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Docket Nos.: 50-413/414

Mr. H. Tucker
Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, NC 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 Reactor 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

Enclosures: As stated

cc: See next page

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SURNAME	KJabbour:eb	MDuncan	EAdensam	TMNovak			
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CATAWBA

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

cc: William L. Porter, Esq.
Duke Power Company
P.O. Box 33189
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Debevoise & Liberman
1200 Seventeenth Street, N.W.
Washington, D. C. 20036

North Carolina MPA-1
P.O. Box 95162
Raleigh, North Carolina 27625

Mr. F. J. Twogood
Power Systems Division
Westinghouse Electric Corp.
P.O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. J. C. Plunkett, Jr.
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

Mr. Jesse L. Riley, President
Carolina Environmental Study Group
854 Henley Place
Charlotte, North Carolina 28208

Richard P. Wilson, Esq.
Assistant Attorney General
S.C. Attorney General's Office
P.O. Box 11549
Columbia, South Carolina 29211

Mr. Henry Presler, Chairman
Charlotte - Mecklenburg Environmental Coalition
943 Henly Place
Charlotte, North Carolina 28207

North Carolina Electric Membership Corp.
3333 North Boulevard
P.O. Box 27306
Raleigh, North Carolina 27611

Saluda River Electric Cooperative, Inc.
207 Sherwood Drive
Laurens, South Carolina 29360

Mr. Peter K. VanDoorn
Route 2, Box 179N
York, South Carolina 29745

James P. O'Reilly, Regional Administrator
U.S. Nuclear Regulatory Commission, Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Robert Guild, Esq.
P.O. Box 12097
Charleston, South Carolina 29412

Palmetto Alliance
2135 1/2 Devine Street
Columbia, South Carolina 29205

5.2.2 Overpressure Protection

Section 5.2.2 (overpressure Protection) for Catawba Units 1 and 2 has been reviewed in accordance with the July 1981 edition of the Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800. A review of each of the areas listed in the Areas of Review section of SRP 5.2.2 was performed according to the guidelines provided in the Review Procedures section of SRP 5.2.2. Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for overpressure protection is acceptable.

Overpressure protection for the reactor coolant pressure boundary (RCPB) is provided by means of the three safety and three relief valves in combination with the reactor protection system, and operating procedures. The combination of these features provides overpressurization protection as required by the General Design Criterion 15, Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, the Appendix G of 10 CFR 50. The above requirements assure RCPB overpressure protection for both power operation and low temperature operation (start up and shutdown). The following is a discussion of both modes of overpressure protection.

5.2.2.1 Overpressure Protection During Power Operation

For this mode, the pressurizer relief valves are sized to limit system pressure to a value not exceeding the safety valve setpoint (2485 psig) to minimize challenges to the safety valves. The pressurizer spray system is designed to maintain the reactor coolant system pressure below the relief valve setpoint of 2350 psig during the step reduction in load of up to 10 percent. The relief valves limit the pressurizer pressure to a value below the high pressure reactor trip setpoint of 2385 psig for all design anticipated transients up to and including the design basis 50 percent step load reduction with steam dump.

Credit is taken only for safety valves in analyzing operational transients and faulted conditions.

Each pressurizer safety valve is spring-loaded and has a relieving capacity of 420,000 pounds mass per hour of saturated steam at 2485 pounds per square inch gauge. The combined capacity of two of these three valves is adequate to prevent the pressurizer pressure from exceeding the ASME Boiler and Pressure Vessel Code Section III limit of 110 percent design pressure following the worst reactor coolant system pressure transient, identified to be a 100 percent load rejection resulting from a turbine trip with concurrent loss of main feedwater. This event was evaluated with no credit taken for operation of reactor coolant system relief valves, steam line relief valves, steam dump system, pressurizer level control system, and pressurizer spray.

The SRP Section 5.2.2 requires that the applicant demonstrate adequate relief protection by assuming that the reactor trip is initiated by the second safety grade signal from the reactor protection system. The applicant has taken credit for a high pressurizer pressure trip (the first safety grade primary system trip). The evaluation is supported by a generic sensitivity study of required safety valve flow rate versus trip parameter presented in WCAP-7769. We have requested additional information on the details of this calculation, and will report our conclusions in a supplement to this SER.

The above analyses were performed using the LOFTRAN code, a digital simulation which includes point neutron kinetics, reactor coolant system (RCS) including the reactor vessel, hot leg, primary side of the steam generator and cold leg, secondary side of the steam generator, pressurizer, and pressurizer surge ine. This code is currently under review by the staff. Our review has progressed to the point that there is reasonable assurance that the conclusions based on these analyses will not be appreciably altered by completion of the analytical review. If the final approval of LOFTRAN indicates that any revisions to the analyses are required, the effect of these changes on Catawba will be evaluated and we will require implementation, if indicated.

The safety valves are designed in accordance with the ASME Code Section III, and periodic testing and inspection are performed in accordance with Section XI of this code. In Chapter 14 of the FSAR, the applicant has described his preoperational test program, which includes testing of the pressure relieving devices discussed in this SER section, and has indicated that these tests would be conducted in full compliance with the intent of Regulatory Guide 1.68. Additionally, Items II.D.1 and II.D.3 of NUREG-0737 require performance testing of relief and safety valves and relief and safety valve position indication. Conformance to these items is addressed in Section * of this SER. With resolution of the above issues by the applicant, we conclude that the overpressure protection provided for Catawba at power operating conditions will comply with the guidelines of Standard Review Plan 5.2.2 and the requirements of General Design Criterion 15.

5.2.2.2 Overpressure Protection During Low Temperature Operation

The SRP Section 5.2.2 requires that the overpressure protection system during low temperature operation of the plant shall be designed in accordance with the requirements of Branch Technical Position (BTP) RSB 5-2.

The low-temperature overpressure protection is primarily provided by the pressurizer relief valves (PORVs). As RCS temperature approaches the temperature setpoint during plant cooldown, an annunciator alerts the operator that plant conditions require low temperature overpressure protection. A key-lock switch for each train of the PORVs is placed to the low pressure position by the operator to enable the PORV low pressure setpoint. Should a pressure excursion occur with the low pressure mode enabled when the plant temperature is below the temperature setpoint, system pressure in excess of the PORV low pressure setpoint would be relieved to the pressurizer relief tank. An annunciator in the control room would alert the operator to system overpressure.

The PORV's and associated block valves are required to have safety grade emergency power supplies in accordance with Item II.G.1 of NUREG-0737. Section * of this SER provides a discussion of Catawba's compliance with this requirement.

As a backup to the low-temperature overpressure protection system, the residual heat removal system (RHRS) has two suction relief valves with a capacity of 900 gpm each at a setpoint pressure of 450 psig. The relieving capacity of

*LPM to provide section numbers.

each valve is adequate to relieve the combined flow of the two centrifugal charging pumps. The RHRS suction relief valves provide overpressure protection after the RHRS is put into operation and the RHRS suction isolation valves are open at RCS pressure less than 425 psig. Also, operating procedures require that the operator lock out the cold leg accumulator isolation valves in the closed position during shutdown.

The applicant has discussed a postulated failure of a DC power bus which would initiate a low-temperature overpressure scenario by both isolating letdown and disabling one train of the Low Temperature Overpressure Protection System, coupled with the single failure (closed) of the PORV in the unaffected train. He has stated that the Reactor Coolant System would be protected by RHR suction side relief valves when the RHR system is in operation, and by alarm-initiated operator action when the RHR system is isolated. To assure at least 10 minutes for operator action, the applicant's operating procedures call for a pressurizer bubble to be maintained when the RHR system is isolated. We find the applicants' discussion of this scenario acceptable.

