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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 2 1981

RECEIVED JUN 04 1981 U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos. 50-413
and 50-414

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

Dear Mr. Parker:

SUBJECT: ACCEPTANCE REVIEW OF APPLICATION FOR OPERATING LICENSES FOR CATAWBA NUCLEAR STATION, UNITS 1 AND 2

On April 7, 1981, you tendered an update of your application for operating licenses for Catawba Nuclear Station, Units 1 and 2 which was filed on March 26, 1979. Your updated application included the General Information Section, an Environmental Report - Operating License Stage (ER) and a Final Safety Analysis Report (FSAR).

We have completed our review of the General Information Section, ER, and FSAR of your tendered application and have concluded that the information filed taken as a whole is sufficiently complete for docketing your application and for initiation of our safety review. It should be noted that substantive deficiencies may exist in some sections that need to be corrected during the review. In addition, our review of the hydrologic engineering information presented in section 2.4 of the FSAR is only partially complete because of our present workload in this area. We expect to complete our acceptance review of this section by July 15, 1981, and we will send you the questions arising from that review separately.

Accordingly, your filing of the application should include three (3) originals signed under oath or affirmation by a duly authorized officer of your organization. In addition, your filing should include fifteen (15) copies of the General Information Section, forty one (41) copies of the Environmental Report and forty (40) copies of the Final Safety Analysis Report. As required by Sections 50.30 and Section 51.21 10 CFR Part 50 and 51, respectively, you should retain an additional ten (10) copies of the General Information Section, one hundred nine (109) copies of the ER and thirty (30) copies of the FSAR for direct distribution in accordance with Enclosure 1 to this letter and further instructions which might be provided later. Within 10 days after filing, you must provide an affidavit that distribution has been made in accordance with this enclosure. All subsequent amendments to the ER and FSAR will require forty-one (41) and sixty (60) copies, respectively for distribution.

THIS DOCUMENT CONTAINS
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On October 28, 1980, the Commission approved a "Clarification of TMI Action Plan Requirements", now contained in NUREG-0737, which supersedes previous NUREGs on this subject. The Catawba FSAR should be amended to satisfy the modified and added requirements contained in NUREG-0737.

On October 9, 1980, the Commission published a notice of proposed rulemaking entitled "Plan to Require Licensees and Applicants to Document Deviations From the Standard Review Plan" 45 Federal Register 67099. As proposed, Duke Power Company would be required to identify and justify, prior to the issuance of the Catawba operating licenses, all deviations in the FSAR from all acceptance criteria contained in the forthcoming revision of the Standard Review Plan. We will keep you informed as to the status and content of these requirements.

You will be advised of key milestones of the review as soon as a schedule is developed. During the course of our preliminary review of your ER and FSAR, the enclosed "Request for Additional Information" (Enclosures 2 and 3 respectively) were generated. In addition to your responses to Enclosures 2 and 3, other additional information is needed to expedite our review. In most cases, Duke was previously informed of this additional information in the form of generic letters, requests for additional information and/or I&E Bulletins. Enclosure 4 contains this request for additional information.

Your response to these requests should be completed as soon as possible for our mutual benefit during the ensuing detailed technical review period. We will prepare the schedule based on the assumption that your responses are received within sixty days from the docketing date. If this milestone cannot be met, it may be necessary for us to revise our review schedule.

We shall follow a revised review procedure whereby only a single set of questions will be transmitted to you for responses. Subsequently after your responses have been received and reviewed, an early draft Safety Evaluation Report (SER) will be prepared. The draft SER will then become the subject of a series of intense meetings designed to close out the identified open items.

If during the course of our review, you should believe there is a need to appeal a staff position because of disagreement, this need should be brought to the staff's attention as early as possible so that an appropriate meeting can be arranged on a timely basis. A written request is not necessary and all such requests should be initiated through our licensing project manager assigned to

the Catawba Nuclear Station, Kahtan Jabbour. His telephone number is (301) 492-7821. The procedure is an informal one designed to allow opportunity for applicants to discuss with management areas of disagreement in the case review.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
See next page

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DISTRIBUTION LIST

FOR LICENSE APPLICATION, FINAL SAFETY ANALYSIS REPORT, AND AMENDMENTS

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(FSAR and amendments only)

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*Number in parentheses indicates number of copies.

