50-269

SEP 8 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 information submitted in Amendments 13 through 17 to the subject application with respect to the reactor internal structures, reactor coolant system, and vibration control measures has been reviewed and evaluated by the DRS Structural Engineering Branch. An evaluation of this information is attached hereto which supersedes the previous report of June 29, 1970.

Original signed by
E. G. Case

Edson G. Case, Director
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Evaluation for Oconee
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OCONEE NUCLEAR STATION UNITS NO. 1, 2, and 3
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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 Febraury 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.

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 correspond 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 have, with the aid of our consultant, reviewed the methods of analyses presented in this report and find them acceptable.

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 actin on the disc. The vibration monitoring system proposed for the preoperational test program on the Oconee 1 plant will consist of four biaxial accelerometers to measure midspan vibratory motions of the surveillance specimen holder tubes and midplane vibratory response of the thermal shield. Each accelerometer will be capable of measuring frequencies over a range of 2-300 HZ and accelerations up to 30 g's. Upon completion of preoperational testing, the reactor internals will be removed and inspected for evidence of fretting, wear, and cracks.

The proposed instrumentation will not measure directly the response of several components of interest, e.g., the control assembly guide tubes, the upper and lower fuel support structures and the core barrel, nor will it provide for the direct measurement of the intensity and frequency of the hydraulic forces which will be driving the system.

The applicant states that the lack of these direct measurements is mitigated by the design of the B & W internals package which results in relatively low parallel flow (as opposed to cross flow) in the annulus between the thermal shield and core barrel and along the guide tubes. While it appears reasonable that these features of the B & W design will assist in decreasing the available driving forces, we are concerned that the proposed instrumentation may not provide the depth of information necessary to successfully correlate the performance of the internals structure with the design analysis or to assure a high degree of reliability in the extrapolation of the performance of the Oconee 1 internal structures to the expected performance of other similar plants of the same B & W product line.

In response to our concerns, the applicant is preparing a summary of the analytical approach taken and a discussion of the manner in which the test data will be processed and evaluated. We are continuing our review in this area and will discuss our findings in a supplemental report.

The feasibility of inservice monitoring for vibration and the detection of loose parts is being explored by the applicant. 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. Additional discussion with consultants and instrumentation vendors is planned in order to determine the feasibility and practicality of such systems in operating PWR systems. The applicant states that when instrumentation for inservice monitoring for vibration and detection of loose parts proves to be feasible, that they will make every effort to install and maintain these instruments in each of the Oconee reactors.

Coolant Pump Replacement

The reactor coolant pumps originally installed in the Oconee I plant have been replaced with Westinghouse Model 93A pumps similar in design to those provided by Westinghouse for other plants presently under construction.

The applicant states that all previous performance analyses are valid for the Oconee I reactor coolant system with the replacement pumps installed; since the replacement pumps offer greater resistance to back flow than the original pumps, the previous analyses for less than four pump operation are considered valid in that they give conservative results.

Pressure boundaries of the replacement pumps are designed and fabricated to Article 4 of Section III of the ASME Boiler and Pressure Vessel Code for Class A vessels. Code Addendum through Summer 1967 will apply. This standard is identical to that employed for the original pumps.

Modification to the Oconee I reactor coolant piping necessary to accommodate the replacement pumps consisted of the addition of a 28 inch ID to 30 inch ID stainless steel transition piece at each pump inlet and a small angle 28 inch ID carbon steel (clad) elbow at each pump outlet.

The stress analysis of the Oconee I reactor coolant system is presently being reviewed by the applicant to confirm compliance with the USAS B31.7 Code for Nuclear Power Piping when the new configuration is considered. All previous design conditions will apply.

The applicant further states that the design of the pumps and connecting piping is such that no furnace sensitized stainless steel will be present in the pressure boundary materials.

Based on the information presented in amendment 17 and our previous reviews of pumps of the Westinghouse line, and compliance with the design and fabrication codes previously accepted for the pump casings and piping, we find this modification acceptable provided the above indicated stress analysis review is satisfactory.