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Docket Nos.: 50-413/414



Mr. William O. Parker, Jr.
Vice President - Steam Production
P.O. Box 33189
Charlotte, North Carolina 28242

Dear Mr. Parker:

Subject: Transmittal of Preliminary Draft SERs - Catawba Nuclear Station

Enclosed for your review and comment are the preliminary draft SERs for the following areas:

1. Auxiliary Systems (Enclosure 1)
2. Quality Assurance (Enclosure 2)
3. Materials Engineering - Component Integrity (Enclosure 3)

In addition, Enclosures 4 and 5 contain items requiring further evaluation in the Auxiliary Systems and Materials Engineering areas. 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 these items. 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 the items requiring further evaluation, are requested within four weeks of this letter.

Sincerely,

Original signed by
Robert L. Tedesco

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosures:
As stated

cc: See next page
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OFFICE	DL:LB#4	LA:DL:LB#4	DE:LB#4	AD:DL		
SURNAME	KJabbour:eb	MDuncan	EAdensam	RTedesco		
DATE	4/19/82	4/19/82	4/23/82	4/23/82		

CATAWBA

Mr. William O. Parker
Vice President - Steam Production
Duke Power Company
P.O. Box 33189
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.
314 Pall Mall
Columbia, South Carolina 29201

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

ENCLOSURE 1

SAFETY EVALUATION REPORT
CATAWBA NUCLEAR STATION
AUXILIARY SYSTEMS BRANCH

The Catawba Nuclear Station was reviewed in accordance with the July 1981 edition of the, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP, NUREG-0800). This Safety Evaluation Report contains the result of the review for the sections of NUREG-0800 which the Auxiliary Systems Branch has primary responsibility. These sections are as follows: 3.4.1, 3.5.1.1, 3.5.1.2, 3.5.1.4, 3.5.2, 3.6.1, 4.6, 5.2.5, 5.4.11, 9.1.1 thru 9.1.5, 9.2.1 thru 9.2.6, 9.3.1, 9.3.3, 9.4.1 thru 9.4.5, 10.3.1, 10.4.5, 10.4.7, and 10.4.9. Conformance with the acceptance criteria listed in the SRP sections forms the basis for concluding that the above SRP Sections satisfy the applicable regulations of 10 CFR 50.

3.4.1 Water Level (Flood) Design

In order to assure conformance with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" we reviewed the overall plant flood protection design including all systems and components whose failure due to flooding could prevent safe shutdown of the plant or result in the uncontrolled release of significant radioactivity. The applicant has provided protection from inundation and the static and dynamic effects for safety-related structures, systems, and components by the "Dry Site" method as defined in Regulatory Guide 1.102, "Flood Protection For Nuclear Power Plants," as described below.

The probable maximum flood (PMF) level has been determined to be 592.4 ft. above mean sea level (msl), in accordance with the guidelines of Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants." (Refer to Section 2.4 of this SER for further discussion on flooding). The plant grade is at elevation 593.5 ft. msl. The plant is provided with a surfact water drainage system that is designed and constructed to seismic Category I criteria and provides protection for all safety-related equipment from flooding. ~~However, in FSAR Section 9.2.1.2.2~~
~~a PMF elevation of 592.6 ft. msl is discussed. Flood level at~~
~~this elevation would be unacceptable as it would reach the ground~~
~~floor elevation of safety related structures (597.0 ft. msl).~~
~~The applicant has been requested to resolve this discrepancy.~~
~~We will report resolution of our concern in a supplement to this~~
~~ser.~~ Within plant structures safety-related equipment is protected against flooding from failures in tanks, vessels and fluid piping systems as identified in the guidelines of Branch Technical Posi-

tion ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," by equipment location and drainage as described under Section 9.3.3 of this SER.

Based on our review of the design criteria and bases, and safety classification of safety-related systems, structures, and components necessary for a safe plant shutdown during and following flood conditions we conclude that the design of the facility for flood protection conforms to the requirements of General Design Criterion 2 with respect to protection against natural phenomena and conforms to the guidelines of Regulatory Guides 1.59 and 1.102 concerning flood protection and is, therefore acceptable. ~~and is~~
~~verification by the licensee that the PMP elevation is 992.~~

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

Protection of plant structures, systems and components outside containment that are required for safe plant shutdown, against postulated internally generated missiles associated with plant operation, such as missiles generated by rotating or pressurized equipment as identified in the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," is provided by any one or a combination of compartmentalization, barriers, separation, and equipment design. The primary means utilized by the applicant to provide protection to safety-related equipment from damage resulting from internally generated missiles is through the use of plant physical arrangement and by the design adequacy of plant equipment to prevent missile generation. Safety-related systems are physically separated from nonsafety-related systems, and redundant components of safety-related systems are physically separated such that a potential missile could not damage both trains of the safety-related system. Stored fuel is protected from damage by internal missiles which could result in radioactive release as identified in the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," by the fuel pool walls and by locating new and spent fuel in an area with no high-energy piping system or rotating machinery in the vicinity.

The applicant has provided an evaluation of potential missile sources from rotating equipment failures and high-energy systems on the basis that a single failure in a system component could result in potential missiles. This evaluation included typical internal

missile sources such as valve stems, valve bonnets, instrument wells, and pump impellers. Based on the design of these components the applicant concluded that none of these are credible missiles. Remote location and separation of safety-related systems trains provides further protection against the effects of potential internally generated missiles. Although the applicant has provided information which indicates that no credible missiles should be postulated outside containment, we have requested that the applicant further demonstrate adequate protection for safety related equipment by responding to our Q410.4. This Question requests that missile sources be postulated and adequate separation and barriers for safety related equipment be verified. Protection of safety-related equipment and stored fuel from the effects of turbine missiles including compliance with the guidelines of Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," is discussed in Section 3.5.1.3 of this SER.

We have reviewed the adequacy of the applicant's design to maintain the capability for a safe plant shutdown in the event of internally generated missiles outside containment. Based on the above, we cannot conclude that design is in conformance with the requirements of General Design Criterion 4 with respect to missile protection nor that it meets the guidelines of Regulatory Guide 1.13 concerning protection of spent fuel from internally generated missiles until the applicant provides a satisfactory response to Q410.4.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Protection of plant structures, systems and component inside containment that are required for safe plant shutdown against postulated internally generated missiles associated with plant operation such as missiles generated by rotating or pressurized equipment as identified in the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," is provided by any one or a combination of barriers, separation, and equipment design. The primary means of providing protection to safety-related equipment from damage resulting from internally generated missiles is provided by shield walls and separation within the containment.

The applicant has provided an evaluation of potential missile sources inside containment. The only credible potential missile sources identified are from high-energy systems as follows:

1. Reactor Vessel - control rod drive mechanism housing plug, drive shaft, and drive shaft and drive mechanisms latched together.
2. Pressurizer:
 - a. safety valves
 - b. spray valves
 - c. relief valves
 - d. relief isolation valves
 - e. heaters
 - f. instrument wells
3. Main coolant piping temperature nozzle with resistance temperature detector.
4. Reactor coolant pump thermowell with resistance temperature detector.

Characteristics were determined for each of the above potential missiles. The applicant's analysis verified that structures, shields, or barriers, and equipment orientation, provide protection for safety-related equipment from the above primary missiles and any secondary missiles generated by their impact or that these missiles are of insufficient energy to cause unacceptable impact or that these missiles are of insufficient energy to cause unacceptable damage. The applicant's analysis also confirmed that no nonseismically supported components within the containment result in gravitational missiles with potentially adverse consequences to safety-related equipment. We have reviewed the applicant's analysis and concur with the applicant's assumptions and evaluation for potential missiles inside containment.

The applicant has analyzed the potential for the reactor coolant pump flywheel to become a missile source as a result of flywheel failures in accordance with the guidelines of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity." The applicant's analysis evaluated the materials integrity of the flywheel under assumed overspeed conditions of the pump as a result of pipe break at the pump discharge. The analysis verified that failure of the flywheel does not occur and thus it is not a postulated missile source. Refer to Section 5.4.1.1 of this SER for further discussion of reactor coolant pump flywheel integrity and compliance with the criteria of Regulatory Guide 1.14.

We have reviewed the adequacy of the applicant's design to maintain the capability for a safe plant shutdown in the event of internally generated missiles inside containment. Based on the above, we conclude that through the use of barriers, separation, and equipment design, the design is in conformance with the requirements of General Design Criterion 4 with respect to missile protection and the guidelines of Regulatory Guide 1.14 concerning reactor coolant pump flywheel integrity and is, therefore, acceptable.

3.5.1.4 Missiles Generated by Natural Phenomena

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components essential to safety to be designed to withstand the effects of natural phenomena, and General Design Criterion 4, "Environmental and Missile Design Bases," requires that these same plant features be protected against missiles. Tornado generated missiles are the only missiles arising from natural phenomena that are of concern. The applicant has identified his plant site in tornado Region I as defined in Regulatory Guide 1.76, ^{Design} ~~Design~~ Basis Tornado for Nuclear Power Plants," and he has selected as the design basis missiles those given ^{for} ~~in~~ Spectrum II of Standard Review Plan (SRP) 3.5.1.4 (Revision 1). The spectrum includes the ^{weight} ~~weight~~, velocity, kinetic energy, impact area, penetration depth and minimum available concrete thickness providing protection ^{and} ~~is~~ in accordance with the Regulatory Guide 1.76. We have reviewed this ^{Spectrum} ~~spectrum~~ and conclude that it is representative of missiles at the site and is, therefore, acceptable. Discussion of the ^{protection} ~~protection~~ (barriers and structures) afforded safety-related equipment from the identified tornado missiles including compliance with the guidelines of Regulatory Guide 1.117, "Tornado Design Classification" is provided in Section 3.5.2 of this SER.

Discussion of the adequacy of barriers and structures designed to withstand the effects of the identified tornado missiles is provided in Section 3.5.3 of this SER. Based on our review of the tornado missile spectrum, we conclude that ^{the spectrum} ~~it~~ was properly selected and

meets the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and missiles and the guidelines of Regulatory Guides 1.76 and 1.117 with respect to identification of missiles generated by natural phenomena and is, therefore acceptable.

3.5.2 Structures, Systems and Components to be Protected from Externally Generated Missiles

General Design Criterion 4, "Environmental and Missile Design Bases," requires that all structures, systems and components essential to the safety of the plant be protected from the effects of externally generated missiles. The spectrum of tornado missiles is discussed in Section 3.5.1.4 of this SER. The applicant has identified all safety related structures, systems and components requiring protection from externally generated missiles. All safety related structures (including containment, ^{the} auxiliary building, ^{and the} fuel handling building) are designed to withstand postulated tornado generated missiles without damage to safety related equipment. However, we are unable to verify that all safety-related components and systems are protected in their entirety by safety-related structures. The applicant has provided insufficient ^{information regarding} ~~description of~~ components exposed to the outside environment such as ventilation system and intake and exhausts, emergency diesel exhausts, freight doors in safety related structures, system and components both safety and non-safety related. We cannot verify that the requirements of General Design Criterion 4 with respect to missile protection and the specific guidance of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," and 1.27, "Ultimate Heat Sink for Nuclear Power Plants," and 1.117, "Tornado Design Classification" concerning tornado missile protection for safety related structures, systems and components including stored fuel and the ultimate heat sink are met. We will report resolution of our concern in a supplement to this SER.

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

General Design Criterion 4, "Environmental and Missile Design Bases," requires that systems and components important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures. Conformance with the recommendations of Branch Technical Position (BTP) ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," serves as the basis for this evaluation. Based on the tendering date of the application for a construction permit, the applicant is required to either: (a) follow the guidance of the A. Giambusso Letter of December 1972 and position B.3 of BTP ASB 3-1, or (b) use BTP ASB 3-1 in its entirety. The applicant has not applied either of these alternatives in a consistent manner. By Q410.7 the applicant has been requested to specify the guidance applied in the analysis.

The applicant has not provided sufficient analysis of the effects on safety-related systems of failures in any high- or moderate-energy piping system. He has not provided the information to correlate interactions between systems. In addition, the applicant has not presented the transient pressure, temperature and humidity effects of postulated pipe ruptures in areas vulnerable to extreme environmental conditions, following pipe breaks such as the doghouse and steam tunnels. ~~general~~, The applicant has provide sufficient environmental information for normal operating conditions.

(Structure housing main steam isolation valves and safety valves)

The applicant has not responded to ^{our} requests for additional information to satisfy the above concerns. (See Q410.6, 410.7 and 410.8). We are unable to adequately evaluate this section of the FSAR until these requests are addressed.

We therefore cannot conclude that the requirements of General Design Criterion 4 and the guidelines of BTP ASB 3-1 are satisfied and therefore, the design for protection against postulated piping failures in fluid systems outside containment is not acceptable. We will report resolution of our concern in a supplement to this SER.

Functional Design of Reactivity Control Systems

The functional design of the Reactivity Control Systems for the facility have been reviewed to confirm that they meet the various reactivity control conditions for all modes of operation. These are:

1. The capability to operate in the unrodded, critical, full power mode throughout plant life.
2. The capability to vary power level from full power to hot shutdown and assure control of power distributions within acceptable limits at any power level.
3. The capability to shut down the reactor in a manner sufficient to mitigate the effects of postulated events discussed in Section 15.0 of this SER.

The control rod drive system (CRDS), the safety injection system (SIS) and the chemical and volume control system (CVCS) constitute the reactivity control systems.

The CRDS is composed of control rod drive mechanisms (CRDMs) to which the rod cluster control assemblies (RCCAs) are attached. The CRDM is a magnetically operated jack. The magnetic jack is an arrangement of three electromagnets which are energized in a controlled

sequence to insert or withdraw RCCAs in discrete steps. The RCCAs are divided into two categories: control and shutdown.

The control category RCCAs may be automatically inserted or withdrawn to compensate for changes in reactivity associated with power level changes and power distribution, variations in moderator temperature or changes in boron concentration. The shutdown category RCCAs, which are fully withdrawn during power operations, are used solely to insert large amounts of negative reactivity to shut down the reactor. Refer to Section 4.3 of this SER for further discussions on these features.

The RCCAs are the primary shutdown mechanisms for normal operation, accidents and transients. They insert automatically upon a reactor trip signal. Concentrated boric acid solution is injected by the SIS in the event of normal feedwater flow, steam generator rupture, or RCCA ejection, thereby complying with the requirements of General Design Criterion 29, "Protection Against Anticipated Operational Occurrences."

Failure of electrical power to an RCCA will result in the insertion of that assembly as well as shearing of the connection between the rod cluster control assembly and control rod drive mechanism. Single failure of a rod cluster control assembly is considered in transient and accident analyses which includes the most reactive rod cluster control assembly stuck outside the core. Analysis of accident withdrawal of a rod cluster control assembly is found to have acceptable results. This conforms to the requirements of General Design Criteria 23, "Protection System Failure Modes," and 25, "Protection System Requirements for Reactivity Control Malfunctions."

The SIS is automatically actuated to inject borated water into the reactor coolant system upon receipt of a safety injection actuation signal (SIAS). The SIS pumps take suction from the refueling water storage tank (RWST). The SIS is discussed further in Section 6.3 of this SER.

The CVCS is designed to accommodate slow or long-term reactivity changes such as those caused by fuel burnup or by variation in the xenon concentration resulting from changes in reactor power level. The CVCS is used to control reactivity by adjusting the dissolved boron concentration in the reactor coolant system. The boron

concentration is controlled to obtain optimum RCCA positioning, to compensate for reactivity changes associated with variations in coolant temperature, core burnup, xenon concentration, and to provide shutdown margin for maintenance and refueling operations or emergencies. A portion of the CVCS (the charging pumps, the boric acid pump discharge, and the boric acid makeup tanks) injects a concentrated boron solution into the reactor coolant system to help ensure plant shutdown in the event of an SIAS. The boric acid concentration in the reactor coolant system is controlled by the charging and letdown portions of the CVCS.

The CVCS can maintain the reactivity of the reactor within required bounds by means of the automatic makeup system to replace minor leakage without significantly changing the boron concentration in the reactor coolant system. Dilution of the reactor coolant system boron concentration required for the reactivity losses occurring as a result of fuel depletion may be accomplished by manual action. The CVCS is discussed further in Section 9.3.4 of this SER. The concentration of boron in the reactor coolant system is changed under the following conditions:

1. Start-up - boron concentration decreased to compensate for moderator temperature and power increase.
2. Load-follow - boron concentration increased or decreased to compensate for xenon transients following load changes.
3. Fuel burnup - boron concentration decreased to compensate for burnup.
4. Cold shutdown - boron concentration increased to compensate for increased moderator density due to cooldown.

Soluble poison concentration is used to control slow operating reactivity changes. If necessary, RCCA movement can also be used to accommodate such changes, but assembly insertion is used mainly to control anticipated operational occurrences even with a single malfunction, such as a stuck rod. In either case, fuel design limits are not exceeded. The soluble poison control is capable of maintaining the core subcritical under conditions of cold shutdown, which conforms to the requirements of General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

The reactivity control systems, including the addition of concentrated boric acid solution by the SIS, are capable of controlling all anticipated operational changes, transients, and accidents, except possibly the small break loss-of-coolant accidents. For further information on performance of the charging and borating portions of the CVCS with respect to small-break loss-of-coolant accident. Refer to Section 6.3 and 15.3 of this SER. All accidents are calculated with the assumption that the most reactive RCCS is stuck out and cannot be inserted, which complies with the requirements of General Design Criteria 27, "Combined Reactivity Control Systems Capability."

Compliance with the requirements of General Design Criterion 28, "Reactivity Limits," is discussed in Sections 4.3 and 15.0 of this SER.

Based on our review, we conclude that the reactivity control systems' functional design meets the requirements of General Design Criteria 23, 25, 26, 27, 28 and 29 with respect to its fail-safe design, malfunction protection design, redundancy and capability, combined systems capability, reactivity limits and protection against anticipated operational occurrences, and is, therefore, acceptable.

5.2.5. Detection of Leakage Through Reactor Coolant Pressure

Boundary

A limited amount of leakage is to be expected from components forming the reactor coolant pressure boundary (RCPB). Means are provided for detecting and identifying this leakage in accordance with the requirements of General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary." Leakage is classified into two types--identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges are not completely leak-tight. Since this leakage is expected it is considered identified leakage and is monitored, limited, and separated from other leakage (unidentified) by directing it to closed systems as identified in the guidelines of Position C.1 of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

Sources, disposition and indication of identified leakage are:

1. Reactor coolant pump seal number 1 leakage will result in excessive flow to the chemical and volume control system. The rate of this leakage from each

pump is indicated and alarmed at the main control board. Leakage through the number 2 seal results in increased flow to the reactor coolant drain tank. This leakage rate is also indicated and alarmed at the main control board.

2. Reactor coolant system (RCS) valves (including manual, motor operated and throttling control) are provided with double stuffing boxes and leakoff connections. All leakoff connections are piped through sight-flow indicators and routed to the pressurizer relief tank. In addition, pressurizer relief and safety valve and reactor coolant pump seal water return relief valve leakage passes to the pressurizer relief tank. The pressure, level, and temperature of the pressurizer relief tank are indicated and alarmed at the main control board.
3. Reactor vessel flange seal leakage is detected by two leakoff connections, one between the inner and outer O-ring, and one outside the outer O-ring. Leakage is indicated and alarmed at the main control board by a surface-mounted resistance thermocouple which monitors the leakage before it is collected in the reactor coolant drain tank.

Unidentified leakage, which induces steam generator tube, isolation valve seat, and intersystem leakage, is monitored by several devices as identified in the guidelines of Positions C.2, C.3, and C.4 of Regulatory Guide 1.45. Leakage is detected by the increasing of interfacing system level, temperature, and pressure or by the lifting of relief valves accompanied by increasing interfacing system level, temperature, and pressure. Specific intersystem leakage detection methods are as follows:

1. Residual heat removal (RHR) system suction side isolation valves are monitored for seat leakage by the lifting of the RHR relief valves which discharge to the recycle holdup tank resulting in increased recycle holdup tank level, pressure and temperature indications, and alarms at the main control board.
2. Safety injection system accumulators are isolated from the reactor coolant system by check valves. Leakage past these valves is detected by redundant accumulator pressure and level indications and alarms at the main control board.

3. Safety injection system/residual heat removal system discharge headers are isolated from the reactor coolant system by check valves and a gate valve. Seat leakage will pressurize the RHR/SIS discharge headers lifting the relief valves which discharge to the recycle holdup tank resulting in increased recycle holdup tank level indication and alarm at the main control board.
4. Letdown heat exchanger, RCP seal water heat exchanger, and excess letdown heat exchanger tube leakage to the component cooling water system is detected by any combination of the component cooling water system radiation monitors and surge tank level. High component cooling water radiation and high surge tank level are alarmed in the main control room.
5. Safety injection system pump discharge subsystem (hot leg injection) is isolated from the RCS by check valves and a gate valve. Leakage past these valves will pressurize the safety injection pump discharge resulting in main control room indication and eventually lifting relief valves with resulting indication and alarm of increasing recycle holdup tank level.

6. Steam generator tube leakage from the RCS to the secondary system will be detected by radiation monitors in the steam generator blowdown system and by the chemical process sampling system. Samples from each steam generator will indicate reduced pH from the presence of boric acid having leaked from the RCS to the secondary system.

