



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

August 6, 1976

ACRS Members

MIDLAND "CATCH-ALL" PARAGRAPH

On July 26, 1976, R. Muller provided you with a copy of a U. S. Court of Appeals ruling on Midland which stated, in part, "...further explication of the ACRS report was necessary..." in connection with the so-called "catch-all" paragraph:

"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 Midland Plant Units 1 and 2."

The Court concluded that the Midland Hearing Board should have returned the ACRS' Report to the Committee for elaboration of the reference to "other problems."

This memo is intended to provide background information in anticipation of the need for such an elaboration.

The Committee's report on the Midland CP application was distributed on June 18, 1970. It stated that the NSSS and ECCS proposed for Midland were essentially identical with those for the previously reviewed Oconee and Rancho Seco Plants. Attachment 1 is a discussion of major similarities and difference excerpts from the Midland PSAR. The Oconee and Rancho Seco reports were written in July 1967 and July 1968, respectively. The list of generic items referred to as "other problems" in the Committee's report on Midland should therefore, be those which were thought to apply to Oconee and Rancho Seco plus any that may have been identified as being generally applicable between July 1968 and June 1970.

The Rancho Seco generic items paragraph referred to a previous ACRS report on Crystal River:

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

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The Crystal River Report cited refers in turn to other, still earlier, ACRS reports on Oconee (7/11/67) and Three Mile Island Unit 1 (1/17/68). Neither of these latter included a "generic items" paragraph. They did include reference to matters of general applicability to water-cooled reactors and to reactors designed by B&W:

Three Mile Island (1/17/68)

1. Diversity of ECCS initiation
2. Improved scram reliability
3. Separation of protection and control instrumentation
4. Development of a failed fuel element monitor
5. R&D aimed at providing assurance that LOCA-related fuel failures will not interfere with ECCS function
6. Potential for axial xenon oscillations
7. The effects of blowdown forces on core internals
8. The effect on Pressure Vessel integrity of ECCS-induced thermal shock
9. The behavior of core-barrel check valves in normal operation.

Oconee (7/11/67)

1. The effect on Pressure Vessel integrity of ECCS-induced thermal shock
2. The effects of blowdown forces on core internals and other primary system components
3. Assurance that LOCA-related fuel rod failures will not interfere with ECCS function
4. Evaluation of the overall effect of the core-barrel check valves
5. Diversity of ECCS initiation
6. Improved QA and in-service inspection of the primary system
7. Fuel integrity during end-of-life transients
8. The potential for xenon oscillations
9. Improved containment liner weld inspection

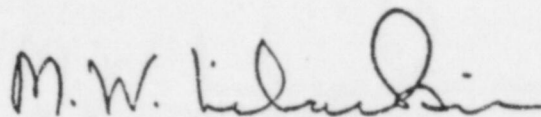
A copy of each of the ACRS reports cited above is attached for information (Attachments 2-6). Also attached (Attachments 7-15) are excerpts from several ACRS reports written during the period 7/68 to 6/70 which identify matters which might have been included in the Committee's reference in the Midland report to "Other problems... cited in previous ACRS reports." Likely candidates include:

1. Continuous monitoring of boron concentration (H. B. Robinson, 4/16/70).
2. The potential consequences of fuel handling accidents (Indian Point 3, 1/15/69; Hutchinson Island, 3/12/70; etc.)
3. Environmental qualification of vital equipment in containment (Palisades, 1/27/70).

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4. Instrumentation to follow the course of an accident (Hutchinson Island, 3/12/70; Beaver Valley, 3/12/70; Point Beach, 4/16/70; H. B. Robinson, 4/16/70; etc.)
5. Vibration and loose parts monitoring (Point Beach, 4/16/70).
6. Failure to scram during anticipated transients - ATWS (Beaver Valley, 3/12/70; Duane Arnold, 12/18/69; etc.)

There seems to be no other guidance available which would help in the requested clarification.

  
M. W. Libarkin, Assistant  
Executive Director for  
Project Review

Attachments:  
As Stated.

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1.3 TABULAR CHARACTERISTICS

Table 1-2 is a comparative list of important design and operating characteristics of the Midland Units 1 and 2, Rancho Seco Unit 1 (Sacramento Municipal Utility District), Oconee Units 1, 2, and 3 (Duke Power Company), and Turkey Point Units 3 and 4 (Florida Power and Light Company). The design and operating parameters of the Rancho Seco, Oconee, and Turkey Point units are close to those of Midland Units 1 and 2. Rancho Seco and Oconee units each have the same rated core power as the Midland units, and are near-duplicates in other respects. The data in Table 1-2 represent information presented in available station descriptions and in Safety Analysis Reports submitted for licensing.



The design of each of these stations is based on information developed from operation of commercial and prototype pressurized water reactors over a number of years. The Midland unit design is based on this existing power reactor technology and has not been extended beyond the boundaries of known information or operating experience.

The similarities and differences of the features of the reactor units listed in Table 1-2 are discussed in the following paragraphs. In each case, the item number used refers to the item numbers used in the table.

