Dockets

FEB 19 1974

Docket Nos. 50-424-427

Voss A. Moore, Assistant Director for Light Water Reactors, Group 2, L

DRAFT SAFETY EVALUATION REPORT - A. N. VOGTLE NUCLEAR PLANT, UNITS 1, 2, 3, AND 4

Plant Name: A. N. Vogtle Nuclear Plant, Units 1, 2, 3, and 4
Docket Nos.: 50-424/425/426/427
Licensing Stage: CP
NSSS Supplier: Westinghouse
Architect Engineer: Bechtel
Containment Type: Dual
Responsible Branch & Project Manager: LWR #2-2; L. Crocker
Requested Completion Date: February 15, 1974
Applicant's Response Date: N/A
Review Status: Incomplete

Enclosed is the draft Safety Evaluation for the Vogtle Nuclear Plant Units 1, 2, 3, and 4, which was prepared by the Containment Systems Branch. This report is based on our review of the Preliminary Safety Analysis Report including submittals up to Amendment 14. Amendment 15 was received on February 4, 1974, and some additional information was verbally received from the applicant on February 13, 1974. As indicated in the enclosure, we are not able to conclude on the adequacy of the design pressures and temperature for the containment or the design pressures for the containment internal structures. The bases for such action are summarized below:

1. <u>Containment Design Pressure and Temperature</u>

   The applicant has recently submitted information regarding the mass and energy flow rate into the containment with consideration of the post-reflood phase of the LOCA. This information was submitted by the applicant in Amendment 15 on February 4, 1974, and supplemented by verbal communication on February 13, 1974. The verbal communication is to be documented by means of another amendment. This recently submitted information requires review of the applicant's method of calculating the post-reflood phase of the LOCA and containment analyses by CSB with this information. In particular, the newly submitted information has resulted in a redefinition of the DBA from the previously assumed case of full safety injection to the case of minimum safety injection, which now results in the maximum containment pressure, according to the applicant.

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2. Subcompartment Design Pressures

Analyses for the subcompartment design pressures were submitted in Amendment 15 on February 4, 1974. These analyses now require review and confirmatory analyses by CSB.

We propose to complete our review of these outstanding items and submit an amendment to the SER by March 15, 1974. Further communication with the applicant may be required to complete our review.

Robert L. Tedesco, Assistant Director
for Containment Safety
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6.2 Containment Systems

6.2.1 Containment Functional Design

The containment system for each of the Vogtle Nuclear Plant Units 1, 2, 3, and 4 includes a reactor containment structure, containment heat removal systems, containment isolation systems, a combustible gas control system and filtration systems for the enclosure building and the mechanical and electrical penetration rooms.

The containment (reactor building) is a steel-lined, prestressed concrete structure with net free volume of 2,750,000 cubic feet. The containment structure 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 is designed for an internal pressure of 52 psig and a temperature of 279°F.

The enclosure building, which encloses the reactor building above grade, and the mechanical and electrical penetration rooms, which are adjacent to the reactor building, are limited leakage structures. These volumes incorporate systems designed to provide for the collection and controlled release to the environment of fission product leakage from the containment following a postulated accident.

The applicant has described the methods used to analyze the containment pressure response for a spectrum of design basis loss-of-coolant accidents, and the results of these analyses, in the Preliminary Safety Analysis Rrport. Various break locations and sizes were evaluated to determine that the double-ended pipe rupture at the pump suction of the reactor coolant system results in the highest containment pressure.

The applicant has analyzed the containment pressure response from postulated loss-of-coolant accidents in the following manner. Mass and energy release rates to the containment were calculated and then used as inputs to the COCO computer program, which is used by the applicant 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 primary coolant, metal, and core stored energy, is released to the containment. To obtain a conservatively high energy release rate to the containment during the blowdown phase, the applicant assumed that the core would remain in nucleate boiling for an extended period of time, so that the energy release rate from the core would be maximized. Under this assumption, the core transfers more heat to the containment than would be predicted by a calculation suitable for core heatup and an emergency core cooling performance evaluation. This additional energy release from the core increases the calculated containment pressure and therefore assures a margin of conservatism in the analysis. The SATAN V computer code has been accepted by the AEC for calculating energy released during a LOCA.

During the core reflood phase of the accident, when the core is again filled with water, mass and energy release rates were calculated by the applicant using a hydraulic model and an energy balance model. The hydraulic model determines the core flooding rate and the entrainment fraction. The energy balance model calculates the core exit conditions and the energy addition from the steam generator. The analysis of the reflood phase of the accident is important with regard to pipe ruptures of the reactor coolant system cold legs since the steam and entrained liquid carried out of the core for these break locations pass through the steam generators which constitute an additional energy source. The steam and entrained water leaving the core and passing through the steam generators 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 and continues until the fuel is recovered with water to about the 8-foot elevation, at which time the fuel clad temperature transient ceases (quenching occurs). The applicant has conservatively assumed quenching of the core at the 10-foot elevation for the containment pressure calculations.

