SAFETY EVALUATION
(CONTAINMENT SYSTEMS)
DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1
DOCKET NO. 50-346

6.2 Containment Systems

6.2.1 Containment Functional Design

The containment system for the Davis-Besse Nuclear Power Station, Unit 1 includes an ASME Code, Section III, Class B, free-standing steel containment vessel surrounded by a reinforced concrete shield building, containment heat removal system, containment isolation system, combustible gas control system, and shield building ventilation system.

The steel containment vessel has a net free volume of 2,834,000 cubic feet. The containment vessel houses the nuclear steam supply system, including the reactor, steam generators, reactor coolant pumps and pressurizer, as well as certain components of the plant's engineered safety feature systems. The containment vessel is designed for an internal pressure of 40 psig and a temperature of 264°F.

The applicant has described in the Safety Analysis Report the methods used to analyze the containment pressure response to postulated loss-of-coolant accidents and reported the results. Various break locations and sizes were evaluated to determine that a 14.1 ft^2 hot leg split results in the highest containment pressure. Minimum containment cooling, assumed in the analysis, included one spray train and one air cooler.

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The applicant has analyzed the containment pressure response to postulated loss-of-coolant accidents in the following manner. The Babcock and Wilcox CRAFT computer code was used to calculate mass and energy releases to the containment during the blowdown, core reflood, and post-reflood phases of the accident. The mass and energy addition rates calculated in this manner were then used as input to the COPATTA computer code to calculate the containment pressure response.

As described above, the CRAFT code was used to calculate blowdown mass and energy releases. The blowdown phase of the accident is the phase during which most of the energy contained in the reactor coolant system, including the stored energy in the water, metal and core, is released to the containment. To obtain a conservatively high energy release rate, the applicant assumed nucleate boiling in the core until the quality of the coolant was approximately 1.0, and full ECCS operation.

The CRAFT program was also used by the applicant to predict mass and energy releases to the containment during the core reflood phase of the accident. The reflood phase is important when analyzing postulated pipe ruptures in the reactor coolant system cold legs since the steam and entrained liquid carried out of the core for these break locations can pass through the steam generators and be superheated to the temperature of the steam generator secondary fluid. During core reflood the carryout rate fraction, which determines the amount of steam and entrained water leaving the core and therefore the amount of energy that can be transferred from the steam generators, is calculated based on a correlation inherent in CRAFT. CRAFT calculates average carryout rate fractions in excess of 0.8. 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, which confirms the CRAFT approach. The rate of energy release to the containment during this phase is proportional to the flow rate into the core, and thus through the steam generators.

After the core is completely covered with water, decay heat generation will produce boiling in the core and a 2-phase mixture of steam and water will exist. This mixture can enter the steam generators and superheated steam will be generated. The applicant's analytical model accounts for this additional energy. About 500 seconds after a large break accident essentially all of the available sensible heat is removed from the primary system and the steam generators.

The CRAFT computer program has been accepted by the NRC for calculating mass and energy releases to the containment during the blowdown phase of the postulated accident. However, in applying the CRAFT code to the reflood and post-reflood phases of postulated loss-of-coolant accidents; i.e., following blowdown, the applicant included the quenching action of the ECCS fluid on the exiting steam. We will require that this effect be neglected. The applicant has committed to providing a reanalysis of the design basis loss-of-coolant accident assuming no quenching. The results of this analysis will be reported in a supplement to the Safety Evaluation Report.

We have performed a confirmatory containment analysis for a postulated cold leg (pump suction) break based on the mass and energy release data for a similar plant which neglected the quenching effect. Using the CONTEMPT computer code (References 1 and 2), we calculated a peak containment pressure of 35.8 psig. The Davis-Besse 1 containment vessel is designed for a maximum pressure of 40 psig. Although we do not expect the peak calculated pressure to change significantly using revised mass and energy release data, we will defer our conclusions on this plant based on the applicant's containment analysis until additional information is received.

The applicant has also analyzed the containment pressure response to a postulated main steam line failure. The applicant calculated a peak containment vessel pressure of about 22 psig for this accident.

