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TELEPHONE: AREA 704
373-4083

June 11, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414

Dear Mr. Denton:

In order to facilitate the completion of the review of the Catawba FSAR, Duke Power Company is transmitting herewith responses, revised responses, or partial responses to the following FSAR questions:

220.21	410.17	440.56
220.22	410.20	440.85
220.25	410.21	460.8
220.30	420.5	480.16
220.46	430.69	730 (A-40)
241.9	430.92	730 (A-46)
410.12	430.102	

These responses will be included in FSAR Revision 6.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr.

ROS/php
Attachment

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

Mr. P. K. Van Doorn
NRC Resident Inspector
Catawba Nuclear Station

B001

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Mr. H. A. Presler, Chairman
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C_i = Constant for the i^{th} mode.

W_i = Natural frequency of the i^{th} mode.

N_i = Constant for the i^{th} mode.

S_{vi} = Maximum velocity from the response spectrum for a single-degree-of-freedom system for a given value of W_i for the i^{th} mode.

N = Number of modes considered.

Equipment located in a structure is used as input to the seismic analysis. The equipment is comparatively rigid by virture of its mass when compared to the mass of the structure. The equipment is connected to the structure in a rigid fashion.

Q 220.21 The equipment masses and locations are used as input to the center of mass calculation for an individual location where mass will be lumped for use in the seismic model. The individual mass points are connected to the seismic model by rigid members and response spectra are generated for each point. The response spectra as generated are then used in the individual equipment design.

3.7.2.1.1.2 Containment Interior Structure, Nuclear Service Water Intake Structure, and Auxiliary Building, Including Diesel Building, Exterior Doghouse, and Upper Head Injection Building

The seismic loads on the Containment Interior Structure, Nuclear Service Water Intake Structure and Auxiliary Building, as a result of a base excitation, are determined by a dynamic analysis. The dynamic analysis is made by idealizing the structure as a series of lumped masses with weightless elastic columns acting as spring restraints. The base of the structure is considered fixed.

The procedure used to lump masses for the seismic structural model is dependent upon the actual mass distribution and structural characteristics of the structure.

Mass locations are established at elevations in the structure where there are concentrations of mass such as floor slabs and/or equipment. Mass locations are also established when there are changes in structural properties such as moments of inertia, shear area or elastic properties.

Q 220.25 Sufficient mass locations of uniform distribution are established for the Diesel Building and Nuclear Service Water Pump Structure to assure consideration of all significant modes with frequencies less than $20H_Z$. Structures with fundamental frequencies greater than $20H_Z$ are designed as rigid structures with a constant acceleration equal to the acceleration corresponding to $20H_Z$ on the response spectrum.

The mass of the equipment is lumped at the elevation at which it is supported such as lateral supports for the steam generators, reactor vessel, reactor coolant pumps, pressurizer, and polar crane. When equipment is supported on a floor slab, the equipment mass is lumped with the structural mass of the slab.

The structural connection between equipment and structure is considered rigid for the seismic analysis of the structure. A response spectrum has been generated as defined in Section 3.7.2.5 at mass locations where equipment or piping is supported. This response spectrum is used for the seismic design of equipment and piping as defined in Section 3.7.3.8.

The mass of vertical structural members is distributed to the adjacent mass locations. Between these mass locations, vertical structural member properties
Q are calculated for moments of inertia, cross-sectional area, effective shear
220.25 area, and length. For those elevations where the center of mass and center of structural rigidity do not coincide or where the centers of structural rigidity change locations, infinitely stiff horizontal members connect these points.

For each of the structures listed above, it has been determined that the structure moves with the ground motion during an earthquake to account for the soil-structure interaction effect (see Section 3.7.3.12).

3.7.2.5 Development of Floor Response Spectra

Figures 2.5.2-5 reflect the time-history spectra and site design spectra.

The synthetic earthquakes used to generate the time-history spectra in Figures 2.5.2-5 are used to generate response spectra at elevations in structures that house systems and components which are required to be designed for seismic excitation.

The analytical technique used to generate the response spectra at specified elevations in a structure is the time-history method. The acceleration time-history of each elevation is retained for the generation of response spectra reflecting the maximum acceleration of a single-degree-of-freedom system for a range of frequencies at the respective elevation.

Q 220.30 Vertical response spectra are not generated. The floor slabs in Category 1 structures were examined for flexibility and were found to be sufficiently stiff to justify the assumption of negligible amplification in the vertical direction.

Damping values for the structural model are selected from Section 3.7.1.3, "Critical Damping Values."

TIME-HISTORY ANALYSIS

The time-history of the specified mass points is determined by the modal method in which the responses in the normal modes are determined separately, then superimposed to provide the total response to a specified base input motion.

Table 3.8.1-2 (Page 7)

effect of these concentrated loads, provided there will be no loss of function of any safety related system.

Load Combinations And Acceptance Criteria For Category I Foundations

In addition to the load combinations and acceptance criteria referenced above, all Category I foundations will also be checked against sliding and overturning due to earthquakes, winds, and tornadoes and against floatation due to floods in accordance with the following:

Load Combination	Minimum Factors of Safety Overturning	Sliding	Floatation
a. D + H + E	1.5	1.5	---
b. D + H + W	1.5	1.5	---
c. D + H + E'	1.1	1.1	---
d. D + H + W_t	1.1	1.1	---
e. D + F'	---	---	1.5
f. D + L + H_w	1.1	1.1	1.1

The minimum factor of safety against flotation of all Category 1 structures used the following procedures:

1. Uplifting/bouyant force - groundwater assumed to rise to grade elevation (593.5).

Q241.9

2. Dead Load - The total weight of the structure was determined including weights of the major pieces of equipment.

The uplifting/bouyant force was then compared to the total dead load.

9.3 PROCESS AUXILIARIES

9.3.1 COMPRESSED AIR SYSTEM

9.3.1.1 Design Bases

The Compressed Air System consists of the Instrument Air, Station Air, and Breathing Air Systems. 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 while performing certain maintenance and cleaning operations. The Compressed Air Systems are shown on Figures 9.3.1-1 thru 9.3.1-8 and the system component design parameters are given in Table 9.3.1-1.

9.3.1.2 System Description

9.3.1.2.1 Instrument Air System

Q410.21 Instrument air is supplied by three non-lubricated instrument air compressors. The compressors' intakes are in the Service Building basement. This area is free of corrosive contaminants and hazardous gases. Downstream of each air compressor, the hot compressed air flows through an aftercooler and water separator before discharging into an instrument air receiver. The aftercooler cools the hot compressed air to within 10°F of the conventional low pressure service water temperature, and the water separator removes any water condensed in the cooling process. The air receivers smooth out any pressure surges. Downstream of the air receivers, the instrument air is dryed to a dewpoint of 35 to 39°F by four refrigerated air dryers. In addition, dessicant air dryers are provided on the lines going outside the building to dry the air to a design dewpoint of -40°F. After the refrigerated air dryers, the air is passed through filters which filter out particles larger than 3 microns. Downstream of the filters, the instrument air headers supply instrument air throughout the plant. At each air operated valve or instrumentation the air is filtered again through a filter-regulator. Q410.21

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 bulk air supply to the Unit 1 and 2 condensate polishing demineralizers will come off the instrument air compressors discharge header upstream of the instrument air dryers. Two check valves with a trap between them will be provided in this supply line to prevent the backflow of water into the Instrument Air System.