The rate of energy release to the containment during the reflood phase is 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. We have compared the mass and energy release to the containment during the reflood phase of the accident, as calculated with our FLOOD computer code, with those values predicted by the applicant. The results of this comparison indicate equivalent predictions of energy release. Therefore, we have accepted the applicant's computer models as a realistic method of computing core reflood for this plant.

The applicant has included consideration of a possible additional energy release to the containment during the post-reflood phase of the large break accident. This postulated additional energy would result from the presence of a two-phase mixture in the steam generator tubes. The analysis performed presents an upper bound on additional energy release since the two-phase mixture is assumed to remain in the tubes until all of the available secondary side energy has been removed to the containment. The analytical procedures used by the applicant are currently under review by the staff. The adequacy of the containment design pressure and temperature cannot be determined until this review is complete.

In our evaluation, we analyze the containment pressure response for the postulated double-ended, cold-leg, pump suction break using a) the CONTEMPT computer code, b) the mass and energy release to the containment provided by the applicant, including the additional energy from the steam generator during the reflood and post-reflood phases of the accident, as described above, c) the containment heat sink and heat removal system, and d) conservative condensing heat transfer coefficients to the structures inside containment.

The applicant also has analyzed the containment pressure response due to a postulated failure of a main steam line within containment. The maximum calculated containment pressure is 35.3 psig which is below the design value for the containment.

The applicant's analyses for the pressure response within the containment interior compartments, such as the reactor vessel cavity, pressurizer compartment, and steam generator cavities are currently under review and we cannot provide an evaluation for the adequacy of the design at this time.

6.2.2 Containment Heat Removal Systems

The containment heat removal system includes two redundant containment spray trains and four containment fan-cooling units.

The containment spray system serves only as an engineered safety feature and performs no normal operating function. It is a seismic Category I system consistingof redundant piping, valves, pumps and spray headers. All active components of the system are located outside the reactor building. Missile protection is provided by direct shielding or physical separation of equipment. The containment spray pump intakes are covered by a screen assembly designed to prevent debris that could clog the spray nozzles from entering. A high-high reactor building pressure on two of four sensors will cause the engineered safety features actuation system to automatically place the containment sprays in operation. The spray pumps and valves also can be operated manually from the control room. The spray pumps initially will take suction from the refueling water storate tank (RWST). When the water in the RWST reaches a low-low level the spray pump suction is manually transferred to the containment sump tc initiate the spray recirculation phase. The applicant's analysis indicates that sufficient water will have been delivered to the containment at that time to provide the required NPSH to the spray pumps.

The containment fan-cooling system consists of four fan-cooler units arranged in two sets of two. Each fan-cooler is sized for one-fourth capacity heat removal under accident conditions. Cooling water to the units is supplied from the nuclear service cooling water system. During normal operation, two of the four fan-coolers operating at high speed are required to provide sufficient cooling. Upon receipt of a safety injection actuation signal, the idle fan-cooling units automatically are started on the low speed setting.

Simultaneously, the running units are switched from high speed to low speed operation. The containment fan-cooling system is a seismic Category I system. The fan-cooling units are located outside the secondary concrete shield for missile protection, and are accessible for periodic testing and inspection during normal plant operation.

We have reviewed the containment heat removal systems for conformance with General Design Criteria 38, 39 and 40, and we find them to be acceptable.

6.2.3 Containment Air Purification and Cleanup Systems

The containment air purification and cleanup systems consist of (1) the normal containment preaccess filtration system, (2) the normal containment purge system, (3) the post LOCA purge system, and (4) the containment spray additive system. Discussed also in this section are the enclosure building filtration system and the mechanical and electrical penetration room filtration systems.

The preaccess filtration system consists of two fan-filter trains, each capable of processing containment atmosphere at 30,000 cfm through a filter bank consisting of a pre-filter, HEPA filter, and a charcoal filter to reduce airborne activity so as to permit safe and continuous access to the containment. This system is not required for post-accident operation.

The normal containment purge system supplies cleaned, conditioned, outside air to the containment where it is circulated and then exhausted through prefilters, HEPA filters and charcoal filters. It is designed for use during normal plant operation and serves no post-accident function.

The post-accident purge system supplies outside air to the containment following a postulated LOCA and exhausts contaminated air from the containment through a demister, pre-filter, electric heater, HEPA filter, charcoal filter and a final HEPA filter. The system components are seismic Category I, are designed to conform to Regulatory Guide 1.52, and are powered from an emergency power bus. The system is redundant to the hydrogen recombiners and is designed to allow the post-accident hydrogen concentration within containment to be maintained below flammable limits.