Indication of unidentified leakage from the reactor coolant pressure boundary into the containment is provided by two sources. The first is containment atmosphere radiation monitor indicators and alarms. The second is containment sump flow with its associated alarms. The containment atmosphere radiation monitor operates continuously to detect particulate, iodine, and gaseous radiation in the containment atmosphere. Indication and alarms are provided in the main control room. The sensitivity of the containment atmosphere radiation monitor is such that leaks of one gallon per minute are detectable in less than one hour. The radiation monitors are seismic Category I and are located in flood and tornado protected structures thus meeting the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification." They are also testable and may be calibrated as identified in the guidelines of Positions C.6, C.7, and C.8

Of Regulatory Guide 1.45. If a break were to occur in the primary system, the resulting coolant flow would pass to the containment atmosphere providing air borne contamination or condense and fall to the floor.

The containment floor and equipment sump pumps, as well as the incore instrumentation room sump pumps, input to a plant computer program designed to detect unidentified leakage inside containment in excess of one gpm in less than an hour. Sump level switches will be set to start pump A1 when 15 gallons has collected in Sump A and start pump A2 and actuate a control room alarm if a high-high level is reached. Upon startup of sump pump A1, a timer is started in the computer; an alarm will be actuated if pump A1 starts again within 15 minutes, indicating that flow into the sump is excessive (pump rated 50 gpm). An identical timer interlock is provided in sump B. Unidentified leakage in excess of one gpm is also indicated if any sump pump operates for two minutes or more. A flow integrator is provided on the combined sump pump discharge for periodic monitoring with an accuracy of the one gpm.

The sump-flow measuring system is testable and can be calibrated as required. Additional sources of indication of unidentified leakage include containment pressure, temperature and humidity indicators, pressurizer level indicators in the main control room.

Based on the above, we conclude that the reactor coolant pressure boundary leakage detection systems are diverse and provide reasonable assurance that primary system leakage (both identified and unidentified) will be detected and meet the requirements of General Design Criteria 2 and 30 with respect to protection against natural phenomena and provisions for reactor coolant pressure boundary leak detection and identification, and the guidelines of Regulatory Guides 1.29 and 1.45 with respect to seismic classification and reactor coolant pressure boundary leakage detection system and are, therefore, acceptable.

5.1-11 Pressurizer Relief Tank (Pressurizer Relief Discharge System)

The pressurizer relief discharge system consists of the pressurizer relief tank, the discharge piping from the pressurizer relief and safety valves, the relief tank internal spray header, the tank nitrogen supply, the vent to containment, and the drain to the liquid rad-waste or boron recycle system. The system is non-safety-related (Quality Group D, non-seismic Category I) and is not part of the reactor coolant pressure boundary since all of its components are downstream of the reactor coolant system safety and relief valves. Therefore, its failure would not affect the integrity of the reactor coolant pressure boundary.

The pressurizer relief tank is sized to absorb the energy content of 110% of the full-power pressurizer steam volume through the primary relief and safety valves. Other relief valves which discharge to the pressurizer relief tank are from the residual heat removal system and from the chemical and volume control system. Releases from these sources are less than the design basis release from the pressurizer. The internal spray and bottom drain on the pressurizer relief tank are used to cool the water within the tank. A nitrogen blanket is also provided in the tank to permit expansion of entering steam and to control the tank internal atmosphere. If a discharge exceeding the design basis should occur, the rupture

discs on the tank would pass the discharge through the tank to the containment.

The contents of the tank can be drained to the waste holdup tank in the liquid radwaste system or the recycle holdup tank in the boron recycle system via the reactor coolant drain tank pumps. The rupture discs on the pressurizer relief tank have a capacity equal to or greater than the combined capacity of the pressurizer safety valves. The tank and the rupture disc holders are designed for full vacuum to prevent collapse if the contents cool following a discharge without nitrogen being added. The pressurizer relief tank is provided with instrumentation to indicate pressure and temperature and alarms for high or low level, high pressure and temperature.

The tank is separated from safety-related equipment so that its failure would not compromise the capability to safely shutdown the plant, and further possible rupture disc fragments do not present a missile hazard when the disc ruptures. Thus, the requirements of General Design Criteria 2, "Design Basis for Protection Against Natural Phenomena," and 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.2 are satisfied.

Based on our review, we conclude that the pressurizer relief discharge system meets the requirements of General Design Criteria 2 and 4 with respect to the need for protection against natural phenomena and internal missile protection as its failure does not affect safety system functions. It meets the guidelines of Regulatory Guide 1.29 concerning its seismic classification and is, therefore, acceptable.

Auxiliary Systems

We have reviewed the design of the auxiliary systems necessary for safe reactor operation, shutdown, fuel storage, or whose failure might affect plant safety, including their safety related objectives and the manner in which these objectives are achieved.

The auxiliary systems necessary for safe reactor operation or shutdown include the essential service water system, component cooling water system, ultimate heat sink, control room chilled water system, the heating, ventilation, and air condition systems for the control room and essential portion of the chemical and volume control system, and the auxiliary feedwater system.

The auxiliary systems necessary to assure the safety of the fuel storage facility include new fuel storage, spent fuel storage, the spent fuel pool cooling and cleanup system, fuel handling systems, and the spent fuel pool area ventilation system.

We have also reviewed other auxiliary systems to verify that their failure will not prevent safe shutdown of the plant or result in unacceptable release of radioactivity to the environment. These systems include:

the nonessential service water system, the demineralized water makeup system, potable and sanitary water system, station heating system, nonessential chilled water systems, nonessential portions of the compressed air system, nonessential portions of the chemical and volume control system, and heating ventilation, and air conditioning systems for nonessential portions of the auxiliary building and the turbine building.

9.1 Fuel Storage Facility

9.1.1 New Fuel Storage

The new fuel storage facility for each unit is completely independent and is located at the extreme end of each fuel building where railroad track access is provided. Each new fuel storage facility provides dry storage for 98 fuel assemblies and includes the new fuel assembly storage racks and the concrete storage vault that contains the storage racks.

The fuel building which houses the facility is designed to seismic-Category-I criteria as are the storage racks and vault. This building is also designed against flooding and tornado missiles with the exception that railroad freight door may not be capable of withstanding tornado missiles. This concern is addressed, in Section 3.5.2 of this SER and ^{was raised} ~~addressed~~ in Q410.10 with regard to tornado missiles. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidance of Regulatory Guide 1.29, "Seismic Design Classification," are satisfied *pending resolution of the concern with the freight door.*

The new fuel storage pit is not located in the vicinity of any high-energy lines or rotating machinery. Physical protection by means of separation is provided for new fuel from internally generated missiles and the effects of pipe ^{breaks} ~~breaks~~, and therefore the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," are met as described in Sections 3.5.1.1 and 3.6.1 of this SER.

Since each unit has its own new fuel storage facility and there is no sharing between units, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

The new fuel storage facility is designed to store unirradiated, low-emission, fuel assemblies. Accidental damage to the fuel would release relatively minor amounts of radioactivity that would be accommodated by the ~~fuel-handling building ventilation system~~. The facility is accessible to plant personnel for inspection. Thus, the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," are satisfied.

The new fuel storage racks are designed to store the fuel assemblies in an array with a minimum center-to-center spacing of 21 inches. However, the applicant has not provided the specific keff values determined in his criticality analysis together with the associated ~~described~~ ^{detailed} assumptions and input parameters producing his optimum moderated condition. In addition, the applicant has not provided sufficiently detailed drawings of the new fuel storage racks to enable evaluation of the capabilities for preventing misplacement of fuel assemblies.

The applicant has not provided sufficient information to determine if he has met the requirements of General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

Based on our review, we conclude that the new fuel storage facility is in conformance with the requirements of General Design Criteria 2 and 61 as they relate to protection against natural phenomena, and radiation protection respectively and the guidelines of Regulatory Guide 1.29 relating to seismic classification and is, therefore, acceptable. However, we cannot conclude that the requirements of General Design Criterion 62 is satisfied until concerns regarding lack of specific keff, and new fuel storage rack design are resolved. We will report resolution of our concern in a supplement to this SER.

9.1.2 Spent Fuel Storage

Each plant unit is provided with an independent spent fuel storage facility located in the fuel building which is structurally part of the auxiliary building. The spent fuel storage facility provides underwater storage for 1418 fuel assemblies including control rods and burnable poison rods. This capacity amounts to approximately 19 normal refueling cycles plus one complete core offload. The facility includes the spent fuel storage racks and the lined spent fuel storage pool that contains the storage racks.

The structure housing the facility (the auxiliary building) is designed to seismic-Category-I criteria as are the stainless steel storage racks and storage pool, fuel transfer canal and cask loading area. We cannot, however, determine the capability

lines to withstand the effects of an SsE without resulting mechanical damage to the spent fuel or damage to due to
of the stainless steel spent fuel pool ~~or damage resulting from~~ *due to* overheating ~~due to~~ *as a result of* blockage of cooling flowpaths. This concern is addressed in Q410.13.

The auxiliary building is designed against flooding and tornado missiles with the exception of the freight door ^{provided to allow truck/train entry into the building} as previously noted (refer to Section 3.5.2 of this SER). Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.29, "Seismic Design Classification," and 1.117, "Tornado Design Classification," are satisfied except for the above concern for the spent fuel pool liner.

The fuel pool is not located in the vicinity of any high-energy lines or rotating machinery. Therefore, protection of spent fuel from internally generated missiles and the effects of pipe breaks by physical separation is provided (refer to Sections 3.5.1.1 and 3.6.1 of this SER). Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.13 concerning missile protection for spent fuel are satisfied.

Since each plant unit has an independent spent fuel storage facility, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

The seismic-Category-I storage rack arrangement provides a fuel storage array adequate to maintain the multiplication factor, k_{eff} , below 0.95 for both normal storage and in case of accidental dropping of a fuel assembly. The design of the storage racks is such as to preclude insertion of fuel assemblies in other than permitted locations. Although the storage racks have been designed to prevent significant lifting forces from being applied to them, we are unable to confirm that they can withstand the impact of a dropped fuel assembly without unacceptable damage to the fuel, nor that they can withstand the maximum uplift forces exerted by the fuel-handling machine. Further, the applicant acknowledges the possibility of storing Ocone or McGuire spent fuel assemblies

in the Catawba spent fuel pool by providing bottom spacers in the spent fuel racks. We are concerned with the consequences of an error in placing Oconee or McGuire fuel assemblies into positions not fitted with spacers and the placement of Catawba fuel into positions with spacers. This concern is addressed in Q410.11. In addition the applicant should provide the following information.

1. The nominal value of the effective multiplication factor of the racks and the uncertainty to be added to this value.
2. The results of the verification of the KENO code used (Note: We have not previously reviewed such calculation with the 22 group cross sections). This should include a description (may be by reference) of the experiments which were calculated and the bias and standard deviation of the calculation results.
3. The Oconee fuel assemblies which are proposed for storage in Catawba rack, are 15 x 15 rather than 17 x 17 assemblies. For the same enrichment there may be small differences between these and the optimized Westinghouse design. Please provide a discussion of such differences.

Thus, the requirements of General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control," and 62, "Prevention of Criticality in Fuel Storage and Handling," and the guidelines of Regulatory Guide 1.13 concerning the protection of fuel from mechanical damage and prevention of criticality are not satisfied.

The design of the storage pool includes a pool water level monitoring system, and radiation and temperature monitoring systems with local indication and alarm in the control room. These features satisfy the requirements of General Design Criterion 63, "Monitoring Fuel and Waste Storage."

Based on our review, we conclude that the spent fuel storage facility is in conformance with the requirements of General Design Criteria 2, 4, and 63 as they relate to protection against natural phenomena, missiles, pipe-break effects, radiation protection, and monitoring provisions, and the guidelines of Regulatory Guides 1.29 and 1.117 concerning the facility's design and seismic classification and protection against tornado ~~missiles~~ ^{missiles} and is, therefore, acceptable, except as noted above. However, we cannot conclude that the design satisfies the requirement of General Design Criterion 62 and the guidance of Regulatory Guide 1.113 until a satisfactory response to Q410.11 is obtained we will report this resolution of our concern in a supplement to this SER.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

Each plant unit has an independent spent fuel pool cooling and cleanup system. It is designed to remove the decay heat generated by the stored spent fuel assemblies and to maintain clarity and purity of the spent fuel pool water. The spent fuel pool cooling system is designed to remove the decay heat generated by the number of spent fuel assemblies that are stored which include: a full core offload, plus the remainder of the capacity of the pool filled with fuel from the previous yearly refuelings. The total capacity is 1410 spent fuel assemblies.

The system includes all components and piping from inlet to exit from the storage pool, piping used for fuel pool makeup, and the cleanup filter/demineralizers to the point of return to the refueling water storage tank or discharge to the radwaste system. The design consists of two essential fuel pool cooling trains each with a fuel pool cooling pump and heat exchanger which are completely redundant. A separate non-safety-related fuel pit skimmer pump and filter is also provided for keeping the pool water surface clean.

The essential portions of the system are housed in the seismic-Category-I; flood- and tornado-protected fuel

building portion of the auxiliary building (refer to Sections 3.4.1 and 3.5.2 of this SER). The system itself, with the exception of the cleanup portion, is designed to Quality-Group-C and seismic-Category-I requirements. Failure of the non-seismic-Category-I, Quality-Group-D cleanup portion will not affect operation of the cooling train as isolation capability of that portion of the piping system is provided, and therefore, no adverse effect on safety-related equipment would result from such a failure. Therefore, the design satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and 1.29, "Seismic Design Classification," with respect to seismic and quality group classification of the spent fuel pool cooling system. Discussion of compliance with Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," for the fuel-handling building is discussed in Section 9.4.2 of this SER.

The various components of the system are located in missile-shielded cubicles within the tornado-missile-

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protected auxiliary building and are separated from other moderate- and high-energy piping systems (refer to Sections 3.5.1.1 and 3.6.1 of this SER). Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," are *satisfied*

Since each plant unit has a separate and independent spent fuel pool cooling and cleanup system, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are *not applicable*.

Either of the two spent fuel pool cooling trains maintains the pool water temperature at 125°F or less under the normal heat load conditions. The normal condition assumes one-third core with full irradiation and 7 days decay, one full core of open spaces and the remainder of the pool filled with fully irradiated fuel from previous yearly refuelings. This normal heat load temperature is below our acceptance criterion of 140°F.

During a total core offload, the reserved open spaces are assumed to be filled by one-third core irradiated 11 days and decayed 7 days, one-third core fully irradiated and decayed 7 days, and one-third core fully irradiated and decayed 25 days with the remainder of the pool filled with fuel from previous yearly refuelings. Under these conditions, both cooling trains must be used to maintain a temperature less than 150°F. These

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"abnormal" heat load temperatures are within acceptable limits and assumes the full-core offload has occurred when the fuel storage racks are full. The heat loads for the above "normal" and "maximum" cases are in accordance with our Branch Technical Position (BTP) ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling." The applicant anticipates the storage of non-Catawba spent fuel in the Catawba spent fuel pool and indicates a slightly higher normal and maximum heat load for this fuel but still within the above mentioned temperature limits.

All connections to the spent fuel pool are either near the normal water level or are provided with anti-siphon holes to preclude possible siphon draining of the pool water. The safety-related component cooling water system provides cooling water to the fuel pool heat exchanger and transfers its heat to the ultimate heat sink (refer to Sections 9.2.1 and 9.2.5 of this SER). The spent fuel pit pumps can be powered from the emergency (Class 1E) power sources. However, the licensee has not provided information on the spent fuel pool water temperature following the loss of one cooling train assuming the offload maximum heat load

condition with either Catawba fuel or non-Catawba fuel. Neither can we determine the disposition of the spent fuel pool heat load to all possible heat sinks under these conditions. We are concerned about the capability of the fuel building to accommodate the high temperature and humidity effects associated with pool high-water temperature and the capacity for the fuel building ventilation system to handle the load. The applicant has not presented an analysis of decay heat load vs. time or pool temperature vs. time. These

concerns have been addressed in Q410.12. Therefore, we cannot conclude that the requirements of General Design Criterion 44, "Cooling Water" are met.

The design of the spent fuel pool cooling system and its accessible location is such that periodic testing and in-service inspection of the system can be accomplished. The active components of the spent fuel pool cooling system are either in continuous or intermittent operation during all plant operating conditions. Thus, the requirements of General Design Criteria 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System," are satisfied.

Normal makeup to the spent fuel pool to replace losses due to evaporation or leakage through the liner and thus maintain proper water level for shielding is provided by the seismic-Category-I refueling water storage tanks. Demineralized water can be supplied to the pool by the reactor make-up water pumps, and emergency make-up water

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can be supplied to the pool from the seismic-Category-1, Quality-Group-C Nuclear Service Water System. The nuclear service water pumps are powered from the emergency (Class 1E) buses. Thus, the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of Regulatory Guide 1.13 concerning fuel pool design are met.

Spent fuel pool temperature and level are alarmed in the control room. Fuel pool cooling heat exchanger parameters as well as fuel pool cooling pump discharge pressure are locally indicated. Pressure and flow indication of the purification loop are also provided as well as fuel building radiation level. Thus, the requirements of General Design Criterion 63, "Monitoring Fuel and Waste Storage," are satisfied.

Based on our review, we conclude that the spent fuel pool cooling and cleanup system is in conformance with the requirements of General Design Criteria 2, 4, 5, 45, 46, 61 and 63 as they relate to protection against natural phenomena, missiles and environmental effects, cooling water capability, in-service inspection, functional testing, fuel cooling and radiation protection and monitoring provisions respectively and the guidelines of Regulatory Guides 1.13, 1.26, and 1.29, relating to the system's design, seismic and quality group classification, and is,

therefore, acceptable. However, we cannot conclude that the requirement of GDC 44 are met with respect to the spent fuel pool heat load disposition. We will report resolution of our concern in a supplement to this SER.

9.1.4 Light Load Handling System (Fuel-Handling System)

The light load handling system is related to refueling and consists of all components and equipment used in handling new fuel from the receiving station to the loading of spent fuel into the shipping cask. The light loads considered are those that, if dropped, would have a kinetic energy on impact less than that of one fuel assembly and its associated handling tool when dropped from the normal handling height above the spent fuel storage racks. The system includes the equipment designed to facilitate the periodic refueling of the reactor, the refueling machine, spent fuel pit bridge crane (fuel-handling machine), new fuel elevator, and the fuel transfer system. The handling of fuel during refueling is controlled by a series of interlocks to assure that fuel-handling procedures are maintained. The design assures that no failure will result in release of radioactivity in excess of that assumed in the design basis fuel-handling accident.

The entire system is housed within the fuel-handling portion of the auxiliary building and reactor building (containment) which are seismic Category I, flood- and tornado-protected structures (refer to Sections 3.4.1

and 3.5.2 of this SER). Although fuel-handling system components are not required to function following an SSE, critical components of the fuel-handling system are designed to seismic-Category-I requirements so that they will not fail in a manner which results in unacceptable consequences, such as fuel damage or damage to safety-related equipment. The refueling machine and fuel-handling machine which handle individual fuel assemblies are designed to seismic-Category-I requirements. The design thus satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification."

Each plant unit has a separate and independent fuel-handling system; thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are ~~not~~ *not applicable*

However, we have identified a new generic concern with respect to dropping of loads lighter than a spent fuel assembly over stored spent fuel in the fuel pool and in the reactor vessel from heights greater than those assumed in the design basis fuel-handling accident. We requested that the applicant review his fuel-handling procedures and identify these loads, their weights and heights when lifted, design features or procedures for assuring the lift height is not exceeded, and the kinetic energy of these loads upon impacting spent fuel. The applicant should verify that the resulting radiological releases are less than those from the design basis fuel-handling accident or take appropriate corrective measures. Thus, we conclude that the requirements of General Design Criteria 61 and 62 are met with the exception of the above concern which has been addressed in Q410.15.

Based on our review, we conclude that the light load fuel-handling system is in conformance with the requirements of General Design Criteria 2 and 5

as they relate to its protection against natural phenomena. The light load fuel-handling system meets the recommendations of Regulatory Guides 1.13 and 1.29 with respect to seismic classification, tornado-generated missiles and interlocks and is therefore, acceptable. However, we cannot conclude that the requirement of General Design Criteria 61 and 62 are met until our concerns with respect to the dropping loads lighter than a fuel assembly are resolved. We will report resolution of our concern in a supplement to this SER.

9.1.5 Overhead Heavy-Load-Handling System

The overhead heavy-load-handling system consists of all components and equipment used in moving all loads weighing more than one fuel assembly and its associated handling device. The equipment includes the containment polar crane and cask handling crane which are used in handling such heavy loads as the reactor vessel head, reactor internals, shield plug segments and spent fuel casks. The containment polar crane is not used for handling fuel assemblies. The cask-handling crane does, however, handle fuel assemblies. Handling of fuel assemblies by the main hoist is limited to the area between the new fuel vault and the spent fuel shipping cask area. Handling of fuel assemblies by the auxiliary hoist is limited to the area between the new fuel vault and the new fuel elevator.