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ENVIRONMENTAL REPORT AND AMENDMENTS

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Request For Additional Information
Environmental Report
Catawba Units 1 And 2
Docket Nos. 50-413/414

240.0 HYDROLOGIC AND GEOTECHNICAL ENGINEERING-HYDROLOGIC ENGINEERING

240.1 (2.4.1.1) The description of low flow periods on the Catawba River at Lake Wylie does not give an adequate picture of the effects of droughts on plant operation. Provide analyses of droughts, including at least the drought of record, showing the effects on water levels in Lake Wylie in relation to minimum required levels at the intake structures. The analyses should indicate both the frequency and duration of shutdowns of the plant due to inadequate water supply or low water levels. (Refer also to Section 3.3.2)

240.2 (2.4.3) Provide your estimate of the frequency of the assumed drought for the SNSWP. Also, Section 2.4.1.1 seems to indicate that an intake from Lake Wylie to the SNSWP is provided, but Section 2.4.3 does not discuss this. Please clarify and, if makeup can be provided, provide the range of water levels at which the intake can operate.

240.3 (2.4.1)

a. Provide descriptions of the floodplains (as defined in Executive Order 11988) of all water bodies, including intermittent water courses; within or adjacent to the site. On a suitable scale map provide delineations of those areas that will be flooded during the one-percent chance flood in the absence of plant effects (i.e., pre-construction floodplain).

b. Provide details of the methods used to determine the floodplains in response to a. above. Include your assumptions of and bases for the pertinent parameters used in the computation of the one-percent flood flow and water elevation. If studies approved by Flood Insurance Administration (FIA), Housing and Urban Development (HUD) or the Corps of Engineers are available for the site or adjoining area, the details of analyses need not be supplied. You can instead provide the reports from which you obtained the floodplain information.

c. Identify, locate on a map, and describe all structures, construction activities and topographic alterations in the floodplains. Indicate the status of each such structure, construction activity and topographic alteration (in terms of start and completion dates) and work presently completed.

d. Discuss the hydrologic effects of all items identified in c. above. Discuss the potential for altered flood flows and levels, both upstream and downstream. Include the potential effect of debris accumulating on the plant structures. Additionally, discuss the effects of debris generated from the site on downstream facilities.

e. Provide the details of your analysis used in response to d. above. The level of detail is similar to that identified in item b. above.

f. Identify non-floodplain alternatives for each of the items (structures, construction activities and topographic alterations) identified in c. above. Alternately, justify why a specific item must be in the floodplain.

g. For each item in f. above that cannot be justified as having to be in the floodplain either show that all non-floodplain alternatives are not practicable or commit to re-locating the structure, construction activity or topographic alteration out of the floodplain.

240.4 (7.1) Calculate the radiological consequences of a liquid pathway release from a postulated core melt accident. The analysis should assume, unless otherwise justified, that there has been a penetration of the reactor basemat by the molten core mass, and that a substantial portion of radioactively contaminated sump water was released to the ground. Doses should be compared to those calculated for the Liquid Pathway Generic Study (NUREG-0440, 1978) small river site. Provide a summary of your analysis procedures and the values of parameters used (such as permeabilities, gradients, populations affected, water use). It is suggested that meetings with the staff of our Hydrologic Engineering Section be arranged so that we may share with you the body of information necessary to perform this analysis.

290.0 ENVIRONMENTAL ENGINEERING-AQUATIC RESOURCES

290.1 Provide a short narrative describing the present status of the application of renewal for the NPDES permit filed with the S.C. Department of Health and Environmental Control on June 11, 1979.

290.2 Make available for examination during the site visit one copy of aerial photographs used to determine forest and land use types along the Catawba transmission corridors (see ER 3.9-2).

290.3 The intake structure has been significantly redesigned since issuance of the CP-EIS. Provide the intake-bay cross sectional area under both full pond and maximum draw down conditions, the size mesh of the traveling screens, a description of traveling screen operation, and the purpose and functioning of the pull-out screen bay. Discuss impact of new design relative to impact of CP-stage design.

290.4 The discharge structure has been significantly redesigned since issuance of the CP-EIS. Provide a description of and purpose of the proposed design change.

290.5 In addition to other requested information provide a summary and brief discussion in table form, by section, of differences between currently projected environmental effects (including those that would degrade, and those that would enhance environmental conditions) and the effects discussed in the environmental report submitted at the construction permit stage.