We requested the applicant to show conformance to Branch Technical Position RSB 5-2. We will report conclusions on this area in a supplement to this SER.

5.2.2.3 Conclusions

Subject to the resolution of the aforementioned concerns, the staff concludes that the overpressure protection system for both normal and low temperature is acceptable and meets the relevant requirements of GDC 15 and 31 and Appendix G to 10 CFR Part 50. This conclusion is based on the following:

(1) The overpressure protection system prevents overpressurization of the reactor coolant pressure boundary under the most severe transients and limits the reactor pressure during normal operational transients. Overpressurization protection is provided by three safety valves. These valves discharge to the pressurizer quench tank through a common header from the pressurizer. The safety and relief valves in the primary, in conjunction with the steam generator safety and relief valves in the

secondary, and the reactor protection system, will protect the primary system against overpressure in the event of a complete loss of heat sink.

(2) The peak primary system pressure following the worst transient is limited to the ASME Code allowable value (110 percent of the design pressure) with no credit taken for nonsafety-grade relief systems. The Catawba plant was assumed to be operating at design conditions (102 percent of rated power) and the reactor is shut down by a high pressurizer pressure scram. The calculated pressure is less than 110 percent of design.

Except for the aforementioned concerns,* the applicant has met GDC 15 and 31 and Appendix G since they have implemented the guidelines of BTP RSB 5-2.

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5.4.7 Residual Heat Removal System

The residual heat removal system (RHRS) for Catawba Units 1 and 2 has been reviewed in accordance with Section 5.4.7 of the Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800. A review of each of the areas listed in the Areas of Review section of SRP Section 5.4.7 was performed according to the guidelines provided in the Review Procedures section of SRP Section 5.4.7. Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for residual heat removal (RHR) is acceptable.

The RHRS is designed to remove heat from the reactor coolant system after the system temperature and pressure have been reduced to approximately 350°F and 425 psig, respectively. The RHRS is capable of reducing the reactor coolant to the cold shutdown temperature and maintaining this temperature until the plant is started up again.

The RHRS operates in the following modes:

(1) <u>Emergency Core Cooling System (ECCS), Injection Mode</u>

Functions in conjunction with the high head portion of the ECCS to provide injection of borated water from the refueling water storage tank (RWST) into the RCS cold legs during the injection phase following a loss-of-coolant accident (LOCA).

(2) <u>Emergency Core Cooling System, Recirculation Mode</u>

Provides long-term cooling during the recirculation phase following a LOCA. This function is accomplished by aligning the RHRS to take fluid from the containment sump, cool it by circulation through the RHR heat exchangers, and supply it to the cold legs of RCS. During this mode of operation, the RHRS discharge flow is also connected to the suctions of the safety injection pumps and charging pumps to provide water supplies for high-head recirculation. Flow paths are also available for hot-leg injection during long-term recirculation mode to prevent boron precipitation in the reactor core.

(3) Refueling

Used to transfer refueling water between the refueling cavity and the refueling water storage tank at the beginning and end of the refueling operations.

(4) Cold Shutdown

Removes RCS decay heat and maintains cold shutdown conditions.

(5) Startup

Connected to the chemical and volume control system (CVCS) via the low pressure letdown line to control reactor coolant pressure.

Design data for the RHRS are as follows:

(1)	Pressure	600 psig
(2)	Temperature	400°F
(3)	Pump capacity	3000 gpm
(4)	Number of independent trains	Two

The RCS cooldown time with one RHR train from initial conditions of 425 psig and 350°F to 200°F is less than 24.4 hours. The two RHR trains are independent in action and powered by separate power supplies to provide redundancy.

The Catawba plants are required to meet Branch Technical Position (BTP) 5-1, Design Requirements of the Residual Heat Removal System. We have requested additional information concerning the Catawba ability to meet these requirements. We will report our findings in a supplement to this report.

5.4.7.1 Functional Requirements

As required by SRP Section 5.4.7, the RHR system for Catawba must meet General Design Criteria (GDC) Items 1 through 5. Items 1 through 4 regarding Quality Standards and Records, Design Bases for Protection Against Natural Phenomena, Fire Protection, and Environmental and Missile Design Bases are covered in Sections *, *, *, and * of this report, respectively. GDC 5, Sharing of Structures, Systems, and Components, is met for the Catawba RHR systems since components are not shared between units.

Redundancy in the RHR system is provided by two trains for each unit. Leak detection for the RHR system is discussed in Section * of this SER. Isolation valve redundancy is discussed in Section 5.4.7.2. The staff has reviewed the description of the residual heat removal system and the piping and instrumentation diagrams to verify that the system can be operated with or without offsite power and assuming a single failure. The two residual heat removal pumps are connected to separate buses which can be powered by separate diesel generators in the event of loss of offsite power.

SRP Section 5.4.7 requires that the RHRS must be operable from the control room in accordance with GDC 19. Limited manual actions are permitted outside the control room assuming a single failure, if justified. The Catawba RHR system is designed to be fully operable from the control room. To assure emergency core cooling system readiness and to protect RHR pumps, valve positions and pump running status indications are provided in the control room.

In a normal cooldown, power to safety injection pumps is locked out when RCS pressure is below 1000 psig. This is to prevent inadvertent operation of the safety injection (SI) pumps which could overpressurize the RHR system when SI pump capacity is larger than RHR relief capacity. We have requested that the applicant address these areas with regard to meeting BTP 5-1 of SRP Section 5.4.7 to show that cold shutdown can be achieved without going outside the control room.

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The cooldown time with one RHR train of 24.4 hours is acceptable. With the stated 4-hour time for cooldown from the standby to RHR conditions, the Catawba plants can be brought to cold shutdown within a reasonable period of time with or without offsite power as specified in SRP Section 5.4.7.

5.4.7.2 RHR System Isolation Requirements

The RHRS valving arrangement is designed to provide adequate protection to the residual heat removal system when the reactor coolant system is at high pressure operation.

(1) There are two separate and redundant motor-operated isolation valves (MOV) between each residual heat removal pump suction and the RCS hot legs. These valves are separately and independently interlocked to prevent valve opening until the reactor coolant system pressure falls to below 425 psig. If the valves are open, they are separately and independently interlocked to close when the reactor coolant system pressure rises above 600 psig. One MOV in each suction line is powered from Power Train B. That is, the loss of one power train will prevent opening of both suction lines and establishing normal shutdown cooling. Should this situation develop, RCS cooling via the steam system can be resumed until power is regained to the failed power train or manual action is taken. Further discussion of this valve configuration is addressed in Section * of this SER.

(2) There are two check valves and an open motor-operated valve on each RHR discharge line. The two check valves protect the system from the reactor coolant system pressure during operation. The applicant has provided design features to permit leak testing of each check valve separately during plant operation to fulfill the staff requirements for high/low pressure isolation with two check valves. This testing is further addressed in Section *.

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5.4.7.3 RHR Pressure Relief Requirements

Overpressure protection of the residual heat removal system is provided by four relief valves, one on each of the suction and discharge lines. Each suction line relief valve has a capacity of 900 gallons per minute (gpm) at 450 psig which is sufficient to discharge the flow from both charging pumps at the relief valve setpoint. Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve to relieve the maximum possible back-leakage through the valves separating the RHRS from the RCS. Each valve has a relief flow capacity of 20 gpm at a set pressure of 600 psig. The fluid discharge by the suction side relief valves is collected in the pressurizer relief tank. The fluid discharged by the discharge side relief valves is collected in the recycle holdup tank.

The applicant will provide a reverse check valve in parallel with the inner RHR suction isolation valve to provide protection against pressure increases due to heating water trapped between the two isolation valves.