The entire system is housed within the fuel-handling portion of the auxiliary building and reactor building (containment) which are seismic-Category I, flood and tornado protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). Although fuel-handling system components are not required to function following an SSE, both the containment polar crane and the cask handling crane are designed to seismic-Category I requirements so that they will not fail in a manner which results in unacceptable consequences such as fuel damage or damage to safety-related equipment. The 125-ton cask handling crane is used for handling the spent fuel shipping cask and is equipped with a 10-ton auxiliary hoist. The 175-ton containment polar crane is used to move the reactor vessel head and has a 25-ton auxiliary hoist. The design thus satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification."

Each plant unit has a separate and independent fuel-handling system; thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are ~~not~~ applicable.

The main hoist of the cask-handling crane is prevented from traveling over the spent fuel pool by mechanical

stops. The spent fuel cask is brought to the cask storage area along a prescribed path and enters the storage area without passing over spent fuel in the pool. The cask is not lifted to an elevation above any structural surface high enough to cause damage which could result in unacceptable radiological release. No safety-related equipment is located along the path of travel of the cask. The walls which surround the cask-loading area rise to the full height of the pool and are structurally designed to withstand the impact force due to a falling cask. Should the cask tip after falling on the guard walls surrounding the cask loading area, its center of gravity is such that it will not fall outside the cask-load area and will thus not affect the fuel in the spent fuel storage pool. It should be noted, however, that the new fuel elevator is located within the spent fuel pool main area, thus requiring that the auxiliary hoist of the cask-handling crane be positioned over the spent fuel pool. However, the applicant should verify that the above discussion concerning a cask drop is also valid for the Ocone and McGuire casks when they are handled at Catawba.

The applicant has not provided a crane load drop analysis required as part of the applicant's response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," which was transmitted to the applicant for action by generic NRC letters dated December 22, 1980 and February 3, 1981. The applicant has been asked to provide an analysis of the effects of dropping heavy

fuel cask. The analysis

should satisfy the evaluation criteria of NUREG-0612, Section 5.1, and consider the consequences of dropping the reactor vessel head and vessel internals during preparation for or completion of fuel handling. In addition, the main and auxiliary hoist load block of both the containment building polar crane and the cask-handling crane should be considered as heavy loads and an analysis of the consequences of their falling included in this analysis. This concern is addressed in Q410.14. Further the applicant has not committed to implement the interim actions of NUREG-0612 prior to final implementation of NUREG-0612 guidelines prior to receipt of the operating license.

In respects other than those related to our evaluation of the applicant's response to NUREG-0612, we find that the requirements of General Design Criteria 4, "Environmental and Missile Design Bases," and 61, "Fuel Storage and Handling and Radioactivity Control," are met.

Based on our review, we conclude that the ~~heavy-load-handling~~ overhead system is in conformance with the requirements of General Design Criteria 2, 4, and 61 as they relate to its protection against natural phenomena, missile protection, and safe handling

of the spent fuel cask, and the guidelines of Regulatory Guides 1.13 and 1.29 with respect to overhead crane interlocks and maintaining plant safety in a seismic event and is therefore acceptable, except as noted above with respect to *handling of the Oconee and McGuire casks and* NUREG-0612 issues. We will report resolution of our concern in a supplement to this SER.

9.2.1 Station Service Water System (Nuclear Service Water System)

The nuclear service water system ^(NSWS) provides cooling water in both normal and emergency situations to safety and non-safety-related heat loads of ^{both units 1 and 2.} ~~each unit~~. Lake Wylie and the standby nuclear service water pond (SNSWP) serve as the normal water source and ultimate heat sink (Refer to Section 9.2.5 of this SER) respectively for the components cooled by the system.

Safety-related components cooled by the nuclear service water system include the component cooling heat exchangers, diesel generator coolers, auxiliary feedwater pump oil coolers, auxiliary building ^{chillers} ~~drillers~~, centrifugal charging pump room coolers, containment spray pump room coolers, safety injection pump room coolers, residual heat removal pump room coolers, auxiliary feedwater pump room cooler and control room air conditioning units. Radiation monitors are provided in the system to detect potential inleakage of radioactivity.

^(“A and B”) of essential equipment re-

The system consists of two redundant trains ~~in each~~ unit.

The “A” trains are supplied by the two “A” NSWS pumps and the “B” trains by the two “B” pumps (four NSWS pumps total). An “A” and “B” pump are designated for each unit. The “A” and “B” pumps are interconnected by a common supply header before splitting into separate loops at each unit's auxiliary building.

~~of each unit.~~ The pumps are powered from separate emergency (Class 1E) power sources ^{the} associated with its unit. [^] The operation of any two of the four pumps on either or both supply trains is sufficient to supply all cooling water requirements for the two unit plant for unit start-up, cooldown, refueling, or post-accident operation. However, ^{the} additional pumps are normally started for unit start-up and cooldown and two pumps per unit will operate during an accident and loss of offsite power if both diesel-generators are in operation. ~~The nuclear service water system function is also assured during a loss of offsite power and LOCA in one unit and simultaneous normal cooldown of the other.~~ Train separation with double valving between main supply and discharge headers, assures both units of having a source of water, two 100-percent-capacity pumps, and two redundant trains of heat exchangers essential for safe shutdown. Trains A and B in each unit are also cross connected together, ~~at five places by crossover piping at each crossover connections~~ ^{which are} ~~valves~~ ^{The} ~~are provided with~~ ^{isolation} ~~connections~~ ^{in emergency conditions} ~~valves~~ ^{are} ~~closed~~ ^{to ensure} train integrity and to meet the single failure criterion. Thus, the requirements of General Design Criterion 44, "Cooling Water" are met.

in its vicinity by causing soil erosion. This concern has been stated in Q410.16. Until this concern is resolved, we can not conclude that the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," are satisfied.

During normal plant operation, one nuclear service water system pump per unit is operating. Availability of the remaining pumps is assured by periodic functional tests and inspections as delineated in plant technical specifications. The system components are located in accessible areas to permit in-service inspection as required. Thus, the requirements of General Design Criteria 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System," are met.

Based on the above, we conclude that the nuclear service water system meets the requirements of General Design Criteria 5, 44, 45 and 46 with respect to the system's ~~protection against natural phenomena~~, shared systems, decay heat removal capability, in-service inspection and functional testing. However, we cannot conclude

The essential portions of the nuclear service water system are common for both Units 1 and 2 up to a point outside the auxiliary building where each train is divided and directed to each unit. The return trains from each unit are combined, both inside and outside the auxiliary building and returned to Lake Wylie under normal condition and to the SNSWP under emergency conditions. ~~The~~ ^{of all interconnections} Isolation between units is accomplished by redundant automatic safety-related valves powered from separate emergency (Class 1E) power supplies. Therefore, General Design Criterion 5, "Sharing of Structures, Systems and Components," is met.

The nuclear service water system is a seismic Category I system essential portion of the system are quality Group C. The system is housed in the seismic Category I, flood and tornado protected nuclear service water pumphouse, diesel generator buildings, and auxiliary building. However, we are unable to determine that buried portions of the nuclear service water system can withstand the effects of a failure of the non-safety-related condenser circulating water system yard piping. We are concerned that the large circulating water ducts may fail during safe shutdown earthquake and jeopardize support of the nuclear service water supply buried ^{pipings} ~~buried~~

that the requirements of General Design Criterion 2
and the guidelines of Regulatory Guide 1.29 with regard
to protection against natural phenomena ^{are met} because of our
concern with the effects of ^a failure ^{in the} condenser circula-
ting water piping as described above. We will report
resolution of our concern in a supplement to this SER.

9.2.2 Reactor Auxiliary Cooling Water Systems Component

Cooling System

The component cooling system (CCS) serves as an intermediate closed cooling water system *picking up heat from* potentially radioactive heat sources. *And rejecting* ~~its heat load~~ via the nuclear service water system to the ultimate heat sink. (Refer to Sections 9.2.1 and 9.2.5 of this SER for a discussion of the nuclear service water system and ultimate heat sink.) This arrangement minimizes the possibility of leakage of radioactive material into the environment. A portion of the system is shared between Units 1 and 2 as described below.

The *component cooling system* supplies cooling to safety-related and non-safety-related plant components during normal operation and safety-related components during postulated accident and emergency conditions. The system serves the residual heat removal heat exchangers and pumps, letdown heat exchanger, excess letdown heat exchanger, reactor vessel support coolers, steam generator blowdown heat exchanger, seal water heat exchanger, fuel pool heat exchangers, sample heat exchangers and various additional subcomponents of the waste gas, waste evaporator and recycle evaporator systems. A radiation monitor is placed at the discharge of each component coolant heat exchanger to detect radioactive leaks into the component cooling system.

The CCS consists of ~~four pumps, two heat exchangers,~~
~~two surge tanks, one drain sump and two drain pumps~~
~~serving~~ two redundant cooling trains in each unit, with
two pumps, one heat exchanger and one surge tank per
train. Each pump is powered from a separate emergency
(Class 1E) power source. On an engineered safety fea-
tures actuation signal both trains of component cooling
equipment are actuated and automatically aligned by
redundant isolation valves to the appropriate trains of
engineered safety equipment. Only one train ~~of com-~~
~~ponent cooling equipment~~ in each unit is necessary to
supply minimum engineered safety ^{feature} equipment requirements
thus assuring adequate cooling capability in the event
of a single failure under all assumed accident condi-
tions including loss of offsite power, LOCA in one unit
and simultaneous safe shutdown of the other. ~~being~~
normal unit operation two pumps and one heat exchanger
are required. Two pumps and one heat exchanger also
provide minimum unit cooldown requirements. ~~However,~~
~~to provide minimum unit cooldown requirements.~~ However,
to provide a more rapid unit cooldown, four component
cooling pumps and two component cooling heat exchangers
are required. The CCS systems for Unit 1 and 2 nor-
mally function as two independent systems. The systems

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are identical except cooling water supplied to shared equipment is normally contained in the Unit 1 subsystem. Crossovers are provided between the two subsystems so that cooling water can be supplied to shared equipment from either train in either unit. Such sharing of components does not degrade the performance or reliability of the essential portion of the component cooling system, as crossovers and non-essential shared components are isolated by redundant automatic valves on engineered ^{safety features} ~~residual~~ actuation signal. Thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," and 44, "Cooling Water" are satisfied.

The applicant indicates a capability to operate the reactor coolant pumps for 10 minutes after a loss of component cooling water. However, the applicant has not provided safety-grade instrumentation with which to indicate this loss of cooling water and allow prompt operator action to prevent a motor bearing failure and possible unacceptable locked rotor condition. The component cooling supply and return piping to the reactor coolant pumps does not meet the single failure criterion since a single supply and return path provides cooling

water to all four reactor coolant pumps and each path contains a single electrically operated containment isolation valve. Thus, cooling water to all four reactor coolant pump motor bearings which require continuous cooling during all modes of operation would be lost on failure of either of these valves to remain open. This concern has been addressed in Q410.17.

The component cooling system is seismic Category I, Quality Group C. The system is housed in the seismic-Category-I, flood- and tornado-protected auxiliary building and containment (refer to Sections 3.4.1 and 3.5.2 of this SER). Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," are met. The component cooling ~~water~~ system operates continuously in all plant operating modes. Pumps are rotated in service on a scheduled basis to obtain even wear or are periodically tested in accordance with plant Technical Specifications. The system is located in accessible areas to permit inservice inspection as required. Thus, the requirements of General Design Criteria 45, "Inspection of Cooling Water System" and 46, "Testing of Cooling Water System," are met.

Based on the above, we conclude that the component cooling system meets the requirements of General Design Criteria 2, 5, 44, 45 and 46 with respect to the system's protection against natural phenomena, shared systems, decay heat removal capability, in-service inspection and functional testing, and the guidelines of Regulatory Guide 1.29, with respect to the system's seismic identification. However, we can not conclude that the system is acceptable until resolution of our concern involving loss of component pumps and the need for safety grade indication of this condition. We will report resolution of this concern in a supplement to this SER.

9.2.3 Demineralized Water Makeup System (Makeup Demineralized Water System)

The non-safety-related (Quality Group D, non-seismic Category I) makeup demineralized water system provides treated and demineralized water to various plant systems and components that include: condensate makeup (to the upper surge tanks), chemical addition tanks, turbine-generator stator cooling makeup, recirculated cooling water makeup, reactor makeup water cooling tanks, component cooling water makeup, auxiliary boiler feed-water, waste disposal system, and the ice condenser storage tank. Lake Wylie provides the source of water to the system.

The system has no safety-related functions. Adequate isolation is provided at all makeup demineralized water connections to safety-related systems. Protection from flooding for safety-related equipment resulting from failure of the system is discussed in Section 9.3.3 of this SER. The system is capable of fulfilling the normal operating requirements of the facility for acceptable makeup water with the necessary component redundancy. At each point of discharge from the system,

check valves prevent contamination of the makeup demineralizer system by backflow from the systems which it supplies. Alarmed instrumentation has been provided in the control room to prevent delivery of off-specification water to safety-related systems. Failure of the system does not affect the capability to safely shut down the plant as described above; thus, the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and 5, "Sharing of Structures, Systems and Components," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.2 are met.

Based on our review, we conclude that the makeup demineralized water system meets the requirements of General Design Criteria 2 and 5 with respect to the need for protection against natural phenomena and shared systems as its failure does not affect safety system functions and meets the guidance of Regulatory Guide 1.29 concerning its seismic classification and is, therefore, acceptable.

9.2.4 Potable and Sanitary Water Systems

The non-safety-related (Quality Group D, non-seismic Category I) potable and sanitary water systems provide clean water for drinking and sanitary purposes and includes all components and piping from the potable supply connection from Lake Wylie to points of discharge.

There are no cross-connections between the potable and sanitary water systems and potentially radioactive systems, and therefore, inadvertent contamination is prevented. Protection from flooding for safety-related equipment resulting from failure of the system is discussed in Section 9.3.3 of this SER. Failure of the system does not affect plant safety as described above. Thus, the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," are met.

Based on our review, we conclude that the potable and sanitary water systems meet the requirements of General Design Criterion 60 with respect to prevention of release of potentially radioactive water, and is, therefore, acceptable.

9.2.5 Ultimate Heat Sink

Two bodies of water serve as the ultimate heat sink (UHS). These sources are separated and protected such that failure of one does not induce failure of the other. Lake Wylie is the normal source of nuclear service water. The emergency source is the standby nuclear service water pond (SNWSP) and consists of an arm of the Lake Wylie reservoir which is retained by a seismic-Category-I dam at an elevation slightly higher (571 ft. msl) than the normal elevation of Lake Wylie (569.4 ft. msl). The UHS provides heat dissipation capability for both Units 1 and 2 through the nuclear service water system (refer to Section 9.2.1 of this SER). The UHS provides a supply of cooling water to dissipate waste heat rejected during a unit LOCA plus a second unit cooldown.

Makeup water can enter the SNWSP from Lake Wylie by alignment of valves from the nuclear service water system or by normal surface runoff. Flood protection is provided by a sloping 5 ft. diameter pipe through the base of the dam into Lake Wylie. The SNWSP dam and overflow drain pipe intake is adequately protected

from tornadoes and missiles. The intake and discharge structures in the SNSWP are submerged and missile-protected, while the safety-related piping (seismic Category I), to the nuclear service water pump house is blow grade and ^{also} adequately protected. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Positions C.2 and C.3 regarding ultimate heat sink protection against natural phenomena and Regulatory Guide 1.29, "Seismic Design Classification" Position C.1, regarding seismic design classification, are met.

The SNSWP is shared between Units 1 and 2 as are the two redundant intake and discharge lines of the nuclear service water system from and to the pond. (Refer to Section 9.2.1 of this SER). A single normal intake and discharge line from and to Lake Wylie are also shared. The safety related redundant lines from and to the SNSWP allow for a single failure of one line as discussed in Section 9.2.1 of this SER. The SNSWP is sized to provide adequate heat removal capability for both Units 1 and 2 and discussed below. A single source of water is acceptable as its passive nature and design features described above demonstrate an extremely low probability of its failure due to natural or site related phenomena. Thus, the requirement of General Design Criterion 5, "Sharing of Structures, Systems and Components" is met.

The applicant's analysis which demonstrates the capability of SNWSP to meet the cooling requirements of the plant modeled utilized Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling." This analysis indicates that sufficient water is available for 30 days without makeup and the maximum acceptable intake temperature to the nuclear service water system is maintained assuming an accident (LOCA) in one unit and simultaneous safe shutdown of the other under the highest 30 day historical ambient temperature conditions. The total heat input to the SNSWP in the analysis includes the fixed heat load due to safety-related pump and motor coolers, air conditioning equipment, and diesel-generator jacket water coolers and the sensible and residual heat loads due to one unit following a LOCA and second unit due to an immediate cooldown.

We are unable to determine in the UHS analysis of residual and auxiliary system heat loads where and at what rate the spent fuel pool cooling is considered. This concern is discussed in Q410.18. In addition, the applicant identified heat loads appear inconsistent with those in FSAR Figure 9.2.5-5. There-

fore, we are unable to conclude that the design meets the requirements of General Design Criterion 44, "Cooling Water," and the guidelines of Regulatory Guide 1.27 Position C-1 regarding the UHS ability to maintain proper system temperature.

The UHS consists of no active components, and thus the requirements of General Design Criteria 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System," are not applicable.

Based on the above, we conclude that the ultimate heat sink meets the requirements of General Design Criteria 2 and 5 with respect to protection against natural phenomena and shared systems and the guidelines of Regulatory Guides 1.27 and 1.29 with respect to seismic classification. However, we are unable to conclude that the requirements of General Design Criterion 44 with respect to decay heat removal capability and the guidelines of Regulatory Guide 1.27 with respect to design capability are satisfied as described above. We will report resolution of our concern in a supplement to this SER.

9.2.6 Condensate Storage Facilities (Condensate Storage System)

The non-safety-related (Quality Group D, non-seismic Category I) condensate storage system provides storage and makeup of deaerated condensate water for the Main Condensate System, the Auxiliary Feedwater System, and the auxiliary electric boilers. It also serves to collect and store miscellaneous system drains.

The condensate storage system for each unit consists of an upper surge tank dome, where makeup from the Makeup Demineralized Water System enters, two upper surge tanks, a condensate storage tank, an auxiliary feedwater condensate storage tank and two condensate storage tank pumps.

The facilities are capable of fulfilling the normal operating requirements of the facility for storage of condensate water with the necessary component redundancy. The facilities were evaluated and found to have no functions necessary for achieving safe reactor shutdown conditions or for accident prevention or accident mitigation. Failure of the facilities will not affect the safety function of safety-related systems.

The ~~Condensate Storage~~ System provides the normal (preferred) supply to the safety-related auxiliary feedwater system. However, this function is not required to maintain plant safety as the safety-related nuclear service water system serves as the assured water source. Refer to Section 10.4.9 of this SER for discussion of compliance with General Design Criterion 44, "Cooling Water," 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water Systems."

The ~~Condensate Storage~~ System is located in the turbine and service buildings and is thus separated from safety-related equipment. However, we are unable to determine that adequate isolation is provided at the interface of the ~~Condensate Storage~~ System with the safety-related ~~Auxiliary Feedwater~~ System condensate storage tank. This concern is addressed in Q410.19. Thus, we are unable to conclude that the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Position C.2 of Regulatory Guide 1.29, "Seismic Design Classification," are satisfied. The ~~Condensate~~

Storage system is not shared between units; thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are not applicable.

Based on our review, we cannot conclude that the condensation stage storage system meets the requirements of General Design Criteria 2 with respect to the need for protection against natural phenomena, as we are unable to verify that its failure does not affect safety system functions and that it satisfied the guidelines of Regulatory Guide 1.29 concerning its seismic classification. We will report resolution of our concern in a supplement to this SER.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

The compressed air system consists of three separate subsystems: instrument air, station air, and breathing air systems. These subsystems are common to both units. The instrument air system supplies clean, oil-free, dried air to all air-operated instrumentation and valves. The station air system supplies compressed air for air-operated tools, miscellaneous equipment, and various maintenance purposes. The breathing air system supplies clean, oil-free, low-pressure air to various locations in the auxiliary building and in the containment for breathing protection against airborne contamination during certain maintenance and cleaning operations. These systems are non-safety-related (Quality Group D, non-seismic Category I) with the exception of containment penetrations which are designated as safety-related (seismic Category I, Quality Group B). The systems are not required to achieve safe reactor shutdown or to mitigate the consequences of an accident. Failure of the station air, instrument air, and breathing air subsystems will not prevent safety-related components or systems from

performing their intended safety functions. Because the compressed air system serves no safety function, the requirements of General Design Criteria, "Quality Standards and Records" are not applicable.

The instrument air system consists of three parallel trains of compressors, coolers, moisture separators and air receivers feeding four parallel dryers and filter banks through a common line. After filtering, the service is divided to Units 1 and 2. In the event of low instrument air pressure, the Station Air System will automatically supply air to the Instrument Air System. This air will be supplied through two oil removal filters to the instrument air compressors' discharge header. The station air system consists of two parallel trains of compressors, coolers, water separators and receivers feeding a manifold to both units. Similarly, the breathing air system has two trains of equipment feeding a common manifold to both units. Failure of the service or instrument air systems will not prevent safety-related components of systems from performing as intended under emergency cooldown conditions or result in unacceptable radioactive releases. All air-operated

safety-related valves and devices are designed for a fail-safe mode on loss of instrument air and do not require a continuous air supply under emergency or abnormal conditions.

