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Docket Nos: 50-413
and 50-414

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Transmittal of Preliminary Draft SER - Catawba Nuclear Station

Enclosed for your review and comment is the preliminary draft SER for Containment Systems (Enclosure).

Your attention is directed in particular to any open items contained within this preliminary draft. A principal objective of this transmittal is to provide for timely identification and resolution of any additional analysis, missing information, clarifications or other work necessary to resolve outstanding issues. Please contact the staff's Project Manager, Kahtan Jabbour, regarding the need for any meetings and telephone conferences to this end.

Your comments, including schedules for completion of any further analyses or other work associated with resolution of open items, are requested within four (4) weeks of this letter.

Sincerely,

Original signed by:
Thomas M. Novak

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

cc: See next page

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CATAWBA

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DRAFT SAFETY EVALUATION REPORT
CATAWBA NUCLEAR STATION, UNITS 1 AND 2
CONTAINMENT SYSTEMS BRANCH
DOCKET NOS. 50-413/414

6.2 Containment Systems

Each nuclear unit of the Catawba Nuclear Station will be housed in a dual containment structure. The primary containment vessel will be provided with a heat removal system, an isolation system and a combustible gas control system; the secondary containment (shield building) will be provided with an annulus ventilation system. Most notably, the plant will utilize an ice condenser type pressure suppression containment similar to the McGuire Nuclear Station, Sequoyah Nuclear Plant, and D.C. Cook Plant which are currently in operation.

The staff has reviewed the applicant's design, design criteria, and design bases for containment systems. The acceptance criteria used as the basis for the staff evaluation are set forth in certain sections of the SRP (NUREG-0800). Specifically, the Catawba application was reviewed against SRP Sections 6.2.1, "Containment Functional Design"; 6.2.2, "Containment Heat Removal Systems"; 6.2.3 "Secondary Containment Functional Design"; 6.2.4 "Containment Isolation System"; 6.2.5, "Combustible Gas Control in Containment"; and 6.2.6, "Containment Leakage Testing." These acceptance criteria include the applicable GDC, Regulatory Guides, Branch

Technical Positions, and industry codes and standards as specified in the above cited sections of the SRP. The results of the staff's review are discussed below.

The design of the Catawba Nuclear Station containment is very similar to the containment design for the McGuire Nuclear Station, which the staff has previously reviewed. Both plants utilize the dual containment concept with a free-standing steel primary containment. Volumes and plant arrangement within the primary containment are similar. Basic differences between the Catawba and the McGuire Nuclear Station plants are slight and are limited to the design of the reactor coolant system and containment internal structures. Table 6.2-1 indicates the differences in principal containment parameters between the two plants.

TABLE 6.2-1
COMPARISON OF CONTAINMENT DESIGN PARAMETERS

	Catawba	McGuire*
Reactor Containment Volumes (net-free volume, cubic feet):		
Upper Compartment	717,000	717,000
Ice Condenser	122,500	111,000
Lower Compartment	345,000	368,000
Total Containment Volume	1,184,500	1,196,000
Reactor Containment Air Compression Ratio:	1.41	1.41
Engineered Safeguards Design Rating (ESDR) Reactor Power Used for Containment Analysis (megawatts, thermal);	3,526	3,579
Design Energy Release to Containment:		
Initial Blowdown Mass Release (pounds)	498,200	493,210
Initial Blowdown Energy Release (Btu)	324.2×10^6	318.4×10^6
Weight of Ice in Ice Condenser (pounds)	2.45×10^6	2.45×10^6
Containment Return Air Fan Flow (cubic feet per minute)	40,000	30,000
Containment Spray Flow (LOCA Analysis, gallons per minute):		
One Spray Train Inoperable		
Upper Compartment	3,400	3,432
Lower Compartment	0	0
One Residual Heat Removal Pump Inoperable		
Upper Compartment	1,575	1,623
Lower Compartment	0	0
Total Spray	4,975	5,055
Containment Design Pressure (pounds per square inch gauge)	15.0	15.0

*These values are taken from the McGuire SER.

The primary reactor containment has a net-free volume of about 1,184,500 cubic feet which is divided into three major subvolumes: (1) 345,000 cubic feet in the lower compartment enclosing the reactor system; (2) 122,500 cubic feet in the ice condenser compartment enclosing the ice condenser; and (3) 717,000 cubic feet in the upper compartment.

The basic performance and design evalation of the ice condenser system have been the subject of both analysis and experimental programs. These efforts are described in a report entitled, "Staff Evaluation of Tests Conducted to Demonstrate the Functional Adequacy of the Ice Condenser Design," dated April 25, 1974, and provide the basis for the staff's evaluation of the containment functional design.

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

The containment for each unit of the Catawba Nuclear Station consists of a primary containment vessel and a shield building. The primary containment vessel is a freestanding, welded steel structure consisting of a vertical cylinder, a hemispherical dome, and a concrete base mat with steel membrane. The shield building is a medium leakage concrete structure enclosing the containment vessel, and is designed to provide for the collection, mixing, holdup, and controlled release of containment vessel fission product leakage following onset of an accident. The interior of the primary containment vessel is divided into three compartments: (1) a lower compartment which houses the reactor and reactor coolant system; (2) the ice condenser compartment which houses the energy-absorbing ice bed in which steam is condensed; and (3) the upper compartment which accommodates the air displaced from the other two volumes during postulated loss-of-coolant and steam line break accidents.

The intermediate, or ice condenser compartment, is an enclosed annular compartment encompassing most of the perimeter of the containment structure. Borated flake ice is stored within the ice condenser compartment in 48-foot long cylindrical perforated metal baskets. The ice contained in the baskets is provided to condense the steam released in the event of a loss-of-coolant accident or a steam line break accident.