5.4.7.4 RHR Pump Protection

Each of the RHR pumps has a miniflow bypass line to prevent overheating and ensure flow to the pump suction. A valve in the line is controlled by flow sensors in the pump discharge header. Pressure sensors in the discharge header provide pressure indication in the control room. We have requested that the applicant address pump protection with the RCS partially drained and that the applicant provide an alarm in the control room to alert the operator to RHR degradation.

5.4.7.5 Tests, Operational Procedures, and Support Systems

The plant preoperational and startup test program provides for demonstrating the operation of the residual heat removal system in conformance with Regulatory Guide (RG) 1.68, Initial Test Programs for Water-Cooled Reactor Power Plants, as specified in SRP Section 5.3.7, subsection III.12.

The staff has reviewed the component cooling water system to assure that sufficient cooling capability is available to the RHRS heat exchangers. The acceptability of this cooling capacity and its conformance to GDC 44, 45, and 46 are discussed in Section *.

The applicant states that the system is housed within a structure that is designed to withstand tornadoes, floods, and seismic phenomena. This area is addressed further in Section *.

The residual heat removal system capability to withstand pipe whip inside containment as required by GDC 4 and RG 1.46 is discussed in Section *. Protection against piping failures outside of containment in accordance with GDC 4 is discussed in Section *.

All residual heat removal lines, including instrument lines, have containment isolation features; their satisfaction of the requirements of GDC 56 and 57 and RG 1.11 is discussed in Section *.

Section * discusses the applicant's compliance with the requirements of Task Action Plan Item III.D.1.1 of NUREG-0737 as it relates to primary coolant sources outside of the containment.

5.4.7.6 Conclusions

The residual heat removal function is accomplished in two phases: the initial cooldown phase and the residual heat removal system operation phase. In the event of loss of offsite power, the initial phase of cooldown is accomplished by use of the auxiliary feedwater system and the atmospheric dump valves. This equipment is used to reduce the reactor coolant system temperature and pressure to values that permit operation of the RHR system. The review of the initial cooldown phase is discussed in Section * of this SER. The review of the RHR system operational phase is discussed below. The residual heat removal (RHR) system removes core decay heat and provides long-term core cooling following

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the initial phase of reactor cooldown. The scope of review of the RHR system for the Catawba plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analysis, and design performance specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RHR system and his analysis of the adequacy of those criteria and bases and the conformance of the design to these criteria and bases.

The staff concludes that, except as noted in the previous paragraphs, the design of the residual heat removal system is acceptable and meets the requirements of GDC 2, 5, 19, and 34. This conclusion is based on the following:

(1) The applicant has met the requirements of GDC 2 with respect to position C-2 of RG 1.29 concerning the seismic design of systems, structures, and components whose failure could cause an unacceptable reduction in the capability of the residual heat removal system, as discussed in Section *.

(2) The applicant has met the requirements of GDC 5 with respect to sharing of structure, systems, and components by demonstrating that such sharing does not significantly impair the ability of the residual heat removal system to perform its safety function including, in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining unit.

(3) Except as noted above, the applicant has met the requirements of GDC 19 with respect to the main control room requirements for normal operations and shutdown, and GDC 34 which specifies requirements for the residual heat removal system by meeting the regulatory position in BTP RSB 5-1.

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6.3 Emergency Core Cooling System

Section 6.3 (Emergency Core Cooling System) for Catawba Units 1 and 2 has been reviewed in accordance with the July 1981 edition of the Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800. A review of each of the areas listed in the Areas of Review section of SRP Section 6.3 was performed according to the guidelines provided in the Review Procedures section of SRP Section 6.3. Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for emergency core cooling is acceptable.

As specified in SRP Section 6.3, subsection I.2, the design of the emergency core cooling system (ECCS) was reviewed to determine that it is capable of performing all of the functions required by the design bases. The ECCS is designed to provide core cooling as well as additional shutdown capability for accidents that result in significant depressurization of the reactor coolant system (RCS). These accidents include mechanical failure of the reactor coolant system piping up to and including the double-ended break of the largest pipe, rupture of a control rod drive, spurious relief valve operation in the primary and secondary fluid systems, and breaks in the steam piping.

The principal bases for the staff's acceptance of this system are conformance to 10 CFR 50.46 and Appendix K to 10 CFR 50, and General Design Criteria (GDC) 2, 5, 17, 27, 35, 36, and 37.

The applicant states that the requirements will be met even with minimum engineered safeguards available, such as the loss of one emergency power bus, with offsite power unavailable.

6.3.1 System Design

As specified in SRP Section 6.3, subsection I.2, the design of the ECCS was reviewed to determine that it is capable of performing all of the functions required by the design bases. The ECCS design is based on the availability of a minimum of three low-pressure cold leg accumulators, the high pressure upper head injection (UHI) accumulator, one charging pump, one safety injection pump, and one residual heat removal (RHR) pump together with associated valves and piping. Following a postulated LOCA, passive (accumulators) and active (injection pumps and associated valves) systems will operate. After the inventory in the refueling water storage tank (RWST) has been depleted, long-term recirculation will be provided by taking suction from the containment sump and discharging to the RCS cold and/or hot legs. The low-pressure passive accumulator system consists of four pressure vessels partially filled with borated water and pressurized with nitrogen gas to approximately 425 psig. Fluid level, boron concentration, and nitrogen pressure can be remotely adjusted in each tank. When RCS pressure is lower than accumulator tank pressure, borated water is injected through the RCS cold legs. The UHI system consists of a borated water filled tank connected to a nitrogen tank that is pressurized to approximately 1250 psia. When the reactor vessel pressure falls below the UHI pressure, water will be injected into the top of the core.

The high-head injection system consists of two centrifugal charging pumps which provide high-pressure injection of boric acid solution into the RCS. In addition to the high-head charging pump system, two intermediate-head safety injection pumps deliver fluid to the RCS. Both high- and intermediate-head pumps are aligned to take suction from the RWST for the injection phase of their operation. Both types of pumps are manually aligned to take suction from the RHRS discharge during the recirculation mode. Low head injection is accomplished by two RHR pump subsystems taking suction from the RWST during the short-term ECCS injection phase and from the containment sump during long-term ECCS recirculation.

The RWST minimum inventory is 350,000 gal of 2000-ppm borated water. To maintain the RWST water above the temperature of boron precipitation and freezing, the applicant has provided the RWST with a heating system. The RWST vent lines and screen over the end of the line are redundantly heat traced to preclude ice blockage during freezing weather. The applicant has addressed concerns about failure of nonseismic piping in lines connected to the RWST by stating that nonseismic portions would be automatically isolated (using seismically qualified valves) upon receipt of a safety injection initiation signal.

We have requested additional information related to the sizing of the RWST and will address resolution in a supplement to this SER.

As specified in SRP Section 6.3, subsection II, the ECCS is initiated either manually or automatically on (a) low pressurizer pressure, (b) high containment pressure, or (c) low pressure in any steam line. This meets the requirements of GDC 20. The ECCS may also be manually actuated, monitored, and controlled from the control room as required by GDC 19. The ECCS is supplemented by instrumentation that will enable the operator to monitor and control the ECCS equipment following a LOCA so that adequate core cooling may be maintained. We have requested additional information from the applicant to insure the installed instrumentation provides sufficient information so the operator can maintain adequate core cooling following an assumed LOCA. The acceptability of the proposed ECCS instrumentation and controls is addressed further in Section *.

As specified in SRP Section 6.3, subsection III.3, the available net positive suction head for all the pumps in the ECCS (the safety injection, centrifugal charging, and RHR pumps) has been shown to provide adequate margin by calculations performed to meet the safety intent of Regulatory Guide 1.1, Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps.