Heater	Extraction Source
A	H-P turbine
B	H-P turbine
C	H-P turbine exhaust
D	L-P turbines
E	L-P turbines
F	L-P turbines
G	L-P turbines

Q430.92 Provided in extraction lines A, B, C, D, and E are piston-assist spring-closed swing check valves. The piston-assist spring-closed actuators are designed to overcome friction and allow the valves to rapidly close on turbine trip. (See Table 10.2.2-2 for closure times) Bleed lines F and G are not provided with check valves since installation in the condenser neck would be impractical. However, heaters F and G are provided with anti-flash baffle plates located below the tube bundles and above the water volume. The extraction design was approved by the turbine manufacturer in regard to turbine overspeed protection. Postulating turbine trip and failure of a single extraction check valve, the manufacturer concluded that unacceptable turbine overspeed would not occur.

Each of the two generators is a 1450 MVA, 1800 rpm, direct connected, 3 phase 60 cycle, 22,000 volt conductor cooled synchronous generator rated at .90 P.F, .50 SCR at a maximum hydrogen pressure of 75 psig. Generator rating, temperature rise, and class of insulation are in accordance with IEEE standards. Excitation is provided by a shaft driven alternator with its output rectified.

Table 10.2.2-2

Turbine Speed Control System

Valve Closure Time

	Valve	Closure Time (Seconds)
Q430.92	Main stop	0.12
	Main control	0.19
	Intermediate stop	0.20
	Intercept	0.17
	Extraction check	2 (maximum)

10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM

10.4.1 MAIN CONDENSER

10.4.1.1 Design Bases

The main condenser is designed to condense turbine exhaust steam for reuse in the steam cycle. The main condenser also serves as a collecting point for various steam cycle vents and drains to conserve condensate which is stored in the condenser hotwell. The condenser also serves as a heat sink for the Turbine Bypass System and is capable of handling 40 percent of rated main steam flow. Rejected heat is removed from the main condenser by the Condenser Circulating Water System.

10.4.1.2 System Description

The main condenser consists of three surface type deaerating condenser shells with each shell condensing the exhaust steam from one of the three low pressure turbines. The condenser shells are of conventional shell and tube design with steam on the shell side and circulating water in the tubes. Each condenser shell is joined to the turbine by a rubber belt type expansion joint. Provisions have been made for mounting two low pressure feedwater heaters in the neck of each of the three condenser shells. The combined hotwells of the three condenser shells have a water storage capability equivalent to approximately 7.5 minutes of full load operation. The internal condenser design provides for the effective condensing of steam, scavenging and removal of noncondensable gases, and the deaeration of the condensate.
Q430.100 The condenser tubes are protected from failure due to high temperature drains and blowdown by spray headers and impingement baffles.

The circulating water side of the main condenser is a triple pass arrangement having two vertically divided water circuits. Each of the two water circuits can be isolated for the repair of leaking tubes by closing the motor operated butterfly valves on the condenser inlet and outlet connections. The main condenser will maintain back pressures of 2.4, 2.9, and 3.7 In. Hg. Abs in the three condenser shells when operating at rated turbine output with 89°F inlet circulating water temperature and 95 percent clean
Q430.101 & Q430.102 tubes. Loss of condenser vacuum due to the accumulation of non-condensable gases is prevented by the steam air ejectors described in Section 10.4.2. However, if vacuum loss should occur, high back pressure signal(s) from turbine exhaust instrumentation will trip the turbine, low vacuum signal(s) from condenser instrumentation will block the condenser steam dump valves (see Section 10.4.4), and the condenser will in effect be isolated from all direct sources of main steam. Loss of condenser vacuum does not affect operation of the main steam isolation valves. The condenser tubes and components are constructed of corrosion/errosion resistant materials. In addition, a continuous tube cleaning system (Amertap) is provided to keep the inside tube surfaces clean and the condenser operating at peak performance.

The main condenser can accept a bypass steam flow of approximately 40% of rated main steam flow without exceeding the turbine high backpressure trip point with design inlet circulating water temperature. This bypass steam

11.2.2.7.1.5 Floor Drain Tank Subsystem Operation

The water in the floor drain tank is sampled to determine the degree of processing required. Normally the contents of the floor drain tank can be sent to Waste Monitor Tank B after filtration. If further processing is required or it is desirable to reclaim distillate, the laundry and hot shower tank subsystem, waste evaporator, and water evaporator condensate demineralized
Q460.8 can be utilized. Unless it is recycled, ultimately the contents of the floor drain tank should be returned to Waste Monitor Tank B for recirculation, sampling, and discharge into the low pressure service water discharge via the Nuclear Service Water System at a rate determined by the dilution flowrate available.

Upon accumulation of ≅3000 gallons of waste in the FDT, valves are aligned for recirculation and the FDT Pump started. At ≅4000 gallons, a sample is taken to gain a rough estimate of activity and concentrations of glycol, boron, iron, detergents, and decon chemicals. The rough sample results indicate the type of treatment needed according to the level of contamination as well as identify the source in the event of unusual leakage. The waste is pumped to WMT B, since WMT A will normally be used for receiving treated LHST waste. For example:

a. Very low activity and low concentration of non-toxic chemicals, i.e. meets discharge requirements: pump through FDT filter to WMT B, recirc., sample, analyze, and discharge through the WL System discharge radiation monitor and automatic control valve. Approximately 30% of the water
Q460.8 from the floor drain tank is diverted upstream of Waste Monitor Tank B for processing using the waste evaporator and waste evaporator condensate demineralizer to reclaim distillate.

b. Low activity detergent-complexed radioactive contaminants: pump through LHST filters and LHST carbon filter to WMT B and recirc., test, and discharge.

c. Low activity ionic radioactive contaminants (salts): pump through LHST filters, carbon filter, and WMT demineralizer to WMT B.

d. Low to high activity particulate radioactive contaminants: treat as in (a) or through LHST filters, if analysis shows 5u filtration necessary.

e. Low activity and moderate levels of elemental chemical contaminants (boron, chromium, iron): pump through LHST primary, secondary filters, and WMT demineralizer to WMT B.

f. Low activity and moderate levels of organic contaminants (glycol, detergents): treat as in (b).

g. High activity in the form of soluble contaminants (detergent-complexed and/or ion-pair): pump to MST in 600 gallon batches, add chemicals to adjust pH and prevent foaming, and pump to the WEFT for processing in the evaporator.

4) Valve stem leakoffs flow directly to the recycle holdup tank at a rate of 200 gpd and at approximately reactor coolant concentration.

5) Based on the above values, the recycle holdup tank has collection time of 29.17 days, assuming 40% capacity. Recycle holdup tank discharge is to the reactor makeup water storage tank through the boron recycle evaporator and evaporator condensate demineralizer, which have a collective DF of 10E3 for iodines and 10E3 for other isotopes. Process time is 4.15 days. Ten percent (≅92,000 gal.) of the boron recycle input is assumed to be released to the environment. This value is considered conservative enough to include any deliberate discharges for tritium control.

6) Waste collected by the waste evaporator feed tank is processed through the waste evaporator and the waste evaporator condensate demineralizer, which have a collective DF of 10E4 for iodine and 10E5 for other isotopes. Collection time is 2.1 days assuming 40% capacity of the waste evaporator feed tank, while process time is 0.2 days for the 7.5 gpm evaporator. Ten percent of the stream is released to the environment.