The staff will require the applicant to weigh the ice in a large statistical sample (approximately 50 percent) of the 1944 ice baskets in each unit, following their initial ice loading. The staff will require that this information be used in statistical analyses to determine (1) the initial distribution of ice in the ice condenser; (2) the minimum amount of ice loaded into the ice condenser at a 95-percent level of confidence; and (3) appropriate subdivision of the ice condenser into groups of bays to be utilized in the periodic ice weight surveillance program. These matters will be resolved in conjunction with the staff's development of the Technical Specifications.

In an effort to provide the earliest possible indication of the actual sublimation rate for the D. C. Cook Unit 1 ice condenser, its licensee, the American Electric Power Company, implemented a program to measure periodically the weight of selected ice baskets in the ice condenser, and has weighed a sample of ice baskets on numerous occasions. The results of the ice basket weighing program have indicated that the average sublimation rate of 2 to 3 percent per year is significantly greater than the expected rate of about 0.5 percent per year, and slightly greater than the maximum design sublimation rate of 2 percent per year. The results also have shown that ice sublimation does not occur uniformly over the cross-sectional area of the ice condenser. Baskets adjacent to the crane and containment wall cooling ducts lose ice at a greater rate than baskets located in the interior of the ice condenser. Interpretation of the data from the ice basket weighing program has been complicated by the fact that variations in original ice loading techniques resulted in three distinct groups of ice weights. The frequent weighing programs conducted at the D. C. Cook facility have provided early identification of the ice condenser loss rates and patterns, the opportunity to develop corrective modifications and procedures, and has ensured the safety of continued operation of the plant.

The ice weighing program at the Sequoyah Nuclear Plant Unit 1 has detected an even greater rate of ice loss, on the order of 4 to 5 percent per year. This high loss rate necessitated, during the first fuel cycle, a reanalysis of the DBAs to permit a reduction in the requirement in the plant Tehnical Specifications for total ice weight.

Based on the above discussion of current ice condenser operating experience, the staff will require the applicant to institute a periodic ice basket weighing program for each unit at Catawba, similar to the programs being conducted at the Sequoyah and McGuire plants. The staff recommends that the applicant continue to evaluate the equipment and techniques available for ice loading to achieve an initial ice inventory that is uniformly distributed. The staff will pursue the development of a suitable periodic ice weighing program with the applicant during the development of Technical Specifications for the operation of the plant and will include appropriate operating limits to ensure an acceptable margin of safety.

During normal plant operation, the ice bed is maintained at about 15 degrees Fahrenheit by a redundant refrigeration system. Refrigeration ducts and insulation on the ice condenser walls serve to minimize heat losses from the ice. Thirty air handling units are provided in the containment, but only 21 of the units are required to operate at any time to maintain the design temperature of 15 degrees Fahrenheit within the ice bed. In the unlikely event that a complete loss of the refrigeration system occurs, the insulation within the ice condenser is sufficient to prevent the ice from melting for a minimum period of seven days, which allows adequate time for safe plant shutdown.

Inlet and outlet doors are provided at the top and bottom of the ice condenser compartment. In the event of a loss-of-coolant accident, the lower inlet doors will open due to the pressure rise in the lower compartment caused by the release of the reactor coolant to the lower compartment. The differential pressure will then cause air, entrained water, and steam to flow from the lower compartment into the ice condenser. The resultng pressure rise, due principally to the air mass in the ice condenser, will cause the doors at the top of the ice condenser to open and allow the air to flow from the ice condenser into the upper compartment. Steam will be condensed as it contacts the ice contained in the ice condenser compartment, and, therefore, does not reach the upper compartment. Complete steam condensation is assured because of the ice mass and geometrical arrangement of the ice columns. Developmental testing by Westinghouse has confirmed this phenomenon. The staff's evaluation of the test programs was completed in conjunction with the review of the D. C. Cook Plant, and was reported in the Safety Evaluation Report for that plant and in the report, "Staff Evaluation of Tests Conducted to Demonstrate the Functional Adequacy of the Ice Condenser Design," dated April 25, 1974.

An operating deck separates the upper and lower compartments and ensures that steam and air flow resulting from a loss-of-coolant accident is directed through the ice condenser to the upper compartment rather than through uncontrolled bypass paths. Following initial blowdown, approximately 1,690,000 pounds of ice (or 69 percent of the initial mass of ice) remains in the ice condenser. Condensation of the steam in the ice limits the containment pressure to approximately 9.0

pounds per square inch gauge between the time reactor blowdown is complete and the time that meltout of the ice bed occurs. Ice meltout is predicted to occur about 58 minutes after onset of a design basis LOCA. Following ice meltout, the rise in the containment pressure due to the release of decay energy from the core is limited by the containment spray system.

The lower compartment is divided into a number of subcompartments formed by internal equipment, structures, and components. The pressure responses within these subcompartments were analyzed by the applicant using the TMD (Transient Mass Distribution) computer code developed by Westinghouse. The code is described in a nonproprietary Westinghouse Topical Report, "Ice Condenser Containment Pressure Transient Analysis Methods," WCAP-8078. The code provides a means for computing pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment. The staff reviewed this code during the review of the D. C. Cook Plant and found it to be acceptable for calculating the short-term pressure response in subcompartments. The pressure response within the subcompartments is different from the overall pressure response of the containment only during the early blowdown phase of the accident; that is, up to about 10 seconds following the occurrence of the break.