However, the applicant has stated that certain valves essential for safe shutdown require instrument air in the event of a control room evacuation coincident with station blackout. We have insufficient information concerning this condition to determine that operation of these valves is assured from the auxiliary shutdown panel. This concern is discussed in Q410.20. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Position C.2 of Regulatory Guide 1.29, "Seismic Design Classification," are not satisfied.

We have evaluated the above subsystems and found it to have no functions necessary for achieving safe reactor shutdown conditions or for accident prevention or accident mitigation. We have determined that the system is capable of providing normal instrument, breathing and station air needs. Discussion of preoperational testing of the compressed air systems and compliance with

Regulatory Guide 1.80, "Preoperational Testing of Instrument Air Systems," is contained in Section 14.0 of this SER.

However, we are unable to determine whether the instrument air system meets the instrument air quality standards defined by ANSI MC 11.1-1976 (ISA S7.3). This concern is addressed in Q410.21.

Insert (A)

All compressed air system containment penetrations are provided with seismic-Category-I, Quality-Group-B isolation valves which are located in seismic-Category-I, flood- and tornado-protected structures. Thus, the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," are satisfied for these portions of the system.

Based on our review, we conclude that the Compressed Air System does not meet the requirements of General Design Criterion 2 with respect to the need for protection against natural phenomena as we can not verify that its failure does not affect safety system functions and the guidelines of Regulatory Guide 1.29 concerning its seismic classification. We conclude

Insert (A)

The three subsystems of the compressed air system are shared between units with the exception of the safety related portions (the containment penetrations) which are not shared. This sharing does not present any safety concerns as identified above; thus the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components", are met.

however, that safety-related portions of the system meet the requirements of General Design Criteria 2 and 5 regarding protection against natural phenomena and shared systems. We also can not conclude that the system meets the air quality standards identified in ANSI MC11.1. Therefore, the system is unacceptable. We will report the resolution of this concern in a supplement to this SER.

9.3.3 Equipment and Floor Drainage System

The equipment and floor drainage system includes all piping from equipment or floor drains to the sump, sump pumps, and piping necessary to carry potentially radioactive and non-potentially radioactive effluents through separate subsystems. Potentially radioactive drainage is collected in floor and equipment drain sumps in each building and discharged to the liquid radwaste processing system, thus satisfying the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment."

Safety-related portions of the equipment floor drainage system are the containment penetration lines and valves for the containment floor and equipment sump and incore instrumentation sump. These portions are seismic Category I, Quality Group B. Also seismic Category-I, Quality Group C check valves which prevent backflooding are used on the discharges of those sump pumps throughout the plant which drain safety-related equipment. All this equipment is located in seismic-Category-I, flood- and tornado-protected structures. All piping in areas housing components needed for safe shutdown and accident mitigation is designed to seismic Category I. Thus, the requirements

of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Positions C.1 and C.2, are satisfied.

Separate sumps are provided for the auxiliary feedwater system pumps so that a break or water jet from one auxiliary feedwater pump will not flood a redundant pump. A similar arrangement exists for the containment spray and residual heat removal pump room sumps. The applicant states that sump pump capacities are sized large enough to handle a credible rupture or the maximum expected flow rate in their respective sumps. However, we are unable to confirm this statement and have addressed this concern in Q410.22. Until the applicant responds, General Design Criterion 4, "Environmental and Missile Design Bases," is not satisfied.

It is our position that the applicant must: (1) demonstrate drainage capability by natural drainage through passive plant features or by failed non-seismic-Category-I drainage systems, (2) equip leak detection sumps with redundant safety-grade alarms in the control room and verify that if operator action is required, unacceptable flooding will not occur within 30 minutes, (3) or provide separate watertight rooms and independent drainage paths with leak detection sumps for each redundant safety-related component.

Based on the above, we conclude that the equipment and drainage system complies with the requirements of General Design Criterion 60 with respect to protection against releases of radioactive material to the environ-

ment and the requirements of General Design Criterion 2 and the guidelines of Regulatory Guidel. 29 Appendix C.1 and C.2 with respect to seismic design classifications. However, we can not conclude that the system complies with the require-

ments of General Design Criterion 4 with respect to protection against internal flooding and it is, therefore, unacceptable. We will report resolution of our concern in a supplement to this SER.

9.4

Heat Ventilation, and Air Conditioning (HVAC) System

9.4.1

Control Room Area Ventilation System

The control room area ventilation system is shared by both Units 1 and 2 and is designed to maintain a suitable environment for equipment operation and safe occupancy of the control room under all plant operating conditions. The control room area ventilation system serves the control room, the electrical penetration rooms, cable rooms, switchgear rooms, battery rooms, and motor control center rooms for both units. (Refer to Section 6.4 of this SER for further discussion of control room habitability).

The system consists of two redundant full-capacity equipment trains each containing intake smoke, radiation and chlorine detectors, prefilters, final filters, supply fans, pressurizing fans, and chilled water cooling units. The system is fully redundant except for some passive interconnecting duct headers.

The control room area is normally maintained at a slightly positive pressure relative to the outdoors by taking makeup air from either or both of two outside intakes located on opposite sides of each reactor building, away from the respective unit vent,

Each outside air intake location is monitored for the presence of radioactivity, chlorine, and products of combustion. Isolation of the outside air intake occurs automatically upon indication of high radiation level, high chlorine concentration or smoke concentration in the intake. Should both intakes close, the operator may override the intake monitors and by inspection of the control room readouts select the least contaminated intake in order to reestablish pressurization of the control room. This will ensure pressurization of the control room at all times.

All essential portions of the system are located in the auxiliary building which is seismic-Category-I, flood- and tornado-protected (refer to Sections 3.4.1 and 5.5.2 of this SER). Essential portions of the system itself are seismic Category I, Quality Group C, and are physically separated from high-energy systems. Each outside air intake is provided with a tornado isolation damper to prevent depressurization of the control room and the control room area during a tornado. These outside air intakes are tornado missile protected. Thus the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory

Guide 1.29, "Seismic Design Classification," are met. Refer to Sections 3.5.1.1, 3.6.1 and 9.3.3 for discussion of protection of essential control room area ventilation system components from internally generated missiles, postulated failures in piping systems and internal flooding.

The control room area ventilation system is an engineered safety feature. Each 100%-capacity redundant train of air-handling units, water chillers, pumps, pressurizing filter trains and fans, and outside air intake isolation valves is served from separate trains of the emergency class 1E power system. This assures the integrity and availability of at least one train of the control room area ventilation system in the event of any single active failure. The control room area ventilation system is designed to maintain temperature, cleanliness and pressurization in the areas served during normal plant operation, shutdown, post-accident conditions, and in all possible weather conditions. The control ~~room area~~ ^{room area} ventilation system is also designed to ensure that the maximum radiation dose received by the control room personnel under accident conditions is within acceptable limits.

Upon detection of high radiation, high chlorine or smoke concentration, the affected intakes isolation valves close automatically and the system is operated entirely on recirculation with no outside air makeup. During control room isolation, additional recirculation flow would be forced through the pressurizing filter train. The above design meets the requirements of General Design Criterion 4, "Environmental and Missile Design Bases and 19, "Control Room" and the guidelines of Regulatory Guides 1.78, on "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous. Chemical Release," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," with respect to the uninterrupted safe occupancy of the control room and associated required manned areas under all normal and accident conditions including LOCA conditions. The control room area ventilation system is shared between both units. However, this sharing does not compromise the systems' safety function because of its redundancy, and, thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are met.

Since the control room is not a source of radioactivity, and the emergency recirculation system only functions following an accident, the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guides 1.52, "Design, Testing and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are not applicable.

However, the applicant has not provided information about the capability of the battery room exhaust fans to prevent accumulation of hydrogen. The applicant should verify that failure of the essential battery room exhaust fan is annunciated in the main control room. The applicant is requested to respond to the above concern.

Based on the above, we conclude that the control room area ventilation system is in conformance with the requirements of General Design Criteria 2, 4, 5, and 19 relating to protection against natural phenomena, maintaining proper environmental limits for equipment operation, shared systems, and protection to permit access for occupancy of the control room under accident conditions, and the guidelines of Regulatory Guides 1.29, 1.78, and 1.95 relating to the system seismic classification, design for protection against hazardous chemical releases, and protection of personnel against chlorine gas release. However, we cannot conclude that the system is acceptable until resolution of our concern with the battery room exhaust system. We will report resolution of our concern in a supplement to this SER.

9.4.2

Spent Fuel Pool Area Ventilation System (Fuel Handling Area Ventilation System)

The fuel handling area ventilation system is designed to maintain a suitable environment for the operation, maintenance, and testing of equipment, personnel access and to limit potential radioactive release to the atmosphere during normal and accident conditions. The fuel handling area ventilation system consists of a non-seismic-Category-I supply subsystem and a seismic-Category-I exhaust subsystem. The supply subsystem is a single train which draws air directly from the outside, through a prefilter, heater, cooler, supply air-handling fan, radiation monitor and ducting to discharge points throughout the fuel-handling and spent fuel pool area. The exhaust subsystem consists of two trains which draw air from various inlets positioned in the fuel-handling area through radiation monitors, and thence normally through exhaust fans, another set of radiation monitors and to a station vent. Upon indication of high radioactivity in the exhaust duct system, dampers will automatically close and filter train inlet dampers will automatically open to direct air flow through four redundant, 50-percent-capacity filter trains before being redirected to the exhaust fans and station vents.

The supply subsystem is not necessary for safe shutdown operations. Its outside air intake opening for the ventilating air supply unit is protected by missile shields above and in front of the opening. The fuel handling area ventilation system is located in the fuel-handling area of the auxiliary building which is a seismic-Category-I, flood- and tornado-protected structure (refer to Sections 3.4.1 and 3.5.2 of this SER). The non-essential supply subsystem is separated from the essential portions such that its failure will not prevent essential portions such that its failure will not prevent essential safety functions. Essential portions of the system itself are seismic Category I, Quality Group C and are physically separated from high-energy systems. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classifications," are met. Refer to Sections 3.5.1.1, 3.6.1 and 9.3.3 for discussion of protection of essential fuel handling area ventilation system components from internally generated missiles, postulated failures in piping systems and internal flooding.

Each station unit has its own independent fuel handling area ventilation system; thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

The exhaust subsystem of the fuel handling area ventilation system is an engineered safety feature. Each of the two redundant sets of exhaust filter train fans and motor-operated dampers is served from separate trains of the emergency Class 1E standby power and thus meets the single failure criterion. Air exhausted from the fuel handling area is monitored by a radioactive gas detector before entering the filter trains. Each of the four 50-percent-capacity filter trains consists of prefilters, absolute filters and carbon filters. An indication of radioactivity above allowable limits will automatically divert exhaust air flow through the filter trains prior to discharge to the atmosphere through the unit vent. Additional monitoring is provided in the unit vent. Outleakage from the fuel-handling area is prevented by maintaining a negative pressure relative to the outside atmosphere. Thus, the requirements of General Design Criteria 60, "Control of Releases of Radioactive

Materials to the Environment," and 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," 1.52, "Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are satisfied.

Based on the above, we conclude that the fuel handling area ventilation system is in conformance with the requirements of General Design Criteria 2, 60 and 61 as they relate to protection against natural phenomena, control of releases of radioactive materials, radioactivity control, and the guidelines of Regulatory Guides 1.13, 1.29, 1.52 and 1.140 relating to protection against radioactive releases, seismic classification, and system design for emergency and normal operation, and is, therefore, acceptable.

9.4.3 Auxiliary and Radwaste Area Ventilation System

The auxiliary and radwaste area ventilation system is designed to maintain a suitable environment for equipment operation and personnel access and ~~the~~^{to} limit potential radioactive releases to the environment during all modes of operation.

The auxiliary building ventilation system consists of both safety and non-safety-related subsystems and serves all areas of the auxiliary building including all engineered safety features within the building excluding the control room and fuel-handling areas. The non-safety-related (non-seismic Category I, Quality Group D) general ventilation supply and unfiltered exhaust subsystems normally operate in conjunction with the safety-related filtered exhaust subsystems. An additional ventilation ^{sub-}system is provided for certain non-safety related equipment rooms and consists of several individual unit coolers serving these rooms. General ventilation air is supplied to both clean and potentially contaminated areas of the auxiliary building. Control of airborne activity is accomplished by exhausting air supplied to clean areas through the potentially contaminated areas. This air in turn is processed by the filtered exhaust subsystem.

This provides a positive flow of air from clean areas to areas of potential contamination. The remaining air supplied to clean areas is exhausted by the unfiltered exhaust subsystem. All air exhausted from the auxiliary building by the filtered exhaust subsystem and the unfiltered exhaust subsystem is directed to the unit vent where it is monitored by the unit vent radiation monitor prior to release to the atmosphere. Upon receipt of an engineered safety feature actuation signal, all auxiliary building ventilation system components automatically shut down. The filtered exhaust subsystem is then automatically operated and all areas of the auxiliary building with the exception of the ECCS pump rooms are automatically isolated from the filtered exhaust subsystem.

The auxiliary building ventilation system is located in the auxiliary building which is a seismic Category I-, flood-, and tornado-protected structure (refer to Sections 3.4.1 and 3.5.2 of this SER). The system is arranged so that both essential and non-essential equipment and areas are cooled normally by non-safety-related equipment with an entirely separate safety-related subsystem, the filtered exhaust subsystem

This provides a positive flow of air from clean areas to areas of potential contamination. The remaining air supplied to clean areas is exhausted by the unfiltered exhaust subsystem. All air exhausted from the auxiliary building by the filtered exhaust subsystem and the unfiltered exhaust subsystem is directed to the unit vent where it is monitored by the unit vent radiation monitor prior to release to the atmosphere. Upon receipt of an engineered safety feature actuation signal, all auxiliary building ventilation system components automatically shut down. The filtered exhaust subsystem is then automatically operated and all areas of the auxiliary building with the exception of the ECCS pump rooms are automatically isolated from the filtered exhaust subsystem.

The auxiliary building ventilation system is located in the auxiliary building which is a seismic Category I-, flood-, and tornado-protected structure (refer to Sections 3.4.1 and 3.5.2 of this SER). The system is arranged so that both essential and non-essential equipment and areas are cooled normally by non-safety-related equipment with an entirely separate safety related subsystem, the filtered exhaust subsystem

brought into service under emergency conditions. The safety-related auxiliary shutdown panel rooms air conditioning subsystem provides ventilation to the auxiliary shutdown panel in both normal and emergency conditions. The failure of any non-safety-related equipment will not affect the essential functions of any safety-related equipment. Essential (safety-related) portions of the system itself are seismic Category I, Quality Group C and are physically separated from high-energy systems. The outside air intakes are tornado missile protected. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," positions C.1 for safety-related and C.2 for non-safety-related portions, are met. Refer to Sections 3.5.1.1, 3.6.1 and 9.3.3 for discussion of protection of essential auxiliary building ventilation system components from internally generated missiles, postulated failures in piping systems and internal flooding.

The radwaste area ventilation system is classified as non-safety-related (non-seismic Category I, Quality Group D). This system supplies air to the hot machine

shop, waste shipping drum storage, office area, laboratory areas, counting room and the environmental laboratory. It is separated from safety related systems, and therefore its failure will not compromise plant safety. Thus, the requirements of General Design Criterion 2 and guidelines of Regulatory Guide 1.29 Position C.2 are met. The non-safety related auxiliary building general supply subsystem unfiltered exhaust subsystem and supplementary ventilation subsystem are independent for each unit. The radwaste area is shared, between Units 1 and 2 and is served by a single ventilation system. This system operates only during normal conditions and performs no safety functions. Thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are met.

The seismic Category I, Quality Group C, auxiliary building filtered exhaust subsystem consists of two redundant, 100 percent-capacity trains. The exhaust filters consist of preheater/demister section, carbon and absolute filters. The filtered exhaust subsystem

performs both a safety and non-safety-related function, and has different modes of operation in each case. During normal plant operation, the two trains of the filtered exhaust system for each unit operate at 50 percent capacity with the filter normally bypassed. Radiation monitoring is provided in each unit's vent. Upon indication of high radioactivity in the unit vent, the bypass dampers will automatically close and the filter train inlet dampers will automatically open to direct air flow through the filter train before being exhausted. During accident conditions the two trains for each unit operate at 100 percent capacity. Upon receipt of a engineered safety feature actuation signal, isolation dampers will close, shutting off air flow from all areas of the auxiliary building except for the rooms which contain safety-related pumps which are part of the emergency core cooling system (ECCS). One of the two 100-percent-capacity exhaust ducts will exhaust air from the pump rooms through the associated preheater/demister section, filter train, and fan to the unit vent. This assures the integrity and availability of one train of the filtered exhaust subsystem in the event of any single active failure.

The two preheater/demister sections, filter trains, centrifugal fans and associated isolation and inlet vane dampers for each unit are connected to separate trains of the Class 1E emergency standby power. Thus, the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment" and the guidelines of Regulatory Guides 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" and 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants" are satisfied.

The seismic Category I, Quality-Group-C, auxiliary shutdown panel rooms air-conditioning subsystem consist of unit room coolers and heaters independent of other ventilation systems and is tornado missile protected. Two 100-percent-capacity trains for each unit consist of prefilters, coolers, heaters and fans. Each train is provided with electrical power from the emergency (Class 1E) power supply associated with the train it

serves. Cooling water is provided from the associated train of the safety-related nuclear service water system (refer to Section 9.2.1 of this SER). Thus, the requirements of General Design Criteria² "Design Bases for Protection Against Natural Phenomena" and the guidelines of Regulatory Guide 1.29 regarding seismic classification are met. The non-safety-related non-seismic Category I unfiltered exhaust subsystem, general ventilation supply system and radwaste area ventilation system all have adequate filtering and radiation detection features. Thus, the requirements of General Design Criterion 60 and the guidelines of Regulatory Guide 1.52, are satisfied.

The applicant has not provided any information concerning cooling of safety related (HPI, RHR, containment spray, charging, spent fuel pool cooling and auxiliary feedwater) pump rooms under accident conditions when the normal ventilation system is not available. This concern is addressed in our Q410.25. Thus, we can not conclude that the requirement of General Design Criterion 4, "Environmental and Missile Design Bases" is met with respect to the capability to maintain a proper operating environment for essential equipment.

Based on the above, we conclude that the auxiliary building ventilation system is in conformance with the requirements of General Design Criteria 2, 5 and 60 as they relate to protection against natural phenomena, shared systems, and control of releases of radioactive materials to the environment, and the guidelines of Regulatory Guide 1.29, 1.52 and 1.140 relating to seismic classification, and system design for emergency and normal operation. However, we can not conclude that the requirements of General Design Criterion 4 as they relate to the design for assuring a proper operating environment for essential equipment are satisfied until resolution of our concern with pump room cooling. We will report the resolution of our concern in a supplement to this SER.

9.4.4

Turbine Area Ventilation System (Turbine Building Ventilation System)

R 2/18/72

The Turbine Building Ventilation System consists of independent, identical systems in each unit. Each consists of exhaust ducting from lower elevations discharging to the environment through roof exhaust fans. Air intake is through louvers in the outside walls. The system is classified as non-safety-related (non-seismic Category I, Quality Group D). The system maintains an acceptable environment for personnel and the non-essential equipment served during normal plant operation. The system has no safety functions. The system is separated from safety-related plant systems and potentially radioactive areas; therefore, failure of the system will not compromise the operation of any essential plant systems or result in an unacceptable release of radioactivity, and, thus, meets the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.2. Conversely, the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment,"

and the guidelines of Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are not applicable.

Based on our review, we conclude that the Turbine Building Ventilation System meets the requirements of General Design Criterion 2 with respect to the need for protection against natural phenomena as its failure does not affect safety system functions, or result in release of radioactive material, and the guidelines of Regulatory Guide 1.29 concerning its seismic classification, and is, therefore, acceptable.

9.4.5 Engineered Safety Features Ventilation (Diesel Building Ventilation System and Nuclear Service Water Pump Structure Ventilation System and Auxiliary Feedwater Pump Room Ventilation)

The engineered safety features ventilation is provided by the diesel building ventilation system, the nuclear service water pump structure ventilation system and the auxiliary feedwater pump room ventilation system. Those engineered safety features housed in the auxiliary building, that is, the emergency core cooling pumps and component cooling water pumps are ventilated as discussed in Section 9.4.3 of this SER and are not evaluated further in this section. The systems that comprise the required safety features ventilation are not required for control of releases of radioactive materials to the environment, and thus, the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guides 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and 1.140, "Design,

Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are not applicable. These three systems are safety related and are seismic Category I, Quality Group C.

The diesel building ventilation system is designed to maintain a suitable environment for equipment operation during normal and emergency operating modes when the diesel is required and consists of a non-nuclear-safety normal ventilation system and a seismic Category I, Quality Group C, emergency ventilation system. The normal ventilation system consists of a fan, heater, filter and shutoff damper for each diesel generator room and operates only during normal plant operation. When an engineered safety features actuation signal is received the two associated diesel generators in that unit are actuated. This same actuation signal de-energizes the normal ventilation systems fans for the two associated diesel generators and actuates the emergency ventilation system for those two diesel generator rooms. The normal ventilation system fans for the remaining two diesel generator rooms in the other unit continue to operate.