Q460.8

7) Floor drain tank contents are normally sampled, then processed through the floor drain tank strainer and filter. Approximately 30% of the influent to the floor drain tank is processed in the waste evaporator and waste evaporator condensate demineralizer to reclaim distillate. The remaining effluent is transferred to Waste Monitor Tank B where it is resampled and, if within specifications for release, the entire batch is discharged from the station. Further processing is performed on the batch if it is out of specifications. This is discussed in Section 11.2.2.7.1.5.

8) Laundry and hot shower tank contents may be discharged through the waste monitor tank demineralizer, as required. However, since use of the demineralizer is not continuous, no credit for reduction of laundry waste activity is taken.

9) Blowdown occurs at a rate of 280 gpm through the steam generator blowdown tank, where liquids are directed upstream of the condensate polishing demineralizers and gases are vented to the "D" heaters. There is no release of wastes from the blowdown system.

11.2.3.1 Release Points

All discharges of detectable radioactivity are through the Low Pressure Service Water System into Lake Wylie. The location of the discharge can be seen on Figure 2.1.1-4.

11.2.3.2 Dilution Factors

Low pressure service water will provide dilution for liquid wastes with a flow that will vary depending, among other things, on the station power output and Lake Wylie water temperatures. For the purpose of dose evaluations, an average dilution with 54,000 gpm is assumed. Estimates of near-field and far-field dilution are discussed in Chapter 5 of the Catawba ER-OLS.

The rate of radioactive discharges will be based on the available dilution and the concentrations of 10CFR20, Appendix B, Table II.

steel barriers in your analyses. The staff's position on allowable ductility ratios is contained in Attachment 3. Indicate your compliance with the staff's position in this matter or provide justification for any deviations therefrom.

Response:

See revised Section 3.5.3.2.

220.20 (3.7.1.3) For concrete and bolted steel structures, you have used damping values which are slightly greater than those given in Regulatory Guide 1.61. Provide the assessment of impact if you have to conform to Regulatory Guide 1.61 values. Also, provide justification for use of higher values and correlation between stress levels and these values.

Response:

An assessment was made to determine the impact should Catawba be required to conform to the lower damping values recommended by Regulatory Guide 1.61. This guide recommends damping values of 4% and 7% for OBE and SSE load cases, respectively, when designing concrete and bolted steel structures. Catawba designs utilize damping values of 5% for both earthquake load cases. Evaluations indicate that a stress increase of approximately 7% would be realized in the OBE load condition if the 4% damping values are used in lieu of the 5% values.

The SSE load condition, however, controls the design of our concrete and steel structures. The SSE load cases create stresses approximately 85% higher than the OBE load cases. Since the controlling SSE design load case utilizes a more conservative damping value than that recommended by Regulatory Guide 1.61, conformance to this guide would have no impact on the Catawba design.

220.21 (3.7.2.1) In Section 3.7.2.1.1.2 of the FSAR you have stated that the mass of the equipment is lumped at the elevation at which it is supported and the structural connection between equipment and structure is considered rigid for the seismic analysis of the structure. Your decoupling criterion between system and subsystem is not clear from above statement. Provide clear definition of your decoupling criterion. One acceptable criterion is outlined in SRP Section 3.7.2.

Response:

See revised Section 3.7.2.1.1.1.

220.22 (3.7.2.1) Clarify, whether or not, your lumped mass model of the containment interior structure contains detailed representations (stiffness,

damping, and mass) of major equipment such as reactor vessels and steam generators. If not, provide justification for not doing so.

Response:

The lumped parameter model used in the seismic analysis of the Catawba Nuclear Station, Reactor Building interior structure contained only the masses of the NSSS components. The stiffness characteristics of the individual items were not used in the model. This method of treating these components was state of the art at that point in time and was excepted when the PSAR was submitted.

In spite of the above, additional work was undertaken. (See the response to SEB Action Item 5, which was transmitted by letter of April 8, 1982, W. O. Parker, Jr. to H. R. Denton). The results of this work indicate that if the NSSS System were to be included in the structural model, two basic changes would result. One, the systems force response would drop from 10% to 15%, indicating that the building is adequately designed and two, the resulting response spectra would decrease but would show a frequency shift of from approximately 10% to 15% toward the low frequency end of the curve. The resulting spectra should fall within the present spectra.

In view of the above it is concluded that the simplification used in the original calculation is justified.

220.23 (3.7.2.1) Provide the description of your procedure used in modeling the hydrodynamic phenomena for the buildings (e.g., Fuel Pool and Fuel Handling building, Nuclear Service Water Pump Structure) where this is a consideration.

Response:

See revised Section 3.7.2.1.1.3.

220.24 (3.7.2) Describe your procedure to compute dynamic lateral earth pressure and hydrodynamic groundwater pressure during seismic event.

Response:

See revised Section 3.7.2.4.

220.25 (3.7.2.1) For structures described in Section 3.7.2.1.1.2 demonstrate that adequate numbers of masses or degrees of freedom in dynamic models are considered. A criterion acceptable to the staff is described in SRP Section 3.7.2.II.1.a(4).

Response:

See revised Section 3.7.2.1.1.2.

220.29 (3.7.2.5) Provide description of your procedure to account for three components of earthquakes in generation of floor response spectra.

Response:

See revised Section 3.7.2.6.

220.30 (3.7.2.5) Provide description of the procedure used in allowing for vertical flexibility of floors in generation of vertical response spectra.

Response:

See revised Section 3.7.2.5.

220.31 (3.7.2.6) In this section, you have stated that the earthquake ground motions are assumed to act in one of the horizontal directions and vertical direction simultaneously. It is not clear how you have combined the responses due to these two motions. Is it by absolute sum method or SRSS method? Also, provide a comparison of your method of combining two components of earthquakes with the currently acceptable procedure of combining all three components with SRSS method. This comparison need only be done for one structure which is essentially non-symmetric; e.g., internal structure.

Response:

See revised Section 3.7.2.6.

220.32 (2.7.2.11) The present technical position of the staff requires that the accidental torsion, based on eccentricity of minimum 5% of the base dimension, be included in the design of structures. This is in addition to that which results from the actual geometry and mass distribution of the building. Either indicate your willingness to comply with this position or provide justification for not doing so.

Response:

Standard Review Plan 3.7.2, Seismic Systems Analysis, Part III, Section II, states that an acceptable method for accounting for accidental torsion is to add an additional 5% of the maximum building base dimension to the eccentricity that exists naturally in the building. This was not done at Catawba because the requirement did not exist at the time of the initial analysis.

The effect of incorporating this requirement into the Catawba analysis has been evaluated using the Auxiliary Building as the case study. The Auxiliary Building was selected because it exhibits the largest existing eccentricities as well as the largest plan dimensions.

Response:

220.43 (3.8.3) Provide analysis details of the ice condenser floor and the lower support structures. Demonstrate that the applicable code provisions are fully met in your design.

Response:

See revised Section 3.8.3.1.10.

220.44 (3.8.3) In FSAR Section 3.8.3.4.4 you stated that the crane wall was analyzed as a space frame. Provide more analysis details, such as computer code, governing loads and load combinations, critical design forces and the design of reinforcing bars.

Response:

See revised Section 3.8.3.4.4.