Following the early blowdown phase of the accident, the pressure and temperature responses of the upper and lower compartments are analyzed with the Westinghouse LOTIC-1 computer program. This program has been described in Westinghouse Topical Report WCAP-8354, "Long Term Ice Condenser Containment Code - LOTIC Code." The staff has completed a generic review of the LOTIC-1 computer program through the NRC's topical report evaluation program and has concluded that the LOTIC-1 code is acceptable for the calculation of the long-term ice condenser containment response to postulated LOCAs (see NRC letter to Westinghouse dated May 3, 1978).

Maximum Pressure and Temperature Analysis

The applicant has performed containment analyses for a spectrum of reactor coolant system (RCS) and secondary system pipe ruptures to verify the containment functional design pressure and temperature, and to establish the pressure and temperature conditions for environmental qualification of safety-related equipment located inside containment. The containment functional analyses include the assumption of the most limiting single active failure and the availability or unavailability of offsite power, depending on which results in the highest containment temperatures and pressures.

For the design basis LOCA, the containment spray system is activated after the completion of blowdown, that is, about 45 seconds after onset of the accident. After about 10 minutes, the return air fans are started and the containment pressure is reduced to approximately 6.5 pounds per square inch gauge as air is returned from the upper volume to the lower volumes. Steam from the reactor coolant system is still being removed almost entirely by the stored ice at this time. After ice meltout, which occurs about 58 minutes after onset of the accident, steam from the reactor coolant system is removed by the containment spray system. The containment pressure will again peak about 1.5 hours after onset of the accident, at which time the energy input equals the minimum heat removal capability of the sprays. The magnitude of this peak pressure is determined by the heat input rate to the containment and heat removal rate of the containment spray system.

The applicant used the LOTIC-1 computer program to calculate the long-term containment pressure and temperature response to a spectrum of LOCAs. LOTIC-1 is a computer program similar to the COCO code, which has been used to analyze the containment pressure transients for other types of containments. The main differences between these computer codes lie in the methods by which the heat removal systems are modeled. LOTIC-1 includes features for modeling the heat removal capabilities of the ice and has provisions to calculate the pressure response of the containment. The containment upper and lower compartments and the ice condenser are modeled as control volumes in the code to represent the physical geometry of the containment. Conservation of mass and energy are applied and equations are solved by appropriate numerical procedures.

The applicant has provided mass and energy data for postulated loss-of-coolant accidents (LOCA) with a spectrum of break sizes and locations in the reactor system. The break sizes are up to and including a double-ended rupture of the largest pipe. The break locations include the cold leg at the suction and discharge sides of the reactor coolant pump, and the hot leg. The effects of single failures on mass and energy release rates were included by bounding the possible effects with two cases, i.e., the maximum safeguards case where no single failures are assumed and the minimum safeguards cases where the single failure assumed is the loss of one emergency diesel. We have reviewed the applicant's spectrum of breaks and the single failures considered and find them acceptable.

The mass and energy released to the containment is considered in terms of the blowdown, refill, reflood, and froth boiling phases. The method used by the applicant to compute the mass and energy release rates from postulated reactor coolant pipe breaks for the containment functional analyses are documented in topical report WCAP-8312A, "Westinghouse Mass and Energy Release Data for Containment Design." This topical report was approved by the staff in a letter dated March 12, 1975. Therefore, the applicant's mass and energy release rate data for postulated reactor coolant pipe breaks are acceptable for use in the containment functional analyses.

The staff has also reviewed the plant containment parameters and has found them to be conservative for the evaluation of the long-term containment response to postulated LOCAs. Using the above mass and energy release data, containment input parameters, and the LOTIC-1 ice condenser containment analysis program, the applicant has calculated a peak containment pressure of 14.4 psig and a peak containment temperature of 237°F for the worst-case LOCA, a double-ended rupture of the RCS pump suction cold leg. Because the peak calculated containment pressure of 14.4 psig is less than the 15.0 psig containment design pressure, the staff finds the applicant's long-term containment response calculations for LOCAs acceptable.

The applicant also has analyzed the effect of steam bypassing the ice condenser on the containment pressure response. Drain lines in the floor of the refueling canal are provided to allow water sprayed into the upper compartment to return to the containment sump. These drains represent a bypass path. The applicant has included in the containment analysis the effect of this bypass area (2.2 ft^2), along with another 2.8 ft^2 of area for margin, for a total area of 5.0 ft^2. The applicant has also provided analyses which indicate that about 40 ft^2 of bypass area can be accommodated in the design without the design pressure of the containment being exceeded.

The staff has reviewed the applicant's analysis of the maximum differential pressures which could exist in the reverse direction (that is, upper compartment to lower compartment) during a LOCA. The applicant's methods of analysis and as·.mptions are conservative. The applicant has calculated a maximum reverse differential pressure of 0.88 psid. This is well within the design reverse differential pressure capability of the operating deck and ice condenser lower inlet doors.

The applicant has calculated the containment response to a spectrum of main steamline breaks (MSLBs) using the LOTIC-3 computer program. This program has been described in Supplement 2 to the Westinghouse Topical Report WCAP-8354. The staff has completed a generic review of the LOTIC-3 code and has concluded that the LOTIC-3 code is acceptable for the calculation of long-term ice condenser containment response to postulated secondary system pipe break accidents (see NRC letter to Westinghouse dated May 3, 1978). At the staff's request, additional small MSLBs were analyzed, extending the spectrum down to a 0.1-ft^2 break size. These analyses were performed by Westinghouse for a "generic" ice condenser plant. Specifically, these analyses concerned the containment response to postulated 0.6 square feet, 0.35 square feet, and 0.1 square feet main steamline split breaks. In all cases the effects of containment spray and return air fan operation were considered in the analyses. In all cases a containment lower compartment pressure high enough to initiate automatic operation of the sprays and fans was calculated in the LOTIC-3 analysis of the postulated event.