290.6 Provide an estimate of the maximum probable yearly recreational harvest of finfish, shellfish and molluscs harvested from waters downstream of the station to the Atlantic Ocean that potentially could be contaminated by radionuclides due to a maximum probably accident. The harvest estimates should be summarized by species and location of capture (water body segment) and provide an explanation of how the estimate was obtained.

290.7 Using data from the last 5 years from the National Marine Fisheries Service provide an estimate of the maximum probable yearly commercial harvest of finfish, shellfish, and molluscs harvested from waters downstream of the station that potentially could be contaminated by radionuclides due to a maximum probable accident. The harvest estimates should be summarized by species and location of capture (water body segment). Provide a generaliz explanation of how the estimate was made.

290.8 Provide a copy of the following references from Section 2.2; 2, 7, 21, 59.

290.9 (ER-OL Sec. 3.6.2) Discuss the plant operational practices, or plant design features that will result in planned 0.1 mg/l maximum total residual chlorine concentration in plant blowdown discharge.

290.10 (ER-OL Sec. 3.6.2) Estimate the time duration that residual chlorine will be present in the plant discharge (for other than sanitary waste discharge) after each application to the cooling towers.

290.11 (ER-OL Sec. 3.6.2) Provide additional information on the type, amount, and frequency of use of the organic biocide control of chlorine resistant organisms. Identify changes in planned usage from that evaluated at the CP stage.

290.12 (ER-OL Sec. 3.7.1) Indicate on a diagram of the site the location of the outfalls from the temporary and permanent sewage treatment systems into Lake Wylie.

290.13 (Table 3.3-1) The average flows cited for station water use do not add up to the total withdrawal cited for the intake from Lake Wylie. Also, some of the average flow values in this table (e.g. LPSW intake, sanitary and potable water) do not coincide with those shown in figure 3.3.1-1. Please clarify these discrepancies.

290.14 (Table 3.6.1-1) Indicate the source of the limits cited in the table.

290.15 (Table 3.6.1-2) The average chemical concentration values cited in the table are based on 7 cycles of concentration. It is stated in the text that the plant will operate at 10 cycles of concentration, but that the optimum value is 8. Indicate the basis for the determination that 8 cycles is the optimum value. Resolve the discrepancy between the table and text values, based on your projected actual operating mode and revise table 3.6.1-2 to reflect the anticipated cycles of concentration.

290.16 (ER-OL Sec. 5.1) Indicate whether and where vegetative screening will be used on-site for attenuation of plant generated noise.

290.17 (ER-OL Sec. 5.1) Identify and provide a discussion of the operational phase noise levels and expected impacts on nearby noise sensitive lands and sampling locations, identified in Sec. 2.7.

291.0 TERRESTIAL RESOURCES

291.1 Describe your method of grounding fences and other metal objects in and along the station's rights-of-way.

291.2 Discuss the anticipated effect of the cooling tower plume on the formation of fog and ice in the vicinity of the station.

291.3 On page 2.2.1-1. Table 2.2.1-1 is missing. Provide the missing table.

310.0 SITING ANALYSIS-REGIONAL IMPACT ANALYSIS

310-1 Are ther any substantial changes in the station external appearance or layout which have been made subsequent to the description in the OL-ER? If so, please describe.

310-2 Are there any new roads, transmission corridors or rail lines or relocations of roads, transmission corridors or rail lines near the plant which have been proposed subsequent to the description in the OL-ER? If so, please describe.

310.3 Section 2.1 of the FES-CP identifies new housing developments around the site including Tega Cay and River Hills Plantation. Both developments are estimated to have about 2,500 people living in each when completed. Have these developments been completed and have their populations been included in the 1980 and later population forecasts of the ER-OL (Tables 2.1.2-3 through 2.1.2-8)?

Are there any additional housing developments planned or in existence within te miles of the site which have not been included in the FES-CP or ER-OL?

If so, have the estimated populations of these developments been included in the population forecasts in the ER-OL?

310.4 Section 2.1.3 of the ER-OL includes a discussion of the Concord Cemetery which is located within the exclusion boundary of the site. Are friends and relatives of those buried there able to visit the graves? Are funerals still held at the cemetery? If so, please identify the arrangements necessary for visitors to gain access to the cemetery.