The enclosure building filtration and vent system (EBFVS) is provided to limit the release to the environment of radioisotopes that may leak from the containment following a postulated accident. The system consists of two, full capacity, redundant, fan-filter and vent subsystems, either of which is capable of reducing the pressure inside the enclosure building to a negative 0.25 inch water gauge and providing multipass filtration of the air inside the enclosure building while discharging to the environment, through filters, sufficient air to maintain the negative pressure differential. The system is designed to seismic Category I criteria and each subsystem is energized from a separate emergency power bus. Each filter train consists of a demister, prefilter, electric heater, HEPA filter, charcoal filter, and final HEPA filter, designed to conform to Regulatory Guide 1.52.

The design concept of the filtration systems for the mechnical and the electrical penetration rooms is identical to that for the enclosure building, although the system sizes are different. Each system consists of two, full capacity, redundant, fan-filter subsystems. The filter bank in each subsystem is design to conform to Regulatory Guide 1.52 and consists of a prefilter, electric heater, HEPA filter, charcoal filter and a final HEPA filter. The subsystems for each room are connected to separate emergency power buses. All subsystems start upon receiptof a containment isolation signal, but one subsystem in each room normally would be manually placed in the standby mode by the operator. Each subsystem is capable of reducing the pressure in its penetration room to a negative 0.25 inch water gauge, and maintaining this pressure differential while subjecting the air in the room to multipass filtration. Modulating dampers will allow filtered discharge to the environment of enough air to maintain the pressure differential.

The applicant has performed analyses to demonstrate that with only one fan operating in either of the penetration rooms or in the enclosure building the design negative pressure can be achieved within about four seconds for the penetration rooms and seven seconds for the enclosure building. We have performed a similar calculation and our results are in agreement with the applicant's.

The applicant will conduct a series of initial preoperational tests to confirm the predicted performance of the filtration systems for the enclosure building and the penetration rooms. We will review the results of this testing program in detail and will require periodic inservice inspection tests as part of our Technical Specifications.

Based on our review of the proposed design and the predicted performance of the enclosure building and the penetration room filtration systems, we conclude that these systems will meet the intent of Regulatory Guide 1.52, General Design Criteria 41, 42, 43 and 64, and are acceptable.

6.2.4 Containment Isolation Systems

The Containment Isolation System is designed to isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of closed systems and isolation valves, is provided so that no single valve or piping failure can result in loss of containment integrity. Reactor building penetration piping up to and including the external isolation valve is designed as seismic Category I equipment, and is protected against missiles that could be generated under accident conditions.

Reactor building isolation will occur automatically upon receipt of a containment isolation signal actuated by high reactor building pressure (4.7 psig). All fluid penetrations not required for operation of the engineered safety features equipment will be isolated. Remotely operated isolation valves will have position indication in the control room.

We have reviewed the containment isolation system for conformance to General Design Criteria 55, 56 and 57. We conclude that the system meets the intent of the General Design Criteria and is acceptable.

6.2.5 Combustible Gas Control Systems

Following a LOCA, hydrogen may accumulate inside the reactor building. The major sources of hydrogen generation include:

1. a chemical reaction between the zirconium fuel rod cladding and water,
2. corrosion of materials of construction, and
3. radiolysis of aqueous solutions in the reactor core and the containment sump.

The applicant's analysis of post-LOCA hydrogen generation, which is consistent with the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-Of-Coolant Accident," indicates that the hydrogen concentration in the containment would not reach the lower flammability limit of 4 volume percent until about 41 days after the postulated LOCA. We have performed a similar analysis of hydrogen generation in the containment following a LOCA and our results are in agreement with the applicant's.

Containment building air cooling and upper dome air circulating systems will be provided to mix the containment atmosphere following an accident so as to avoid possible problems of hydrogen stratification. Two, full capacity, electric, hydrogen recombiners located inside containment also will be provided, either of which will be capable of limiting the hydrogen concentrations to below the guidelines of Regulatory Guide 1.7. The proposed recombiner system incorporates several design features that are intended to assure the capability of the system to be operable in the event of an accident. Among these are: (1) seismic Category I design, (2) protection from the containment spray system, (3) protection from missile and jet impingement and (4) redundance to the extent that no single component failure disables both recombiners.

The staff previously has reviewed and accepted the design and prototype unit tests for the proposed recombiners. The recombiner manufacturer has completed preliminary qualification test on a production recombiner which is identical to the units proposed for the Vogtle plant. Further long-term environmental testing presently is being conducted. We believe that these final test results will provide additional verification of the design adequacy of the recombiner units.

A post-accident purge system, as described in Section 6.2.3, also will be provided to serve as a backup to the redundant hydrogen recombiner units.

Redundant monitoring systems, located outside the containment, will be provided to allow periodic sampling and analysis of the hydrogen concentration in the containment.

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