220.45 (3.8.4) With respect to FSAR Section 3.8.4.1, Spent Fuel Building, describe in detail your design and analysis of spent fuel building structures and fuel pool racks. Enclosed is a copy of staff position on "the minimum requirements for design of spent fuel pool racks" (Attachment 5). Indicate compliance with this position or justify deviations therefrom.

Response:

The design and analysis of the Spent Fuel Building structures is described in Sections 3.8.4.4, 9.1.2.1, 9.1.2.2, and 9.1.2.3. The design and analysis of the spent fuel racks is described in Section 9.1.2.2 and 9.1.2.3. Spent fuel rack plans and details are provided in Figures 9.1.2-7 and 9.1.2-8. The spent fuel pool liner plate is described in Section 3.8.4.1 (A.2). The spent fuel storage racks are described in Sections 9.1.2.1 (8), 9.1.2.2, and 9.1.2.3.

Refer also to revised Sections 9.1.2.1 (1), 9.1.2.2, and 9.1.2.3.

220.46 (3.8.4) In FSAR Section 3.8.4.4 you stated that masonry construction is designed and reinforced to remain functional under the applicable loading conditions. Enclosed is a copy of design criteria for safety-related masonry wall evaluation (Attachment 6). Identify any difference in requirements of materials, testing, analysis, design and construction between Catawba design and the staff position. Provide justification for these differences or indicate your compliance with

them. Provide a general description of the masonry walls at Catawba indicating number of walls, usage, buildings in which they are located, types (e.g., single wythe, double wythe), construction practices, loads and load combinations, and general design procedures. Provide sample calculations for each different type of wall.

Response:

A response was provided in W. O. Parker's letter of April 8, 1982 to H. R. Denton in response to SEB Action Item 34.

static loads and uplift in the design of subsurface walls and foundation mats are justified and conservative.

(b) Provide material gradation curves for the in-situ and backfill soils, and fine and coarse filter materials, and also confirm (with calculations) that the filter criterion has been met between: (a) insitu rock and back fill soils; (b) fine and coarse filter materials, and (c) coarse filter material and the perforated pipe drain.

(c) Provide the settlement (total and differential) data and piezometric pressure readings taken on the Standby Nuclear Service Water (SNSW) dam, and compare the measured data with estimated values.

Response:

See revised Sections 2.4.13.5, 2.5.4.13, and 2.5.6.10.

241.8 (2.5.6.8) Describe the procedures to be used in performing periodic inspections of the SNSW Dam and discharge facility to conform to the provisions of Regulatory Guide 1.127. Provide a report on the results of any recent dam inspection which may have been performed, particularly regarding erosion, excessive seepage, 'piping,' etc.

Response:

See revised Section 2.5.6.8.2.

241.9 (2.5.6.8) In Table 3.8.1-2 (page 7) in FSAR, Vol. 4, the minimum factor of safety against floatation for all Category I foundations is given as 1.1. Describe briefly the procedure used in determining this factor of safety.

Response:

See revised Table 3.8.1-2.

reading will indicate approximately 2200 lbs (assembly plus mast-wet condition) and for the dry condition, approximately 2500 lbs (assembly plus mast total weight). Should the assembly movement fall outside the range of the suspended weight ±200 lbs, movement should stop.

(iii) Once the guide cable becomes slack, the operator verifies the correct position of the assembly, using a z-z tape which identifies the depth of the assembly. Verification is made within ±¼ inch according to operational procedures.

1b. The following precautions are taken to prevent loading an Oconee fuel assembly into a spent fuel rack designated for Catawba fuel (without spacer):

(i) A visual examination will verify that the rack is empty.

(ii) As stated in the McGuire and Oconee operating procedures and shown in the applicable drawings, to separate the fuel handling tool from the fuel assembly, a roll pin must be turned by hand, in a counterclockwise direction 350°. Should the handling tool and guide line be under any form of tension, the roll pin would be difficult to turn the necessary rotation. As such, the bridge operator can observe and correct the situation.

(iii) The B&W fuel assembly can be inserted into the Catawba spent fuel rack. However, due to the design and necessary clearance required to remove the grapples, it is considered impossible to remove the handling tool. Thus, the assembly cannot disengage in the spent fuel rack.

2. As per the operational procedures, the Reactor Engineer or his designee, is given the completed procedure process record form, labeling exactly which spent fuel racks are equipped with spacers.

3. As stated in Section 9.1.2.4 the storage of the Oconee spent fuel in the Catawba spent fuel pool is within the Safety Evaluation presented in Section 9.1.2.3. Sections 9.1.2.3.1 and 9.1.2.4 state that the analysis of an infinite array of Oconee assemblies (3.3 wt% - U-235) stored in Catawba's spent fuel pools will have a K_{eff} of less than 0.95 (1). Oconee fresh fuel will never be stored in the Catawba pool; therefore, the criticality analysis is conservative.

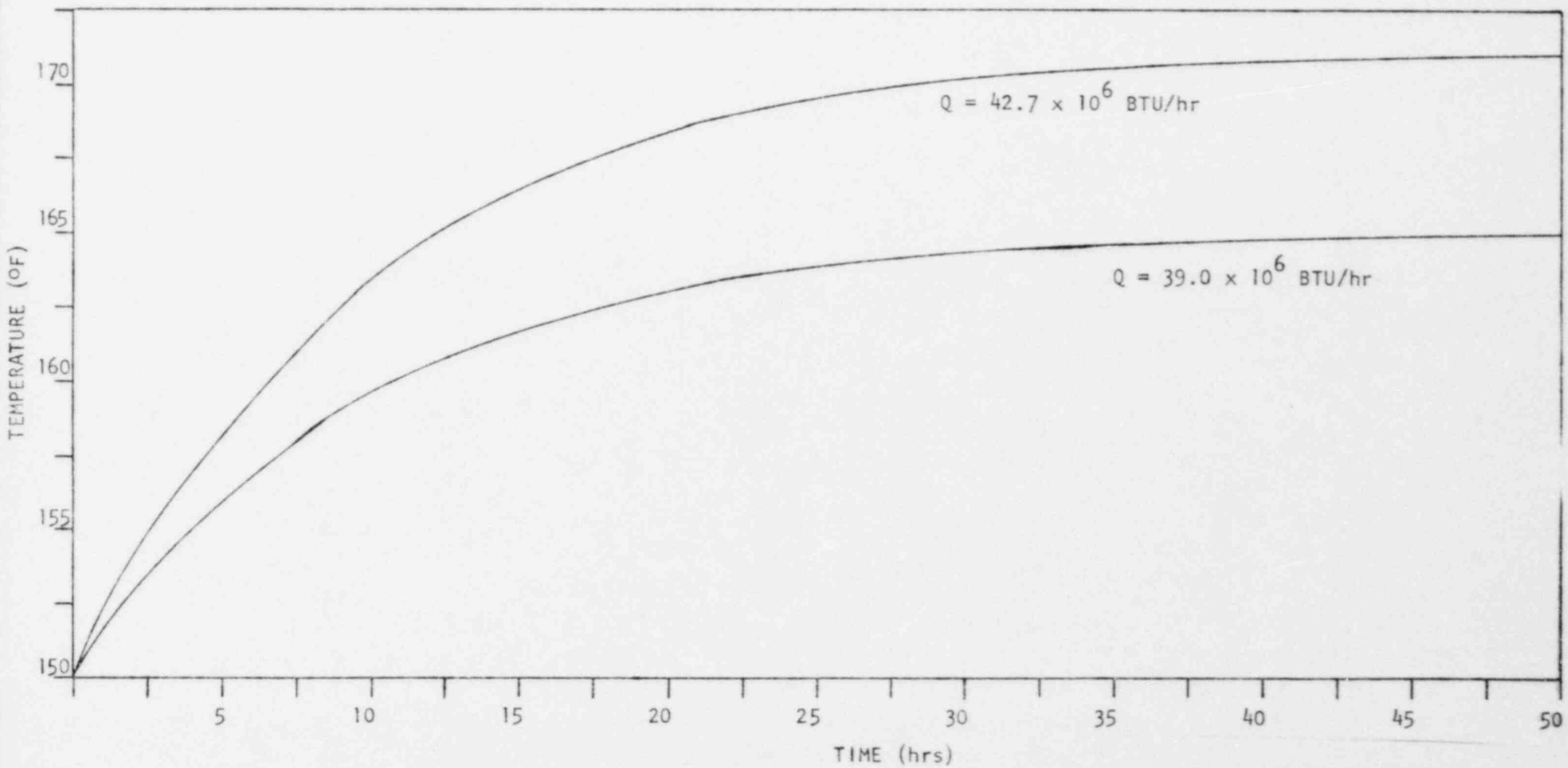
410.12 (9.1.3) Provide information on the spent fuel pool water temperature following the loss of one cooling train assuming the maximum heat load condition with Catawba fuel and with non-Catawba fuel. Present an analysis of temperature vs. time for these two conditions. Describe

