance of the Midland design to the guidelines of this Regulatory Guide has not yet been evaluated by the staff. Such evaluation will occur during the staff review of the operating license application.

IEEE Standard 383-1974 pertains to the type testing of cables, splices and connections for nuclear power plants. It is a sub-element of IEEE Standard 323-1974, which is endorsed by Regulatory Guide 1.89.

Acceptability of the final implementation of the guides and standards requirements for environmental qualification of vital equipment will be determined during the operating license review for the Midland Plant.

2.10 Instrumentation to Follow the Course of an Accident

"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)."

During and following a loss-of-coolant accident, hydrogen is generated by radiolysis and water-metal reactions, a system such as a hydrogen recombiner is required to assure the hydrogen concentration within the containment remains below the flammability limit.

General Design Criterion 41 requires that systems to control hydrogen, oxygen and other substances which may be released into the reactor containment be provided as necessary to control their concentrations following postulated accident to assure that containment integrity is maintained. Regulatory Guide 1.7 (Safety Guide 7) "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident", describes a method acceptable to the staff for implementing General Design Criterion 41. The issuance of the regulation 10 CFR Part 50 Appendix A which provides the General Design Criteria for light-water reactors occurred after the Midland ACRS letters were issued. These criteria were not issued as a Commission regulation at the time the Midland application was under review but were used to evaluate the Midland design.

In a letter to the licensee dated September 29, 1976, the staff noted that the Licensee has committed to comply with the design guidance and assumptions for analysis contained in Regulatory Guide 1.7 as supplemented by Standard Review Plan Section 6.2.5 and Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations for Containment Following a LOCA." The staff found this design approach to be acceptable, but noted that we will review the combustible gas control system design and supporting analyses in conjunction with the application for an operating license.

Acceptability of the final implementation of hydrogen control system requirements will be determined during the operating license review for the Midland Plant.

The matter of instrumentation to follow the course of an accident still is carried by the Committee in the "Resolution Pending" category of concerns. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," was distributed for comment in December 1975. Comments now have been received, the guide is being revised as deemed appropriate by the staff and by the Committee, and the present schedule calls for publication in 1977. The Licensee will be required to meet the requirements of the Regulatory Guide 1.97 or provide staff approved alternatives. Acceptability of the final implementation of the Regulatory Guide 1.97 requirements will be determined during the operating license review for the Midland Plant.

2.11 Improved Quality Assurance and In-Service Inspection of Primary System

"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)."

This concern is satisfactorily covered by Appendix B to 10 CFR Part 50 which specifies the requirements for a quality assurance program for design, construction and operation of a plant and Regulatory Guides 1.28, 1.30, 1.37, 1.38, 1.39, 1.58, 1.64, 1.74, 1.88 and 1.94 which describe procedures for implementing the requirements of Appendix B. The quality assurance program for the Midland Plant meets these requirements.

During a recent review by the staff to determine the extent of conformance of the Midland Plant Units 1 and 2 to these various Regulatory Guides, the Licensee elected to upgrade the quality assurance program to meet the requirements indicated above. In a letter to the Licensee dated September 24, 1976, the staff reported that it had reviewed the revised quality assurance program description submitted by the Licensee in March of 1976, which incorporates Consumers Power Company Topical Report CPC-1, Bechtel Topical Report BQ-TOP-1, Revision 1A dated May 1, 1975, and Babcock and

Wilcox Topical Report BAW-10096A, Revision 3, of Consumers Power Company, the Bechtel Corporation, and the Babcock and Wilcox Company. They replace the quality assurance program described in the Preliminary Safety Analysis Report for the Midland Plant. We therefore consider the quality assurance program for the Midland Plant to comply with Appendix B of 10 CFR Part 50 and is acceptable.

The Safety Evaluation Report for the Midland Plant, dated November 12, 1970, states on page 25 that in-service inspection will comply with the draft ASME Code for the In-Service Inspection of Nuclear Reactor Coolant Systems (N-45) which is equivalent to Section XI of the ASME Boiler and Pressure Vessel Code. By letter to the Licensee dated September 24, 1976, the staff concluded that the degree of conformance to Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessels Closure Studs," is acceptable. Recently the requirements of the ASME Code Section XI was incorporated into the regulation 10 CFR Part 50.55 and the Midland Plant will be required to meet the ASME code Section XI requirements or justify and request relief for any nonconformance.