The emergency ventilation system consists of separate ventilation subsystems for each diesel-generator room. The system is housed in the seismic Category I, flood and tornado protected auxiliary building. All the outside air intakes are tornado missile protected. (Refer to Sections 3.4.1 and 3.5.2 of this SER). Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification" are met. The requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components" are not applicable as noted above.

The emergency ventilation subsystem for the diesel building consists of two 50-percent-capacity trains for each of the two diesel-generator enclosures for each unit. Each train includes a birdscreen, an inlet plenum, damper, fan and ductwork designed to withstand an SSE and to mix recirculated and outdoor air for temperature control. Ventilation is provided also to the oil storage day tank which is within the diesel building. Combustion air for the diesel-generators is taken from a common plenum along with the normal ventilation air. Refer to Section 9.5.8 of this SER for discussion of the combustion air system.

Each diesel-generator emergency ventilation subsystem is powered from the emergency (Class 1E) bus corresponding to the diesel-generator it serves. Thus, operation of at least one diesel generator per unit is assured in the event of a single failure in any system component. However, the design does not meet the requirements of General Design Criterion 4, "Environmental and Missile Design Bases" and 17, "Electric Power Systems," since the design does not comply with the recommendations of NUREG/CR-0660 for assuring a proper operating environment for the diesel generator. The ventilation air intakes are low to the ground, thus air filters should be provided to avoid unacceptable dust accumulation on essential equipment within the diesel enclosures. Further recirculation of exhaust gases into the ventilation air intake appears possible due to the proximity of the diesel generator engine exhaust to the ventilation intake openings. In addition, we cannot determine if the essential long term fuel oil storage tanks which are located remotely from the diesel generator building are buried or located in an open pit. If tanks are in a pit, additional information on the ventilation provided to control oil fumes, for prevention of potential fires is necessary.

These concerns have not been addressed previously and the applicant is requested to respond to them.

The nuclear service water pump structure ventilation system is housed in the remotely located nuclear service water pump structure and is designed to maintain a suitable operating environment for the nuclear service water pumps during all operating modes. It consists of two 100 percent capacity subsystems for each train of the service water system and is common for both plant units, that is, each ventilation subsystem serves service water system pump trains for both plant units. Each ventilation train includes a birdscreen, dampers, vane axial fan and duct work. The system is seismic Category I, Quality Group C except for the non-essential maintenance fan. Temperature is controlled by recirculation of the pump structure air with additional outside air. A passive portion of the ductwork is common to both trains. Switchover between each 100 percent capacity subsystem is done manually. All essential fans, dampers, ductwork and supports are designed to withstand the safe shutdown earthquake. Essential electrical components of each train required for ventilation of the building during accident conditions are

connected to separate emergency Class 1E standby power supplies, thus, assuring system function in the event of a single failure.

The nuclear service water pump structure ventilation system is located completely within a seismic Category I structure and all essential components are protected from tornado missile damage (refer to Sections 3.4.1 and 3.5.2 of this SER). The outside air intakes are tornado missile protected. Thus, the requirements of General Design Criteria 2 and 5 with respect to protection against natural phenomena, sharing between units and the guidelines of Regulatory Guide 1.29 with respect to seismic classification are met.

The ventilation system for the auxiliary feedwater pump rooms is included in the auxiliary building ventilation system discussed in Section 9.4.3 of this SER. However, it is not part of the safety related filter exhaust subsystem of the auxiliary building ventilation system. As an engineered safety feature the auxiliary feedwater pump rooms and controls must receive assured ventilation. The applicant has not

indicated that a proper operating environment is maintained for the auxiliary feedwater pump on a loss of ventilation due to failure of the non-seismic Category I unfiltered auxiliary building exhaust system in accident conditions (including loss of offsite power. This concern has not been addressed previously and the applicant is requested to respond.

Based on the above, we conclude the engineered safety features ventilation is in conformance with the requirements of General Design Criteria 2 and 5 as they relate to protection against natural phenomena and shared systems and the guidelines of Regulatory Guide 1.29 concerning seismic classification. However, we cannot conclude that the system is in conformance with the requirement of General Design Criteria 4 and 17 for the diesel generator building ventilation system and the auxiliary feedwater pump room ventilation system as they relate to assurance of the capability to maintain a proper operating environment in view of the concerns identified above. We will report resolution of our concerns in a supplement to this SER.

10.3. Main Steam Supply System

The function of the main steam supply system is to convey steam from the steam generators to the high-pressure turbine and other auxiliary equipment for power generation. The steam produced in the four steam generators is conveyed in separate lines from the steam generators through the main steam isolation valves. The four individual main steam lines each contains one main steam isolation valve (MSIV). The portions of the main steam lines outside the containment, including the main steam isolation valves, the main steam safety valves and the atmospheric relief valves are located in seismic Category I, flood-and tornado-protected structures (main steam doghouses) (refer to Sections 3.4.1 and 3.5.2 of this SER) and are Quality Group B and seismic Category I, thereby satisfying the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification."

Main steam isolation is provided by a spring-loaded, air-piston operated valve in each steam line located just outside the containment. The MSIVs automatically close on high or low steam line pressure or on a high-high containment pressure signal, and can be operated from the main control room or local panels. The main steam isolation valves are designed to close in less than five seconds and are also designed to stop steam flow from either direction. A steam line break upstream or downstream

of the MSIVs coupled with an MSIV failure to close will not result in the blowdown of more than one steam generator. In the event of a steam-line break upstream of an MSIV and a failure of an MSIV to close on an unaffected steam generator, blowdown of the unaffected steam generator through the break is prevented by the closure of the MSIVs for the affected steam generator. Blowdown through the turbine and condenser is prevented by closure of the non-seismic Category I turbine stop valves and turbine bypass valves which serve as an acceptable backup for this accident in accordance with the guidelines of NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 memorandum from Director, NRR to NRR Staff."

One seismic Category I, Quality Group B, air piston operated, atmospheric relief valve is provided on each main steam line upstream of the MSIV. These valves will automatically operate to prevent lifting of the lowest set safety valve during small pressure transients. The valves also remove heat from the nuclear steam supply system during periods when the main turbine or condenser are not in service. The valves fail closed (safe position) on loss of air supply. The combined capacity of the power operated relief valves is sufficient to effect a 50 degree Fahrenheit per hour cooldown rate.

Twenty seismic Category I, Quality Group B, safety valves (five on each main steam line) are also provided. The safety valves

have a combined capacity of 105 percent of the total steam flow at a pressure not exceeding 110 percent of the system design pressure. The safety valves and atmospheric relief valves are located outside containment and upstream of the MSIVs in the seismic Category I doghouses between the steam tunnel and containment which are accessible areas. The main steam supply system is fully tested and inspected before initial startup. However, the applicant has not committed to provide the capability to operate the power-operated atmospheric relief valves remotely from the control room on a loss of offsite power condition. Further, the applicant has not committed to perform a local operability verification test if these valves are not controllable from the control room following an SSE. These concerns are addressed in Q410.27. Thus, the design of the main steam supply system does not meet the requirements of General Design Criterion 34, "Residual Heat Removal," and the applicable guidelines of Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System."

The equipment required to function in order to assure main steam isolation when called upon is protected against the effects of high energy pipe breaks (refer to Section 3.6.1 of this SER). This equipment is located in tornado-missile protected structures and is located such that it is unaffected by internal generated missiles (refer to Section 3.5.1.1 of this SER). Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide

1.117, "Tornado Design Classification," and Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," are satisfied. Protection *against* low trajectory turbine missiles is discussed in Section 3.5.1.3 of this SER. There is no sharing between Units 1 and 2 of any portion of the main steam supply system; thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

Based on the above, we conclude that the main steam supply system from the steam generators through the main steam isolation valves meets the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena, missiles and environmental effects and the guidelines of Regulatory Guides 1.29, and 1.117 and Branch Technical Position ASB 3-1 relating to the system's seismic and quality group classification, protection against high energy pipe breaks. However, we cannot conclude that the requirements of General Design Criterion 34 and the guidelines of Branch Technical Position RSB 5-1 are met until satisfactory resolution of our concern for loss of control room operability of the power operated atmospheric relief valves during a loss of offsite power and SSE as discussed above. We will report resolution of our concern in a supplement to this SER.

10.4.5 Circulating Water System (Condenser Circulating Water System)

The non-safety¹ related (Quality Group D, non-seismic Category I) condenser circulating water system (CCW) supplies cooling water to the main and feedwater pump turbine condensers to condense turbine exhaust steam. This water is circulated to three parallel mechanical draft cooling ~~systems~~ ^{Towers} per unit where heat is rejected to the atmosphere. Circulating water from the cooling tower basins is directed to the low-, intermediate- and high-pressure condensers in series. The feedwater pump turbine condensers are supplied cooling water by a parallel piping system after the high-pressure condenser. Four parallel condenser circulating water pumps return the heated water to the cooling towers. The condenser circulating water system is not required to maintain the reactor in a safe shutdown condition or mitigate the consequences of accidents.

The applicant has analyzed the flooding consequences resulting from failure of the circulating water system. Flooding of the turbine and service buildings will occur. However, there is no equipment essential to plant safety in either the turbine or service buildings, and all penetrations and passageways from the turbine or service buildings to the auxiliary buildings which house safety related equipment will be watertight to elevation 576.0 feet. There are no other penetrations to safety related plant areas from the turbine and service buildings. The maximum water level due to a simultaneous failure of the CCW system on

both units and the subsequent draining of all water in the two closed loop cooling systems back to their respective turbine buildings results in a maximum water elevation of 575.4 feet. The available storage volume in both turbine and service buildings is 1,690,000 cubic feet. The Volume of condenser cooling water from both systems that could flood into the turbine and service buildings is 1,340,000 cubic feet. Thus, flood water is contained within nonessential plant areas, and therefore the requirements of General Design Criterion 4, "Environmental and Missile Design Bases" are met.

Based on our review, we conclude that the condenser circulating water system meets the requirements of General Design Criterion 4 with respect to protection against environmental effects (flooding) on safety-related equipment as its failure does not compromise plant safety, and is therefore, acceptable.

10.4.7 Condensate and Feedwater System (Condensate and Feedwater Systems)

The condensate and feedwater system provides feedwater from the condenser to the steam generators and includes the piping and components arranged in stages. Starting at the condenser hotwell, the condensate passes through hotwell pumps, polishing demineralizers, air ejector condensers, gland steam condenser, low-pressure heaters, condensate booster pumps, main feedwater pumps, high pressure heaters and containment isolation valves to the four steam generators. The steam generators are the preheat type; therefore, two separate feedwater connections are provided at the steam generators: a tempering line connection which is located on the upper part of the steam generator above the normal water level used during startup, shutdown, and light load conditions, and a preheat line connection to the preheat section which is located on the lower part of the steam generator, used only during normal power operation.

The system serves no safety function (with the exception of containment isolation integrity) and is, therefore, classified as non-safety related (Quality Group D, non-seismic Category I). Adequate isolation is provided at connections between seismic and non-seismic Category I systems, and, therefore, failure of non-safety related portions of the condensate and feedwater system will not affect safe plant shutdown.

The condensate and feedwater system is designed with features to preclude the potential for damaging flow instabilities (water hammer). The conditions necessary to produce water hammer in the main feedwater piping and/or steam generators must occur simultaneously as either low steam generator temperature and extremely low steam generator level (below the level which initiates the auxiliary feedwater system) or low steam generator temperature and low steam generator pressure.

Design features are incorporated to minimize the potential and severity of any possible water hammer event. Loop seals in the feedwater piping minimize the volume of possible steam voids in the unlikely event that the steam generator water level falls below the main feedwater nozzle.

It should be noted that the Catawba preheat model steam generator (Westinghouse design) does not include a feedring and therefore, the guidance in Branch Technical Position ASB 10-2 "Design Guidelines for Water Hammers in Steam Generators with Top Feedring Designs," is not applicable.

However, our evaluation of the Catawba preheat model steam generator for water hammer potential indicates that the applicant should perform a plant specific verification test as indicated in our Q410.29 to demonstrate that no damaging water hammer will occur. It is our position that the applicant perform this preoperational test using the standard plant operating procedures to demonstrate

the ability to transfer main feedwater flow from the top feed nozzle to the main feed nozzle without unacceptable water hammer as recommended in NUREG/CR-1606, "An Evaluation of Condensation Induced Water Hammer in Preheat Steam Generators." Preoperational testing of the auxiliary feedwater system (AFWS) will verify that no unacceptable water hammer occurs on automatic initiation of AFWS.

Based on the above, we conclude that the safety-related portion of the condensate and feedwater system meets the requirements of General Design Criteria 2, 44, 45 and 46 with respect to its protection against natural phenomena, missiles and environmental effects, decay heat removal function, in-service inspection and testing, and the guidelines of Regulatory Guide 1.29 with respect to its seismic classification. However, the applicant has not committed to perform verification testing with respect to prevention of damaging water hammer in accordance with the recommendations of NUREG/CR-1606, and the system is, therefore, unacceptable. We will report resolution of our concern in a supplement to this SER.

10.4.9 Auxiliary Feedwater System

We reviewed the auxiliary feedwater system (AFWS) against the specific acceptance criteria of Standard Review Plan (SRP) Section 10.4.9 as follows:

1. General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," as related to structures housing the system and the system itself being capable of withstanding the effects of earthquakes. Acceptability is based on meeting position C.1 of Regulatory Guide 1.29, "Seismic Design Classification," for safety-related portions, and position C.2 for non-safety-related portions.
2. General Design Criterion 4, "Environmental and Missile Design Bases," with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks. The basis for acceptance for meeting this criterion is set forth in the SRP Section 3.5 and 3.6 series.

3. General Design Criterion 5, "Sharing of Structures, Systems and Components," as related to the capability of shared systems and components important to safety to perform required safety functions.
4. General Design Criterion 19, "Control Room," as related to the design capability of system instrumentation and controls for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown. Acceptance is based on meeting Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," with regard to cold shutdown from the control room using only safety-grade equipment.
5. General Design Criteria 34, "Decay Heat Removal," and 44, "Cooling Water," to assure:
 - a. The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions.
 - b. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure.
(This may be coincident with the loss of offsite power for certain events.) Branch Technical

Position ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactors," as it relates to AFWS pump drive and power supply diversity shall be used in meeting these criteria.

- c. The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

In meeting these criteria, the recommendations of NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," and NUREG-0635, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants," shall also be met. An acceptable AFWS should have an unreliability in the range of 10^{-4} to 10^{-5} per demand based on an analysis using methods and data presented in NUREG-0611 and NUREG-0635. Compensating factors such as other methods of accomplishing the safety functions of the AFWS or other reliable methods for cooling the reactor core during abnormal conditions may be considered to justify a larger unavailability of the AFWS.

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6. General Design Criterion 45, "Inspection of Cooling Water System," as related to design provisions made to permit periodic in service inspection of system components and equipment.

7. General Design Criterion 46, "Testing of Cooling Water System," as related to design provisions made to permit appropriate functional testing of the system and components to assure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions. In meeting this criterion, the technical specifications should specify that the monthly AFWS pump test shall be performed on a staggered test basis to reduce the likelihood of leaving more than one pump in a test mode following the tests.

The following evaluation discusses the implementation of the above acceptance criteria and follows the format of the Review Procedures identified in SRP Section 10.4.9 (NUREG-0800). This evaluation also incorporates our review of the applicant's response to Item II.E.1.1, "Auxiliary Feedwater System Reliability," of NUREG-0737, "Clarification of TMI-Action Plan Requirements." This

includes:

1. An evaluation against the deterministic criteria of the Standard Review Plan.
2. An evaluation against the generic recommendations of NUREG-0611 and 0635.
3. The evaluation of system reliability based on the applicant's reliability study.
4. An evaluation of the design basis for the flow capability for the system.

~~This includes an evaluation against the systematic criteria of the Standard Review Plan, an evaluation against the generic recommendations of NUREG-0611 and NUREG-0635, evaluation of the design bases for the flow capability based on the applicant's reliability study.~~

The AFWS is designed to supply an independent source of water to the steam generator in the event of a loss of main feedwater supply. The system consists of three redundant, safety-related essential trains each with its own pump. Two full-capacity (500 gpm each) motor-driven pumps are powered from two separate trains of emergency on-site electrical power, each normally supplying feedwater to a separate pair of the four steam generators. Additionally 1000-gpm turbine-driven pump which supplies feedwater normally to steam

generators is driven from steam contained in either of the two steam generators being fed by this pump. All AFWS pumps can be valved to supply feedwater to any of the four steam generators. Only one of these three pumps is necessary to supply the minimum total feedwater requirements to at least two intact steam generators in order to ensure safe shutdown. The AFWS water supply is normally provided by three non-safety-related sources of condensate-quality water. These are: (1) the auxiliary feedwater condensate storage ~~tank~~^{tank}, (2) the upper surge tanks and (3) the condenser hotwell. The essential safety-related source of water for the AFWS is the standby nuclear service water pond. An additional non-safety-related water source exists in the buried piping of the condenser circulating water system and is included as part of the standby shutdown system. With the exception of the auxiliary feedwater storage tank and the common standby nuclear service water pond, the AFWS is not shared between units and, therefore, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable. Sharing of these water sources is discussed in Sections 9.2.6 and 9.2.1 of this SER.

1. We have reviewed the AFWS design in order to verify its acceptability with respect to its classification and operating characteristics.
 - a. Minimum performance requirements for the AFWS have been identified and are found not sufficient for the various functions of the system. This is discussed in more detail in the following section of the SER.
 - b. Adequate isolation of essential portions of the AFWS from non-essential portions and from other non-essential systems is included in the system design. Seismic Category I electrically operated control-room-actuated isolation valves and check valves are provided on the non-safety-related auxiliary feedwater condensate storage tank, upper surge tank and condenser hotwell supply lines to the header supplying the AFWS pumps upstream of the connection from the essential nuclear service water system. A seismic Category I check valve is provided in the common line from all three non-seismic Category-I condensate-quality feedwater sources. Similarly, seismic Category I, electrically operated isolation valves and a check valve are provided in the non-seismic Category I feedwater supply in the buried condenser circulating water.

system piping. These valves will prevent flow loss in the event of a failure of any of these non-seismic Category I tanks or pipes. The essential AFWS lines to each steam generator connect to the non-essential main feedwater bypass feed and main feedwater tempering feed lines through seismic Category I fail-closed isolation valves and check valves. These valves close automatically on receipt of a safety-injection signal. Other points of interface with non-essential systems are the auxiliary feedwater pump discharge lines to the upper surge tank which are provided for minimum flow and testing purposes. Seismic Category I self-contained automatic recirculation valves or orifices are provided to assure individual pump minimum flow and loss of flow in the event of downstream pipe failure. Interfaces with the Chemical Addition system are provided with seismic Category I normally closed manual valves and check valves. The above features provide sufficient isolation to assure that system function is not impaired in the event of failure of a non-essential component. Therefore, we conclude that the AFWS meets the isolation requirements of General Design Criteria 2 and 44 and the guidelines of Position C.2 of Regulatory Guide 1.29.

- c. Essential portions of the AFWS are designed to seismic Category I, and Quality Group C requirements with the exception of the portions of the main feedwater bypass feed, main feedwater tempering flow, chemical addition and auxiliary feedwater line in the doghouse structure area as they combine to penetrate containment. These portions form the containment penetration and isolation and are seismic Category I, Quality Group B.

However, the applicant indicates (Fig. 10.3.2-2, note 24) that the turbine drive for the turbine driven AFW pump is not ASME Boiler and Pressure Vessel Code, Section III, Class 3 and seismic Category I. This applies also to the stop and control steam valves to the turbine. It is our position that this component be qualified to function during the SSE. This concern has not been previously addressed. Therefore, we conclude that the AFWS does not meet the requirements of General Design Criteria 2 and the guidelines of Regulatory Guide 1.29 with respect to its seismic classification.

- d. Provisions for AFWS testing and inspection are included in the design. Each essential AFWS pump is equipped with a recirculation line to the upper surge tank for periodic functional testing purposes.

A continuous recirculation during pump operation is provided through a fixed orifice and local realignment of valves is required for periodic performance testing. However, ~~two~~ ^{Two} trains are operable when one is being tested. Periodic surveillance testing of the essential pumps and their associated flow trains is identified in the plant Technical Specifications Subsection 3.7.1.2 which states that an inoperable AFWS pump will be restored to operable status within 72 hours or the plant will be taken into a hot shutdown condition within six hours. The AFWS is tested each month for pump capacity and valve positioning and each 18 months for automatic start-up capability. However, the applicant's response to Recommendation GS-6 of NUREG-0611 is not satisfactory with respect to flow path verification testing. We require that the applicant revise plant technical specifications to incorporate an AFW flow path verification test where water is pumped from the primary water source to the steam generators before startup after any cold

shutdown of 30 days or longer. Therefore, we cannot conclude that the AFWS meets the requirements of General Design Criterion 46 and the recommendations of NUREG-0611 with respect to functional testing. The AFWS components are located in areas that are accessible during normal plant operation to permit periodic in-service inspection. The applicant has committed to provide a second (independent) operator verification of proper AFWS valve position following restoration of an AFWS train to service after periodic testing or maintenance. Therefore, we conclude that the AFWS meets the requirements of General Design Criterion 45 and the recommendations of NUREG-0611 regarding provisions for in-service inspection.