the disposition of the heat load to all heat sinks including the spent fuel pool heat exchanger, the spent fuel pool water purification system, any waste or drains, and the environment of the fuel building. Discuss the effects of the heat and humidity load on the fuel building ventilation system.

Response:

An analysis of temperature versus time for Q = 39.0×10^6 BTU/hr (Catawba fuel) and Q = 42.7×10^6 BTU/hr (non-Catawba fuel) was performed using only one spent fuel pool heat exchanger and the pool water as heat sinks. The results of this analysis are presented in Figure Q410.12-1.

The additional heat and humidity load for these two cases would not affect the fuel building ventilation system.



SPENT FUEL POOL
TEMPERATURE VERSUS TIME
CATAWBA NUCLEAR STATION
Figure Q410.12-1

410.15 (9.1.4) Provide a listing of all light loads (those of weight less than one fuel assembly) carried over the open reactor vessel or the spent fuel pool including their kinetic energy on impact with spent fuel and discuss the consequences of dropping of these loads on stored fuel. It is our position that dropping of these loads not result in release of radioactivity in excess of that assumed in the design basis fuel handling accident.

Response:

410.16 (9.2.1) Figure 9.5.4-10 indicates that the nuclear service water yard piping crosses the non-safety-related condenser circulating water system piping. Describe the protective features provided for the nuclear service water piping at these crossover points against the effects resulting from failed condenser circulating water pipe in a SSE. It is our position that the integrity of the nuclear service water system be maintained in a SSE.

Response:

See revised Section 9.2.1.2.1.

410.17 (9.2.2) Your component cooling water system has a single common supply line and a single common return line to the reactor building components cooled by this system. These lines contain electrically-operated containment isolation valves. Cooling water to all four reactor coolant pump (RCP) seals and motor bearings which require continuous cooling during all modes of operation would be lost on failure of one of these valves. Loss of cooling could result in motor bearing failure and locking of all pump rotors. Indicate the consequences of the loss of motor bearing and seal thermal barrier cooling to all pumps and show that, without operator intervention, this event will not cause fuel damage or damage to the reactor coolant system pressure boundary with consequences greater than those assumed in the design basis RCP locked rotor accident.

If this can not be demonstrated, then it is our position that you verify by test that the RCPs can function satisfactorily for 30 minutes without component cooling water flow and provide redundant safety grade indication of the loss of component cooling water to the pumps in order to assure that the operator will have sufficient time to trip them. Alternatively, you may provide redundant component cooling water supply and return lines to/from the RCPs, or an automatic trip of the RCPs on loss of component cooling water flow.

Response:

Redundant safety grade indication of the loss of component cooling water to the reactor coolant pumps (RCP) will be provided to assure that the operator will have sufficient time to trip them. Two safety grade transmitters will be provided on the common header supplying the reactor coolant pump motor upper and lower bearings, and the reactor coolant pump thermal barriers. These transmitters will alarm on low flow in the control room. It has been verified, by testing that the pumps can function satisfactorily for 10 minutes without component cooling water flow. If component cooling water flow cannot be re-established to the RCP's within 10 minutes following a low flow alarm, the reactor and then the RCP's will be tripped by the operator.

410.18 (9.2.5) Indicate the effect of including the maximum worst case spent fuel pool cooling load on your analysis of the heat rejection capabilities of the standby nuclear service water pond (SNSWP). It is unclear if this load has been considered in your analysis of the 30-day heat inputs to the SNSWP from the station auxiliary systems. Also provide a drawing showing the physical locations of the SNSWP intake and discharge structures, the nuclear service water pump house, and the interconnecting pipe routing.

Response:

The fuel pool cooling heat exchangers are on the non-essential KC header and are isolated from the KC and RN Systems following a safety injection (SS) signal. The fuel pool is designed to reject decay heat during this time by boiling off and is therefore not included in the 30 day heat rejection load on the nuclear service water pond.

The location of the SNSWP intake and discharge structures and the nuclear service water pump house is shown on Figure 2.4.1-1. Interconnecting pipe routing is shown on Figures 9.5.4-10 and Q410.18-1.

410.19 (9.2.6) Provide drawings that indicate the relationship of the auxiliary feedwater condensate storage tank to the condensate storage system and to the auxiliary feedwater system. FSAR Figures 9.2.6-3 and 10.4.9-1 concerning this tank and system do not agree.

Response:

See revised Figure 9.2.6-3.

410.20 (9.3.1) Provide additional information concerning the discussion in FSAR Section 9.3.1.3 which describes the need for air operated valve operation in the event of a control room evacuation coincident with a station blackout in order to bring the station to a hot standby condition from the auxiliary shutdown panel. Describe any special provisions made to assure the reliable delivery of instrument air to these valves under this coincident event. Identify the valves requiring this capability.

Response:

As discussed in Sections 9.3.1 and 10.3 a non-safety related air supply and control system is sufficient and acceptable for the steam generator PORV's based on the following:

1. The steam generator PORV's are equipped with redundant safety related solenoid valves (Train A & B) which are deenergized to vent air off the spring loaded PORV's to close them upon receipt of a main steam isolation signal. Continued heat rejection following PORV closure will be provided by the main steam safety valves.

2. Once hot standby is reached there will be ample time to call in additional personnel or expertise to assess the situation and take the necessary corrective action. The plant can then be taken to a cold shutdown condition by manually operating the steam generator PORV's using local handwheels in the event instrument air is not available and cannot be restored.

3. Instrument air can be provided by any of the three instrument air compressors or either of the two station air compressors which automatically back up instrument air. The instrument air compressors and dryers can be manually loaded on the blackout bus during sequence #13 after 12 minutes in the event of a station blackout. Based on the above, complete unrestorable loss of instrument air is very unlikely and manual operation of the PORV's is acceptable.

4. Catawba is a Class 2 plant and limited operator action is allowed outside the control room in BTP 5-1.

410.21 (9.3.1) Describe the means provided for assuring that instrument air quality is within the necessary limits to assure proper functioning of all air operated valves and instrumentatic. in safety related systems.