The applicant has not completed the pressure response analysis of the containment vessel interior compartments, such as the reactor vessel cavity, the steam generator compartments and the primary shield pipe penetration annulus. The applicant has committed to providing the results of the analysis and the applicable mass and energy release data. We will report on this in a supplement to the Safety Evaluation Report.

We have evaluated the containment system functional design in accordance with the General Design Criteria stated in 10 CFR Part 50 of the Commission's Regulations and, in particular, Criteria 16 and 50. However, before we can conclude that the containment vessel and interior compartment design pressures are adequate, we will need revised mass and energy release data for the containment vessel analysis which does not include the quenching action of the ECCS water on the exiting steam following blowdown, revised mass and energy release data for the containment vessel interior compartment analysis that adequately describes the blowdown for each postulated pipe break over the time scale of interest, and the results of the containment vessel and interior compartment analyses based on the revised mass and energy release data. We will report our conclusions on the acceptability of analyses and adequacy of design pressures in a supplement to the Safety Evaluation Report.

6.2.2 Containment Heat Removal Systems

The containment spray system and the containment air cooling system are provided to reduce the containment vessel pressure following postulated high energy line break accidents. The [illegible] air cooling system is also used during normal plant operation, whereas the containment spray system has no normal operating function.

The containment spray system consists of two separate spray trains of equal capacity. All active components of the system are located outside the containment vessel to facilitate maintenance operations. Missile protection is provided by direct shielding or physical separation of equipment. The system is seismic Category I. The containment spray pump recirculation intakes from the containment emergency sump are enclosed by a screen assembly to prevent the entry of debris which could clog the spray nozzles.

A high containment pressure signal from the safety features actuation system will automatically actuate the containment spray system. The system pumps and valves can also be manually operated from the control room. The spray pumps initially take suction from the borated water storage tank. When the water in the tank reaches a low level, a switchover from injection to recirculation is manually initiated.

The applicant has provided an analysis which demonstrates that sufficient net positive suction head will be available to the spray pumps for both the injection and recirculation modes of operation. The analysis performed is consistent with the guidelines of Regulatory Guide 1.1.

The containment air cooling system consists of three equal capacity air cooler units. The system components and equipment required to remain operable following an accident are located outside the secondary concrete shield for missile protection at an elevation that precludes flooding, are designed to withstand the differential pressures resulting from a loss-of-coolant accident, and are seismic Category I.

A high containment pressure signal or a low reactor coolant system pressure signal from the safety features actuation system will automatically actuate the containment air cooling system. The system can also be manually operated from the control room.

Based on our review of the containment heat removal systems, we conclude that the system designs are consistent with the requirements of General Design Criteria 38, 39, and 40, and are therefore acceptable.

6.2.3 Secondary Containment Functional Design

The secondary containment (shield building) is a reinforced concrete structure surrounding the steel containment vessel. Potential leakage from the containment vessel to the shield building and adjoining penetration rooms is collected and processed by the emergency ventilation system, which is a seismic Category I system. The emergency ventilation system consists of redundant trains, each capable of the functional requirement. Following an accident the emergency ventilation system will maintain the areas it serves at a negative pressure to assure the collection of leakage from the containment vessel.

The applicant has attempted to identify potential leak paths from the containment vessel which bypass the volumes treated by the emergency ventilation system. The bypass leak paths identified by the applicant and the total allowable leakage from these bypass leak paths have been included in the plant technical specifications. However, all potential bypass leak paths have not been identified. The applicant has committed to provide additional information regarding this matter. We will conclude on the acceptability of identified potential bypass leak paths in a supplement to the Safety Evaluation Report.

The applicant has analyzed the pressure response of the shield building following a postulated loss-of-coolant accident. Based on our review, we conclude that the applicant has underestimated the time required to depressurize the shield building and reach a negative pressure of 0.25 in. w.g. The applicant calculates that about 20 seconds is required to establish a negative pressure after the emergency ventilation system becomes operational which is about 45 seconds after the accident; our calculations indicate that this will not occur until about 80 seconds after the emergency ventilation system becomes operational assuming only one train is operable. As part of the preoperational and periodic in-service inspection and test programs the applicant will confirm the operability of the system components and equipment, and the functional capability of the system to maintain a negative pressure within prescribed limits. We will also require the applicant to verify the time required to depressurize the shield building and establish a negative pressure.

