POOR ORIGINAL

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CRYSTAL RIVER NUCLEAR GENERATING PLANT

REACTOR INTERNALS

The reactor internals and core will be designed to meet normal mechanical, hydraulic, and thermal design loads plus operational basis earthquake loads (OBE).* Stress limit criteria used will be as established in Section III of the ASME Boiler and Pressure Vessel Code with the exception of fuel rod cladding which is not covered by the code. Seismic stresses will be combined in the most conservative way and will be considered as primary stresses.

The reactor internals will be designed to withstand also the concurrent blowdown and design basis earthquake (DBE)** loads, as indicated in Supplement No. 1, Section 9.11. Primary tensile stresses under such load combination will not exceed 2/3 of the stresses corresponding to the uniform strain value at operating temperature. Since this criterion results in allowable stresses and strains lower than those corresponding to the 20% of uniform strain criterion recently accepted (Diable Canyon plant), we consider the stress criterion proposed to provide an adequate margin of safety. The proposed deformation limits of reactor internals are shown on pages 9.11-10 through 9.11-12 of Supplement No. 1. The allowable deformation limits are about 50% of the estimated no-loss-of-function deformations for the specific components, and we believe these limits provide an adequate margin of safety. The applicant has stated (page 9.11-2) that the final calculated pressure differential time histories within the reactor vessel, as well as the final response and inertial loads, will not be available until later. Preliminary estimates of these response and inertial loads due to LOCA and earthquake are expected to be available before mid-1968 and the applicant has agreed to submit these to DRL as soon as practical.

Where simultaneous occurrence of LOCA and the DBE is considered, the applicant intends that both excitations will be input to the system simultaneously. Relative starting times will be changed until maximum structural motions, indicating maximum stresses, are obtained. The dynamic loading resulting from the pressure oscillations caused by a loss-of-coolant accident will not prevent scramming of the control rods.

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* OBE - The smaller earthquake. The plant will be designed to continue normal operation during an operational basis earthquake.

** DBE - The larger earthquake. The Class I (seismic) items will be designed to retain their functional capability during a DBE.

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We have concluded that the reactor internals design criteria are acceptable as proposed.

REACTOR COOLANT SYSTEM

Section III of the ASME Pressure Vessel Code will be used to design the reactor vessel, pressurizer, pump casings, and the steam generators. The vessel and its internals will be constructed so as to permit removal of the internals during plant life.

The reactor coolant piping will be fabricated and analyzed in accordance with the requirements of USA S.I. B31.1 Code for Pressure Piping. The applicant stated that total stresses resulting from thermal expansion, pressure, mechanical and seismic loadings will be considered in the design of the reactor coolant piping. The pressurizer surge line connection and the high pressure injection connections will be equipped with thermal sleeves to limit stresses from thermal shock.

The reactor coolant system, and all other Class I (seismic) mechanical systems, will be designed to withstand normal design loads of mechanical, hydraulic and thermal origin plus operational basis earthquake (OBE) loads within normal code allowable stresses. In addition, as stated in Supplement 1, Class I systems and components will be designed to withstand the concurrent blowdown and design basis earthquake (DBE) loads. Primary membrane stress intensities under such load combinations will not exceed 2/3 of the stresses corresponding to the uniform strain value at operating temperature. Since this criterion results in the allowable strains and stresses lower than those corresponding to the 20% of uniform strain criterion recently accepted (Diablo Canyon plant), we consider the stress criterion proposed to provide an adequate margin of safety.

THERMAL SHOCK ON REACTOR VESSEL

We are continuing our general review of the thermal shock effects on reactor components, induced by operation of the emergency core cooling system (ECCS). Our review includes the four major water reactor designers, i.e., Babcock & Wilcox, Combustion Engineering, General Electric, and Westinghouse.

Two modes of a potential failure are being considered: ductile yielding and brittle fracture. The latter is being analyzed by some applicants using both the Pellini-Puzak fracture diagram approach and fracture mechanics. Our preliminary evaluation indicates that the ductile yielding analysis is generally realistic, and that ductile yielding is not likely to cause a vessel failure.

On the basis of our preliminary evaluation of the brittle fracture analyses submitted by several applicants, we conclude that the transition

temperature approach is not quite adequate to treat crack propagation problems in plates with section thicknesses greater than 2 in. The detailed reasons for this conclusion are well summarized in the report ORNL-NSIC-21, "Technology of Steel Pressure Vessels for Water-Cooled Nuclear Reactors," Section 7.6.4. Specifically, while the transition temperature diagram provides a good and conservative guide for a fracture-safe design using low-strength steels of about 1 in. section thickness, it is based on only gross estimates of stress level and does not consider factors such as loading speed, crack geometry or the plane strain condition in the thick vessel plates. For this reason we generally require that a fracture mechanics analysis be performed to show that a crack would not propagate through the entire thickness of the vessel, when subjected to the thermal shock stresses.

The information submitted by the applicant in connection with Crystal River application is included in Supplement 1, page 3.11-1. It refers to the information submitted in connection with the Three Mile Island application (Docket No. 50-289) according to which a crack could not propagate any further than 35 per cent of the wall thickness.

In addition to this information, the applicant stated in Supplement No. 1 to the Crystal River application, that the fracture mechanics analysis currently under way will be based on the method outlined by Prof. P. C. Paris. This method consists of a plane strain two-dimensional approach utilizing superposition of pure tension and wedge force, with correction factors for the finite thickness of the plate. The critical stress intensity factor to be used in the analysis will be 30 Ksi-in.$^{1/2}$. We find this value to be conservative.

The fact that the outer layer of the vessel shell would be subjected to a compressive stress is an additional factor supporting a position that a crack should arrest without propagating through the vessel. In general, however, the information submitted so far is insufficient to conclusively show that the vessel will maintain its integrity when subjected to the thermal shock stresses and we remain seriously concerned about the potential for vessel failure through crack propagation.

The thermal shock problem is not unique to the Crystal River plant, and we intend to continue the general review until the integrity of the reactor vessel is proven by either analytical or experimental data.