2. We have reviewed the AFWS design for protection against the effects of natural phenomena, pipe breaks or cracks in fluid systems outside containment, single system component failures, loss of an onsite motive power source, or loss of offsite power.

a. All essential AFWS components are located in seismic Category I structures. Protection against failure of non-seismic Category I plant features is provided. Failure of non-seismic Category I systems, components or structures will not adversely affect AFWS function as adequate isolation or separation is provided. In the event of failure of the non-seismic Category I condensate-quality water sources and the buried piping of the condenser circulating water system, the transfer to the seismic Category I nuclear service water system occurs automatically on low suction pressure. However, we require the applicant to commit to perform a test of this feature in order to assure adequate suction is available to prevent AFW pump damage during the ~~transfer~~ ^{transfer}. In addition, the applicant should verify that air-binding of the AFWS pumps does not occur prior to transferring pump suction supply from condensate-quality water sources (auxiliary feedwater condensate hotwell) to the essential nuclear service water system on failure of any of these non-safety-related sources.

Therefore, we cannot conclude that the essential portions of the AFWS are adequately protected from earthquakes. We can not conclude that the requirements of General Design Criterion 2 and recommendations of NUREG-0611 are met pending satisfactory responses to our concerns regarding indication of loss of normal auxiliary feedwater supply, and air-binding of AFWS pumps .

- b. Protection against missiles, tornadoes and floods is provided. Essential portions of the AFWS are located in the tornado-missile- and flood-proof auxiliary building doghouse and nuclear service water pump house. All AFWS components are located above design flood level. The non-essential condensate-quality water sources are located in the service and turbine buildings and are not tornado-missile-protected. Automatic transfer of AFW pump suction is provided in the event of failure of the three due to tornado missiles as well as earthquakes identified above. Thus, an AFWS water supply is assured. Each essential AFWS pump is located in a separate area within the auxiliary building, which is provided with adequate drainage and provides protection against internally generated missiles. (Refer to Sections 3.4.1, 3.5.1.1 and 3.5.2 of this

report for further discussion.) Environmental qualification of AFWS components is discussed in Section 3.11 of this report. Therefore, we conclude that the essential portions of the AFWS are protected from floods, tornadoes, and missiles and meet the requirements of General Design Criteria 2 and 4 and the recommendations of NUREG-0611 subject to the automatic suction supply transfer test previously mentioned.

- c. The AFWS trains are not used during start-up and shutdown; therefore, they are not designed as high-energy lines. Protection against moderate-energy pipe cracks in the AFWS is provided by separation and redundancy of equipment. Essential portions of the AFWS are separated from the effects of high- and moderate-energy line breaks in other systems. These include the effects of pipe whip, jet impingement and flooding. High-energy piping systems are not located in the area of essential AFWS components. Therefore, we conclude that the essential portions of the AFWS are protected against the effects of pipe whip, jet impingement and flooding associated with pipe breaks and meet the requirements of General Design Criterion 4 with respect to pipe breaks outside containment. Protection against

the effects of pipe breaks is discussed further in this SER under Section 3.6.1. Environmental qualification of AFWS components with respect to pipe breaks is discussed in Section 3.11 of this SER.

- d. The essential AFWS trains can function automatically as required in the event of a loss of offsite power. The heat transfer path from the steam generators under this condition is to the atmosphere via the atmospheric relief valves. (Refer to Section 10.3.1 of this SER for further discussion.) The essential turbine driven pump functions independently on any offsite **AC** power as discussed subsequently in this CER section and this is not affected by a loss of offsite power. Power for the motor-driven pumps is normally provided by the station auxiliary power system. Each motor-driven pump is provided emergency power from one of the two onsite emergency diesel-generators. The power supply train for each pump is physically separated from that of the other pump. Driving steam for the turbine-driven pump is provided from either the B or C steam generator main steam lines upstream of the main steam isolation valves and is discharged to the atmosphere from the turbine. Each steam supply line is provided with an air piston-operated valve that opens on a signal to start the turbine-driven pump. Redundant control systems are provided to assure opening of each valve on a turbine-driven pump start signal. Any power or air failure will result in the valve failing open. A check valve is provided in each

steam supply to prevent flow reversal. Each auxiliary feedwater pump discharge line is provided with a normally open motor operated isolation valve, a normally open air operated fail open flow control valve and a check valve in individual feedlines to each steam generator. The discharge from each AFW also has a loop for full flow pump testing. Self-contained automatic recirculation valves are provided to assure individual pump minimum flow when needed during operation. These pump recirculation valves are self-regulating self-contained control valves.

Therefore, we conclude that requirements of General Design Criteria 34 and 44 and the recommendations of NUREG-0611 with respect to the AFWS ability to transfer decay heat from the reactor coolant system under a loss of offsite power are satisfied.

- e. The AFWS is designed to accommodate a single failure in any active system component without loss of function. The essential portion of the AFWS consists of three redundant, 100%-capacity trains. All three trains are capable of supplying any steam generator. The trains are

powered from separate and diverse sources. The essential AFWS pumps are provided with three suction supply connections to the condensate-quality sources through separate normally open electrically operated valves. The essential supply of nuclear service water system is redundant from two drains, each provided with motor operated, normally closed valves. Thus, adequate feedwater is assured to two steam generators in the event of a postulated design basis accident concurrent with a single failure. Adequate isolation is provided for all essential portions of the AFWS from non-essential portions and systems (see Item 1b above). Therefore, we conclude that the AFWS meets the requirements of General Design Criteria 34 and 44 and the recommendations of NUREG-0611 with respect to single failure.

f. Adequate auxiliary feedwater flow is assured to the steam generators in the event of the loss of offsite and all emergency onsite ac power by relying upon the safety-related turbine-driven pump subsystem which can perform its safety function independent of ac power. Loss of all ac power will not adversely affect the position of motor-operated valves in the turbine-driven pump

subsystem. All electrically operated valves in the normal turbine-driven pump discharge path to B and C steam generators are normally open and fail as is on loss of ac power. The motor-operated turbine-driven pump suction isolation valve and hotwell source isolation valve are normally open and fail as is on loss of all ac power. The motor-operated supply valves from the condenser circulating water system are supplied dc power from the standby shutdown facility, batteries. Therefore, we conclude that the AFWS meets the requirements of General Design Criteria 34 and 44 and the guidelines of BTP ASB 10-1 and recommendations of NUREG-0611 with regard to AFWS power diversity.

g. The motor-driven pumps will automatically start and provide the minimum required feedwater flow within one minute following any of these conditions:

1. Two-out-of-four low-low level alarms in any one of the four steam generators
2. Loss of both main feedwater pumps
3. Initiation of the safety injection signal

4. Loss of station normal auxiliary electric (offsite) power.

The turbine driven pump will automatically start and provide the minimum required feedwater flow within one minute following either of these conditions:

1. Two out of four low-low level alarms in any two of the four steam generators,
2. Loss of station normal auxiliary electric power.

Flow of auxiliary feedwater to each steam generator is monitored and controlled manually from the control room.

Further discussion of automatic AFWS initiation and flow indication including compliance with the recommendations of Item II.E.1.2 of NUREG-0737 is contained in Section 7.3 of this SER. Manual capability to initiate and control the AFWS pumps/valves and isolate AFWS train is provided in the control room. This capability is also provided at the local

auxiliary shutdown panel. In addition, the standby shutdown facility provides complete operating capability for the turbine driven pump using water from the buried piping of the condenser circulating water system. Local manual control at the individual components is also available. Therefore, we conclude that the AFWS provides adequate instrumentation and control for prompt initiation of a shutdown using safety related equipment in accordance with the requirements of General Design Criterion 19 and the guidelines of BTP RSB 5-1 and the recommendations of NUREG-0611.

The applicant has described the design of the AFWS to prevent excessive pump runout following a main steam or feedwater line break (steam generator depressurization and still maintain a minimum AFWS flow to at least two intact steam generators. Certain automatic isolation functions are provided in the event of a steam generator depressurization. Operator action is not required for a minimum of 30 minutes following such an event. Further the design of the AFWS flow control valves includes travel stops set at a predetermined position to provide pump runout protection and optimize system resistance for various accidents. We are unable to verify that the automatic isolation

features and valve travel stops will not adversely affect the capability of the AFWS to deliver required flow when considering AFWS component failures during the more probable occurrences such as feedwater transients or loss of offsite power. The applicant should address this concern in order to assure maximum AFWS reliability in accordance with NUREG-0611 Criteria.

AFWS flow is not throttled to avoid the occurrence of water hammer as system design provisions minimize the possibility of such a condition. We cannot conclude that the AFWS meets the requirements of General Design Criteria 34 and 44 with respect to its ability to transfer heat under accident conditions and provide isolation to assure system function as described above. We conclude that the system meets the recommendations of NUREG-0611 concerning throttling for water hammer prevention.

- h. Each AFWS pump is designed to provide 100% of the flow necessary for residual heat removal over the entire range of accidents in accordance with conservatism assumed in the accident analysis. A minimum of 200,000 gallons of water is reserved by Technical Specification contained in the upper surge tank, condenser hot well and auxiliary feedwater condensate storage tank. This volume assures a cooldown of the

reactor coolant system to the shutdown cooling system cut in temperature. An additional long term backup source of AFWS supply is provided by the safety related nuclear service water system.

The turbine driven AFW pump can also be supplied from the condenser ~~incubating~~ ^{circulating} water system to meet postulated fire and sabotage events. However, the applicant has not provided redundant level indicators on the primary water sources, as recommended in NUREG-0611. We require the applicant to provide this feature. Also the applicant ^{should provide} procedures as guidance for the operator for transferring AFWS water supply to the backup sources in accordance with NUREG-0611 recommendations.

Therefore we cannot conclude that the AFWS meets the decay heat removal requirements of General Design Criteria 34 and 44 and the recommendations of NUREG-0611.

3. The generic recommendations of NUREG-0611 as they relate to improvements in AFWS design, procedures and technical specifications have been discussed in the ~~preceding~~ ^{preceding} paragraphs. The applicant has committed to perform a 48 hour endurance test on each AFW pump prior to initial fuel loads. This commitment is acceptable.

4. By letter dated October 12, 1981, the applicant has submitted an auxiliary feedwater system reliability analysis in accordance with Enclosure 1 of the March 10, 1980 generic NRC letter concerning Item II.E.1.1 of NUREG-0737. Our review of this analysis is not yet complete, and thus the requirements of General Design Criteria 34 and 44 concerning the reliability of decay heat removal systems are not satisfied. We will report on conclusions of the study in a supplement to this SER.

5. We have reviewed the applicant's response to the staff request in Enclosure 2 of the letter dated March 10, 1980 regarding the design basis for the AFWS flow requirements. We conclude that the applicant's design basis for AFWS flow requirements is acceptable.

Based upon our review, we conclude that the auxiliary feedwater system meets the requirements of General Design Criteria 4, 5, 19 and 44 with respect to protection against missiles and environmental effects, shared systems and operational capability from the control room, and inspection inservice and the guidelines of Branch Technical Positions ASB 10-1 and RSB 5-1 concerning power diversity and design of decay heat removal systems are met. However, we cannot conclude that the AFWS fully conforms to the requirements of General Design Criteria 2, 34, 44

and 46 concerning protection against natural phenomena,
decay heat and cooling water capability and function testing
and the guidelines of recommendations of NUREG-0611,
concerning generic improvements of the AFWS design,
procedures and technical specifications and AFWS reliability pending satisfactory resolution of the above identified concerns. We will report resolutions of our concerns in a supplement to this SER.

SAFETY EVALUATION REPORT

Catawba Nuclear Station, Units 1 & 2Operations Phase Quality Assurance Program17 Quality Assurance17.1 General

The description of the quality assurance (QA) program for the operations phase of the Catawba Nuclear Station, Units 1 & 2 is contained in Section 17.2 of the FSAR which includes a reference to the Duke Power Company topical report entitled "Quality Assurance Program Duke 1-A." Our evaluation of this QA program is based on a review of this information and discussions with representatives from Duke Power Company (Duke). We assessed Duke's QA program for the operations phase to determine if it complies with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," the applicable QA-related Regulatory Guides listed in Table 1, and the Standard Review Plan, Section 17.2, Rev. 1, dated February 1979, "Quality Assurance During the Operations Phase."

Since this review effort, the Standard Review Plan was updated to Revision 2. An additional review, conducted to determine the extent to which the Duke QA program meets Revision 2, shows that the QA program complies with Revision 2 of the Standard Review Plan except for the following controls to which no clear commitment is provided.

The program does not specify:

1. Criteria for determining the size of the QA organization including the inspection staff (Ref. SRP 17.1, item 1A5);
2. That designated QA individuals are involved in day-to-day plant activities important to safety (i.e., the QA organization routinely attends and participates in daily plant work schedule and status meetings to assure they are kept abreast of day-to-day work assignments throughout the plant and that there is adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments) (Ref. SRP 17.1, item 1B6);
3. That the QA organization reviews and documents concurrence with quality-related procedures (Ref. SRP 17.1, item 2B1b);
4. That the QA organization participates early in the QA program definition stage to determine and identify the extent QA controls are to be applied to specific structures, systems, and components. This effort involves applying a defined graded approach to certain structures, systems, and components in accordance with their importance to safety and affects such disciplines as design, procurement, document control, inspection tests, special processes, records, audits, and others described in 10 CFR Part 50, Appendix B (Ref. SRP 17.1, item 2B3 partial);
5. That procedures are established and described requiring a documented

check to verify the dimensional accuracy and completeness of design drawings and specifications (Ref. SRP 17.1, item 3E1);

6. That procedures are established and described requiring that design drawings and specifications be reviewed by the QA organization to assure that the documents are prepared, reviewed, and approved in accordance with company procedures and that the documents contain the necessary quality assurance requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results (Ref. SRP 17.1, item 3E2);
7. That for commercial "off-the-shelf" items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verification requirements shall be established and described to provide the necessary assurance of an acceptable item by the purchaser (Ref. SRP 17.1, item 7B4); and
8. That program procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or define how and when inspections are performed. The QA organization participates in the above functions (Ref. SRP 17.1, item 10A partial).

The above quality assurance controls generally provide for increased involvement of the QA organization in certain quality-related activities, and increased emphasis on the need for procedures, documentation, and requirements for other quality-related activities.

As such, we believe that adequate quality assurance controls in these areas are already included in the QA program for the Catawba Nuclear Station, Units 1 & 2, and that the above listed items described in the SRP, Revision 2, need not be mandatory at this time.

17.2 Organization

The structure of the organization responsible for the operation of the Catawba Nuclear Station, and for the establishment and implementation of the operations phase quality assurance program is shown in Figure 17-1. The President has overall responsibility for planning, design, construction, and operation of the company's generation and transmission facilities. The Senior Vice-President, Engineering & Construction, who reports directly to the President, is responsible for the company's engineering, construction, and quality assurance activities. The Senior Vice-President, Production and Transmission, who is responsible for directing the operation of the company's generation and transmission facilities, also reports directly to the President.

The Corporate Quality Assurance Manager, who reports directly to the Senior Vice-President, Engineering & Construction, directs the Quality Assurance Department and has the sole responsibility for implementing and executing quality assurance policies, goals, and objectives.

The Quality Assurance Department is responsible for all quality assurance activities related to Duke Power Company nuclear stations, including quality assurance for the operations phase of the Catawba Nuclear Station. Duke Power Company has committed that the Corporate Quality Assurance

Manager is independent of influences and responsibilities for schedules and costs. The organization chart of the Quality Assurance Department is presented in Figure 17-2. Quality Assurance Department personnel are organizationally separate and independent from those persons responsible for performing engineering, construction, operational, and procurement activities.

Quality assurance personnel have the freedom and responsibility to identify quality problems; to initiate, recommend, or provide solutions; and to verify and report directly to management the implementation of such solutions. These personnel have written authority and responsibility to stop work when the continuance of work would produce results adverse to quality. A resident Senior Quality Assurance Engineer, who reports to the Corporate Quality Assurance Manager through the Quality Assurance Manager, Operations, is assigned to the Catawba Nuclear Station. The resident Senior Quality Assurance Engineer is responsible for all Quality Assurance Department activities at the Catawba Nuclear Station. He is supported by quality assurance engineers and technicians and by a quality control staff.

Specific responsibilities of the Quality Assurance Department with regard to nuclear station operational activities are identified in the aforementioned topical report on quality assurance. In general, the Quality Assurance Department performs checking, auditing, and inspecting functions to verify that activities have been correctly performed. The corporate organizational structure of the Duke Power

Company is such that the individuals performing such verifications are independent of the personnel directly responsible for performing the activities being checked, inspected, or audited.

The Catawba Nuclear Station Manager reports to the Vice-President, Steam Production through the Manager, Nuclear Production, as shown in Figure 17-3. The Catawba Nuclear Station Manager is directly responsible for the safe operation of the facility, and directs the activities of the station organizations as shown typically in Figure 17-4.

Activities affecting the operational quality assurance program and department interfaces, including the quality of nuclear safety-related structures, systems, and components are performed by or under the cognizance of the Steam Production Department and the Quality Assurance Department. If a disagreement arises between members of these departments, resolution is sought at successively higher levels of management, as necessary, up to and including the President.

17.3 QA Program

In addition to descriptive material contained in the Duke Power Company topical report on quality assurance, the operations phase of the quality assurance program is detailed in company procedures. A summary of the topics addressed in these procedures and their relationship to the quality assurance requirements of Appendix B to 10 CFR Part 50 is presented in the topical report.

Procedures and work instructions necessary to implement the requirements of the operations phase program are developed by the organization responsible for the activity. Lower tier procedures and instructions are contained in manuals, station procedures and directives, administrative instructions, and/or other documents. Onsite implementation of procedures and work instructions is the responsibility of the Catawba Nuclear Station Manager. Quality Assurance Department personnel verify that the procedures are followed by means of inspections, audits, and other surveillance. Procedures for such inspections, audits, surveillance are developed, approved, and implemented by the Quality Assurance Department.

Inspections are performed using preplanned checklists in accordance with written and approved inspection plans. The qualifications of inspectors and their current status to conduct inspections, tests, and examinations are based on applicable codes, standards, and Duke Power Company training programs.

The quality assurance organization is responsible for the content and control of the audit program. Audits are performed in accordance with written procedures or checklists by appropriately trained quality assurance personnel not having direct responsibility in the area being audited. The audit activities described in the topical report are conducted at least annually, or on a more frequent basis as determined by the quality assurance organization. These include an objective evaluation of quality assurance practices, procedures, and instructions;

work areas, activities, processes, and items; effectiveness of implementation of the quality assurance program; and compliance with policy directives.

The quality assurance program requires both documentation of audit results and formal notification of the audit findings to the Quality Assurance Manager and to management of the audited function. Audit findings, which indicate quality trends and the effectiveness of the quality assurance program, are also reported to the Senior Vice-President, Engineering and Construction. Management for the area audited implements any corrective action needed. Follow-up audits are performed to determine that nonconformances are effectively corrected, and that the corrective action precludes repetitive occurrences.

An indoctrination and training program is established to assure that persons involved in quality-related activities are knowledgeable in quality assurance instructions and requirements, and demonstrate a high level of competence and skill in the performance of their quality-related activities. A program for retraining of such persons is provided to assure maintenance of their proficiency.

17.4 Conclusion

Based on our detailed review and evaluation of the quality assurance program description contained in Section 17 of the Final Safety Analysis Report for the Catawba Nuclear Station and the topical report referenced therein, we conclude that the quality assurance program for operations

is acceptable with the exception that certain additional information and clarifications may be necessary regarding the list of items that are under the control of the QA program. This list of items has been identified in the Final Safety Analysis Report and is undergoing review by NRR technical review branches (inputs are still needed from CSB and the Hydrologic Engineering Section of HGEB as of 3/26/82). At the completion of this review, additional information may be required from the Duke Power Company. This SER will be amended to reflect subsequent FSAR action.