The matter of in-service inspection, therefore, is adequately resolved for the Midland Plant and, the quality assurance program is acceptable as noted above.

Acceptability of the final implementation of the requirements of the ASME Code Section XI will be determined during the operating license review for the Midland Plant.

3.0 CONCLUSIONS

Based on our review of the eleven items referred to in the Advisory Committee on Reactor Safeguards letter dated June 18, 1970 we find that our conclusions stated in Section 19.0 of our Safety Evaluation Report dated November 12, 1970 are unchanged. As discussed above, the ACRS has clarified the problems identified in their Midland letter and this report provides the status and resolution by the staff for the Midland Plant for the eleven identified items.

4.0 REFERENCES

The following references were used to provide the basis for acceptance and resolution of the eleven matters in Section 2.0 of this report and are available to the public at the Public Document Room, 1717 H St. NW Washington DC.

(1) ACRS report of April 16, 1976, "Status of Generic Items Relating to Light-Water Reactors: Report No. 4".

(2) Appendix A to 10 CFR Part 50, General Design Criteria for Nuclear Power Plants

(3) Institute of Electrical and Electronic Engineers Standard (IEEE-279) Criteria for Protection Systems for Nuclear Power Generating Stations

(4) Safety Evaluation Report on Midland Plant Units No. 1 and 2, Docket Nos. 50-329 and 50-330 dated November 12, 1970

(5) Supplemental Safety Evaluation Report on Midland Plant Units No. 1 and 2, dated January 14, 1972

(6) Letter from Nuclear Regulatory Commission to Duke Power Company dated January 30, 1976, with attachment entitled "Report Evaluation: B&W Operating Experience of Reactor Internals Vent Valves".

(7) Regulatory Guide 1.13 - "Spent Fuel Storage Facility Design Basis".

(8) Regulatory Guide 1.20 - "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing"

(9) Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, RMS-50-1

(10) Appendix K to 10,CFR Part 50, Emergency Core Cooling Systems Evaluation Models

(11) Regulatory Guide 1.2 - Thermal Shock to Reactor Pressure Vessels

(12) Regulatory Guide 1.40 - Qualification Tests of Continuous-Duty Motor Installed Inside the Containment of Water-Cooled Nuclear Power Plants

(13) Regulatory Guide 1.63 - Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants

(14) Regulatory Guide 1.73 - Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

(15) Regulatory Guide 1.89 - Qualification of Class IE Equipment for Nuclear Power Plants

(16) Letters from Nuclear Regulatory Commission to Consumers Power Company dated September 29, 1976 entitled "Midland Plant Units 1 and 2 - Regulatory Guide Review".

(17) Regulatory Guide 1.7 - Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

(18) Standard Review Plan - NUREG 75/087

(19) Regulatory Guide 1.97 - "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions and Following an Accident" under review

(20) Regulatory Guide 1.28 - Quality Assurance Program Requirements (Design and Construction)

(21) Regulatory Guide 1.30 - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment

(22) Regulatory Guide 1.37 - Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

(23) Regulatory Guide 1.38 - Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants

(24) Regulatory Guide 1.39 - Housekeeping Requirements for Water-Cooled Nuclear Power Plants

(25) Regulatory Guide 1.58 - Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel

(26) Regulatory Guide 1.64 - Quality Assurance Requirements for Design Nuclear Power Plants

(27) Regulatory Guide 1.88 - Collection, Storage, and Maintenance of Nuclear Power Plants Quality Assurance Records

(28) Regulatory Guide 1.94 - Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

(29) Appendix B to 10 CFR Part 50 - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