310.5 Transient populations are discussed in section 2.1.2-3. That discussion mentions the annual and average daily attendance figures for Carowinds Theme Park. The data indicate the park is not open year round. Please identify the usual operating dates for the park.

310.6 The first paragraph of page 2.1-4 describes the "existing land use within 5 mi. of the site "as" predominately rural nonfarm with residential and recreational development bodering Lake Wylie". Please present the extent of the land use categories within 5 mi. of the site by percentage of the area.

310.7 It is estimated that the purchase of local and regional products and services averaged $2,000,000 a year during construction. (8.1.2.2.3)

Please provide information on local purchases of goods and services expected to be made by the plants during a typical year of operation. (For these purposes, local may be defined as either the host county or the host county and one or more contiguous counties.) To the extent possible, identify specific types of dollar amounts of these purchases. If it appears that there will be no significant local purchases, explain why.

310.8 Section 3.1.2-2. estimates that 346 full-time employees will be required to operate the station. The same section reports that 17% of the construction workers moved into York County, 28% were local laborers and 55% commute from beyond York County. Are the operating workers expected to have the same commuting patterns. If so, please give a more detailed estimate of domiciles for those residing outside of York County. If not, please provide new estimates for the operating workers.

During operation, are any contractual workers expected to be employed at the site along with the other employees? If so please provide their number and average annual operating payroll.

310.9 The sections covering tax revenues (8.1.2.2.1 and 11.2.2.1) speak of "payments made in lieu of taxes". Please define these payments and their calculations.

320.0 UTILITY FINANCE

320.1 Please provide production cost analyses which show system operating cost associated with both the availability and unavailability of the proposed nuclear facility. Costs resulting from this type of analysis generally represent the variable or incremental expenditures (fuel, operation, and maintenance) necessary to supply system load. If, in your analysis, other factors influence the cost of energy production, explain in detail.

a) The analyses should assume electrical energy load grows at (1) the system's latest forecasted growth rate, and (2) zero growth from the latest actual annual energy load experienced.

b) The analysis should provide results on an annual basis covering the period from initial operation of the first unit through five full years of operation of the last unit.

c) For each year (and for each growth rate scenario) the following results should be clearly stated: (1) system production costs with the proposed nuclear addition available as scheduled; (2) system production costs without the proposed nuclear addition available; (3) the capacity factor assumed for the nuclear addition; (4) the average fuel cost and variable O & M for the nuclear addition and the sources of replacement energy (by fuel type) - both expressed in mills per KwH; and (5) the proportion

of replacement energy assumed to be provided by coal, oil, gas, etc.

d. Where more than one utility shares ownership in the proposed nuclear addition, the analysis should include results for the aggregate of all participants.

e. All underlying assumptions should be explicitly identified and explained.

451.0 ACCIDENT EVALUATION - METEOROLOGY

451.1 (2.3) a. To expedite the meteorological review, provide hour-by-hour meteorological data from the onsite measurements program for the period 1975-1977 on magnetic tape using the enclosed guidance on format and tape attributes. Additional data collected since 1977 should also be included if available.

b. One year of hour-by-hour meteorological data are necessary for evaluating the environmental consequences of Class 9 accidents. From the period of record of hour-by-hour meteorological data provided on magnetic tape, select a representative continuous one-year period whe data recovery for wind direction, wind speed, atmospheric stability indicator, and precipitation is high. (All missing data should be properly identified on the magnetic tape.) Substitutions are necessary for each of the four parameters missing for any hour during the selected one-year period. Substituted data should not be incorporate onto the magnetic tape, but rather provided as a separate listing. The listing of substituted data should identify the date, time and the value of the parameter to be substituted. Also identify the source of the substituted data, and provide a brief description of the bases for selecting the substituted data. In the selection of substitutions for long periods of missing data, the diurnal and synoptic cycles should be considered.

451.2 (2.3) Explain the statement made on page 2.3-2 that "changes in the numerical values of X/Q and D/Q estimates result from the correction of a coefficient in the calculation of stable plume rise pertaining to the respective codes used for these purposes." What X/Q and D/Q values have changed? What computer codes were previously incorrect?

451.3 (2.3) Identify the sources and periods of record of the data presented in Table 2.3.0-1. Update extreme values as appropriate to reflect meteorological events occurring since 1960.