As required in SRP Section 6.3, subection III.11, the valve arrangement on the ECCS discharge lines has been reviewed with respect to adequate isolation between the RCS and the low pressure ECCS. In some lines, this isolation is provided by two check valves in series with a closed isolation valve (high-head injection discharge, intermediate- and low-head injection discharge to the hot legs).

Other discharge lines have only two check valves in series. This arrangement is acceptable since periodic leak detection across each check valve is performed during plant operation. Test lines are provided for periodic leakage checks of

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reactor coolant past the check valves forming the reactor coolant system pressure boundaries. This is discussed further in Section *.

Containment isolation features for all ECCS lines, including instrument lines, the requirements of GDC 56 and Regulatory Guide (RG) 1.11, Instrument Lines Penetrating Primary Reactor Containment, are discussed in Section *. The effects of primary coolant sources outside containment, NUREG-0737, Item III.D.1.1, are discussed in Section *. The safety injection lines are protected from intersystem leakage by relief valves in both suction header and discharge lines. Intersystem leakage detection is described in Section * for the RHR and safety injection pump systems.

As specified in SRP Section 6.3, subsection II.B, no ECCS components are shared between units, which meets the requirements of GDC 5.

6.3.2 Evaluation of Single Failures

As specified in SRP Section 6.3, subsection II, the staff has reviewed the system description and piping and instrumentation diagrams to verify that sufficient core cooling will be provided during the initial injection phase with and without availability of offsite power, assuming a single failure. The cold leg accumulators have normally open motor-operated isolation valves in their discharge lines. One accumulator is attached to each of the RCS cold legs. These isolation valves will have control power removed to preclude inadvertent valve movement that could result in degraded accumulator performance. The upper head injection subsystem is aligned for injection, through two parallel lines with normally open isolation valves, when the primary pressure drops below the upper head injection set pressure. An inadvertent valve closure in either discharge line will not preclude upper head injection. Each upper head injection discharge line has two isolation valves in series which are closed automatically when a low level in the upper head injection accumulator is reached. Failure of a single valve to close will not prevent isolation of the upper head injection accumulator.

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Three active injection systems are available, each system having two pumps. The pumps in each system are connected to separate power buses and are powered from separate diesel generators in the event of loss of offsite power, as required by GDC 17. Thus, at least one pump in each injection train would be actuated. The high-head injection systems contain parallel valves in the suction and discharge lines, thus ensuring operability of one train even in the event that one valve fails to open. The low- and intermediate-head injection systems are normally aligned so that valve actuation is not required during the injection phase.

The staff has requested that the applicant address the spurious movement of valves whose mispositioning could cause degradation of emergency core cooling system. This includes but may not be limited to the following valves:

(1) Accumulator discharge isolation valves (spurious closure)

(2) Safety injection pumps cold leg discharge isolation valve (spurious closure)

(3) RWST to safety injection pumps suction valves (spurious closure)

(4) Safety injection pumps miniflow line isolation valve (spurious closure)

(5) Safety injection pump hot leg discharge isolation valves (spurious opening)

(6) RHR pumps hot leg discharge isolation valve (spurious opening)

(7) RHR cold leg discharge isolation valves (spurious closure).

The applicant has provided the following interlocks to address various single failures.

Cold Leg Accumulator Isolation Valves - To assure valves are open during power operation.

RHR Suction From RWST - To prevent valves from opening during post-accident recirculation operation of ECCS.

RHR Pump Discharge to CCP - To prevent flow of recirculation sump fluid to RWST, prevent possible overpressure of pipe during cooldown, permit alignment to supply pumps only during recirculation.

Containment Sump Valve - The interlocks to prevent the control room operator from opening the sump valves and flooding containment with fluid from the reactor coolant system or the RWST. The automatic features override the interlocks and open the valve if the RWST level is low and an "S" signal has been generated (this prevents the sump valve from opening and flooding containment during refueling as the RWST is emptied into the refueling cavity).

Charging Pump Normal Suction - To isolate normal charging sources after RWST is available to pumps.

RCS to RHR Isolation Valves - Interlocks to prevent flow from RCS to RWST spill of RCS to containment sump, potentially overpressuring charging pump and SI pump suction lines, spraying RCS to containment via residual spray headers. Pressure interlocks and automatic feature prevent overpressure of the RHR pump suction line.

Safety Injection Pump Miniflow - Interlocks to prevent recirculation sump fluid from being pumped to RWST.

Containment Spray Suction from RWST - To prevent spill of RWST fluid to containment sump via ND piping.

Containment Spray Suction from Sump - To prevent spill of RWST fluid to containment sump and prevent containment spray with reactor coolant.

Residual Containment Spray - To prevent residual containment spray with reactor coolant.

We require the applicant to justify the compatibility of these interlocks with the functional requirements discussed in other SER sections (e.g., 5.4.7).

The applicant addressed single failures and deadheading conditions that could cause the safety injection and charging pumps to overheat and subsequently fail by removing the automatic-isolation-on-"S"-signal of the miniflow line. We require that the applicant provide plans to improve his design with automatic features to address this concern.

The applicant has proposed a partially automatic system with operator action to switch the low-head system from the injection to the recirculation mode. The automatic function of the system opens the RHR pump suction valves from the containment sump, with operator action required to isolate the RWST. Several valves that would have to be actuated during the switchover are interlocked to other components to prevent out-of-sequence operation. SRP Section 6.3, subsection III.19, states where manual action is used in the switch to recirculation, a sufficient time (greater than 20 minutes) is available for the operator to respond. The staff has requested that the applicant address this concern and the sizing of the RWST.

The staff has reviewed the plant's capability for hot-leg injection during the recirculation phase to preclude excessive buildup of boron concentration in the pressure vessel. The staff has concluded that there is sufficient redundancy in injection lines and pumps to ensure adequate hot leg injection after 15 hours of cold leg injection. This meets the requirements of SRP Section 6.3, subsection III.6.

The applicant has addressed a single failure scenario which postulates a failure in volume control tank level instrumentation, diverting letdown away from the volume control tank, and permitting continued charging pump suction from the volume control tank, with eventual cavitation of the charging pump(s). The applicant has addressed this scenario indicating that diversion of letdown flow to a holdup tank (on high level in the volume control tank (VCT) rather than to the VCT) and automatic opening of a charging pump suction path from the RWST (on low VCT level) are both initiated independently by either of two diverse VCT level transmitters. In addition, the applicant has indicated that for this

We require the applicant to justify the compatibility of these interlocks with the functional requirements discussed in other SER sections (e.g., 5.4.7).

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During the long-term recirculation cooling phase of ECCS, leak detection is require to identify passive ECCS failures outside of containment, such as pump seal failures. The applicant has provided a system of water-level monitors. With this system, the limiting leak (assumed to be 50 gpm) would be detected and isolated within 30 minutes. The applicant has calculated that the total leakage in 30 minutes would not compromise long-term cooling. Leak rates of less than 50 gpm would result in scenarios in which the detection (alarm) time would be longer, but the time available for operator response would also be longer. We have requested that the applicant provide additional information to show that there would not be an unacceptable loss of circulating coolant inventory for this scenario.

We have requested that the applicant address nonseismic piping on miniflow lines, provide procedures for resetting the ECCS after a safety injection signal, and evaluate the effects of flooding valves and instrumentation. We will report our evaluation of these issues in a supplement to this report.

Based on staff review of the design features and with satisfactory resolution of the items discussed above, the staff concludes that the ECCS complies with the single-failure criterion of GDC 35.

6.3.5 Qualification of Emergency Core Cooling System

The ECCS design to seismic Category I requirements, in compliance with RG 1.29, Seismic Design Classification, and its housing in structures designed to withstand a safe shutdown earthquake and other natural phenomena, as required by GDC 2, are discussed in Section *. The equipment design to Quality Group B, in

*LPM to provide section numbers.

compliance with RG 1.26, Quality Group Classification and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants, is discussed in Section *.