However, the applicant has not presented sufficient data comparing the containment input parameters assumed in the analysis of the "generic" plant with the same parameters for the Catawba Nuclear Plant. This information is necessary in order for the staff to conclude that the "generic" plant parameters are equivalent to, or more conservative than, the Catawba parameters pertinent to these analyses. Therefore, the staff cannot conclude that the "generic" plant MSLB analyses are applicable to Catawba.

For large MSLBs, the applicant assumed that the mass released from the breaks was dry saturated steam (i.e., no liquid entrainment). Although this is an acceptable assumption, the applicant assumed there would be complete revaporization of the water which condenses on passive heat sinks in the containment during periods when superheated conditions exist inside containment. This assumption is not acceptable for releases with no liquid entrainment, under the terms of the staff's acceptance of the LOTIC-3 code, as expressed in the

NRC letter to Westinghouse, dated May 3, 1978. For MSLB releases with no liquid entrainment, the applicant must assume no revaporization of condensate from passive heat sinks. The applicant has done this for small MSLBs, but not for large MSLBs; therefore, the staff requires a reanalysis of large MSLBs with LOTIC-3 using the required input assumptions detailed above.

The mass and energy release for postulated MSLBs are calculated using the Westinghouse MARVEL code. However, during the course of the staff's review, Westinghouse made several model changes. One of these changes, accounting for additional heat transfer to steam during tube bundle uncovery in the steam generator, could have significant impact on the containment temperature response for ice condenser containments. Based on the data from a sensitivity study performed by Westinghouse (letter from E. Rahe, Jr. (Westinghouse) to J. Miller (NRC) dated February 17, 1982, Westinghouse NS-EPR-2563), it is estimated that the peak containment temperature response of the containment lower compartment may exceed the temperature profile currently calculated by the applicant. Therefore, it is required that a refined main steam line break analysis be done, taking into account the change in the heat transfer model, to determine the adequacy of the temperature profile in the containment lower compartment.

Based on the above review of the applicant's containment pressure and temperature functional analyses, the staff concludes that the applicant has satisfactorily demonstrated the adequacy of the containment functional design for LOCAs. However, the staff is not able to complete its review of the containment temperature and pressure response for postulated MSLBs. Pending receipt of the additional information and analyses specified above, the staff will report its conclusion in the final SER.

<u>Protection Against Damage From External Pressure</u>

The containment vessel is designed for an external pressure of 1.5 pounds per square inch gauge. Inadvertent operation of the spray system or the return air fan systems during normal plant operation would cause a reduction in the containment pressure. The applicant has provided a Containment Pressure

Control System to prevent the containment from being depressurized to below its design pressure; this system is activated when the containment internal pressure is less than 0.25 pounds per square inch gauge. The system is composed of eight independent pressure sensors (four per train) which interlock the controls of the spray and fan systems so as to prevent their operation when the containment pressure is less than 0.25 pounds per square inch gauge. The staff concludes that the use of this system, in lieu of a vacuum relief system, is acceptable.

Also, all high-energy lines penetrating containment are provided with guard pipes in the annulus space between the primary and secondary containments. These guard pipes assure that any steam released by a pipe break in the annulus is directed back into primary containment, so as not to pressurize the annulus and apply an external pressure on the containment vessel.

Therefore, the staff concludes that the Catawba design adequately protects the containment against damage from external pressure.

6.2.1.2 Subcompartment Analysis

Following the onset of a postulated reactor coolant pipe rupture, differential and local pressures build up in the subcompartments of the lower containment compartment as high-energy fluid is released and transported throughout the various regions. The pressure magnitudes depend upon the volumes of the subcompartments, interconnecting vent flow paths, mass flow behavior, and the thermodynamic behavior within the pressure nodes. During this phase of the transient, flow to the upper contianment compartment is rising but pressure is still near its initial pressure. It is during this time that the peak operating deck differential pressure and peak subcompartment differential pressures would be experienced. As the blowdown continues, the pressure in the upper compartment rises, and about 10 seconds after the start of blowdown the upper compartment reaches a peak pressure approximately equal to the lower compartment pressure, that is, about 8.0 psig. The primary factor in producing this upper compartment peak is the displacement of air from the lower compartment through the ice columns into the upper compartment.

The LOCA mass and energy release rate data used in the subcompartment analyses were developed from the computer program SATAN described in Westinghouse Topical Report WCAP-8312A, which was approved by the staff in a letter dated March 12, 1975. The mass and energy release rates for the steam line break in the steam generator subcompartment were calculated by the MARVEL code. We have reviewed the code with a detailed evaluation in Section 6.2.1.1 for containment analysis. For steam generator subcompartment analyses, we have found the method along with the conservative assumptions used to be acceptable.

The staff concludes that the methodology for computing the LOCA and steam line break mass and energy release for subcompartment analyses is acceptable.

The applicant used the TMD computer program to calculate the short-term pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment, including the containment compartments, following either a LOCA or MSLB. The model includes a nodalization scheme of 53 volumes representing the containment to analyze the pressure response of the subcompartments within the lower compartment (including dead-ended compartments), the ice condenser compartment, and the upper compartment.