451.4 (2.3) Tables 2.3.0-2 and 2.3.0-3 are somewhat misleading. Although the tables are entitled "1975-1977 wind occurrences" for the 40m and 10m levels, respectively, implying 3 years of data, examination of the tables indicates the data are for just 2 years. Identify the period of record for Tables 2.3.0-2 and 2.3.0-3.

451.5 (2.3) Table 2.3.0-3 indicates that wind speeds less than 5.5 mph occur about 58% of the time at the 10m level, however, only about 0.7% apparently can be classified as "calm." Define the threshold for calm winds and indicate the reasonableness of such a small fraction of calm conditions.

451.6 (2.3) The first part of table 2.3.0-6 identifies annual average X/Q values at various intake vents around the site. Provide the distance and direction from the release points to each intake. For distances less than 100m, describe how appropriate σ_z values were determined.

451.7 (2.3) The second part of Table 2.3.0-6 identifies a single population X/Q value to 50 miles from the plant. Explain the significance of such a single value of X/Q to represent all directions out to a distance of 50 miles. Provide arrays of annual average X/Q and D/Q values by distance and direction out to 50 miles from the plant for use in determining population exposure from routine releases of radioactive material to the atmosphere.

451.8 (2.3) The analyses of atmospheric effects of the mechanical draft cooling towers at Catawba are apparently based solely on empirical data for "a mechanical draft cooling tower at the Duke Power Cliffside Plant (p. 5.1-4).

a. Discuss the validity of extrapolating from observations of the operation of one tower to six considering differences in number of tower units, tower shape, orientation, topography, meteorology, heat load, and evaporation.

b. Figures 5.1.4-1 and 5.1.4-2 depict frequencies of visible plumes estimated in the vicinity of the Catawba site for summer and winter. Annual mean values at meteorological parameters considered representative of the Cliffside and Catawba facilities are presented on pages 5.1-5 and 5.1-6. Discuss seasonal variations in meteorological conditions at the Cliffside and Catawba facilities, and indicate how seasonal differences were considered in developing Figures 5.1.4-1 and 5.1.4-2.

451.9 (6.1.3) An inconsistency appears to exist between the preoperational meteorological measurements program described in Section 6.1.3 and the preoperational program described in Section 6.2.4 "which will continue during the operational program" (page 6.2-2) with respect to monitoring of visibility. Describe the preoperational program designed to provide "baseline" measurements of meteorological conditions affected by operation of the mechanical draft cooling towers, and describe the operational program to assess the affects of the cooling towers. Identify the type(s) and locations of visibility monitoring equipment, data reduction procedures, and calibration and maintenance schedules.

451.10 (6.1.3) Describe the status of the onsite meteorological measurements program since December 1977.

451.11 (6.1.3) Assuming that meteorological measurements are made on or near the microwave tower identified in Figure 6.1.3-1, the measurements are made at a distance of about 700 feet from the western edge of the Unit 1 turbine building about 1100 feet from the southwestern edge of the Unit 1 reactor building, and about 1400 feet west of the mechanical draft cooling tower complex. Provide the heights of these structures and discuss possible building influences on meteorological measurements. Also discuss the possible effects of the condensate plume, humidity plume, and drift from the cooling towers on meteorological sensors and data recovery.

451.12 (6.1.3) Provide the percent recovery for each of the following parameters for the period December 17, 1975 to December 16, 1977: wind speed at the 40m and 10m levels; wind direction at the 40m and 10m level delta temperature (10m to 40m); dry bulb temperature and dew point temperature at 10m; and precipitation.

451.13 (6.1.3) The discussion of the rationale for not adjusting the straight-line gaussian dispersion model to consider spatial and temporal variations in airflow (page 6.1-14) requires further elaboration. a) Explain how measurements made at an elevation 90 feet above Lake Wylie and 70 feet above plant grade on nearly the highest elevation near the plant can be expected to identify nocturnal downslope airflow. b) The definition of stagnation (i.e., winds less than 1 mph) appears to be unnecessarily restrictive. Stagnation conditions can be accompanied by wind speeds considerably higher than 1 mph. Plume "meander" can also occur with wind speeds higher than 1 mph. Wind speeds less than 5.5 mph occur nearly 60% of the time at the 10m level of the onsite meteorological measurements program. Discuss the behavior of effluent plumes in the vicinity of the Catawba site during conditions with wind speeds less than 5.5 mph, and indicate if "recirculation" of the plume is possible during these conditions.