APPENDIX A

CHRONOLOGY OF MILESTONES SINCE ISSUANCE OF THE SAFETY EVALUATION REPORT

Safety Evaluation Report issued	November 12, 1970
First prehearing conference held to outline hearing agenda.	November 17, 1970
In compliance with AEC environmental regulations, Consumers Power files responses to agency comments.	November 25, 1970
Construction permit hearing begins and adjourns after nine hour session.	December 1, 1970
AEC issues "final" regulations to implement NEPA.	December 4, 1970
Delay in resumption of hearing requested by attorney for Saginaw Intervenors.	January 7, 1970
Consumers Power submits additional environmental data to AEC.	January 19, 1971
Prehearing conference held.	January 21, 1971
Prehearing conference scheduled for January 30 cancelled at the request of attorney for Saginaw Intervenors.	January 26, 1971
AEC issues draft environmental statement on Midland plant for comment by federal, state and local agencies and the public.	February 5, 1971
Scheduled May 17 resumption of hearing cancelled at the request of attorney for Saginaw Intervenors.	May 3, 1971

U.S. Environmental Protection Agency issues evaluation of environmental effects of Midland plant, concluding the site is "suitable for the facility as planned."	May 20, 1971
Prehearing conference orders resumption of hearing after request for postponement by attorney for Saginaw Intervenors.	June 7, 1971
Construction permit hearing resumes.	June 21, 1971
Hearing adjourns for board members to attend conference on emergency core cooling systems.	June 26, 1971
AEC issues interim standards for performance of emergency core cooling systems.	June 29, 1971
Hearing resumes.	July 7, 1971
Hearing adjourns as previously scheduled. Calvert Cliffs-NEPA court decision announced. AEC ordered to revise its environmental regulations.	July 23, 1971
Saginaw Intervenors request manufacture of Midland plant components be prohibited.	August 5, 1971
Mapleton Intervenors request application or construction permit be dismissed.	August 13, 1971
AEC issues new environmental regulations to comply with Calvert Cliffs decision.	September 9, 1971
Consumers Power submits 1100 page supplemental environmental report in compliance with new AEC environmental regulations.	October 20, 1971
Computer analysis of Midland plant's emergency core cooling system is submitted to AEC.	November 1, 1971

Prehearing conference held to formulate guidelines on environmental issues.	November 23, 1971
AEC determines Midland emergency core cooling system meets interim criteria.	December 18, 1971
Draft detailed environmental statement issued by AEC. Concludes environmental benefits of Midland plant outweigh the costs.	December 18, 1971
Supplemental Safety Evaluation Report issued.	January 14, 1972
National hearing on emergency core cooling system regulations begins in Washington, D.C.	January 28, 1972
Final detailed environmental statement issued by AEC. Concludes environmental benefits to be derived from plant outweigh the adverse effects.	March 31, 1972
Prehearing conference to finalize hearing agenda held. Attorneys for Saginaw Intervenors and Mapleton Intervenors request for indefinite postponement of the hearing is denied.	April 28, 1972
Construction permit hearing resumes.	May 17, 1972
Construction permit hearing ends.	June 15, 1972
Atomic Safety and Licensing Board issues initial decision authorizing construction permits.	December 14, 1972
Construction permits issued by Atomic Energy Commission.	December 15, 1972
Exceptions to initial decision filed by Mapleton and Saginaw intervenors.	January 1973
Atomic Safety and Licensing Appeal Board issues preliminary order imposing additional quality assurance reporting requirements on construction.	March 26, 1973
Atomic Safety and Licensing Appeal Board affirms initial decision authorizing issuance of construction permits.	May 18, 1973