TMD was developed specifically to analyze the short-term pressure response of the ice condenser system. The mathematical modeling in TMD is similar to that of the SATAN-V blowdown code in that the analytical solution is developed by considering the conservation equations of mass, momentum, and energy, and the equation of state, and uses the control volume technique for simulating spatial variation. The governing equations for TMD are somewhat different from those in SATAN-V in that a two-phase (liquid water droplets and steam-air vapor), two component (air-water) system is considered. The staff has reviewed these mathematical differences between SATAN-V and TMD and concurs that TMD has maintained the conservatisms incorporated in SATAN-V. The TMD calculates the critical flow of a two-component, two-phase fluid (air, steam, and water) assuming a thermal equilibrium condition. However, a correction factor, which was determined by Westinghouse to account for experimental data on applicable flow regimes, is then applied to the calculated critical flow. The correction factor as used in the code increases the critical flow up to 20 percent through the compartments as the quality of the fluid decreases. This increased critical flow is refered to as "augmented" flow. The net effect results in a lower differential pressure between compartments when compared to an unaugmented flow regime. The use of the augmented flow factor results in less conservatisms than use of the thermal equilibrium correlation.

Following the staff's review of the experimental data and analysis performed during the review of the D. C. Cook facility, the staff determined that the use of this correction factor for the type of analysis being performed could not be justified. In accordance with this determination, the applicant has analyzed the short-term containment pressure response using the latest version of the TMD code with an unaugmented or unity flow correlation.

The applicant's heat transfer model of the ice condenser used in the latest version of the TMD code is based on the results of full-scale testing done by Westinghouse during 1973 and 1974. Results of the staff review of the 1973 - 1974 full-scale ice condenser tests have been presented in an April 1974 report, "Staff Evaluation of Tests Conducted To Demonstrate the

Functional Adequacy of the Ice Condenser Design." The latest version of the TMD code also includes a compressibility factor which is used with the subsonic incompressible flow equations to include the effects of compressible fluid flow. These code provisions are acceptable to the staff.

The 53-node containment model was used to calculate maximum differential pressure on the operating deck and on the containment outer wall in the ice condenser inlet plenum and the lower compartment loop or dead-ended compartments. The calculated differential pressures acting on these structures do not exceed the corresponding design values. The staff, therefore, finds the applicant's method of analysis, modeling assumptions, and results to be acceptable for use for the structural analysis of the structures mentioned above. Separate analyses were done for pipe breaks in the steam generator enclosures, pressurizer enclosure, and reactor cavity, as discussed below.

Steam Generator Enclosure

The applicant has used the TMD code with the compressibility factor and assuming unaugmented critical flow to perform the transient analysis of the steam generator enclosure. The applicant has performed the analysis for the only possible high energy line break within the enclosure, a 3.05 square foot limited displacement rupture of the main steam line at the top of the steam generator. The size and location of the steamline break has been limited by enclosing the steam line by a continuous guard pipe over the entire length of the steam line within the enclosure. The applicant has performed a nodalization sensitivity study on the steam generator enclosure which resulted in a nine-node model of the enclosure. However, FSAR Table 6.2.1-20 was to present information concerning the nodalization sensitivity study, but does not. This information is required so that the staff may complete its review. Also, the applicant has not provided the design differential pressure values for the steam generator enclosure walls. The staff also requires this information in order to complete its review.

Pressurizer Enclosure

The applicant has analyzed the response of the pressurizer enclosure to the postulated rupture of the largest line within the enclosure; i.e., a double-ended rupture of the six-inch pressurizer spray line at the pressurizer vessel nozzle to spray line piping weld. The applicant has used the TMD code without the augmented critical flow correlation and with the compressibilty factor to perform the pressurizer enclosure analysis. He has also performed a nodalization sensitivity study, using both two-node and four-node models of the enclosure. The peak calculated differential pressure acting across the enclosure structure was 18.3 psi for the two-node model and 18.4 for the four node model. However, the applicant has not provided the design differential pressure values for the pressurizer enclosure walls. The staff requires this information in order to complete its review.

Reactor Cavity

The applicant used the TMD code with the compressibility factor and without the augmented critical flow correlation to analyze the response of the reactor cavity structures to a LOCA. The reactor cavity was modeled by forty-four nodes within the cavity and ten nodes external to the cavity. The annulus between the reactor vessel and the shield wall was divided into axial and circumferential nodes. The maximum credible break size and its corresponding location were identified by the applicant as an 85-square inch limited displacement rupture of a cold leg pipe at the pressure vessel nozzle-to-pipe-weld. The maximum differential pressures calculated are less than the design pressure for all nodes of the reactor cavity structure. The staff finds the applicant's method of analysis, modeling assumptions, and results acceptable for the evaluation of reactor cavity structures. However, the applicant has not provided sufficient information concerning the calculation of asymmetric blowdown pressure forces and moments on the reactor vessel. Specifically, the applicant should show conformance with the provisions of Section 3.2.2.4 of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," dated January 1981.

In summary, the staff finds acceptable the applicant's subcompartment analysis for the 53-node model of the entire containment. However, additional information, as stated above, is required for the steam generator enclosure, pressurizer enclosure, and reactor cavity subcompartment analyses.

6.2.1.3 Minimum Containment Pressure Analysis for Performance Capability Studies of the ECCS

Appendix K to 10 CFR 50 of the Commission's regulations requires that the effect of operation of all the pressure-reducing systems and processes installed in the containment be included in the ECCS evaluation. For the purpose of this evaluation, it is conservative to minimize the containment pressure. The reflood rate in the core will then be reduced because of the resistance to steam flow in the reactor primary system.

Following onset of a LOCA, the pressure in the containment building will be increased by the addition of steam and water from the reactor primary system to the containment atmosphere. After initial blowdown, heat transfer from the core, primary metal structure, and steam generators to the emergency core cooling water will produce additional steam. This steam, together with any emergency core cooling water spilled from the primary system, will flow through the postulated break into the containment. This energy will be released to the containment during both the blowdown and later operational phases (that is, the reflood and post-reflood phases).

