

JUN 29 1970

Peter A. Morris, Director
Division of Reactor Licensing

OCONEE NUCLEAR STATION UNITS 1, 2, AND 3, DOCKET NOS. 50-269, 270
AND 287

The enclosed evaluation of the reactor pressure vessel, reactor
internal structures, reactor coolant systems, and Class I mechanical
equipment of the Oconee Nuclear Station was prepared by the DRS
Structural Engineering Branch.

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Distribution:
Suppl. Docket File:
50-269
50-270
50-237

DR RF
DRS RF
SEB RF

OFFICE ▶	DRS:SEB	DRS:SEB	DRS:AD/E	DRS:DIR		<i>memo</i>
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DATE ▶	6-26-70	6-26-70	6-26-70	6-26-70		

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OCONEE NUCLEAR STATION UNITS NO. 1, 2, AND 3
DOCKET NOS. 50-269, 270, AND 267

Reactor Coolant System

The reactor coolant system has been designed as a Class I (seismic) system to withstand the normal loads of mechanical, hydraulic, and thermal origin including anticipated plant transients and the operational basis earthquake within the stress limits of the appropriate codes given below.

The steam generator, pressurizer, and reactor coolant pump casings have been designed to Class A requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, including the Summer 1967 Addenda. Safety and relief valves are in accordance with the requirements of Article 9 of the above edition and addenda of Section III.

Piping which is part of the reactor coolant system has been designed to the ANSI B31.7 Code for Nuclear Power Piping, dated February 1968, including the June 1968 Errata.

Nondestructive examination requirements for reactor coolant system pumps and valves are given in Table 4-12 of the FSAR. These examinations include radiography of castings, ultrasonic testing of forgings, dye penetrant inspection of pump and valve body surfaces, and radiography of circumferential weldments. This program upgrades the nondestructive testing of pumps and valves within the reactor coolant pressure boundary to essentially that of the ASME Code for Pumps and Valves for Nuclear Power.

The applicant states that earthquake loads for the OBE and DBE have been determined by dynamic analyses. We are currently examining the analytical techniques employed in these analyses in conjunction with our consultants. We expect to have this matter resolved prior to the ACRS meeting for this plant.

Reactor Vessel

The reactor vessels have been designed and fabricated in accordance with Class A requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, including the Summer 1967 Addenda. Applicable Code Cases are 1332, 1335, and 1336.

The vessels are essentially identical to those intended for the Russelville, Crystal River 3 and 4, Rancho Seco 1, Midland 1 and 2, and Oyster Creek 2 plants, and have been designed to permit complete removal of the vessel internals. Fabrication materials are low alloy steel plates Type SA-533, Grade B, Class 1, and forging steel Type SA-508-64, Class 2. The vessel interior is clad with Type 304 austenitic stainless steel applied by weld overlay technique. The applicant has informed us that furnace sensitization of stainless steel vessel material has been limited to the non-pressure-bearing interior cladding. The requirements for nondestructive examinations have been limited to those required by Section III, except that head and shell plate material and flange forgings have been given a 100% volumetric examination using both longitudinal and shear wave UT techniques.

The Oconee reactors are the prototype vessels of the Babcock & Wilcox supplied 850 MWe Class of reactor vessels. However, no unusual design or fabrication problems either before or during manufacture have been identified. We conclude that the reactor vessels as designed and fabricated are acceptable.

Reactor Internals

For normal design loads of mechanical, hydraulic, and thermal origin, including the operational basis earthquake and anticipated transients, the reactor internals have been designed to operate within the allowable stress intensity limits of Section III of the ASME Boiler and Pressure Vessel Code.

All internals components are designated as Class I seismic items, and will be designed to withstand loads resulting from a combined design basis earthquake and loss-of-coolant accident. Strain limits for the internals under this combined load will be held to less than 20% of the uniform ultimate strain for this material (304S.S.) corresponding to an elastically calculated stress limit of not greater than 2/3 of the ultimate tensile strength. Allowable deflection limits will generally be within 50% of loss-of-function deformation limits. We consider these design limits to be acceptable.

Topical Report BAW-10008, Parts 1 and 2, is referenced in the FSAR as outlining the methods of analysis to be employed for the internals and fuel assemblies under loss-of-coolant and design basis earthquake loadings for skirt supported reactor vessels. We are presently, with the aid of our consultant, reviewing the analyses presented in the topical report; however, completion of this review awaits the submittal of the applicant's responses to our concerns. We expect to receive these responses early in July.

Other Class I (Seismic) Mechanical Equipment

Quality control standards for engineered safety features are summarized below:

All welding procedures and operators concerned with the fabrication of pumps and valves have been qualified to Section IX of the ASME Boiler and Pressure Vessel Code.

Hydrostatic tests of valve bodies and valve seats were conducted in accordance with ANSI B16.5 and MSS SP-61. Pumps have been hydrostatically tested to the requirements of UG-99 of Section VIII - Division 1 of the ASME Code.

The quality control standards for pumps and valves require inspection of raw material and review of material certification in conformance to ANSI B31.7 requirements. In addition, radiography and liquid penetrant tests of valve bodies, valve bonnets, and pump casings are performed to meet ANSI B31.7 acceptance standards.

These requirements result in a fabrication and inspection program which contain the essential elements of the ASME Code for Pumps and Valves for Nuclear Power. We find these requirements acceptable.

The codes and standards applicable to other Class I systems have been reviewed and are considered adequate from a safety standpoint.

All Class I equipment has been designed to withstand the design basis earthquake without loss of function. We are presently, with the aid of our consultant, reviewing the analytical procedures used to calculate the seismic loadings on Class I equipment.

In conjunction with this effort we are also reviewing the method of specifying the seismic design requirements for purchased Class I equipment, the adequacy of the applicant's check on the vendors' methods of certification, the design organizations involved in seismic design and their responsibilities, and documented procedures to provide for the interchange of design information between the involved organizations. We plan to report on these matters prior to the ACRS meeting for this plant.

Vibration Control

Flow induced vibration analyses have been made for reactor internals such as the thermal shield, fuel assemblies, fuel rods, surveillance tube and specimen holder assembly, control rod guide tube assembly, and piping for the in-core monitors. The thermal shield analysis for vibration problems showed that the flow induced pressure fluctuations acting on the

surface of the shield resulted in modal amplitudes less than 0.002 inch. These analyses considered inlet flow impingement and turbulent flow, as well as natural frequency calculations, to establish that a factor of at least two exists between conditions of possible resonance and excitation frequencies. It has also been determined that the flow induced pressure fluctuations acting on the disc of the vent valve are such that for normal operation there is always a positive net closing force acting on the disc.

We understand that the applicant will present a program to the staff which outlines his plans for confirmatory vibration monitoring during preoperational testing of the Oconee Plant. We expect to have this information prior to the ACRS meeting for this plant.

The feasibility of inservice monitoring for vibration and the detection of loose parts is being explored by the nuclear steam system supplier, B&W. They have investigated the application of such sensors as accelerometers, strain gages and load cells to monitor vibration of internals, and of inertially loaded-force pickups to monitor for loose parts. B&W plans additional discussion with consultants and instrumentation vendors in order to determine the feasibility and practicality of such systems in operating PWR systems.