Atomic Energy Commission issues amendment to construction permits incorporating quality assurance reporting requirements.	May 23, 1973
Construction of plant resumes.	June 15, 1973
Mapleton Intervenors petition US Court of Appeals for review of AEC ASLB and ASLAB decisions approving issuance of construction permits.	July 15, 1973
Saginaw Intervenors petition US Court of Appeals for review of ASLAB decision approving issuance of construction permits.	August 6, 1973
Saginaw Intervenors ask Appeal Board to revoke construction permits.	August 21, 1973
Consumers Power and Dow authorized by US Court of Appeals to intervene Mapleton appeal.	August 23, 1973
Saginaw Intervenors' motion to revoke construction permits denied by ASLAB.	September 18, 1973
AEC inspection questions adequacy of procedure used to maintain cadwelding work on containment building foundation. Company voluntarily halts cadwelding operations until question can be resolved.	November 6-8, 1973
Saginaw intervenors file motion to reopen license hearing to consider energy conservation issues which they claim were not adequately investigated.	November 20, 1973
AEC Director of Regulation orders Consumers Power to show cause why construction should not be suspended pending a showing that the Company is in full compliance with AEC quality assurance regulations. Order also continues suspension of cadwelding work.	December 3, 1973
AEC inspection determines that revised cadweld measuring procedures are adequate and all cadwelds have been properly made.	December 6-7, 1973

AEC Director of Regulation modifies show cause order to permit resumption of cadwelding operations.	December 17, 1973
Saginaw Intervenors file "emergency petition" asking the AEC to void the action of the Director of Regulation which permitted resumption of cadwelding.	December 18, 1973
Saginaw Intervenors file petition with AEC to revoke Midland plant construction permits.	December 18, 1973
AEC denies emergency petition, thus permitting cadwelding operations to be resumed.	December 20, 1973
Consumers Power files response to show cause order detailing the adequacy of its quality assurance program. Company also filed motion to dismiss show cause order and requests a public hearing on its quality assurance program if motion is denied.	December 24, 1973
Saginaw Intervenors file request for public hearing in connection with the show cause order.	December 24, 1973
AEC issues order, 1) denying Saginaw Intervenors' December 18 petition to revoke the construction permits, 2) denying the Company's motion to dismiss show cause order, and 3) ordering a public hearing on the show cause order and appointing a hearing board.	January 21, 1974
Atomic Safety and Licensing Appeal Board for the Midland construction permit proceeding issues memorandum disqualifying themselves from any participation in the show cause hearing.	January 22, 1974
Saginaw Intervenors petition AEC to order a hearing for reassessment of the Midland cost-benefit analysis, citing the increased cost of the plant for support.	January 23, 1974
AEC denies Saginaw Intervnors November 20 motion to reopen construction permit hearing to consider energy conservation issues.	January 24, 1974
AEC denies Saginaw Intervenors petition for reassessment of the Midland cost-benefit analysis.	February 5, 1974

Prehearing conference on show cause proceeding held.	March 28, 1974
Second prehearing conference on show cause proceeding.	May 30, 1974
AEC denies Saginaw Intervenors request to pay for lawyers and expert witness fees.	July 10, 1974
Show cause hearing starts in Midland.	July 16, 1974
AEC concludes show cause hearings in Midland.	July 18, 1974
AEC reports "We find that Consumers Power Company is financially qualified to continue construction of Midland plant since it has reasonable assurance of obtaining the necessary construction funds."	September 13, 1974
Atomic Safety and Licensing Board issues findings from its show cause hearing. The report concludes: 1) Consumers Power Company is implementing its quality assurance program in compliance with AEC regulations, 2) There is reasonable assurance that such implementation will continue through the construction process, 3) Construction permits should not be suspended, modified or revoked.	September 25, 1974
First of two steam generators arrives at plant site.	October 31, 1974
In service dates of the two units delayed by one year each, to 1980 and 1981 respectively. Unit 2: Fuel load Nov. 1980; Commercial operation 3/81; Unit 1 Fuel load 11/81; Commercial operation 3/82.	November 14, 1974
AEC Safety and Licensing Board hears oral argument in Chicago on Saginaw Intervenors motion that show cause hearing record be reopened.	November 18, 1974
Small fire caused by hot slag from a welder's torch damaged some electrical cables in the reactor building and caused damage to a small area of liner plate. Damage estimated at about $10,000.	November 21, 1974
Oral argument in U.S. District Court of Appeals in Washington, D.C., on petition by Saginaw Intervenors and Mapleton Intervenors to reopen construction permit hearing. The two groups	November 27, 1974

allege the AEC did not adhere to provisions of the Atomic Energy Act and National Environmental Protection Act in granting the Midland construction permits.