6.2.4 Containment Isolation System

The containment isolation system is designed to automatically isolate piping systems that penetrate the containment to prevent outleakage of the containment atmosphere following postulated accidents. Double barrier protection, in the form of closed systems and isolation valves, are provided to assure that no single active failure will result in the loss of containment integrity. The containment isolation provisions, including the isolation valving and penetration piping, are seismic Category I.

Containment isolation will automatically occur upon receipt of containment high pressure signals or reactor coolant system low pressure signals from the safety features actuation system. High radiation signals are also used to isolate the containment vessel purge system lines.

Based on our review, we conclude that the containment isolation system design conforms to General Design Criteria 54, 55, 56 and 57, and the guidelines of Regulatory Guide 1.11, and is acceptable.

6.2.5 Combustible Gas Control System

Following a loss-of-coolant accident, hydrogen may accumulate inside the containment as a result of (1) a chemical reaction between the fuel rod cladding and the steam resulting from vaporization of emergency core cooling water, (2) corrosion of construction materials by the spray solution, and (3) radiolytic decomposition of the cooling water in the reactor core and the containment sump.

The combustible gas control system is designed to control the concentration of hydrogen within the containment vessel following a loss-of-coolant accident. The system consists of the containment hydrogen dilution system, hydrogen purge system, recirculation system, and gas analyzer system.

The containment hydrogen dilution system controls the hydrogen concentration within the containment vessel by the addition of air. The system is seismic Category I and consists of redundant trains. The system blowers have a 100 SCFM capacity. The maximum pressure that the system blowers are capable of repressurizing the containment vessel to is 18 psig.

The hydrogen purge system serves as a backup to the hydrogen dilution system, and consists of a single train. It releases the containment vessel atmosphere through HEPA and charcoal filters to the station vent. The system is seismic Category I.

The containment recirculation system is designed to draw air from the containment vessel dome and discharge it toward the containment air coolers, to provide a more uniform dispersion of hydrogen. The system is seismic Category I and consists of redundant trains. The gas analyzer system is designed to monitor the hydrogen concentration within the containment vessel following a loss-of-coolant accident. The system is seismic Category I and consists of redundant trains. Samples can be drawn from four points in the containment vessel.

The applicant has performed an analysis of the post-loss-of-coolant accident hydrogen generation in the containment vessel following a loss-of-coolant accident that is consistent with the guidelines of Regulatory Guide 1.7. The applicant calculated that the hydrogen concentration in the containment will not reach the lower flammability limit of four volume percent until about 44 days after the accident, and that the control limit of three volume percent will occur about 24 days after the accident. The hydrogen concentration in the containment will be maintained below three volume percent by actuating one of the trains of the hydrogen dilution system when the control limit is reached. We have performed similar calculations for the hydrogen generation in the containment following a loss-of-coolant accident and our results have confirmed those of the applicant.

Based on our review of the systems provided for combustible gas control following a postulated loss-of-coolant accident, we conclude that the systems conform to the guidelines of Regulatory Guide 1.7 and the requirements of General Design Criteria 41, 42 and 43, and are therefore acceptable.

6.2.6 Containment Leakage Testing Program

The containment design includes provisions and features which satisfy the testing requirements of Appendix J to 10 CFR Part 50. The design of the containment penetrations and isolation valves will permit periodic leakage rate testing at the pressure specified in Appendix J. Included will be those penetrations that have resilient seals and expansion bellows, such as personnel airlocks, equipment hatch, refueling tube blind flange, hot process line penetrations and electrical penetrations.

The proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR Part 50. Such compliance provides adequate assurance that containment integrity can be verified throughout the service lifetime of the plant and that the leakage rates will be periodically checked on a timely basis to assure that they are within specified limits. Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment vessel, the loss-of-containment atmosphere through potential leak paths will not be in excess of acceptable limits specified for the site.

We have concluded that the containment leakage testing program complies with the requirements of Appendix J to 10 CFR Part 50, and that such compliance constitutes an acceptable basis for satisfying the requirements of General Design Criteria 52, 53, and 54.

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