Energy removal occurs within the containment by several means. Steam condensation on the containment walls and on internal structures serves as a passive energy heat sink that becomes effective early in the blowdown transient. Subsequently, the operation of the containment heat removal systems such as containment sprays will remove steam from the containment atmosphere. In an ice condenser-type containment, energy is removed as the mixture of steam, air, and water passes through the ice condenser (that is, when the mixture is forced from the containment lower compartment to the upper compartment).

The ECCS containment pressure calculations for Catawba were done using the Westinghouse ECCS evaluation model. The containment response calculations were performed using the Westinghouse LOTIC-2 containment code. The staff has reviewed the LOTIC-2 code and has concluded that the LOTIC-2 code is acceptable for the calculation of minimum containment pressure response for ice condenser plants. Although the staff has accepted the methods used to calculate containment pressure response, justification of the plant-dependent input parameters used in the analysis of containment pressure response are required to be submitted for review on a plant-by-plant basis. This information was submitted in the Catawba FSAR. The applicant has evaluated the containment net-free volume, the passive heat sinks, operation of the containment heat removal systems, and containment initial conditions with regard to conservatism for the ECCS analysis. The containment heat removal systems were assumed to operate at their maximum capacities, and maximum operational values for the spray water and service water temperatures were assumed. The staff finds these assumptions to be acceptable and in accordance with BTP CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." However, contrary to the provisions of BTP CSB 6-1, the applicant has not considered the effect of containment air lost through containment purge or vent lines open at the beginning of the accident. This loss of air would reduce the minimum calculated containment pressure, and the staff requires that this effect be considered in the analysis.

The mass and energy release rates for minimum containment pressure analysis were calculated using the method described in Section 15.6.5 of the FSAR. This method is evaluated in Chapter 15 of this SER.

The staff has concluded that the plant-dependent information used for the ECCS containment pressure analysis for Catawba is reasonably conservative; however, additional information is required concerning the effect of open containment purge/vent lines on the analysis, in order for the staff to complete its review.

6.2.1.4 Summary and Conclusions

Based on the preceeding evaluations, the staff concludes that the containment functional design is acceptable and meets the requirements of GDC 16, 50, 52, and 53 for LOCA, steam bypass, maximum differential pressure in the reverse direction acting on the operating deck, protection against damage from external pressure, and the 53-node subcompartment analysis for the entire containment. Additional information is required concerning the MSLB analyses, subcompartment analyses for the reactor cavity and steam generator and pressurizer enclosures, and minimum containment pressure analysis for performance capability studies of the ECCS. Pending receipt of the additional information, the staff will report its conclusions in the final SER.

6.2.2 Containment Heat Removal Systems

The energy released to the containment following a design basis loss-of-coolant accident will be initially absorbed by the ice condenser. After the ice bed has melted, mass and energy will continue to be released to the containment, at which time the containment spray systems will maintain the containment pressure in the long-term below the containment design pressure, and eventually reduce the containment pressure to about atmospheric pressure.

The containment spray for each unit of the Catawba Nuclear Station is provided by two spray trains, each designed to provide the cooling capacity required to maintain the peak containment pressure at less than design pressure for the full spectrum of break sizes. Each spray system delivers 3400 gallons per minute of borated water to the containment from one containment spray pump and heat exchanger. The containment spray pump is started by a containment presure signal set at approximately 3.0 pounds per square inch gauge, and containment spray starts at about 45 seconds after the onset of the accident. An additional 1,575 gallons per minute of containment spray from the residual heat removal pump may be manually initiated after the change over from the injection to recirculation mode of operation. This would be done no earlier than one hour after onset of the accident.

The containment is equipped with two 100 percent capacity return air fan systems, each of which uses a 40,000 cubic feet per minute fan to force air from the upper compartment back to the lower compartment after the reactor coolant system blowdown and subsequent reactor reflooding are completed.

The return air fans are utilized to return air from the upper compartment to the lower compartment after the compression peak is reached, and thus provide a homogeneous mixture of steam and air throughout the containment during the long-term pressure peak. Although the return air fans are started by the containment pressure signal, fan startup is delayed for 10 minutes to provide an increased backpressure during core reflood.

The applicant has provided a malfunction analysis and other information which demonstrates independence of the redundant spray trains and return air fan systems. Each spray train has its own recirculation piping suction inlet from a common sump. The sump is protected by grating and screening to prevent debris from passing into the suction lines. The spray nozzles are the limiting component in the containment spray systems and are not subject to clogging by particles less than 1/4 inch. The applicant has provided a pump net positive suction head analysis which shows that adequate suction head is available at both the containment spray pump and residual heat removal pump inlets during both the injection and recirculation phases without taking credit for increased containment pressure, as recommended by Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps" (Safety Guide 1). The staff finds the net positive suction head analysis to be acceptable.

The applicant used the LOTIC-1 code to demonstrate the long-term capability of minimum containment heat removal systems (one complete train of spray and return air systems) to maintain the containment pressure below design pressure for the design basis loss-of-coolant accident. The staff has reviewed the applicant's containment pressure and temperature response as calculated by the LOTIC-1 code and concludes that the design of the containment heat removal systems is acceptable. Provisions are made in the containment spray system and the return air system to permit inservice inspection of the system components and functional testing of active components in both systems.

The staff therefore concludes that the design of the containment heat removal system is acceptable and meets the requirements of General Design Criteria 38, 39 and 40.