#### Item 1. Hydraulic and Thermal Design Parameters

The rated power of each Midland unit is the same as that for the Oconee and Rancho Seco units. The slight variation in other parameters between the Midland units and the other B&W units is due to the utilization of canless fuel assemblies in place of the canned fuel assemblies. The canless assembly allows a slightly larger fuel rod which increases fuel loading and heat transfer surface area. Elimination of the can wall results in a slightly lower power peaking factor and the more open lattice of the canless assembly increases coolant flow area. The reactor coolant flow rate, operating pressure, and operating coolant temperature are the same for the Midland, Oconee, and Rancho Seco units. The conservatism of design of the Midland units is evidenced by the DNBR of 1.71 (W-3) at the overpower condition compared to essentially the same value for the other B&W units and to a lower value for the other reactor presented.

#### Item 2. Core Mechanical Design Parameters

The table presents comparable mechanical design data for the canless fuel assembly for the Midland units, the canned fuel assembly for the Oconee and Rancho Seco units, and the canless fuel assembly used for the Turkey Point units. The dimensions, materials, and technology for each of these reactors are similar. Differences between the B&W units and the Turkey Point units are related to differences in power levels.

The small differences in fuel rod dimensions between the Midland units and the other B&W reactors result from the utilization of larger fuel rods and a larger fuel rod pitch to match fuel assembly pitch of the canless and canned fuel assemblies. The number of fuel rods per core is unchanged. The lesser number of control rod assemblies in the Midland units compared to the other B&W units results from a reduction in the requirements for inserted control rod assemblies for equilibrium xenon and transient xenon control. Burnable poison rod assemblies described for the Midland units result from the higher first cycle burnup shown in the Preliminary Nuclear Design Data, Item 3.

#### Item 3. Preliminary Nuclear Design Data

The core size, number of fuel assemblies, and number of fuel rods are the same for all of the B&W units and differ from the Turkey Point units primarily due to the difference in power level. Fuel enrichments differ between the B&W units primarily due to the different fuel cycle burnup requirements. Enrichment increase for the higher first cycle burnup of the Midland units is partially offset by the reduced amount of structural steel in the canless

assembly. The excess reactivity requirements for each reactor also vary with fuel cycle burnup; the higher burnup of Midland units is reflected in the higher initial excess reactivity. The Midland units have fewer control rod assemblies than do the Oconee and Rancho Seco units and more control rods than the Turkey Point units. The reduction in the number of control rod assemblies and control rod worth for the Midland units is due to less control rod insertion in the core during operation for compensation of equilibrium and transient xenon reactivity changes. The movable control rod worth for shutdown is not changed. The utilization of burnable poison as a part of the control balance allows for a reduction of the soluble poison concentration to obtain moderator coefficients within a desired range. The Doppler coefficient for all cases shown is negative over the core life.

#### Item 4. Principal Design Parameters of the Reactor Coolant System

Most of the features in this section are directly related to material properties and the amount of heat produced in the reactor core. Note that the B&W units are identical. The parameters are scaled in proportion to the power of the reactor. The major difference is the number of coolant loops required to remove the heat produced.

For the B&W units, only two loops are required since once-through steam generators are used instead of the U-tubes-in-shell design. The greater cooling capacity of these steam generators permits a reduction in the number of cooling loops for an equivalent amount of heat removed.

#### Item 5. Reactor Coolant System - Code Requirements

The B&W units are identical. Code requirements for the shell side of the steam generator conform to the ASME III Class A Specification. This is considered to be a contribution to the safety of the vessel. It enhances the integrity because of the more stringent ASME III Class A design, material, and quality-control requirements.

#### Item 6. Principal Design Parameters of the Reactor Vessel

The B&W units are identical. These vessel designs are characterized by a thinner thermal shield and a relatively larger shell diameter. The larger diameter provides for additional water between the edge of the core and the vessel which leads to additional neutron attenuation.

#### Item 7. Principal Design Features of the Steam Generators

The steam generators in the B&W units are the same. They are basically different from the Turkey Point units since they are a once-through design incorporating an integral superheat section.

#### Item 8. Principal Design Parameters of the Reactor Coolant Pumps

The B&W designs are the same. In each specific tabular parameter the relative number or size is in proportion to the total amount of heat removed from

the core. The B&W reactor pumps have higher head and horsepower requirements than the Turkey Point units have for approximately the same flow because of differences in system pressure drops.

Item 9. Principal Design Parameters of the Reactor Coolant Piping

The B&W designs are the same. They utilize carbon steel clad with stainless steel.

Item 10. Reactor Building Parameters

All reactor buildings are basically of the same design and construction. The differences are physical dimensions, amount of concrete shielding needed and design incident pressures, which are a direct result of plant layout, engineered safeguards, system capacities, and site location. The reactor building design and shielding offer satisfactory protection to the surrounding population in case of accident and during normal operation of the generating units.

Item 11. Engineered Safeguards

Engineered safeguards are generally similar.