451.14 (6.1.3) The starting thresholds for the wind direction and wind speed sensors are 0.7 mph and 0.6 mph, respectively (page 6.1-8). However, in the discussion of minimum wind speed to be used in dispersion calculations, a value of 0.45m/sec (1.0 mph) is selected. Discuss this inconsistency.

451.15 (6.1.3) Discuss the rationale for using meteorological data from the 40m level (762' elevation) in the calculation of annual average atmospheric dispersion conditions when the height of the station vent is at an elevation of 719 feet.

451.16 (6.1.3) The onsite meteorological measurements program used to provide data for the Construction Permit Review was apparently in a different location than the present program. Identify on Figure 6.1.3-1 the location (including base elevation) of the meteorological towers used to provide data for the CP review, and compare data from the period 7/71-6/72 with more recent data from the present measurements program. Data from the earlier period showed strong secondary airflow from the northeast which is not discernable from data presented in the ER. Discuss the rationale for siting the current meteorological measurements system in its present location, and discuss the representativeness of the data collected at this location.

470.0 RADIOLOGICAL ASSESSMENT- RADIOLOGICAL IMPACT

470.1 Although Table 5.2.4-1 of the ER compares the estimated doses from the Catawba Station with the Appendix I dose design objectives, it does not compare the estimated quantities of non-tritium liquid effluents and I-131 airborne releases with the curie limits contained in the Annex to 10 CFR 50 Appendix I. If a cost-benefit analysis is not going to be performed, then the estimated quantities of the preceding effluents should be compared with the curie limits in the Annex to 10 CFR 50 Appendix I.

470.2 Section 5.2.4.4.1 of the ER disscusses population doses from ingestion of drinking water. However, the population data used in §5.2.4.4.1 is not consistent with the population data in ER Table 2.1.3-5. For example, Table 2.1.3-5 lists 330,000 persons using the Catawba River as a drinking water site, whereas §5.2.4.4.1 lists only 210,000 presons for all populations served by Lake Wylie and the Catawba River. Resolve this apparent discrepanc and provide the population size ingesting water from the major sources of water.

470.3 On p. 5.2-10 of the ER, it is stated that the GASPAR and LADTAP computer codes were used to estimate doses from exposure to radioactive effluents. Provide a listing of input parameters that were used in the GASPAR and LADTAP computer runs.

470.4 ER Table 2.1.3-5 lists the locations of surface water users in terms of river distance miles. Please provide the location of the plant discharge point in river distance miles.

Request for Additional Information
Final Safety Analysis Report
Catawba, Units 1 and 2
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100.0 Miscellaneous

100.1 (1.2.2.3) Per Regulatory Guide 1.70, provide a brief discussion of the NSSS safety considerations.

100.2 (1.4) Provide a list of outside service organizations providing audits of the quality assurance program.

100.3 (1.8) Per Regulatory Guide 1.70, Table 1.8-1 should indicate proposed exceptions to the regulatory position.

210.0 MECHANICAL ENGINEERING

210.1 (3.2.1) Reference or provide a discussion of your compliance with Regulatory Guide 1.29.

210.2 (3.9.5.4) Indicate the extent to which the design and construction of the core support structures is in accordance with Subsection NG of the ASME code. Indicate the extent to which the design of other reactor internals will be consistent with Article NG-3000.

210.3 (3.9.6.3) Per the requirements of Regulatory Guide 1.70, Rev. 3, provide the information requested in Section 3.9.6.3 of this guide.

220.0 STRUCTURAL ENGINEERING

220.1 (3.7.1.1) Provide or reference a discussion of your compliance with Regulatory Guide 1.60.

220.2 (3.8.1.3) Provide or reference a discussion of your compliance with Article CC-3000 of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments".

220.3 (3.8.1.4) Provide or reference a discussion of your compliance with Article CC-3000 of the ASME Code Section III, Division 2.

220.4 (3.8.1.5) Provide or reference a discussion of your compliance with Article CC-3000 of the ASME Boiler and Pressure Vessel Code, Section III, Division 2.