6.2.3 Secondary Containment Functional Design

The secondary containment system consists of a shield building enclosing the primary containment structure. An annulus ventilation system is provided for the annulus formeed by these structures. The annulus ventilation system collects and filters airborne radioactivity that may leak from the primary containment following a loss-of-coolant accident. The annulus ventilation system consists of two independent, 100 percent capacity fan/filter trains. Each train consist of filters, ducting, supports, valves, fan (9000 cfm), and instrumentation and controls.

In the event of a LOCA, the annulus ventilation system is started by a containment high pressure signal (three psig). The annulus ventilation system is aligned to exhaust at 9000 cfm until the annulus reaches a pressure of minus 0.5 inches of water gauge. Upon reaching the prescribed negative pressure, the system is modulated to exhaust air as necessary to maintain the pressure within the annulus at minus 0.5 inches of water gauge.

The applicant has analyzed the performance of the annulus ventilation system using the CANVENT computer code. The analysis has considered the inleakage of air to the annulus, the compression of the annulus volume due to expansion of the steel containment vessel, and the transfer of heat to the annulus volume from the heated containment vessel. The applicant's analysis also assumes no fan flow until 23 seconds after the onset of the postulated accident, and full flow (9000 cfm) thereafter. The applicant calculated a negative pressure would first occur in the annulus 51 seconds after the onset of the accident and a pressure of minus 0.5 inches of water gauge would occur 63 seconds after the onset of the accident.

As mention in Section 6.2.1.1 of this SER, high energy lines which pass through the annulus are equipped with guard pipes which will prevent the pressurization of the annulus due to a high energy pipe break in the annulus.

Access to the annulus is under administrative control. Both upper and lower doors to the annulus are normally locked and the keys under the control of the shift supervisor. Personnel entry into the annulus must be authorized and recorded. However, contrary to the provisions of SRP 6.2.3, there are no remote alarms or indications of annulus door position in the main control room. The staff will require door position indicators and alarms having readout and alarm capability in the main control room to be provided, unless the applicant can provide adequate justification for not doing so.

The staff has not completed its review of the applicant's provisions for accounting for containment leakage which may bypass the annulus, nor of the applicant's compliance with the provisions of BTP CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants." The staff will report its conclusions concerning bypass leakage in an update to this draft SER.

Therefore, except for the two issues concerning position indication and alarms for access doors to the annulus, and bypass leakage, the staff concludes that the secondary containment functional design is acceptable and meets the requirements of GDC 4, 16, and 43. The staff will report its conclusions of the outstanding items in a future revision of the SER.

6.2.4 Containment Isolation System

There are at least two barriers between the atmosphere outside the containment and the reactor coolant system or the containment atmosphere. No manual operation is required for immediate isolation of the containment. Automatic isolation valves are provided in those lines which must be isolated immediately following an accident. Each automatic trip valve is provided with backup remote manual capability, and the valve position is displayed in the main control room. Isolation valves inside the containment are located between the crane wall and the containment wall for missile protection. The containment isolation valves have been designed to safety class 2 (Quality Group B), but the applicant has not stated that the valves meet Seismic Category I design requirements. When closed systems inside or outside containment are considered to be isolation barriers, the applicant has not indicated whether the systems meet Seismic Category I design requirements. It is also unclear whether certain of these systems may be Quality Group C rather than the required Quality Group B. It is not stated whether closed systems inside containment are protected from pipe whip. The applicant has not indiciated whether closed systems outside containment are protected from high energy line breaks outside containment. The staff requires this information in order to complete its review.

The staff requires additional information concerning the Containment Purge System and its conformance to the provisions of BTP CSB 6-4 (Rev. 2), "Containment Purging During Normal Plant Operations." The applicant should address, point-by-point, each of the provisions of BTP CSB 6-4. This should also be done for the Containment Air Release and Addition System and the Containment Hydrogen Sample and Purge System.

The staff has reviewed the containment isolation signals for the isolation valves, including the containment purge system isolation valves. Containment isolation will automatically occur upon receipt of safety injection or high containment pressure signals. In addition, the containment purge system isolation valves are also closed by high radiation level in the containment. The staff concludes that the containment isolation signals provide acceptable diversity.

The classification of systems as essential and non-essential for the purpose of prompt containment isolation, requires, further review. As stated in NUREG=0737, "Clarification of TMI Action Plan Requirements," Section II.E.4.2, "Containment Isolation Dependability," and in SRP 6.2.4, all non-essential systems shall be automatically isolated by the diverse containment isolation signal. Further consideration by the applicant of the definitions of essential and non-essential is needed and the basis for classifying each system into these two categories also needs additional justification.

In conformance with the provisions of NUREG-0737, Section II.E.4.2, the applicant states that the design of control systems for automatic containment isolation valves is such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Also, the containment pressure setpoint (approximately 1.0 psig) that initiates containment isolation for non-essential penetrations (Phase A isolation) has been set at the minimum compatible with normal operating conditions.

Pending receipt of the additional information required as stated above, the staff will report its conclusions concerning containment isolation in the final SER.

6.2.5 Combustible Gas Control System

Following a loss-of-coolant accident, hydrogen may accumulate within the containment as a result of (1) metal-water reaction between the fuel cladding and the reactor coolant, (2) radiolytic decomposition of the post-accident emergency cooling water, or (3) corrosion of certain construction materials by emergency core cooling and containment spray solutions. The applicant has analyzed the production and accumulation of hydrogen within containment from the above sources using the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident".