Response:

Section 9.3.1.3 was revised to clarify that the Instrument Air System is not safety related. Failure of this system will not render any safety system equipment or its function inoperable. However, a means of operating the Instrument Air System during a station black-out has been provided to facilitate shutdown.

See revised Sections 9.3.1.2.1 and 9.3.1.5 for further discussion on instrument air quality.

410.22 (9.3.3) Provide 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 these safety-related components or systems. The analysis must show that drainage by natural routes such as stairwells of equipment hatches or by the non-seismic Category I drainage system under failed conditions is adequate to prevent the loss of function of safety-related components and systems. Indicate how interconnected drains serving redundant saftey-related equipment or cubicles can be prevented from allowing leakage from one failed redundant train from backflowing and flooding out the other train. In those cases where separate drains are provided for redundant safety-related components or systems, provide an analysis that demonstrates that the component and/or area drains serving these components or systems have been sized for maximum leakage flow conditions.

420.5 On November 7, 1979, Westinghouse notified the Commission of a potential undetectable failure which could exist in the engineered safeguards P-4 interlocks. Test procedures were developed to detect failures which might occur. The procedures require the use of voltage measurements at the terminal blocks of the reactor trip breaker cabinets.

Concern:

In order to minimize the possibility of accidental shorting or grounding of safety system circuits during testing, the staff believes that suitable test jacks should be provided to facilitate testing of the P-4 interlocks. Provide a discussion on how the above issue will be resolved for Catawba.

Response:

In order to implement the Westinghouse recommended procedures, a voltage indicator will be wired to the reactor trip breaker terminal blocks. This will allow operating personnel to check the status of the P-4 interlock. This modification will be completed prior to fuel load.

height these lines are terminated above finished ground grade. If these lines are located outdoors discuss the provisions made in your design to prevent entrance of water into the storage tank during adverse environmental conditions.

Response:

See revised Section 9.5.4.2.1.

430.69
(9.5.5)
Recent licensee event reports have shown that tube leaks are being experienced in the heat exchangers of the jacket cooling water system. Provide a discussion on the means used to detect tube leakage and the corrective measures that will be taken. Provide the permissible inleakage or outleakage for each subsystem cooled by the cooling water system and the time of operation with leakage to assure coolant or cooled liquid quality can be maintained within safe limits. Also provide the results of a failure mode and effects analysis to show that failure of a piping connection between subsystems (engine water jacket, lube oil cooler, governor lube oil cooler, and engine air inter-cooler) does not cause total degradation of the diesel generator cooling water system. (SRP 9.5.5, Part III, Item 1a.)

Response:

See revised Section 9.5.5.2.2.

430.70
(9.5.5)
FSAR Section 9.5.5.2 refers apparently to two kinds of "standby operation." The first paragraph states that the engine-driven cooling water circulation pump operates during engine standby and assures that the system is completely filled with water. The third paragraph states that a keep-warm circulating pump keeps "the engine warm during standby." Describe the meaning of standby operation for both of these cases. Modify the FSAR accordingly.

Response:

See revised Section 9.5.5.2.

radioactive release to the atmosphere from the faulty unit. The recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam line. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily. Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the isolation procedure can be completed within 30 minutes of accident initiation. Included in this 30 minute time period would be an allowance of 5 minutes to trip the reactor and actuate the safety injection system (automatic actions), 10 minutes to identify the accident as a steam generator tube rupture and 15 minutes to isolate the faulted steam generator.

Immediately apparent symptoms of a tube rupture accident such as falling pressurizer pressure and level and increased charging pump flow are also symptoms of small steam line breaks and loss of coolant accidents. It is therefore important for the operator to determine that the accident is a rupture of a steam generator tube in order that he may carry out the correct recovery procedure. The accident under discussion can be identified by the following method. In the event of a complete tube rupture, it will be clear soon after the trip that the level in one steam generator is rising more rapidly than in the others.

Also this accident could be identified by either a condenser air ejector exhaust high radiation alarm or a steam generator blowdown radiation alarm.

The operator carries out the following major operator actions subsequent to reactor trip which lead to isolation of the faulted steam generator and minimizing primary to secondary leakage.

1. Identification of the faulted steam generator.

2. Isolation of the faulted steam generator.

3. Subcooling of NC system fluid to 50° below no-load temperature.

4. Depressurization of the NC system to terminate breakflow, and

5. Terminating safety injection.

Loss of Coolant Accident: See Table 440.56-4

No manual actions are required of the operator for proper operation of the ECCS during the injection mode of operation. Only limited manual actions are required by the operator to realign the system for the cold leg recirculation mode of operation, and, at approximately 24 hours, for the hot leg recirculation mode of operation. These actions are delineated in Table 440.56-4.

The changeover from the injection mode to recirculation mode is initiated automatically and completed manually by operator action from the control room. Protection logic is provided to automatically open the two Safety Injection System (NI) recirculation sump isolation valves when two of four refueling water storage tank level channels indicate a refueling water storage tank level less than a low level setpoint in conjunction with the initiation of the engineered safeguards actuation signal ("S" signal). This automatic action would align the two residual heat removal pumps to take suction from the containment sump and to deliver directly to the NC system. It should be noted that the residual heat removal pumps would continue to operate during this changeover from injection mode to recirculation mode.

The two charging pumps and the two safety injection pumps would continue to take suction from the refueling water storage tank, following the above automatic action, until manual operator action is taken to align these pumps in series with the residual heat removal pumps.

The refueling water storage tank low level protection logic consists of four level channels with each level assigned to a separate process control protection set. Four refueling water storage tank level transmitters provide level signals to corresponding normally deenergized level channel bistables. Each level channel bistable would be energized on receipt of a refueling water storage tank level signal less than the low level setpoint.

A two out of four coincident logic is utilized in both protection cabinets A and B to ensure a trip signal in the event that two of the four level channel bistables are energized. This trip signal, in conjunction with the "S" signal, provides the actuation signal to automatically open the corresponding containment sump isolation valves.

The low refueling water storage tank level signal is also alarmed to inform the operator to initiate the manual action required to realign the charging and safety injection pumps for the recirculation mode. The manual switchover sequence that must be performed by the operator is delineated in Table 440.56-4. Following the automatic and manual switchover sequence, the two residual heat removal pumps would take suction from the containment sump and deliver borated water directly to the NC system cold legs. A portion of residual heat removal pump A discharge flow would be used to provide suction to the two charging pumps which would also deliver directly to the NC system cold legs. A portion of the discharge flow from residual heat removal pump B would be used to provide suction to the two safety injection pumps which would also deliver directly to the NC system cold legs. As part of the manual switchover procedure, the suctions of the safety injection and charging pumps are cross connected so that one residual heat removal pump can deliver flow to the NC system and both safety injection and charging pumps, in the event of the failure of the second residual heat removal pump.