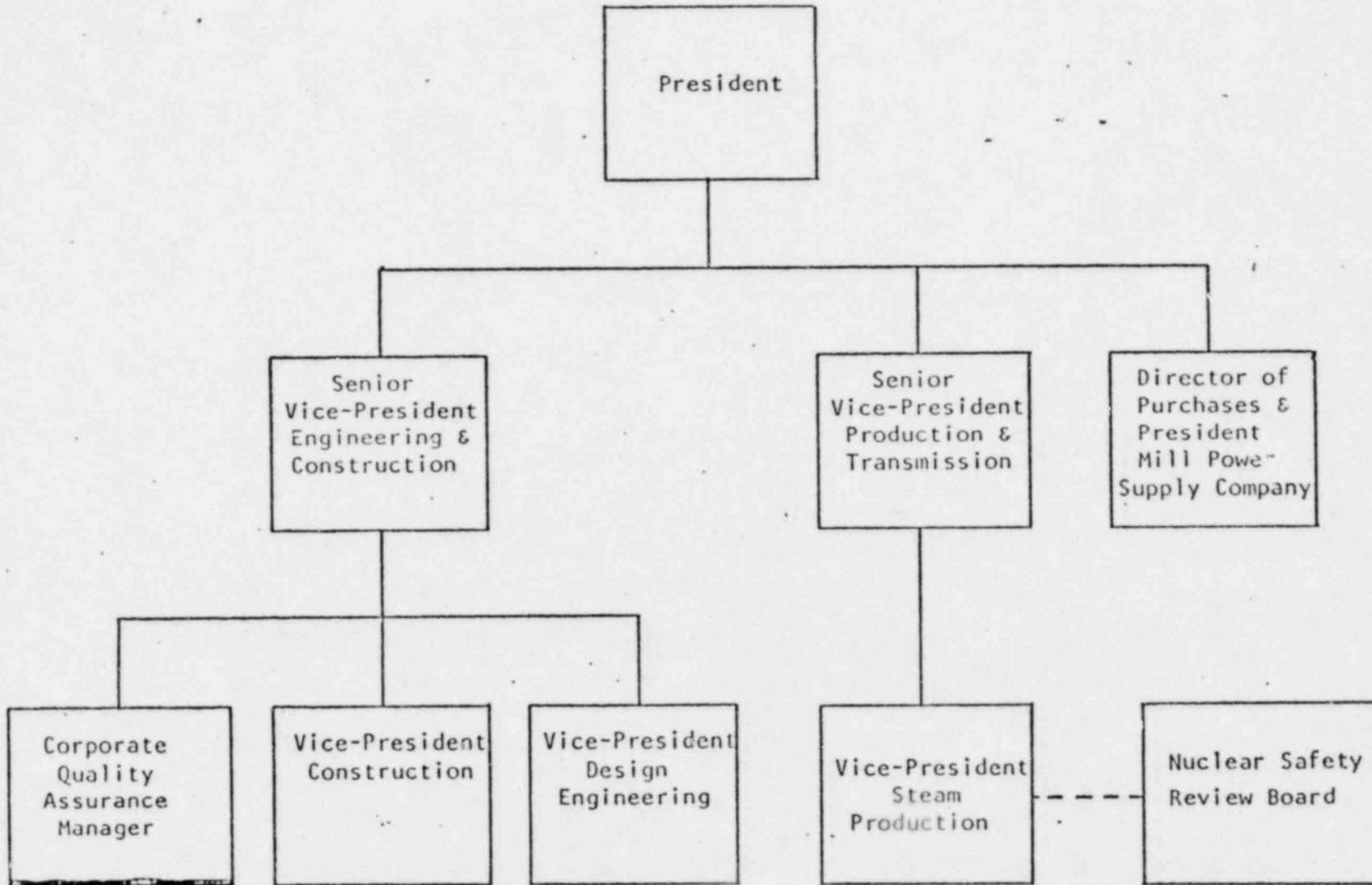
TABLE 17.1

REGULATORY GUIDANCE APPLICABLE TO QUALITY ASSURANCE PROGRAM

1. Regulatory Guide 1.8-Rev. 1, "Personnel Selection and Training."
2. Regulatory Guide 1.30, August 1972, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment."
3. Regulatory Guide 1.33-Rev. 2, "Quality Assurance Program Requirements (Operation)."
4. Regulatory Guide 1.37, March 1973, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."
5. Regulatory Guide 1.38-Rev. 2, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants."
6. Regulatory Guide 1.39-Rev. 2, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."
7. Regulatory Guide 1.58-Rev. 1, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel."

8. Regulatory Guide 1.64-Rev. 2, "Quality Assurance Requirements for the Design of Nuclear Power Plants."
9. Regulatory Guide 1.74, February 1974, "Quality Assurance Terms and Definitions."
10. Regulatory Guide 1.88-Rev. 2, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records."
11. Regulatory Guide 1.94-Rev. 1, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."
12. Regulatory Guide 1.116-Rev. 0-R, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems."
13. Regulatory Guide 1.123-Rev. 1, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants."
14. Regulatory Guide 1.144-Rev. 1, "Auditing of Quality Assurance Programs for Nuclear Power Plants."
15. Regulatory Guide 1.146, August 1980, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants."

DUKE POWER COMPANY
CORPORATE ORGANIZATION
FOR QUALITY ASSURANCE



TOPICAL REPORT
QUALITY ASSURANCE PROGRAM

Figure 17-1

QUALITY ASSURANCE DEPARTMENT ORGANIZATION CHART

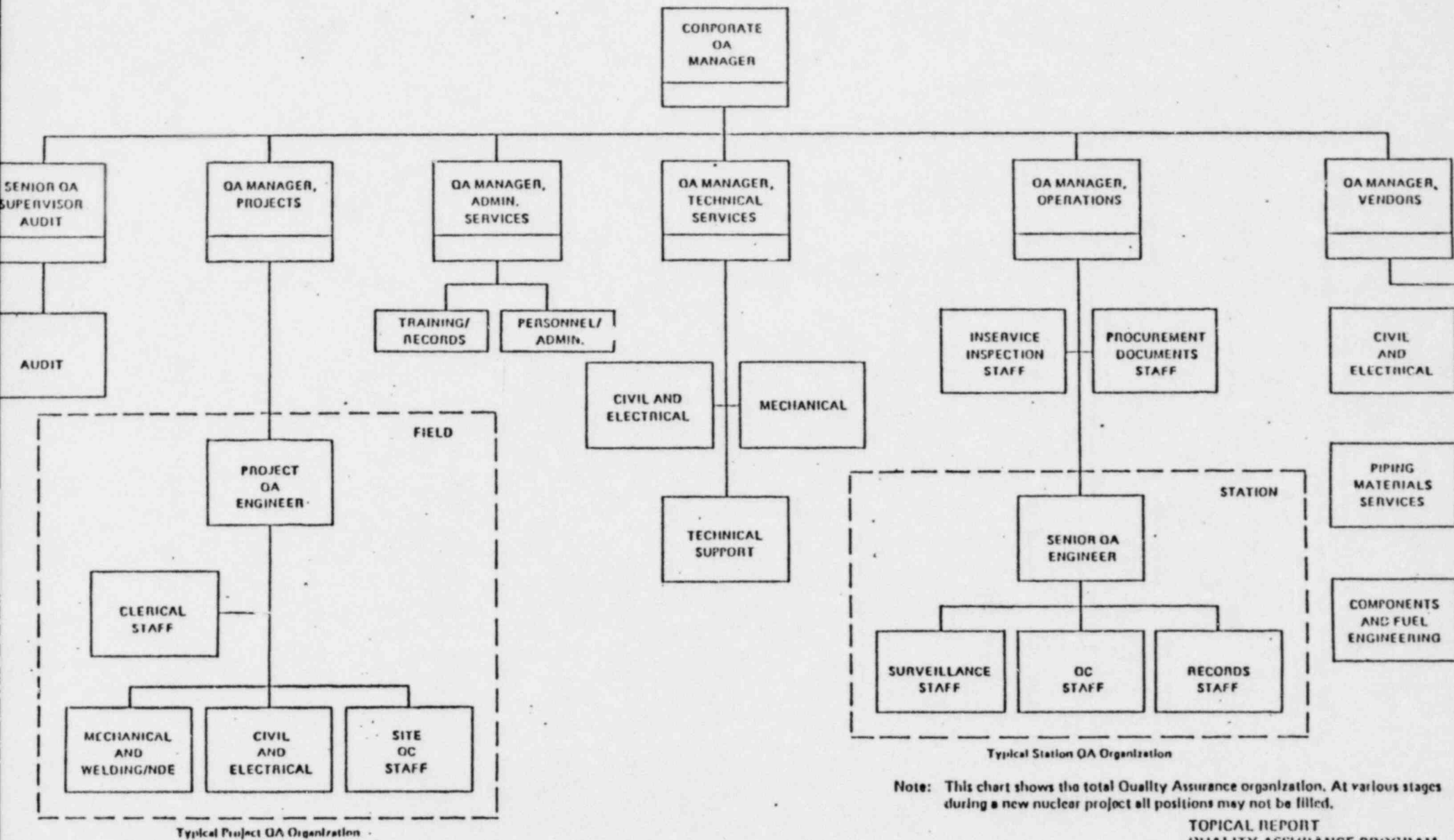


Figure 17-2

OPERATIONAL QUALITY ASSURANCE
 STEAM PRODUCTION DEPARTMENT ORGANIZATION

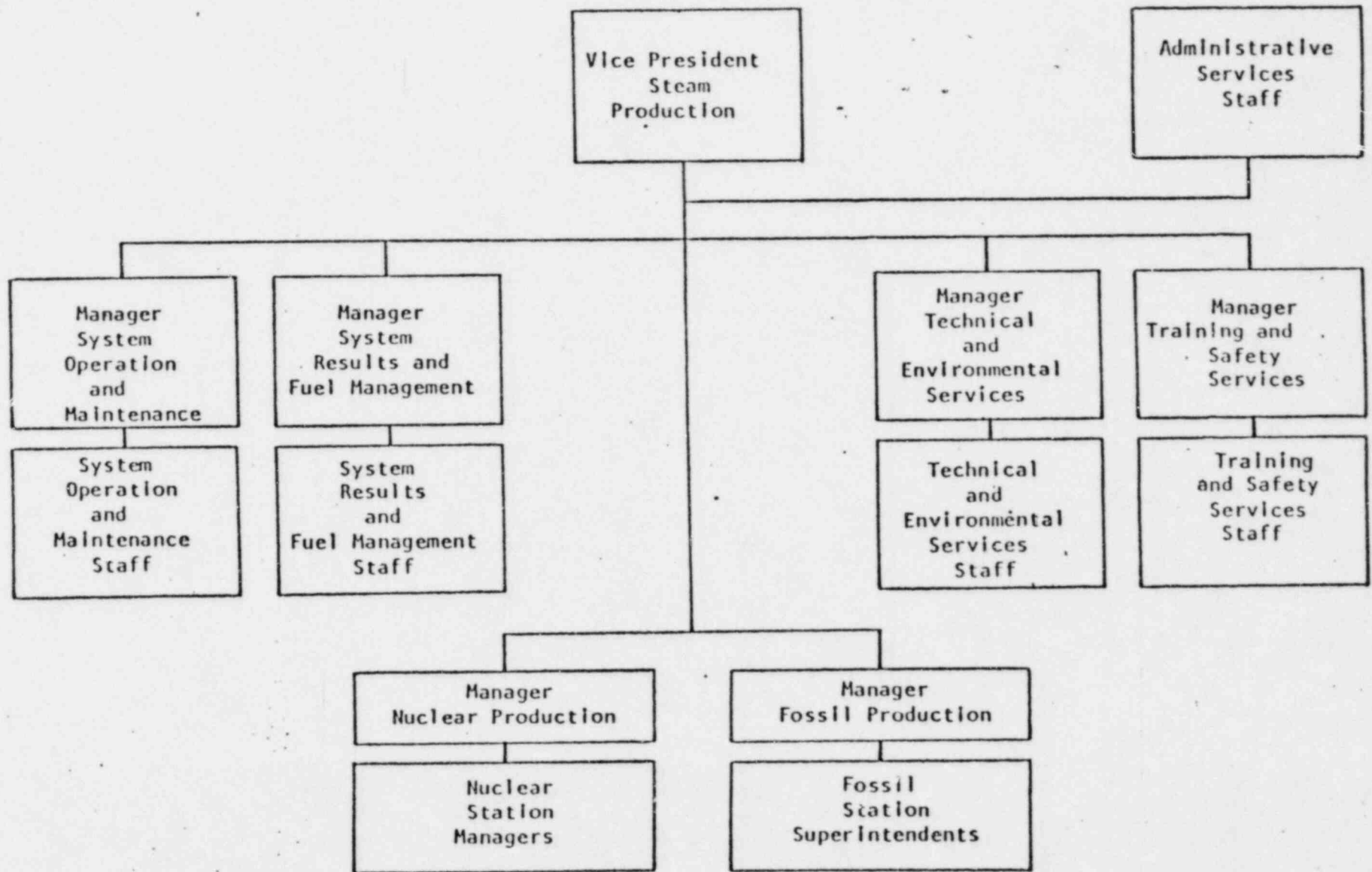


FIGURE 17-3

OPERATIONAL QUALITY ASSURANCE
 TYPICAL NUCLEAR STATION ORGANIZATION

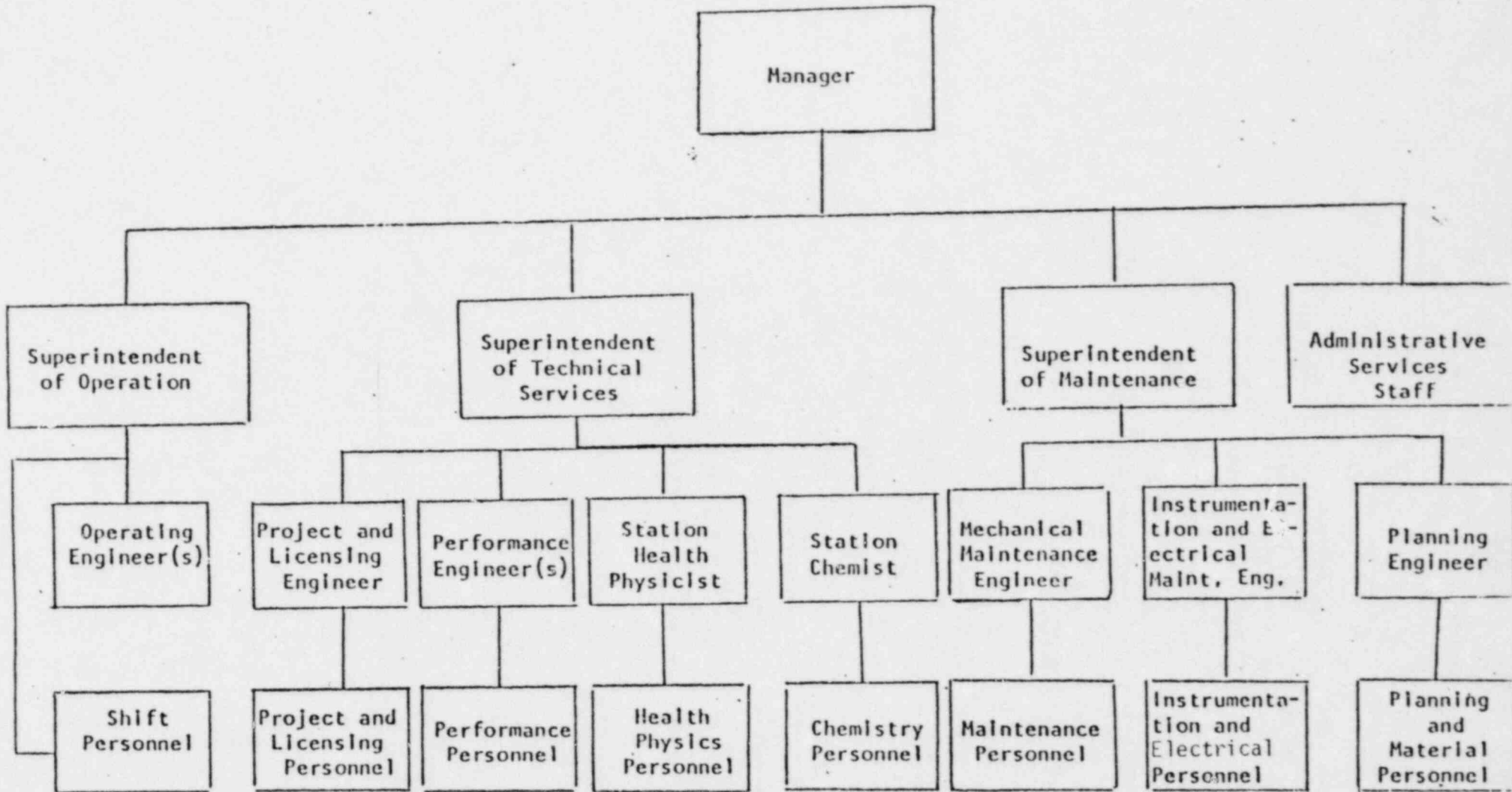


FIGURE 17-4

DUKE POWER COMPANY
Catawba Units 1 and 2
Docket No. STN 50-413/414

SAFETY EVALUATION REPORT
MATERIALS ENGINEERING BRANCH
COMPONENT INTEGRITY SECTION

5.3.1 Reactor Vessel Materials

The staff of EG&G, Idaho National Engineering Laboratory has reviewed the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references which are the basis for this evaluation are set forth in Paragraph II.3.a of Standard Review Plan (SRP) Section 5.2.3 and Paragraph II.5, II.6, and II.7 (Appendices G and H, 10 CFR Part 50) of SRP Section 5.3.1 in NUREG 0800 Rev. 1 dated July 1981. A discussion of this review follows.

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, and testing conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires, in part, that the reactor coolant pressure boundary be designed to permit an appropriate material surveillance program for the reactor coolant pressure boundary. Materials selection, toughness requirements and extent of material testing were reviewed in accordance with the above criteria subject to the rules and requirements of 10 CFR Part 50 Paragraph 50.55a--"Codes and Standards," 10 CFR Part 50 Appendix G--"Fracture Toughness Requirements," and 10 CFR Part 50 Appendix H--"Reactor Vessel Materials Surveillance Program Requirements."

Compliance with Paragraph 50.55a, 10 CFR Part 50

The Catawba Unit 1^{and Unit 2} (hereinafter CNS-1) construction permit, ^{were} ~~was~~ issued in August 1975. Based upon the construction permit date, 10 CFR Part 50, Paragraph 50.55(a) requires that ferritic reactor coolant pressure boundary (RCPB) materials used for vessels be constructed to Section III of the ASME Code no earlier than the Summer 72 Addenda of the 1971 edition and that ferritic RCPB materials used for pressure retaining piping, pump and valve components be constructed to Section III of the ASME Code no earlier than the Winter 72 Addenda of the 1971 edition. Ferritic RCPB materials used for fabrication of the CNS-1 reactor pressure vessels were constructed to the 1971 Edition, Winter 1971 Addenda of the Code. Therefore, the ferritic materials in the reactor pressure vessels do not meet the requirements of Paragraph 50.55a of 10 CFR Part 50. However, we will evaluate the applicant's reactor pressure vessel ferritic materials to Appendix G of 10 CFR Part 50 which will ensure that material properties are equivalent or superior to those specified in Section 50.55a, 10 CFR Part 50. Ferritic RCPB materials used for fabrication or piping, pump and valve components were constructed to ASME Code Edition and Addenda which satisfy the requirements of 10 CFR Part 50 Paragraph 50.55(a).

Compliance with Appendix G, 10 CFR Part 50

We have evaluated the applicant's FSAR to determine the degree of compliance with fracture toughness requirements of Appendix G, 10 CFR Part 50. Our evaluation indicates that the applicant complied with Appendix G, 10 CFR Part 50, except for Paragraphs III.B.4, IV.A.1, IV.A.3, and IV.B, which will remain open items until the applicant submits the requested data. Our evaluation of each of these areas follows.

Paragraph III.B.4 requires individuals performing fracture toughness tests be qualified by training and experience and that individuals demonstrate competency to perform tests in accordance with written procedures. The applicant has not provided any information that demonstrates compliance with these fracture toughness tests requirements. The applicant must provide the required information or present another method of qualifying personnel which is equivalent to the requirements of Paragraph III.B.4.

Paragraph IV.A.1 requires that all ferritic material used in vessels that are part of the RCPB be tested to the requirements of NB-2300 of the ASME Code. The ASME Code and Addenda to which the CNS-1^{a-d-2}V RCPB vessel was fabricated require a reference temperature, RT_{NDT} , be determined for all base metal, weld metal and heat-affected zone materials. The applicant has reported the RT_{NDT} for all base metals and one weld used in the fabrication of CNS-1^{a-d-2}V ferritic RCPB vessels.

The applicant has not submitted any fracture toughness data for the intermediate to lower shell weld (weld control number P710). The applicant has estimated the RT_{NDT} per NRC Standard Review Plan (hereinafter SRP) Section 5.3.2, but has not furnished CVN test data (impact energy absorbed and test temperature) to corroborate the estimation technique. To demonstrate compliance with the requirements of Paragraph IV.A.1 of Appendix G, 10 CFR Part 50, the applicant must prove by actual material test data, analysis, or data from the literature that the intermediate to lower shell weld (P710) has RT_{NDT} value of 0°F.

To demonstrate that the RCPB vessel welds comply with the requirements of Paragraph IV.A.1, the applicant must identify the weld RT_{NDT} which will be limiting for operation during startup and at end of life. To demonstrate that RCPB heat-affected zones comply with the requirements of Paragraph IV.A.1, the applicant must indicate that all RCPB welds were fabricated using materials and welding procedure combinations that will not be deleterious to the fracture toughness of the heat-affected zone material.