The ECCS protection against missiles inside and outside containment by the design of suitable reinforced concrete barriers which include reinforced concrete walls and slabs (conformance to GDC 4) is discussed in Section *. The protection of the ECCS from pipe whip inside and outside of containment is discussed in Section *.

The active components of the ECCS design to function under the most severe duty loads including safe shutdown earthquake is discussed in Section *. The ECCS design to permit periodic inspection in accordance with ASME Code Section XI, which constitutes compliance with GDC 36, is discussed in Section *. This meets the intent of SRP Section 6.3, subsection III.23.c.

The ECCS incorporates two subsystems which serve other functions. The RHR system provides for decay heat removal during reactor shutdown, while at other times the RHR system is aligned for ECCS operation. The centrifugal charging pumps are utilized for maintaining the required volume of primary fluid in the RCS. On an ECCS actuation signal, the system is aligned to ECCS operation and the CVCS function is isolated. In neither case (RHR or centrifugal charging) does the normal system use impair its capability to function as an integral portion of the ECCS.

6.3.4 Testing

The applicant has committed to demonstrate the operability of the ECCS by subjecting all components to preoperational and periodic testing, as required by RG 1.68, Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors. The applicant has stated that recirculation sump tests as per RG 1.79, Preoperational Testing of Emergency Core Cooling System for Pressurized Water Reactors, will not be performed at Catawba. The applicant has referenced

*LPM to provide section numbers.

McGuire sump tests to demonstrate acceptable ECCS sump design at Catawba. We require the applicant to provide further information to justify the applicability of McGuire sump test results to Catawba. We will report our evaluation of this issue in a supplement to this report.

6.3.4.1 Preoperational Tests

One of these tests is to verify system actuation, namely, the operability of all ECCS valves initiated by the safety injection signal, the operability of all safeguard pump circuitry down through the pump breaker control circuits, and the proper operation of all valve interlocks.

Another test is to check the cold leg accumulator system and injection line to verify that the lines are free of obstructions and that the accumulator check valves and isolation valves operate correctly. The applicant will perform a low pressure blowdown of each accumulator to confirm the line is clear and check the operation of the check valves.

Two blowdown tests of the UHI system are performed: one with low pressure (about 100 psi) and one with the gas pressure in the normal operating range. The low pressure test verifies the resistance of the piping from the accumulator to the reactor vessel and allows the setpoint from the water level detectors to be determined. The high pressure blowdown test simulates the performance of the system during a large cold-leg break of the reactor coolant piping and confirms proper operation of the isolation valves.

The applicant will use the results of the preoperational tests to evaluate the hydraulic and mechanical performance of ECCS pumps delivering through the flow paths for emergency core cooling. The pumps will be operated under both miniflow (through test lines) and full-flow (through the actual piping) conditions.

The applicant has been requested to commit to (a) by measuring the flow in each pipe, make the adjustments necessary to ensure that no one branch has an unacceptably low or high resistance, (b) analyze the results to ensure there is sufficient total line resistance to prevent excessive runout of the pumps and adequate net positive suction head (NPSH) under the most limiting system alignment, (c) verify that the maximum flow rate from the test results confirms the maximum flow rate used in the NPSH calculations under the most limiting conditions and (d) confirm that the minimum acceptable flow used in the LOCA analysis is met by the measured total pump flow and a relative flow between the branch lines.

Subject to resolution of the above concerns, the staff concludes that the preoperational test program conforms to the recommendations of RG 1.68 and 1.79 and is acceptable pending successful completion of the program. Additional discussion of the preoperational test program is presented in Section * of this SER.

6.3.4.2 Periodic Component Tests

Routine periodic testing of the ECCS components and all necessary support systems at power will be performed. Valves that actuate after a LOCA are operated through a complete cycle. Pumps are operated individually in this test on their miniflow lines except the charging pumps which are tested by their normal charging function. The applicant has stated that these tests will be performed in accordance with ASME Code Section XI.

6.3.5 Performance Evaluation

The ECCS has been designed to deliver fluid to the RCS to limit the fuel cladding temperature following transients and accidents that require ECCS actuation. The ECCS is also designed to remove the decay and sensible heat during the recirculation mode. 10 CFR 50.46 lists the acceptance criteria for an ECCS. These criteria include the following:

(1) The calculated maximum fuel cladding temperature does not exceed 2200°F.

(2) The calculated total oxidation of the cladding does not exceed 0.17 times the total cladding thickness before oxidation.

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(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

In addition, 10 CFR 50.46 states: ECCS cooling performance shall be calculated in accordance with a.. acceptable model, and shall be calculated for a number of postulated loss-of-coolant accidents. Appendix K to 10 CFR 50, ECCS Evaluation Models, sets forth certain required and acceptable features of evaluation models.

The applicant has examined a spectrum of large breaks in RCS piping and these analyses indicate that the most limiting event is a double-ended cold-leg guillotine (DECLG) break with a discharge coefficient of 1.0. The applicant took credit for one train of active ECCS components, the UHI accumulator, and three of the four cold leg accumulators in the analysis. This most limiting break is different than previously reviewed UHI designs which identified a DECLG break with a discharge coefficient of 0.6. We have requested that the applicant provide the reason for this difference, justify that previous sensitivity studies apply to Catawba, provide adequate treatment of the cladding swelling and rupture model in the LOCA analysis, verify that methods used to determine cold-leg accumulator settings and assumptions are similar to previous UHI analyses and justify their applicability in light of the differences in Catawba LOCA analysis results, and provide an analysis of the transients resulting from a break in the ECCS injection lines. We have requested additional information concerning the adequacy of the ECCS during shutdown/startup situation when portions of the ECCS are isolated to verify compliance with SRP Section 6.3.22.e.

The emergency core cooling system must provide abundant core cooling to minimize fuel and clad damage in accordance with the requirements of 10 CFR 50.46. Topical Report WCAP-8479, "Westinghouse Emergency Core Cooling System Evaluation Model Application to Plants Equipped with Upper Head Injection," describes the Westinghouse calculational model for a pressurized water reactor with ice condenser containment and upper head injection systems. We have reviewed and approved the Westinghouse evaluation model for analyzing loss-of-coolant accidents in UHI plants. We require further information to justify the adequacy of the break spectrum sensitivity analyses for Catawba. We will report our findings on this issue in a supplement to this report.

Containment parameters are chosen to minimize containment pressure so that core reflood calculations are conservative. Fuel rod initial conditions are chosen to maximize clad temperature and oxidation. Calculations of core geometry are carried out past the point where temperatures are decreasing. The most limiting break with respect to peak clad temperature is the double-ended guillotine break in the pump discharge leg with a $C_D = 1.0$. The peak clad temperature is 2195°F, which is below the 2200°F limit. The limiting local and core-wide clad oxidation values calculated by the applicant were 6.9 percent and less than 0.3 percent, respectively.

The amount of bypass flow into the upper head region has been predicted by the applicant to be sufficient to maintain the upper head region at cold leg temperatures. Similar calculations performed for similar Westinghouse plants have shown good agreement with measured values of upper head temperatures. Assurance that upper head temperatures can be maintained in the cold leg temperature zone has been provided by a verified analytical technique.

6.3.5.2 Small-Break LOCA

The applicant has submitted analyses for a spectrum of small-break LOCA analyses (4-in., 6-in., 8-in.). These identify that the 8-in. break is the limiting small break, the calculated peak cladding temperature is 1218°F, the local metal-water reaction is 0.077 percent, and the core-wide oxidation is less than 0.3 percent. None of these small-break analyses were analyzed with a model that properly accounted for fuel cladding strain and rupture. Because of the magnitude of cladding temperatures involved, the error is small. A "corrected" peak cladding temperature for these small breaks would be far below that for large breaks and clearly would not be limiting.