220.5 (3.8.1.6) Provide or reference a discussion of your compliance with Article CC-2000, CC-4000 and CC-5000 of the ASME Code, Section III, Division 2.

220.6 (3.8.1.7) Provide or reference a discussion of your compliance with Articles CC-6000 and CC-9000 of the ASME Code, Section III, Division 2, and Regulatory Guides 1.18, 1.35 and 1.90.

220.7 (3.8.2.3) Provide or reference a discussion of your compliance with Regulatory Guide 1.57.

220.8 (3.8.2.5) Provide or reference a discussion of your compliance with Regulatory Guide 1.57.

220.9 (3.8.3.3) Provide or reference a discussion of your compliance with ACI-349, AISC Specification for "Design, Fabrication, and Erection of Structural Steel for Buildings", Regulatory Guide 1.57, Article CC-3000 of the ASME Code, Section III, Division 2 and Subsection NF of the ASME Code, Section III, Division 1.

220.10 (3.8.3.4) Provide or reference a discussion of the extent to which the design and analysis procedures comply with ACI 349, the AISC Specification for Concrete and Steel Structures, Subsection NF of the ASME Code Section III, Division 1, Regulatory Guide 1.57 and Article CC-3000 of the ASME Code, Section III, Division 2.

220.11 (3.8.3.6) Provide or reference a discussion of your compliance with ACI-349 with AISC Specifications for Steel, Subsection NF of the ASME Code, Section III, Division 1, ANSI N45.2.5, and Regulatory Guide 1.55.

220.12 (3.8.4.4 & 3.8.4.5) Per the requirements of Regulatory Guide 1.70, address your extent of compliance with ACI-349.

230.0 GEOSCIENCES - SEISMOLOGY

230.1 (2.5.2.4) Per Regulatory Guide 1.70, determine ground motion at the site assuming seismic energy effects are constant over the region under the specified assumptions. The conditions describing the occurrence of the potential earthquake that would produce the largest vibratory ground motion at the site should be defined. For the largest earthquake possible, provide information relative to the nature of the faulting.

240.0 HYDROLOGIC AND GEOTECHNICAL ENGINEERING - HYDROLOGIC ENGINEERING

240.1
(2.4.2.3) Per Regulatory Guide 1.70, sufficient details of the site drainage system should be provided (a) to allow an independent review of rainfall and runoff effects on safety-related facilities, (b) to judge the adequacy of the design criteria, and (c) to allow an independent review of the potential for blockage of site drainage.

240.2
(2.4.3) Per Regulatory Guide 1.70, indicate whether, and if so how, the guidance given in Appendix A of Regulatory Guide 1.59 has been followed.

240.3
(2.4.3.4) Per Regulatory Guide 1.70, provide the estimated PMF discharge hydrograph at the site and if available a similar hydrograph without upstream reservoir effects.

240.4
(2.4.4.3) Per Regulatory Guide 1.70, discuss the verification and reliability of the water elevation estimate for the most critical upstream dam failure or failures.

240.5
(2.4.5.2) Per Regulatory Guide 1.70, identify the parameters used in the analysis of water heights associated with hurricanes and a detailed description of the methods and models used, and the results of the computation of the probable maximum surge hydrograph (graphical presentation).

240.6
(2.4.11.1) Per Regulatory Guide 1.70, for non-safety-related water supplies demonstrate that the water supply will be adequate during a 100-year drought.

240.7
(2.4.11.3) Per Regulatory Guide 1.70, provide a discussion of historical low water flow and the probabilities associated with historical low water levels and flows.

240.8
(2.4.11.5) Per Regulatory Guide 1.70, discuss the capability of cooling water pumps to supply sufficient water during periods of low water resulting from the 100-year drought.

240.9 (2.4.13.2) Per Regulatory Guide 1.70, indicate the range of values and method of determination for effective and total porosity (specific yield) for each relevant geological formation below the site.

240.10 (3.4.1.1) Provide or reference a discussion of your compliance with Regulatory Guides 1.59 and 1.102.

252.0 MATERIALS ENGINEERING - MATERIALS APPLICATION

252.1 (4.5.1) Provide or reference a discussion of your compliance with Regulatory Guides 1.31 and 1.37.

252.2 (5.2.1.2) Provide or reference a discussion of your compliance with Regulatory Guide 1.84.