Paragraph IV.A.3 requires, in part, that the materials for bolting and fasteners meet the requirements of Paragraph NB-2333 of the ASME Code. This paragraph requires that bolting having a diameter of more than four inches, have three CVN impact tests at the preload temperature or lowest metal service temperature (whichever is less) of 45 ft-lb energy absorbed and 25 mils lateral expansion (LE). Some bolts from heat number 35674 exhibit CVN values of 29 to 38 ft-lb with 8 to 12 mils LE. To demonstrate compliance to Paragraph IV.A.3, the applicant must supply data from the literature or an analysis which proves that at the preload temperature or lowest metal service temperature (whichever is less) that the requirements in NB-2333 of the ASME Code are met.

Paragraph IV.B requires unirradiated reactor vessel beltline materials have a minimum upper shelf charpy V-notch energy of 75 ft-lbs. The applicant in Table 5.3.1-2 of the CNS-1Y^{ad-2}FSAR showed that all the beltline base material CVN results exceeded 75 ft-lb. However, there were no CVN impact data for the lower to intermediate shell weld root (P710). To demonstrate compliance with Paragraph IV.B, the applicant must show by actual material test data, analysis, or data from the literature that the lower to intermediate shell weld (P710) meets 75 ft-lb CVN impact test requirement.

Compliance With Appendix H, 10 CFR Part 50

The materials surveillance program at CNS-1Y^{ad-2} will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment. Under CNS-1Y^{ad-2} surveillance program, fracture toughness data will be obtained from material specimens that are representative of the limiting base, weld, and heat-affected zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

The fracture toughness properties of reactor vessel beltline materials must be monitored throughout the service life of CNS-1Y^{ad-2} by a materials surveillance program that meets the requirements of ASTM Standard E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" and Appendix H of 10 CFR Part 50.

We have evaluated the applicant's information for degree of compliance to these requirements. Based on our evaluation we conclude that the applicant has met all the requirements of Appendix H, 10 CFR Part 50 with the exception of Paragraphs II.B and II.C.3.

Paragraph II.B of Appendix H requires that the surveillance program comply with ASTM E-185-73. ASTM E-185-73 requires the surveillance capsule materials be removed from beltline reactor vessel base metals and weld samples which

represent the material that may limit operation of the reactor vessel during its lifetime and must provide a sketch indicating the azimuthal location for each surveillance capsule. The applicant has identified from which samples the material surveillance specimens were removed but has not provided a sketch showing the surveillance capsule locations. To demonstrate compliance with Paragraph II.B of Appendix H, the applicant must provide a sketch which indicates the azimuthal location for each capsule relative to the reactor core.

Paragraph II.C.3 requires that the basis for the surveillance capsule withdrawal schedule is the adjusted reference temperature at the end of the service life of the reactor vessel. The applicant has indicated in Paragraph 5.3.1.6 of the CNS-1^{ad-2}VFSAR that there will be six surveillance capsules in the reactor vessel surveillance program but has not indicated the lead factors and the withdrawal schedule for each capsule. The applicant must supply this information in order for us to determine whether the applicant complies with Paragraph II.C.3 of Appendix H.

Conclusions for Compliance with Appendices G and H, 10 CFR Part 50

Based on our evaluation of compliance with Appendices G and H, 10 CFR Part 50, we conclude that the applicant has not supplied sufficient information to demonstrate compliance with all the fracture toughness requirements of Appendix G and surveillance program requirements of Appendix H. The areas in which additional information is required include Paragraphs III.B.4, IV.A.1, IV.A.3, and IV.B of Appendix G and Paragraphs II.B and II.C.3 of Appendix H; these items will remain open in our safety evaluation report until the applicant submits the necessary data.

Appendix G, "Protection Against Nonductile Failure," Section III of the ASME Boiler and Pressure Vessel Code, will be used, together with the fracture toughness test results required by Appendices G and H, 10 CFR Part 50, to calculate the reactor coolant pressure boundary pressure-temperature limitations for CNS-1^{ad-2}.

The fracture toughness tests required by the ASME Code and the Appendix G of 10 CFR Part 50 will provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the reactor coolant boundary. The use of Appendix G of Section III of the ASME Code as a guide in establishing a safe operating procedures, and use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations, will provide adequate safety margins during operation, maintenance, and testing conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the fracture toughness requirements of General Design Criterion 31.

The materials surveillance program, required by Appendix H, 10 CFR Part 50, will provide information on material properties and the effects of irradiation on material properties so that changes in fracture toughness of material in CNS-1^{ad-2} reactor vessel beltlines caused by exposure to neutron radiation can be properly assessed, and adequate safety margins against the possibility of vessel failure can be provided.

Compliance with ASTM E-185-73 and Appendix H, 10 CFR Part 50 assures that the surveillance program constitutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material and satisfies the materials surveillance requirements of General Design Criteria 31 and 32.

5.3.2 Pressure Temperature Limits

The staff of EG&G, Idaho National Engineering Laboratory has reviewed the applicant's pressure temperature limits for operation of their reactor vessels. The acceptance criteria and list of references which are the basis for this evaluation are set forth in the Standard Review Plan (SRP) Section 5.3.2 of NUREG 0800 Rev. 1 dated July 1981. A discussion of this review follows.

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," 10 CFR Part 50, describe the

conditions that require pressure-temperature limits for the reactor coolant pressure boundary and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins for the reactor coolant pressure boundary at least as great as the safety margins recommended in the ASME Boiler and Pressure Vessel Code, Section III. Appendix G, "Protection Against Nonductile Failure." Appendix G, 10 CFR Part 50, requires additional safety margins whenever the reactor core is critical, except for low-level physics tests.

The following pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests are reviewed to ensure that they provide adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components as required by General Design Criterion 31.

1. Preservice hydrostatic tests,
2. Inservice leak and hydrostatic tests,
3. Heatup and cooldown operations, and
4. Core operation.

The applicant has not submitted pressure-temperature limits for CNS-1^{ad-2} but has indicated that technical specifications will be developed to establish pressure-temperature limits for normal operation and testing. The applicant must provide the actual pressure-temperature limits for CNS-1^{ad-2} based upon the fracture toughness of the limiting reactor vessel material and predicted shift in the adjusted reference temperature, RT_{NDT} , resulting from radiation damage of the limiting beltline material. The pressure-temperature limits for the above identified conditions must be included in the technical specifications.

The applicant has proposed the use of Westinghouse trend curves as an alternative method for predicting the shift in the adjusted reference temperature, RT_{NDT} . The method for predicting the shift in the adjusted reference temperature, RT_{NDT} , presently recommended by the NRC for predicting the shift in RT_{NDT} , is Regulatory Guide 1.99, "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The curves used for

predicting the shift in RT_{NDT} in Regulatory Guide 1.99 are based on data from surveillance test specimens and will be used by the NRC for evaluating the applicant's initial pressure-temperature limit curves. Subsequent to operation, predictions of radiation damage can be based on the actual measured shift in reference temperature that is obtained from the results of the surveillance program at CNS-1~~ad-2~~.

The NRC will review the applicant's pressure-temperature limits based upon the radiation damage predicted in Regulatory Guide 1.99 and the calculation methodology of Standard Review Plan Section 5.3.2, "Pressure-Temperature Limits."

5.3.3 Reactor Vessel Integrity

We have reviewed the following FSAR sections related to the reactor vessels' integrity for CNS-1~~ad-2~~. Although most areas are reviewed separately in accordance with other review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted.

The staff of EG&G, Idaho National Engineering Laboratory has reviewed the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, the pressure temperature limits for operation of the reactor vessels, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references which are the basis for the evaluation are set forth in Paragraphs II.1, II.6 and II.7 (Appendices G and H, 10 CFR Part 50) of Standard Review Plan (SRP) Section 5.3.3 in NUREG 0000 Rev. 1 dated July 1981. A discussion of this review follows.

We have reviewed the information in each area to ensure that it is complete and that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed are:

1. Design (SER Paragraph 5.3.1)
2. Materials of construction (SER Paragraph 5.3.1)
3. Fabrication methods (SER Paragraph 5.3.1)
4. Operating conditions (SER Paragraph 5.3.2)

We have reviewed the above factors contributing to the structural integrity of the reactor vessel and conclude that the applicant has complied with Appendices G and H, 10 CFR Part 50, except for the following items:

Paragraph III.B.4, Appendix G: The applicant has not provided sufficient information to determine whether individuals performing fracture toughness tests were qualified by training and experience and that the individuals had demonstrated competency to perform tests in accordance with a written procedure.

Paragraph IV.A.1, Appendix G: The applicant has not provided sufficient information to define the reference temperature, RT_{NDT} , for all ferritic reactor coolant pressure boundary materials.

Paragraph IV.A.3; Appendix G: The applicant has not demonstrated that the bolting and fasteners conform to the requirements of Paragraph NB-2333 of the ASME Code.

Paragraph IV.B, Appendix G: The applicant has not provided sufficient information to define the upper-shelf energy for all beltline materials.

Paragraph II.B, Appendix H: The surveillance capsule identification data per ASTM E-185-73 have not been included in the FSAR.

Paragraph II.C.3, Appendix H: The applicant has not provided sufficient information to define the capsule withdrawal sequence and lead factors for the material surveillance program.

In addition, the applicant has not submitted pressure temperature limit curves to ensure that safe operation of the reactor vessel during normal operation and testing.

Until the applicant has supplied the information necessary to complete our evaluation of compliance with Appendices G and H, 10 CFR Part 50, and reactor pressure temperature limits, we cannot complete our evaluation of the structural integrity of the reactor vessels of CNS-1 and -2.

5.4.1.1 Pump Flywheel Integrity

The staff of EG&G, Idaho National Engineering Laboratory, has reviewed the applicant's pump flywheel design, material selection, fracture toughness, preservice and inservice inspection program and overspeed test procedure. The acceptance criteria and references which are the basis for this evaluation are set forth in the Standard Review Plan (SRP) Section 5.4.1.1 of NUREG-0800 Rev. 1 dated July 1981. A discussion of this review follows.

General Design Criterion 4, "Environmental and Missile Design Bases," of Appendix A, 10 CFR Part 50, requires that nuclear power plant structures, systems and components important to safety be protected against the effects of missiles that might result from equipment failures. Because flywheels have large masses and rotate at speeds of approximately 1200 revolutions per minute during normal operation, a loss of flywheel integrity could result in high energy missiles and excessive vibration of the reactor coolant pump assembly. The safety consequences could be significant because of possible damage to the reactor coolant system, the containment, or the engineered safety features. Adequate margins of safety and protection against the potential for damage from flywheel missiles can be achieved by the use of suitable material, adequate design, and inspection.

According to Section 5.4.1.5.2.1 of the FSAR, the material used to manufacture the pump flywheels is SA-533 Grade B Class 1 steel plate. The applicant further states that the NDTT, nil-ductility transition temperature, of the flywheel material is no higher than +10°F, and that the Charpy upper shelf energy level in the "weak" direction is no less than 50 ft-lbs at 70°F.

Paragraph C.1.c(2) of Safety Guide 14 requires that the adjusted fracture energy, as read from the adjusted CVN curve at normal operating temperatures

of the flywheel can be demonstrated to be equivalent to a K_{1C} (dynamic) value of at least 100 ksi in.^{1/2} by using appropriate correlation of data. Our analysis indicates that the normal operating temperature of the flywheel material must be greater than 100°F above the RT_{NDT} to ensure the material has a K_{1C} (dynamic) value of at least 100 ksi in.^{1/2}. The applicant has indicated that the RT_{NDT} of the flywheels is at least 100°F less than their normal operating temperature. Therefore, the applicant satisfies Paragraph C.1.c(2) of Safety Guide 14 for CNS-1^{ad-2}.

The pump flywheels are designed to the requirements of Paragraph C.2 of Safety Guide 14 and the flywheel assembly are given a preoperational test at the design overspeed of the flywheel.

The Technical Specification for inservice inspection and testing of the pump flywheel has been submitted by the applicant for CNS-1^{ad-2}. We find that it complies with Paragraph C.4 of Safety Guide 14 dated 10/27/71.

Based on the data provided by the applicant, we conclude that CNS-1^{ad-2} possess a margin of safety against flywheel missiles equivalent to that recommended in Safety Guide 14. Compliance with Safety Guide 14 will provide a basis acceptable to the staff for satisfying the requirements of General Design Criterion 4.

AUXILIARY SYSTEMS ITEMS REQUIRING
FURTHER EVALUATION

1. Section 3.5.1.1 - Internally Generated Missiles (Outside Containment) -

Although the applicant has provided information which indicates that no credible missiles should be postulated outside the containment, we have requested that the applicant provide an analysis which discusses the protection provided for safety related equipment from missiles generated by non-safety-related sources.

2. Section 3.5.2 - Structures, Systems and Components to be Protected from Externally Generated Missiles - The applicant has not provided a description of structures, systems and components which are subject to tornado missile damage such as ventilation system air intakes and exhausts, emergency diesel generator exhausts, freight doors in safety related structures, and outdoor piping.

3. Section 3.6.1 - Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment - The applicant has not provided sufficient analysis of the effects on safety-related systems of failures in high-or moderate-energy piping systems. In addition, the applicant has not presented the transient pressure, temperature and humidity effects of postulated pipe ruptures in areas vulnerable to extreme environmental conditions following pipe breaks such as the main steam doghouse and steam tunnels.

4. Section 9.1.1 - New Fuel Storage - The applicant has not provided the specific K_{eff} values determined in his criticality analysis for the new fuel storage arrangement with the associated assumptions and input parameters.

5. Section 9.1.2 - Spent Fuel Storage - The applicant has not verified whether the spent fuel pool liner is designed to remain in place and retain its leak tight integrity in a SSE, thus precluding damage to spent fuel. In addition the applicant should provide certain information concerning the capability and safety aspects (criticality concerns) of storing Ocone or McGuire spent fuel assemblies in the Catawba spent fuel pool.

6. Section 9.1.3 - Spent Fuel Pool Cooling and Cleanup System - The applicant has not provided information concerning the spent fuel pool water temperature following the loss of one cooling train, assuming a full core offload (maximum heat load condition) with either Catawba fuel or non-Catawba fuel. Additionally the applicant has not presented an analysis of decay heat load vs. time or pool temperature vs. time for the above condition.

7. Section 9.1.4 - Light Load Handling System (Fuel-Handling System) - The applicant is required to verify that the radiological releases resulting from dropping of light loads are less than those from the design basis fuel-handling accident.

8. Section 9.1.5 - Overhead Heavy-Load-Handling System - The applicant has not provided an analysis of the effects of dropping heavy loads other than a spent fuel cask (such as the reactor vessel head and internals) to satisfy the evaluation criteria of NUREG-0612, Section 5.1.

9. Section 9.2.1 - Station Service Water System (Nuclear Service Water System) - The applicant is requested to verify that the integrity of the buried portion of the nuclear service water system is maintained in the event of failure of the buried condenser circulating water piping in the event of an SSE.
10. Section 9.2.2 - Reactor Auxiliary Cooling Water System (Component Cooling System) - The applicant has not indicated agreement to provide safety-grade instrumentation with which to indicate the loss of cooling water to the reactor coolant pumps and allow prompt operator action to prevent a motor bearing failure and possible unacceptable locked rotor condition.
11. Section 9.2.5 - Ultimate Heat Sink - The applicant is requested to clarify in his analysis of the ultimate heat sink capacity and performance how the spent fuel pool cooling load was considered.
12. Section 9.2.6 - Condensate Storage Facilities (Condensate Storage System) - The applicant has not verified that adequate isolation is provided at the interface of the non-safety-related condensate storage system with the safety-related auxiliary feedwater system condensate storage tank.
13. Section 9.3.1 - Compressed Air System - The applicant is requested to provide additional information to describe any special provisions made to assure the reliable delivery of instrument air to the essential valves under the station blackout situation. Additionally, the applicant is required to verify that the instrument air system meets the instrument air quality standards defined by ANSI MC 11.1-1976.
14. Section 9.3.3 - Equipment and Floor Drainage System - The applicant has not provided an analysis to demonstrate that drainage of leakage water away from safety-related components or systems is adequate for worst case flooding resulting from pipe breaks or cracks in high-or-moderate energy piping or postulated failure in all non-seismic Category I piping near safety-related components or systems.
15. Section 9.4.1 - Heat Ventilation and Air Conditioning (HVAC) System - The applicant has not provided any information regarding the capability of the battery room exhaust fans to prevent accumulation of hydrogen.
16. Section 9.4.3 - Auxiliary and Radwaste Area Ventilation System - The applicant has not provided any information concerning cooling of safety related pump rooms under accident conditions when the normal ventilation system is not available.

17. Section 9.4.5 - Engineered-Safety Features Ventilation System - The applicant has not discussed the environment which is maintained for the auxiliary feedwater pumps under accident conditions when the normal ventilation provided by the non-seismic Category I unfiltered auxiliary building exhaust system is not available.
18. Section 10.3 - Main Steam Supply System - The applicant has not committed to provide the capability to operate the power operated atmospheric relief valves remotely from the control room on a loss of offsite power condition. Additionally, the applicant has not committed to perform a local operability verification test for the atmospheric dump valves if these valves are not controllable from the control room following an SSE.
19. Section 10.4.7 - Condensate and Feedwater System - The applicant has not committed to perform a plant specific verification test to demonstrate that no damaging feedwater water hammer will occur.
20. Section 10.4.9 - Auxiliary Feedwater System - The applicant has not verified (1) that the turbine drive for the turbine driven AFW pump is qualified to function in an SSE and is built to proper ASME Code requirements; (2) that the plant technical specification has been revised to incorporate an AFW flow path verification test. (Recommendation GS-6 of NUREG-0611); (3) that adequate suction is available to prevent AFWS pump damage during transfer to the nuclear service water backup water source (Recommendation GL-4 of NUREG-0611); (4) that air binding of AFWS pumps does not occur prior to transferring pump suction supply; (5) that redundant primary AFWS water source level indicators will be provided in the control room (Additional Short Term Recommendation 1 of NUREG-0611); and (6) that AFW pump runout protection is provided which does not affect system reliability. We have not completed our evaluation of the AFWS reliability study as identified in Item II.E.1.1 of NUREG-0737.

MATERIALS ENGINEERING ITEMS
REQUIRING FURTHER EVALUATION
CATAWBA UNIT NOS. 1 AND 2

123.0 MATERIALS ENGINEERING BRANCH--COMPONENT INTEGRITY SECTION

123.1 Indicate whether the individuals performing the fracture toughness tests are qualified by training and experience and whether their competency was demonstrated in accordance with a written procedure. If the above information cannot be provided, state why the information cannot be provided and identify why the method used for qualifying individuals is equivalent to those of Paragraph III.B.4 Appendix G, 10 CFR Part 50.

123.2 To demonstrate compliance with the fracture toughness requirements of Paragraph IV.A.1 of Appendix G, 10 CFR Part 50:

- a. Provide the RT_{NDT} for all RCPB welds which may be limiting for operation of the reactor vessels.
- b. Indicate whether there are any RCPB heat-affected zones which require CVN impact testing per paragraph NB-4335.2 of the 1977 ASME Code. Provide CVN impact test data for these heat-affected zones which may be limiting for operation of the reactor vessels.
- c. Indicate that there are no ferritic RCPB base metals other than in vessels which require fracture toughness testing to NB-2300 of the ASME Code. If there are ferritic RCPB base metals other than in vessels which require fracture toughness testing to NB-2300 of the ASME Code, provide CVN impact and drop weight data for all materials which will be limiting for operation of the reactor vessels.

123.3 Provide material test data, analysis or data from the literature to demonstrate that bolts from heat number 35674 which had CVN values of 29 to 38 ft-lbs and 8 to 12 miles LE at 10°F, would meet the requirements of NB-2333

of the ASME Code (45 ft-lbs and 25 mils L.E. at the preload or lowest service metal temperature which ever is less). The sample material which demonstrates that the heat no. 35674 bolts will comply with the CVN requirements of NB-2333 of the ASME Code, must have been fabricated to an equivalent material specification and heat treated to an equivalent metallurgical condition as the material from heat no. 35674 bolts.

123.4 Provide material test data, analysis or data from the literature which demonstrates that the intermediate to lower shell weld (P710) has an RT_{NDT} of $0^{\circ}F$ and an upper shelf greater than 75 ft-lbs. The additional data should be from similar welds, i.e., those having the same type of weld wire and flux and thermal treatment as weld (P710). The information should include a comparison of the significant weld parameters (e.g., weld wire, flux and thermal treatment) and mechanical properties from the sample and (P710) beltline weld.

123.5 Provide actual pressure-temperature limits for CNS-1^{and 2} based upon the limiting fracture toughness of the reactor vessel material and the predicted shift in the adjusted reference temperature, RT_{NDT} resulting from radiation damage. The pressure-temperature limits for the following conditions must be included in the technical specifications when they are submitted:

1. Preservice hydrostatic tests,
2. Inservice leak and hydrostatic tests,
3. Heatup and cooldown operations, and
4. Core operation.

123.6 Provide full CVN impact curves for each weld and plate in the beltline region. Provide the data in tabulated and graphical form.

123.7 To demonstrate the surveillance capsule program complies with Paragraphs II.B and II.C.3 of Appendix H.

- a. Provide the withdrawal schedule for each capsule.

- b. Provide the lead factors for each capsule.
- c. Provide a sketch which indicates the azimuthal location for each capsule relative to the reactor core.