Letter from applicant providing schedule for implementation of regulatory guides with regard to operating license review for the Midland plant.	July 3, 1975
Letter from Consumer Power Company concerning in depth review of Regulatory Guide Implementation on the Midland Project.	July 21, 1975
Summary of Meeting on Implementation of Quality Assurance Regulatory Guides.	August 4, 1975
Letters from applicant transmitting responses to regulatory Guides 1.10, 1.61, 1.15, 1.18,, 1.19, 1.27, 1.35, 1.55, 1.57, 1.59, 1.60, 1.90 and 1.92.	August 19, 1975
Amendment 30 to Preliminary Safety Analysis Report containing design imformation regarding pipe lines from ultimate heat sink to service water pump structure.	September 3, 1975
Letter from applicant transmitting response to Regulatory Guides 1.2, 1.14, 1.31, 1.34, 1.36, 1.43, 1.44, 1.50, 1.65, 1.66, 1.71.	September 9, 1975
Summary of Meeting on (Structural) Regulatory Guides	September 30, 1975
Letter to applicant from NRC indicating analytical procedures and criteria described in Amendment 30 are acceptable.	October 9, 1975
Letter from applicant addressing implementation of Regulatory Guides 1.20, 1.26, 1.29, 1.46, 1.67, 1.72 and 1.40 for the Midland Project.	October 10, 1975
Letter from applicant responding to Regulatory Guide 1.28, 1.30, 1.37, 1.38, 1.39, 1.58, 1.64, 1.74, 1.88 and 1.94.	October 15, 1975

Letter from applicant responding to their positions regarding Regulatory Guides 1.1, 1.4, 1.7, 1.13, 1.25, 1.42, 1.49, 1.52, 1.54 and 1.70 for the Midland Project.	November 11, 1975
Letter to applicant from NRC advising of a potential safety problem regarding design of pressure vessel support systems.	November 14, 1975
Additional Information Request from NRC to applicant on implementation of Regulatory Guides.	November 19, 1975
Summary of Meeting on Regulatory Guides (Electrical)	November 21, 1975
Summary of Meeting on Regulatory Guides (Quality Assurance)	November 25, 1975
Summary of Meeting on Regulatory Guide (Mechanical)	November 26, 1975
Letter from applicant providing information regarding design of pressure vessel support system.	December 11, 1975
Summary of Meeting on Regulatory Guides (Quality Assurance)	December 24, 1975
Summary of Meeting on Regulatory Guides	January 6, 1976
Amendment 31 to Preliminary Safety Analysis Report providing updated information relating to maximum flooding conditions for the Midland Project.	January 9, 1976
Summary of Meeting held on Format Content and Schedule For the Final Safety Analysis Report.	January 23, 1976
Additional Information requested by NRC regarding implementation of Regulatory Guides 1.26, 1.29 and 1.94 for the Midland Plant.	January 26, 1975
Letter from applicant providing additional information on implementation of Regulatory Guides concerning quality group, seismic classification and concrete placement.	February 5, 1976
Letter to applicant from NRC addressing Appendix I to 10 CFR Part 50 requirements.	February 23, 1976

Summary of Meeting to Discuss Criteria to be Used for Analysis of Breaks in High-Energy Lines.	July 6, 1976
Letter to applicant from NRC regarding acceptance of Regulatory Guides for Midland Plant.	September 24, 1976
Letter to applicant from NRC regarding acceptance of Regulatory Guides for Midland plant.	September 29, 1976
Letter to applicant from NRC providing guidance regarding information required to evaluate fire protection systems	September 30, 1976
Summary of Meeting on Regulatory Guide Positions	October 20, 1976
Start of Atomic Safety and Licensing Board hearing to decide if Midland Plant construction permits should be modified, suspended, or continued.	November 30, 1976

APPENDIX B

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).

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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
3. Report on Hutchinson Island Unit No. 1, dated 3/12/70
4. Report on Palisades Plant, dated 1/27/70
5. Report on Three Mile Island Nuclear Station Unit 1, dated 1/17/68
6. Report on Oconee Nuclear Station, Units 1, 2, and 3, dated 7/11/67