Q440.56-4
TABLE LOSS OF COOLANT ACCIDENT

REQUIRED OPERATOR ACTION	ALARMS TO ALERT THE OPERATOR TO INITIATE A PARTICULAR ACTION	DELAY TIME ASSUMED	INSTRUCTIONS GIVEN TO THE OPERATOR FOR PERFORMING THE REQUIRED ACTION	COMPONENTS AND INSTRUMENTATION NECESSARY TO COMPLETE INDICATED ACTION	IMPACT OF SINGLE ACTIVE COMPONENT FAILURE	IMPACT OF THE OPERATOR'S FAILURE TO TAKE ACTION OR THE OPERATOR TAKING A CLOSELY RELATED BUT ERRONEOUS ACTION
A. The operator must manually complete the changeover of the ECCS system from the injection mode to the cold leg recirculation mode. * Please Refer to Section 6.3.	Refer to Sections 6.3.5.4 and 9.2.6	Long Term	1. Verify that the containment sump isolation valves are open. 2. Verify the isolation valve in each RHR suction line from the RWST is closed. 3. Close the isolation valve in each SI pump miniflow line. 4. Close the two isolation valves in the crossover line downstream of the RHR heat exchangers. 5. Open the two parallel valves in the common suction line between the centrifugal charging pump suction and the SI pump. 6. Open the valve in the discharge line	1. RWST Level Indicator: (PAMS) 2. All valves required to be operated for the switchover sequence.	1. None 2. Refer to Table 6.3.2-5.	A&B The plant emergency operating procedures include instructions and verification steps to ensure proper manual realignment of the ECCS for recirculation by the operator. The failure to perform one step or the performance of one step out of order, as a single failure, should not reduce ECCS recirculation capability below minimum safeguards. Should the operator fail to take any action following automatic ECCS switchover initiation, the consequences will be the loss of the safety injection and charging pumps. The residual heat removal pumps will be protected from damage by automatic ECCS switchover initiation. For small break LOCA in the

Q440.56-4 (Cont'd)
TABLE LOSS OF COOLANT ACCIDENT

REQUIRED OPERATOR ACTION	ALARMS TO ALERT THE OPERATOR TO INITIATE A PARTICULAR ACTION	DELAY TIME ASSUMED	INSTRUCTIONS GIVEN TO THE OPERATOR FOR PERFORMING THE REQUIRED ACTION	COMPONENTS AND INSTRUMENTATION NECESSARY TO COMPLETE INDICATED ACTION	IMPACT OF SINGLE ACTIVE COMPONENT FAILURE	IMPACT OF THE OPERATOR'S FAILURE TO TAKE ACTION OR THE OPERATOR TAKING A CLOSELY RELATED BUT ERRONEOUS ACTION
			from the number 1 RHR heat exchanger to the suction of the centrifugal charging pumps; open the valve in the discharge line from the number 2 RHR heat exchanger to the suction of the SI pumps. 7. Close the isolation valve in the SI pump suction line from the RWST. Close the two parallel isolation valves in the centrifugal charging pump suction line from the RWST.			unlikely event of losing all high head pump delivery capability this situation could lead to core uncovery and inadequate core cooling. This situation is addressed in WCAP-9691 as the loss of the Emergency Coolant Recirculation (ECR) function following a small break. LOCA Analyses have been performed for loss of high head safety injection for small LOCA which are presented in WCAP-9573. Inadequate Core Cooling guidelines are being developed which would instruct the operator on the appropriate operator actions to be taken for this event.
B. At approximately 24 hours after the transient,	None Required	N/A	1. Insure that the "S" signals have been reset and defeated if not previously	Refer to Section 6.3	Refer to Table 6.3.2-5	For large break LOCA the residual heat removal pump delivery to the RCS would be

TABLE LOSS OF COOLANT ACCIDENT

REQUIRED OPERATOR ACTION	ALARMS TO ALERT THE OPERATOR TO INITIATE A PARTICULAR ACTION	DELAY TIME ASSUMED	INSTRUCTIONS GIVEN TO THE OPERATOR FOR PERFORMING THE REQUIRED ACTION	COMPONENTS AND INSTRUMENTATION NECESSARY TO COMPLETE INDICATED ACTION	IMPACT OF SINGLE ACTIVE COMPONENT FAILURE	IMPACT OF THE OPERATOR'S FAILURE TO TAKE ACTION OR THE OPERATOR TAKING A CLOSELY RELATED BUT ERRONEOUS ACTION
the operator must manually switch over to the hot leg recirculation mode.			accomplished. 2. Terminate RHR pump flow to the RCS cold legs and establish RHR pump flow to the RCS hot legs by: a) closing the RHR cold leg header isolation valves b) opening the two isolation valves in the crossover line downstream of the RHR heat exchangers c) opening the RHR hot leg header isolation valve 3. Terminate SI pump flow to the RCS cold legs and establish SI pump flow to the RCS hot legs by: a) stop SI pump no. 1 and close its corresponding			sufficient to provide adequate core cooling during recirculation.

Q440.56-4
TABLE LOSS OF COOLANT ACCIDENT

REQUIRED OPERATOR ACTION	ALARMS TO ALERT THE OPERATOR TO INITIATE A PARTICULAR ACTION	DELAY TIME ASSUMED	INSTRUCTIONS GIVEN TO THE OPERATOR FOR PERFORMING THE REQUIRED ACTION	COMPONENTS AND INSTRUMENTATION NECESSARY TO COMPLETE INDICATED ACTION	IMPACT OF SINGLE ACTIVE COMPONENT FAILURE	IMPACT OF THE OPERATOR'S FAILURE TO TAKE ACTION OR THE OPERATOR TAKING A CLOSELY RELATED BUT ERRONEOUS ACTION
			cold leg cross-over header isolation valve b) open its corresponding hot leg header isolation valve c) restart SI pump no. 1 d) stop SI pump No. 2 and close its corresponding cold leg cross-over header isolation valve e) close the SI common cold leg header isolation valve f) open SI pump No. 2's hot leg header isolation valve g) restart SI pump No. 2			
C. Check Reactor Coolant Pump trip criteria	None Required		The operator is instructed to trip all RCP's when the RCS pressure reaches a specified pressure and	C.1 RCP's C.2 RCS pressure indicators	If a single active failure prevents tripping RCP's	Discussed in WCAP-9584.

Q440.56-4 (Cont'd)
TABLE LOSS OF COOLANT ACCIDENT

REQUIRED OPERATOR ACTION	ALARMS TO ALERT THE OPERATOR TO INITIATE A PARTICULAR ACTION	DELAY TIME ASSUMED	INSTRUCTIONS GIVEN TO THE OPERATOR FOR PERFORMING THE REQUIRED ACTION	COMPONENTS AND INSTRUMENTATION NECESSARY TO COMPLETE INDICATED ACTION	IMPACT OF SINGLE ACTIVE COMPONENT FAILURE	IMPACT OF THE OPERATOR'S FAILURE TO TAKE ACTION OR THE OPERATOR TAKING A CLOSELY RELATED BUT ERRONEOUS ACTION
			SI operation is verified.	C.3 Safety injection flow indicators.	the impact is discussed in WCAP-9584.	

*Time of RCP trip is a function of break size. It is assumed that RCP's are tripped prior to or at the RCS pressure specified in the plant procedures.

440.84 (15.3.3)

Provide an analysis, with justification, of the extent of DNBR and fuel damage. Show that the extent of fuel damage is sufficiently limited so that the core will remain in place and intact with no loss of core cooling capability. Has the calculation of the $Zr-H^2O$ reaction included the reaction on the inner cladding surface when the cladding fails? Provide an analysis of the percent of $Zr-H_2O$ reaction for the core and discuss the consequences of H_2 generated. Provide figures for three loops in operation that are consistent with those for four loops in operation, for example, Figure 15.3.3-6 is labelled for two loops in operation.