The applicant has used the same assumptions as Regulatory Guide 1.7 to calculate the rate of hydrogen released by radiolysis and corrosion of metals. The applicant has also assumed an immediate release of hydrogen to the containment from the metal-water reaction between the fuel cladding and the reactor coolant. The applicant assumed a core wide average depth of reaction into the cladding of 0.00023 inch (about 1 percent of the core cladding). 10 CFR 50.44 requires that the hydrogen produced by metal-water reaction of the cladding be five times the amount of the maximum calculated reaction predicted in the evaluation to satisfy the emergency core cooling system acceptance criteria under 10 CFR 50.46, or that amount would be evolved from a core-wide average depth of reaction into the original cladding of 0.00023 inch, whichever is greater. However, the FSAR indicates that the maximum calculated metal-water reaction predicted by the ECCS analysis done under 10 CFR 50.46 is 0.3 percent. Therefore, the applicant must revise the hydrogen production and accumulation analysis to include a metal-water reaction of 1.5 percent.

The applicant will provide redundant electrical thermal hydrogen recombiners designed to limit the hydrogen concentration within the containment to conform with Regulatory Guide 1.7.

The Westinghouse electric thermal hydrogen recombiner system incorporates several design features that are intended to ensure the capability of the system to be operated in the event of an accident. Among these are:

(1) seismic Category I design;

(2) Quality Group B design;

(3) protection from missile and jet impingement from broken pipes;

(4) redundancy to the extent that no single component failure can disable both recombiners; and

(5) separate power supplies for each heater;

Each of the two 100-pecent-capacity electric recombiners is capable of processing 100 scfm of containment atmosphere for postaccident hydrogen control. The staff has reviewed tests that have been conducted for a full-scale prototype and a production recombiner. The tests consisted of proof-of-principle tests, testing on a prototype recombiner, environmental qualification testing, and functional tests for a production recombiner. (These tests are described in WCAP-7820 and its Supplements 1-4.) The results of these tests demonstrated that the recombiner should be capable of properly recombining hydrogen in a post-LOCA containment environment. Because these recombiners are situated inside containment, the requirements of Item II.E.4.1 of NUREG-0737, "Dedicated Hydrogen

Penetrations," do not apply; that is, no penetrations are used.

The containment return air fans, which begin operation 10 mintues after the onset of an accident, provide adequate mixing of the upper and lower containment compartment volumes. Two redundant hydrogen skimmer systems are provided to prevent the accumulation of hydrogen in the containment dome or in dead-ended subcompartments in the lower compartment. These areas are continuously vented by diverting a portion of the return air fan flow through the skimmer system; therefore, the potential for local hydrogen pocketing is limited.

In accordance with Regulatory Guide 1.7, the applicant also has provided a containment hydrogen purge system for post-accident containment atmosphere cleanup.

Capability is provided for periodic inservice inspection, operability testing, and leak rate testing of these systems and components.

Although the plant has a system to allow samples of containment atmosphere to be taken and analyzed for their hydrogen content, the applicant has not stated that the plant also has a hydrogen monitoring system with continuous indication in the control room, such as one that satifies the provisions of Item II.F.1 of NUREG-0737, "Additional Accident Monitoring Instrumentation" (Attachment 6, "Containment Hydrogen Monitor"). The staff requires such information in order to complete its review.

The staff has requested that the applicant describe the measures to be taken at the Catawba Station to control the substantial amounts of hydrogen that would be produced by an accident involving a severely degraded reactor core and a metal-water reaction of up to 75% of the active cladding. This information is required to complete the review.

In summary, the staff requires additional information concerning (1) a reanalysis of hydrogen production and accumulation using a 1.5 percent metal-water reaction; (2) a hydrogen monitoring system; and (3) degraded core hydrogen control. The remaining aspects of the design of the combustible gas control system are acceptable and meet the requirements of 10 CFR 50.44 and 50.46, and GDC 5, 41, 42, and 43. The staff will report its conclusions regarding the open items in the final SER.

6.2.6 Containment Leakage Testing

The Catawba Nuclear Station containment design includes provisions and features to satisfy the testing requirements of Appendix J to 10 CFR Part 50. Included are those penetrations that have resilient seals and expansion bellows; i.e., airlocks, emergency hatches, refueling tube blind flanges, and electrical penetrations.

The applicant intends to conduct periodic Type A tests with a duration of less than 24 hours, provided certain conditions are met, as specified in the FSAR. The specified conditions are insufficient for the purpose. The applicant should provide more conservative criteria for early termination of a Type A test.

The applicant proposes to locally test containment penetration bellows at a pressure which is less than the peak calculated accident pressure (Pa). The staff requires justification of this proposed testing practice in order to conclude on its acceptability.

The FSAR states that, for Type C testing of containment isolation valves, if a column of water exists on the auxiliary building side of the outside isolation valve for a particular penetration, the test pressure will be increased by an equal amount to ensure the required test pressure across the valve. This practice is not acceptable to the staff; the outer side of the valve should be drained of water and vented to the atmosphere.

The staff has requested that the applicant provide justification for not Type C testing certain containment isolation valves, whereas such tests may be required by Appendix J. This information is required in order for the staff's review to be completed.

With the exceptions noted in this section, the proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR 50. Such compliance provides adequate assurance that containment leak-tight integrity can be verified periodically throughout service lifetime on a timely basis to maintain such leakage within the limits of the Technical Specifications.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through leak paths will not be in excess of acceptable limits specified for the site.

The staff therefore concludes that, with the exceptions of the Type A test duration, bellows testing, and Type C testing procedures and determination of valves to be tested, the containment leak testing program is acceptable and meets the requirements of GDC 52, 53, and 54; Appendix J to 10 CFR 50; and 10 CFR 100. The staff will report its resolutions of the open items in the final SER.