The core geometry remains amenable to cooling throughout both types of LOCAs discussed above. The ECCS is designed to remove decay heat for an extended time following a LOCA. The staff concludes, subject to satisfactory resolution of our concerns discussed above, that the applicant's analyses of the LOCA meet the acceptance criteria and, therefore, are acceptable.

The applicant has analyzed the performance of the ECCS in accordance with the criteria set forth in Section 50.46 and Appendix K to 10 CFR 50. The staff has reviewed the applicant's evaluation and concludes that it is acceptable and meets the criteria of 10 CFR 50.46 subject to satisfactory resolution of our concerns discussed above.

6.3.5.3 Conclusions

The emergency core cooling system (ECCS) includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to transfer heat from the core following a loss-of-coolant accident. The scope of review of the ECCS for the Catawba plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analyses, and design specifications for essential components. The staff review has included the applicant's proposed design criteria and design bases for the ECCS and the manner in which the design conforms to these criteria and bases.

Pending resolution of the aforementioned concerns, the staff concludes that the design of the emergency core cooling system is acceptable and meets the requirements of GDC 2, 5, 17, 27, 35, 36, and 37. This conclusion is based on the following:

(1) The applicant has met the requirements of GDC 2 with regard to the seismic design of nonsafety systems or portions thereof which could have an adverse

effect on ECCS by meeting position C.2 of RG 1.29, as discussed in Section *.

(2) The applicant has met the requirements of GDC 5 with respect to sharing of structures, systems, and components by demonstrating that such sharing does not significantly impair the ability of the ECCS to perform its safety function including, in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining units.

(3) The applicant has met the requirements of GDC 17 with respect to providing sufficient capacity and capability to assure that (a) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, and (b) the core is cooled and vital functions are maintained in the event of postulated accidents.

(4) The applicant has met the requirements of GDC 27 with regard to providing combined reactivity control system capability to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained and the applicant's design meets the guidelines of RG 1.47.

(5) The applicant has met the requirements of GDC 35 to provide abundant cooling for ECCS by providing redundant safety-grade systems that meet the recommendations of RG 1.1.

(6) The applicant has met the requirements of GDC 36 with respect to the design of ECCS to permit appropriate periodic inspection of important components of the system.

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(7) The applicant has met the requirements of GDC 37 with respect to designing the ECCS to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation.

(8) The applicant has provided an analysis of the proposed ECCS relative to the acceptance criteria of 10 CFR Part 50, 50.46, and Appendix K to demonstrate that their ECCS designs for peak cladding temperature, maximum calculated cladding oxidation, maximum hydrogen generation, coolable core geometry, and long-term cooling are in accordance with the acceptable evaluation model.

As discussed in Section *, the applicant has met the requirements of Task Action Plan Item III.D.1.1 of NUREG-0737 which involves primary coolant sources outside of the containment.

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15 ACCIDENT ANALYSIS

The accident analyses for the Catawba Units 1 and 2 have been reviewed in accordance with the July 1981 edition of the Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800. A review of each of the areas listed in the Areas of Review portion of the appropriate SRP section was performed according to the guidelines provided in the Review Procedures portion of the appropriate SRP Section. Conformance with the acceptance criteria, except as noted for each of the sections, formed the basis for concluding that the design of the facility for each of the areas reviewed was found to be acceptable for Catawba.

15.1 General Discussion

The applicant evaluated the ability of Catawba to withstand normal and abnormal transients and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. The results of these analyses are used to show conformance with GDC 10 and 15.

During its review of the transients and accidents analyses of Sectin 15, the staff has considered GDC 21, 26, and 28 and Regulatory Guides 1.53 and 1.105 as they apply to the events analyzed to ensure that the applicable requirements have been met. For each event analyzed, conservative operating conditions were assumed, and credit was taken for minimum engineering safeguards response. For Chapter 15 events per staff request, the applicant has identified the single active component failure or operator error that is the most limiting, has provided an analysis of the incident in combination with the identified failure, and has described the long-term events and assessed the operator's role. Generic Task actions (e.g., Task A-17, Systems Interaction, and A-47, Safety Implications of Control Systems) will address related concerns. If these tasks identify additional requirements we will require compliance from Catawba. Parameters specific to individual events were conservatively selected. Two types of events were analyzed:

(1) Those incidents that might be expected to occur during the lifetime of the reactor (anticipated transients).

(2) Those incidents not expected to occur that have the potential to result in significant radioactive material release (accidents).

The nuclear feedback coefficients were conservatively chosen to produce the most adverse core response. The reactivity insertion curve, used to represent the control insertion, accounts for a stuck rod, in accordance with GDC 26. We have requested additional information concerning response times and discharge rates for various components assumed to function during the analysis of Chapter 15 events. We have also requested information regarding the allowed number of operating loops as specified in the technical specifications. We will report our evaluation of this issue in a supplement to this SER.

Review of thermal hydraulic code THINC-IV is described in Section * of this SER. The staff reviews of the FACTRAN and LOFTRAN codes have progressed to the point that there is reasonable assurance that analyses results dependent on the codes will not be appreciably altered by any revisions that may be required by the staff. For some events analyzed in Chapter 15, the applicant utilized an improved thermal design method (described in Section *). We have requested that the applicant clearly identify the incidents for which this method is utilized and show that implementation of this method conforms to appropriate restrictions and limitations. We will report our evaluation of this issue in a supplement to the SER.

The applicant accounts for variations in initial conditions by making the following assumptions as appropriate for the event being considered:

1. Core power, 3427 MWT, +2 percent

2. Average reactor vessel temperature (T_{avg}), 590.8 ± 4.0°F

3. Pressure (at pressurizer), 2250 ± 30 psia.

Pending a satisfactory response to the aforementioned concerns, the staff concludes that the assumptions for initial conditions are acceptable because

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they are conservatively applied to produce the most adverse effects. For transients and accidents used to verify the ESF design, the applicant has utilized the safeguards power design value of 3581 MW(t).

15.2 Normal Operation and Operational Transients

The applicant has analyzed several events expected to occur one or more times in the life of the plant. A number of transients can be expected to occur with moderate frequency as a result of equipment malfunctions or operator error in the course of refueling and power operation during the plant lifetime. Specific events were reviewed to ensure conformance with the acceptance criteria provided in the Standard Review Plan (SRP).

The acceptance criteria for transients of moderate frequency in the Standard Review Plan include the following considerations:

(1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of design values (Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code).

(2) Fuel clad integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) will remain above the 95/95 DNBR limit for PWRs. (The 95/95 criterion discussed in Section * provides a 95 percent probability, at a 95 percent confidence level, that no fuel rod in the core experiences a departure for nucleate boiling.)

(3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

(4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than

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fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

Conformance with SRP acceptance criteria constitutes compliance with GDC 10, 15 and 26 of Appendix A to 10 CFR 50.

The transients analyzed are protected by the following reactor trips:

(1) Power range high neutron flux

(2) High pressure

(3) Low pressure

(4) Over power ΔT

(5) Overtemperature ΔT

(6) Low coolant flow

(7) Pump undervoltage/underfrequency

(8) Low steam generator water level

(9) High steam generator water level

Time delays to trip, calculated for each trip signal, are included in the analyses. See Section * of this SER for a discussion of the staff review of reactivity control system functional design.