Response:

The locked rotor analysis in Section 15.3.3 shows no fuel failures occuring with calculated peak clad temperatures well below the limit temperature of 2700°F. With four loops operating initially, a peak clad temperature of 2200°F was calculated. With three loops operating initially, a peak clad temperature of 2331°F was calculated.

A maximum of 9% of the rods experience DNB ratios less than the limit value. DNB ratios less than the limit values do not necessarily imply fuel failures. However, for the purpose of conservatively calculting environmental radioactivity dose releases only, the percentage of rods in DNB was assumed to be equal to the percentage of fuel activity released.

Because no clad failures occur, zirc-water reaction on the inner clad surfaces is not applicable. With four loops operating initially, 1.4% by weight of the outer clad surface will undergo zirc-water reaction, with three loops initially operating, 1.9% by weight of the outer clad surface will undergo zirc-water reaction. The maximum amount of H_2 release is calculated to be less than 1 Ft^3. Even if all of this non-condensable gas were to migrate and collect at the top of the steam generator U-tubes (volume of U-tubes is $\sim$ 70 Ft^3/SG). This would not be enough to disrupt core cooling. This is substantiated by natural circulation work done in WCAP-9600 (Vol. 1), page 2.9-7.

The title for Figure 15.3.3-6 was corrected to read "...Three Loops In Operation, 1 Locked Rotor." In addition, figures of nuclear power, heat flux, and clad temperature with three loops in operation are added to the FSAR.

440.85 (15.3.3, 15.3.4)

The acceptance criteria in the Standard Review Plan for Sections 15.3.3 and 15.3.4 state that the rotor seizure and shaft break transients should be analyzed assuming turbine trip and coincident loss of offsite power and coastdown of undamaged pumps. Describe how your analysis has considered these assumptions.

Response:

See response to Question 440.115.

440.86 (15.3.3) Demonstrate that a rotor seizure and shaft break in a reactor coolant pump will not generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

Response:

The peak RCS pressure observed during a locked rotor or a shaft break in a reactor coolant pump was 2570 psia. This is well below 110% of design pressure (2750 psia). During the short time that RCS pressure is high enough to open pressurizer safety valves, the relief rate was within the maximum capacity of the safety valves. Thus, a loss of function of the reactor coolant system or containment barriers will not occur.

440.87 (15.3.3, 15.3.4) You classify the reactor coolant pump shaft break and locked rotor accidents as ANS Condition IV (limiting fault). The Standard Review Plan for Sections 15.3.3 and 15.3.4 classify these accidents as Condition III (infrequent incident).

Show that the transients meet the acceptance criteria for an infrequent incident.

Response:

The reactor coolant pump locked rotor and reactor coolant pump shaft break events are classified according to ANS as Condition IV events - limiting faults. Westinghouse follows this classification in Chapter 15 safety analysis. The results of the Catawba locked rotor shaft break events meet acceptance criteria for an infrequent event. The peak RCS pressure is maintained well below 110% of design pressure (2750 psia). The peak clad temperature is well below 2700°F, thus no clad failures are expected to occur.

440.88 (15.4.6) Reference or describe the analytical model used for obtaining the results in Section 15.4.6.2. Discuss the degree for conservatism incorporated in this model.

Response:

460.7 (Tables 1.8-1 11.2, 11.3, 11.4)

Table 1.8-1 indicates that Regulatory Guide 1.143, Rev. 1, 10/79 (formerly Branch Technical Position ETSB II-1, Rev. 1), "Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light-Water-Cooled Nuclear Power Plants" is not applicable to Catawba. This is not acceptable. Compare your design of liquid, gaseous and solid radwaste systems to each position in Regulatory Guide 1.143 and list the items of non-compliance and the justification for it for the purpose of evaluating your design.

Response:

See revised Table 1.8-1.

460.8 (11.2)

In Section II.2.3 for estimating the liquid releases, credit is taken for waste evaporator and condensate demineralizer in processing the floor drain tank contents. From the description of the floor drain tank subsystem operation (11.2.2.7.1.5) it is not clear that evaporator and condensate demineralizers will be used frequently. Full credit may not be taken unless both of these pieces of equipment are used continuously. Similarly, the full credit should not be taken for waste evaporator condensate demineralizer for processing waste collected by waste evaporator feed tank. Please justify your assumption.

Response:

Catawba Nuclear Station anticipates never having to process the contents of the floor drain tank using the waste evaporator or waste evaporator condensate demineralizer, to meet effluent requirements. However, these components will be used to process approximately 30% of the floor drain tank's contents to reclaim distillate.

See revised Sections 11.2.3(7) and 11.2.2.7.1.5.

Residual containment spray operation is initiated manually when required by the operator and only if the following two criteria are met:

1. Emergency Core Cooling System and Containment Spray System are operating in the recirculation mode.

2. At least one hour has passed since the initiation of the Safety Injection signal.

The Residual Spray may be required due to high containment pressure. High containment pressure would be indicated to the operator via the pressure sensors in the Containment Pressure Control System, which has pressure indicators in the control room. Refer to FSAR Section 7.6.5 for Containment Pressure Control System.

It is not intended that RHR spray be limited to a maximum value of 1575 gpm. Therefore, no flow control elements are required to adjust to a particular flowrate.

480.16 (6.2.3) Identify all secondary containment openings such as personnel access doors and equipment hatches. Discuss the administrative control to be exercised over these openings.

Response:

Access to the annulus is under the control of the Station Operations Department. Both upper and lower doors to the annulus are kept locked and the keys are under the control of the shift supervisor. If an authorized individual must gain entry to the annulus he or she obtains the key from the shift supervisor and is "logged" in. Anytime someone is inside the annulus another person is placed at the annulus door to prevent unauthorized entry of other individuals. Station Directive 3.1.2 is the procedural document that establishes the above controls. There are no remote alarms or indications of annulus door position or activation.

480.17 (6.2.4) For each containment penetration, specify whether the line is essential or nonessential, in accordance with the provisions of NUREG-0737, "Clarification of TMI Action Plan Requirements," II.E.4.2, Containment Isolation Dependability.

A-40 - Seismic Design Criteria

NRC regulations require that nuclear power structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in NRC regulations and in regulatory guides issued by the Commission. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, rereviews of the seismic design of various plants are being undertaken to assure that these plants do not present an undue risk to the public. Task A-40 is, in effect, a compendium of short-term efforts to support such reevaluation efforts of the NRC staff, especially those related to older operating plants. In addition, some revisions to the Standard Review Plan sections and regulatory guides to bring them more in line with the state-of-the art are expected.

The seismic design basis for this facility is described in FSAR Section 3.7. Pending the identification of new issues by the NRC Staff that could alter the plant design basis, we do not expect the results of Task A-40 to affect our conclusions. Accordingly, we conclude that Catawba Nuclear Station can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-46 - Seismic Qualification of Equipment in Operating Plants

The design criteria and methods for the seismic qualification of mechanical and electrical equipment at the Catawba Nuclear Station are presented in Sections 3.7, 3.9, 3.10, and 5.2 of the FSAR. These criteria and methods are under review by the Staff as a part of the overall operating license review, and any outstanding items in this area will be resolved to the satisfaction of the Staff prior to issuance of the operating license. Therefore, we conclude that the Catawba Nuclear Station can be operated before resolution of this generic issue without undue risk to the health and safety of the public.