Docket No. 50-305

MAY 1 1973

R. C. DeYoung, Assistant Director for Pressurized Water Reactors, L

SUPPLEMENTAL SAFETY EVALUATION FOR KEWAUNEE NUCLEAR POWER PLANT, DOCKET NO. 50-305

Plant Name: Kewaunee Nuclear Power Plant
Licensing Stage: OL
Responsible Branch & Project Manager: PWR #2: L. Crocker
Requested Completion Date: N/S
Applicant's Response Date: N/S
Description of Response: N/A
Review Status: Incomplete

Enclosed is our evaluation of the containment pressure and containment subcompartment pressure analysis for the subject plant. Note that subcompartment pressures were calculated higher than design. The Structural Engineering Branch should confirm the acceptability of the compartments to withstand these pressures.

We still have not received the containment pressure response resulting from the main steam line breaks within the containment. This information was to have been provided us by April 11, 1973. We have no schedule.

Robert L. Tedesco, Assistant Director
for Containment Safety
Directorate of Licensing

Enclosure:
As stated

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Docket file 50-305
L Reading
L:CS Reading
L:CSB Reading

Docket File

OFFICE ▶	L:CSB	L:CSB	L:CS			Memo
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DATE ▶	4-30-73	4-30-73	5/1/73			

SUPPLEMENT TO SAFETY EVALUATION

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

Primary Containment

The primary containment structure for the Kewaunee Nuclear Power Plant is a free-standing cylindrical steel vessel with a net free volume of approximately 1,320,000 ft^3. The structure houses the reactor and the primary system including the pressurizer and steam generators, as well as certain components of the engineered safety features provided for the facility. The primary containment structure is designed for a maximum internal pressure of 46 psig and a temperature of 268°F.

The report of the ACRS in its review of the Kewaunee plant (August 17, 1972) recommended that the staff confirm the adequacy of the applicant's analysis of peak overall accident pressures during postulated loss-of-coolant accidents. The reason for the ACRS concern is the potential for increased energy addition to the containment following a LOCA as a result of carryover water transferring heat from the steam generators to the containment.

As a result of the expressed concern of the ACRS, the staff has requested and the applicant has furnished (Appendix 14C and the response to Q 5.86) additional information to confirm the adequacy of the containment and the compartment walls within the containment.

We have evaluated the containment system in comparison to the Commission's General Design Criteria stated in Appendix A to 10 CFR Part 50 of the Commission's Regulations and, in particular, to Criteria 16 and 50. As a result of our evaluation, we have concluded that the calculated pressure and temperature conditions resulting from any design basis loss-of-coolant accident will not exceed the design conditions of the containment structure. The highest calculated containment pressure and temperature are about 40 psig and 263°F, respectively, which are calculated for the loss-of-coolant accident resulting from a postulated 3.0 square foot rupture in the suction leg of the reactor coolant system. We also have evaluated the capability of the compartment walls within containment to withstand the dynamic forces associated with blowdown and have concluded that the walls are adequately designed.

The applicant has described in the Safety Analysis Report for the Kewaunee Nuclear Power Plant, as amended by Amendments 23, 27, and 28), the results and methods used to analyze the containment pressure response for a number of design basis loss-of-coolant accidents. Break locations and sizes were varied to determine that the 3.0 square foot pipe rupture at the pump suction of the reactor coolant system results in the highest containment pressure. As discussed below, we have reviewed these analyses, and verified by our own analyses that the methods used by the applicant were conservative.

The applicant has analyzed the containment pressure response from postulated loss-of-coolant accidents in the following manner. Mass and energy release rates were calculated using the SATAN V, LOCTA, and REFLOOD Computer Codes. These mass and energy addition rates were then used as inputs to COCO which is the applicant's computer program used to calculate the containment pressure response. The SATAN V Computer Code was used by the applicant to determine the mass and energy addition rates to the containment during the blowdown phase of the accident, i.e., the phase of the accident during which most of the energy contained in the reactor coolant system including the coolant or water, metal and core stored energy is released to the containment. The applicant has, however, increased the energy release rate to the containment during the blowdown phase by extending the time that the core would remain in nucleate boiling, i.e., the time when the energy release rate from the core is highest. This energy was determined by using the LOCTA computer program. By using this method, the core would transfer more heat to the containment for containment analyses rather than storing the energy in the core for subsequent core heat-up analyses suitable for emergency core cooling studies. In our opinion this additional energy release from the core will increase the containment pressure and therefore assure that the calculation is conservative. Both the SATAN V Computer Code and the LOCTA Computer Code have been approved by the AEC for calculating energy release during a LOCA.

The mass and energy release rates to the containment were calculated by the applicant during the reflood phase of the accident using the computer code REFLOOD, i.e., following blowdown. The analysis of the reflood phase of the accident is important with regard to pipe ruptures of the reactor coolant system cold legs because the amount of the steam and liquid entrainment carried out from the core for these break locations passes through the steam generator which becomes an additional energy source. The water leaving the core and passing through the steam generator will be evaporated and/or superheated to the temperature of the steam generator secondary fluid. Results of the FLECHT experiments indicate that the carryout fraction of fluid leaving the core during reflood is about 80% of the incoming flow to the core. The rate of energy release to the containment during this phase becomes proportional to the flow rate into the core. The rupture of the cold leg at the pump suction results in the highest mass flow through the core and thus through the steam generators and therefore contributes to calculation of the highest containment pressure. To determine the mass and energy release to the containment during the reflood phase of the accident, we have compared the results using our FLOOD Code with those predicted by the applicant's using the REFLOOD computer program. The results of this comparison indicate equivalent prediction of energy release. Therefore, we have accepted the REFLOOD program as a realistic method of computing core flow for this plant.

We have analyzed the containment pressure response for a 3.0 square foot rupture in the suction leg of the reactor coolant system using the CONTEMPT-LT Computer Code which includes the energy addition to the containment from the steam generators and have calculated a peak containment pressure of about 40 psig as compared to the applicant's calculation of 40.5 psig using the COCO Computer Code.

We conclude that the maximum containment pressure is correctly calculated to be below the design pressure and that there is sufficient margin between the maximum containment pressure and the design pressure of the containment structure to assure the integrity of the leak tight containment barrier in the unlikely event of a design basis loss-of-coolant accident.

Subcompartment Pressure Analysis

As a part of our ongoing review of containment pressure calculations for pressurized water reactors, we requested additional information regarding the applicant's final design calculations for the subcompartment pressure analysis. In addition, we have performed confirmatory calculations of the peak compartment pressures following a design basis LOCA.

The results of the applicant's calculations and of our calculations regarding the pressure difference in psi between the compartments and the free space in the containment following a LOCA are in reasonably good agreement and are listed below:

		Reactor Compartment	
	Steam Generator Compartment	Reactor Nozzle Cavity	Reactor Vessel Cavity
Design Pressure, psi	25	475	100
Peak Calculated pressure, psi			
(1) Applicant	27.5	350	85.3
(2) Staff	31.4	485	115