All of the transients which are expected to occur with moderate frequency can be grouped according to, (a) increase or decrease in heat removal by the secondary system, (b) decrease in reactor coolant flow rate, (c) reactivity and

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power distribution anomalies, and (d) increase or decrease in reactor coolant inventory. Design-basis accidents have been evaluated separately as indicated in Section *.

15.2.1 Increase in Heat Removal by the Secondary System

The applicant has analyzed the following events that produced increased heat removal by the secondary system:

(1) Decrease in feedwater temperature (SRP Section 15.1.1),

(2) Increase in feedwater flow (SRP Section 15.1.2),

(3) Excessive increase in steam flow (SRP Section 15.1.3), and

(4) Inadvertent opening of a steam generator relief valve or safety valve (SRP Section 15.1.4).

The transient which is most limiting of these with respect to fuel performance is the excessive increase in steam flow for the case with minimum moderator feedback and automatic reactor control. The reactor does not trip and the plant reaches a stabilized condition rapidly following the load increase.

The transient which is most limiting of these with respect to the peak pressure is the increase in feedwater flow transient. The applicant has calculated a peak pressure of 2390 psia during this transient.

15.2.2 Decrease in Heat Removal by the Secondary System

The applicant has analyzed the following events which result in a decrease in heat removal by the secondary system:

(1) Loss of external load (SRP Section 15.2.1),

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(2) Turbine trip (SRP Section 15.2.2),

(3) Loss of condenser vacuum (SRP Section 15.2.3),

(4) Inadvertent closure of main steam isolation valves (SRP Section 15.2.4),

(5) Steam pressure regulator failure (SRP Section 15.2.5),

(6) Loss of nonemergency power to the station auxiliaries (SRP Section 15.2.6),

(7) Loss of normal feedwater flow (SRP Section 15.2.7).

Plant transients which result in an unplanned decrease in heat removal by the secondary system that might be expected to occur with moderate frequency are identified in the above list. All these postulated transients have been reviewed. It was found that the most limiting event in this group of events in regard to the maximum pressure within the reactor coolant and main steam systems was the turbine trip at full power without credit taken for the presurizer spray, PORVs, or steam dump. The reactor is tripped on the high pressurizer pressure signal and the peak pressure during the transient is 2550 psia, well below the ASME requirements for maximum pressure to be limited to 110 percent of design pressure.

The most limiting event in regard to fuel performance is the loss of nonemergency AC power to the station auxiliaries transient. In this transient, the loss of offite power is closely followed by a turbine trip and reactor trip. The reactor trip occurs on steam geneator low-low water level. The emergency feedwater system is automatically started. Since only safety grade equipment is used to mitigate the event, residual heat is removed through the steam generator safety valves. The minimum DNBR is approximately 1.7 during the transient.

15.2.3 Decrease in Reactor Coolant Flow Rate

The applicant has analyzed the total loss of forced reactor coolant flow and the partial loss of forced reactor coolant flow events. These events are reviewed using the review procedures and acceptance criteria set forth in SRP Section 15.3.1 and 15.3.2.

The loss of off-site power and resulting loss of all forced coolant flow through the reactor core is the most limiting and causes an increase in the average coolant temperature and a decrease in the margin to DNB. The reactor is tripped from an undervoltage trip monitoring the RCP power supply and a minimum DNBR of 1.55 is reached 4 seconds into the transient. The maximum calculated pressurizer pressure is 2450 psia during the transient.

15.2.4 Reactivity and Power Distribution Anomalies

15.2.4.1 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

In Section 15.4.4 of the FSAR, the applicant provides the results of an analysis for startup of an inactive reactor coolant pump event. This event is reviewed using the review procedures and acceptance criteria set forth in SRP Section 15.4.4.

During the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. Reactivity addition for the inactive loop startup event is due to the decrease in core inlet water temperature. The maximum calculated pressurizer pressure is 2440 psia and the minimum DNBR is above 2.5 during the transient.

15.2.4.2 Inadvertent Boron Dilution

Section 15.4.6 of the Standard Review Plan requires that at least 15 minutes is available from the time the operator is made aware of an unplanned boron dilution event to the time a loss of shutdown margin occurs during power operation, startup, hot standby, hot shutdown, and cold shutdown. Thirty minutes warning is required during refueling. The staff has requested that control room alarms be available to alert the operating staff to boron dilution events in all modes of operation. The staff requires that the applicant provide an analysis for all possible boron dilution events in each of the six operational modes and confirm that time intervals which meet the SRP criteria from the time of the first alarm to the time when the core would go critical is available. Also, technical specifications should be established to restrict when alarms can be taken out of service. We will report our evaluation of these issues in a supplement to this SER.

We require that the applicant show that equipment used to mitigate this event meets single failure criteria. We also requested that the applicant describe the model used in the analysis of boron dilution events and discuss the conservatism incorporated into this model. We will report our evaluation of this issue in a supplement to this SER.

15.2.5 Increase in Reactor Coolant System Inventory

The applicant has analyzed the following events that result in increase in the primary system inventory:

(1) Actuation of emergency core cooling system (SRP Section 15.5.1)

(2) Chemical and volume control system malfunction (SRP Section 15.5.2)

Emergency core cooling system operation could be initiated by a spurious signal or operator error. Reactor trip occurs due to low pressurizer pressure. The reactor pressure decreases during the initial phase of the transient and reaches the peak pressure of 2350 at 175 seconds into the transient. The DNBR never drops below its initial value.

The applicant's evaluation of the chemical and volume control system malfunction event is presented in Section 15.4.6 and the staff evaluation is addressed in Section 15.2.4.2 of this SER.

15.2.6 Decrease in Reactor Coolant Inventory

In Section 15.6.1 of the FSAR, the applicant provides the results of an analysis for inadvertent opening of a pressurizer safety or relief valve.

During this event, nuclear power is maintained at the initial value until reactor trip occurs on low pressurizer pressure. The DNBR decreases initially, but increases rapidly following the trip. The minimum DNBR of 2.0 occurred at 35 seconds into the transient. The RCS pressure decreases throughout the transient.

15.3 Design-Basis Accidents

The staff has reviewed the postulated events with regard to the facility design basis. These events have been classified in the Standard Review Plan as postulated accidents. The acceptance criteria specified in the SRP for evaluation of the consequences of the postulated accidents include the following:

(1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures, except that higher calculated pressures may be permitted for very low probability events (<120 percent of design).

(2) The potential for core damage should be evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 limit discussed in SRP Section 4.4. If the DNBR falls below these values, fuel damage (rod perforation) should be assumed unless it can be shown, based on an acceptable fuel damage model, that no fuel failure results. If fuel damage is calculated to occur, it should be of sufficiently limited extent so that the core will remain in place and geometrically intact with no loss of core cooling capability.

(3) Any activity release must be such that the calculated doses at the site boundary are within the guidelines of 10 CFR 100 (see Section *). Conformance with the SRP acceptance criteria constitutes compliance with GDC 27, 28, and 31.

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Other aspects of the staff review included evaluation of protection against conditions which might lead to brittle fracture of the reactor system pressure boundary during low-temperature operation for compliance with GDC 31 (see SER Section *). Staff review of emergency core cooling system functional design for compliance with GDC 35 is discussed in Section * of this SER. The staff coordinated its review of Chapter 15 events with the review of the auxiliary feedwater system. Section * of the SER discusses compliance of the AFW design with the requirements in Item II.E.1.1 of NUREG-0737 and Section * discusses compliance with Item II.E.1.2.

In the analysis of the events, the applicant investigated a broad spectrum of related events to determine the bounding case, including the worst single active failure unless otherwise noted. Sensitivity studies were performed to identify parameters for initial conditions and appropriate credit for systems and their performance during the limiting events in terms of protection of various barriers.

15.3.1 Loss of Coolant Accident (SRP Section 15.6.5)

The applicant has analyzed the double-ended cold-leg guillotine (DECLG) break with a discharge coefficient of 1.0 as the most limiting large break LOCA. In this analysis peak clad temperature reached is 2195°F. For the small break LOCA the applicant analyzed 4 in., 6 in., and 8 in. diameter breaks. The results show that the 8 in. diameter break is the worst case small break and it results in a peak clad temperature of 1218°F. Only safety grade equipment is assumed to mitigate the accident.

Pending resolution of the concerns discussed in Sections 6.3 and 15.1 of this SER, the staff concludes that the loss-of-coolant analysis resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary is acceptable and meets the relevant requirements of 10 CFR Part 50.46, and Appendix K, GDC 35, and 10 CFR Part 100. This conclusion is based on the following discussion.

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* LPM to provide section numbers.

The applicant has performed analyses of the performance of the emergency core cooling system (ECCS) in accordance with the Commission's regulations (10 CFR Part 50, 50.46, and Appendix K to 10 CFR Part 50). The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model which had been previously reviewed and approved by the staff as described in NUREG-0390 and Safety Evaluations Reports for licensing the Sequoyah (NUREG-0011) and McGuire (NUREG-0422) plants. The results of the analyses show that the ECCS satisfy the following criteria:

(1) The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

(2) The calculated maximum local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

(6) The applicant has met the requirements of TMI Action Plan Items II.K.3.5, II.K.3.25, II.K.3.30, and II.K.3.31.

Pending resolution of the aforementioned concerns, the staff concludes that the calculated performance of the emergency core cooling system following a postulated loss-of-coolant accident and the conservatively calculated radiological consequences of such an accident conform to the Commission's regulations and to applicable regulatory guides and staff technical positions and, the ECCS is considered acceptable.

15.3.2 Steamline Rupture (SRP Section 15.1.5)

The applicant has submitted analyses of postulated steamline breaks that show no fuel failure attributed to the accident. These results are similar to those obtained for previously reviewed Westinghouse four-loop plants.

A postulated double-ended rupture at zero power and no decay heat with and without offsite power were analyzed. The applicant referenced WCAP-9226 as justification for this selection. WCAP-9226 is currently under review by the staff. The review of WCAP-9226 has progressed to the point that there is reasonable assurance that analyses results will not be appreciably altered by any revisions that may be required by the staff. The double-ended rupture would cause an increase in reactivity due to the decrease in reactor coolant temperature. The most reactive control rod assembly was assumed to be fully withdrawn. The worst single failure, which is the loss of one safety injection train, was assumed in the analysis. Credit was taken for operator action within 10 minutes to control the high head pump to reestablish normal pressure control and to isolate the affected steam generator. Although a return to criticality occurs, there is no fuel damage since the minimum DNBR ratio is greater than 1.3.

The staff concludes that the consequences of postulated steam line breaks meet the relevant requirements set forth in the General Design Criteria 27, 18, 31, and 35 regarding control rod insertability and core coolability and TMI Action Plan Items. This conclusion is based upon the following:

(1) The applicant has met the requirements of GDC 27 and 28 by demonstrating tht the resultant fuel damage was limited such that control rod insertability would be maintained, and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (DNBR) experienced by any fuel rod was > 1.30, resulting in no rods experiencing cladding perforation.

(2) The applicant has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(3) The applicant has met the requirements of GDC 35 with respect to demonstrating the adequacy of the emergency cooling systems to provide abundant core cooling and reactivity control (via boron injection), pending resolution of concerns addressed in Section 6.3.

(4) The analyses and effects of steam line break accidents inside and outside containment, during various modes of operation with and without offsite power, have been reviewed and were evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

(5) The parameters used as input to this model were reviewed and found to be suitably conservative.

(6) The applicant has met the requirements of Task Action Plan Items II.E.1 and II.E.1.2, with respect to demonstrating the adequacy of the auxiliary feedwater design to remove decay heat following steam system piping failures, as discussed in Section *.

(7) The applicant has met the requirements of Task Action Plan Item II.K.3.25 with respect to demonstrating the integrity and operation of the reactor coolant pumps to withstand the postulated accident.

(8) The applicant has met the requirements of Task Action Plan Item II.K.3.5 with respect to the operation and tripping of the reactor coolant pumps.

15.3.3 Feedwater System Pipe Break (SRP Section 15.2.8)

The applicant has provided a feedwater line break analysis for Catawba using assumptions that would minimize secondary system heat removal capability, maximize heat addition to the primary system coolant, and maximize the

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* LPM to provide section numbers.

calculated primary system pressure. A double-ended rupture of the largest feedwater line was assumed. The turbine driven auxiliary feedwater pump is assumed to fail and all flow from one of the two motor driven pumps spills out through the break. Two cases were analyzed: one with offsite power and the other without offsite power.

The system code used to perform these analyses is LOFTRAN (discussed in Section *). Emergency feedwater flow is supplied to only two intact steam generators. This is sufficient feedwater flow to adequately remove the residual heat after reactor shutdown. The use of only safety grade equipment was assumed to mitigate this accident. No fuel damage was calculated to occur, and the peak calculated pressurizer pressure was about 2510 psia.

The staff concludes that the consequences of postulated feedwater line breaks meet the requirements set forth in the General Design Criteria 27, 28, 31 and 35 regarding control rod insertability and core coolability, 10 CFR Part 100 guidelines regarding radiological dose at the site boundary, and applicable TMI Action Items. This conclusion is based upon the following:

(1) The applicant has met the requirements of GDC 27 and 28 by demonstrating that the resultant fuel damage was minimal, control rod inserbability would be maintained and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ration (MDNBR) experienced by any fuel rod was 1.8, resulting in no rods experiencing clad perforation.

(2) The applicant has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(3) The applicant has met the requirements of GDC 35 with respect to demonstrating the adequacy of the emergency cooling systems to provide abundant core cooling and reactivity control (via boron injection), pending resolution of concerns addressed in Section 6.3.

(4) The analyses and effects of feedwater line break accidents inside and outside containment, during various modes of operation and with and without offsite power, have been reviewed and evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

(5) The parameters used as input to this model were reviewed and found to be suitably conservative.

(6) The applicant met the requirements of TMI Action Plan Items II.E.1 and II.E.1.2 with respect to demonstrating the adequacy of the auxiliary feedwater design to remove decay heat following feedwater piping failures, as discussed in Section *.

(7) The applicant met the requirements of TMI Action Plan Item II.I.3.25 with respect to demonstrating the integrity and operation of the reactor coolant pumps to withstand the postulated accident.

(8) The applicant met the requirements of TMI Action Plan Item II.K.3.5 with respect to the operation and tripping of the reactor coolant pumps.

15.3.4 Rector Pump Rotor Seizure and Shaft Break (SRP 15.3.3/15.3.4)

The applicant's analyses for locked reactor coolant pump rotor and a sheared reactor coolant pump shaft assumes the availability of offsite power throughout the event. In accordance with Standard Review Plan 15.3.3, 15.3.4, adn GDC 17, we require that these events be analyzed assuming turbine trip and consequential loss of offsite power to the plant auxiliaries and resulting coastdown of all undamaged pumps. Appropriate delay times may be assumed for loss of offsite power if suitably justified.

The event should also be analyzed assuming the worst single failure of a safety system active component. Maximum technical specification primary

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*LPM to provide section numbers.

system activity and steam generator tube leakage at the rate specified in the Technical Specifications should be assumed. The results of the analyses should demonstrate that offsite doses following the accident are less than the 10 CFR 100 guidelines values.

In response to the staff request, the applicant has committed to reanalyze the event. We will report our resolution of this issue in a supplement to this SER.