252.3 (5.2.3.3) Provide or reference a discussion of your compliance with Regulatory Guide 1.43.

252.4 (5.3.1.4) Per the requirements of Regulatory Guide 1.70, address Regulatory Guides 1.31 and 1.34.

260.0 QUALITY ASSURANCE

260.1 In Chapter 17.0 of the FSAR, Duke Power Company has referenced Duke Power Company Topical Report, Duke-1A entitled, "Quality Assurance Program." This report was recently revised (Amendment 5 letter to Haass from Wells dated March 30, 1981) and is currently under review by the Quality Assurance Branch. It is not clear in Section 17.0 of the FSAR which revision of the Duke Topical Report Duke Power Company intends to apply to the Catawba Nuclear Station. Therefore, clearly describe, by revision number and date, which issue of the Duke topical report will be applicable to the Catawba Nuclear Station.

260.2 Identify or reference in Chapter 17.0 of the Catawba FSAR, those safety-related structures, systems, and components under the control of the QA program.

280.0 CHEMICAL ENGINEERING

280.1 (9.5.1.1.3) Provide or reference a discussion that addresses how the fire protection system for any particular plant safety system or area minimizes radioactive releases to the environment in the event of a fire.

280.2 (9.5.1.1.5) Provide or reference a discussion that addresses how the consequences of inadvertant operation of, or a crack in, a moderate-energy line in the fir suppression system meets the guidelines specified for moderate-energy systems outside containment.

280.3 (9.5.1.3.4) Provide or reference a discussion that addresses a failure mode and effects analysis that demonstrates that operation of the fire protection system in areas containing engineered safety features would not produce an unsafe condition or preclude safe shutdown of the plant. Also, the effects of fire fighting activities and fire suppression agents on safety systems should be discussed. Also, an evaluation of the effects of failure of any portion of the fire protection system not designed to Seismic Category I requirements should be provided with regard to the possibility of damaging other Seismic Category I equipment.

280.4 (9.5.1.5.4) Provide or reference a discussion that addresses the Emergency Response Plan with respect to fire protection. The need for good organization, training, and equipping of fire brigades at nuclear power plants requires that effective measures be implemented to ensure proper discharge of these functions. The guidance in Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," should be followed as applicable. Successful firefighting requires both proper equipment and personnel capable of using it efficiently.

311.0 SITING ANALYSIS

311.1 (2.1.1.1) Per Regulatory Guide 1.70, specification is to be made by longitude and latitude of each reactor at the site.

311.2 (2.1.1.3) Per Regulatory Guide 1.70, describe how access to the plant site is controlled for radiation protection purposes.

311.3 (2.1.3.5) Per Regulatory Guide 1.70, indicate the extent to which transient population has been considered in establishing the population center.

311.4 (2.2.3.1) Per Regulatory Guide 1.70, design basis events coming under the category of fires should be addressed or justification should be provided for lack of consideration of fires.

311.5 (2.2.3.1.2) Per Regulatory Guide 1.70, the evaluation of the effects of a flammable vapor cloud on the plant should be made relative to "worst-case" meteorological conditions.

410.0 AUXILIARY SYSTEMS

410.1 (3.4.1.1) Identify those safety-related systems or components, if any, that are capable of normal function while completely or partially flooded.

410.2 (10.4.7) Provide or reference a discussion that addressed the following items for the condensate and feedwater systems:

a. The design codes to be applied, criteria for isolation from the steam generator or reactor coolant system, supply of condensate available for emergency purposes, inservice inspection requirements, and environmental design requirements.

b. The evaluation of the condensate and feedwater systems should include an analysis of component failure, effects of equipment malfunction on the reactor coolant system, and an analysis of isolation provisions to preclude release of radioactivity to the environment in the event of a pipe leak or break.

430.0 POWER SYSTEMS

430.1 (9.5.8.2) Provide or reference a discussion that addresses the results of a failure mode and effects analysis of the Diesel Generator Combustion Air Intake and Exhaust System.

430.2 (10.2.1) Provide or reference a discussion that addresses the turbine generator performance requirements under normal, upset, emergency, and faulted conditions and the design codes that apply to these units.

440.0 REACTOR SYSTEMS

440.1 (5.4.7.1) Provide or reference a discussion of the design bases of the RHR system with respect to General Design Criterion 5.

440.2 (6.2.4.2) Per the requirements of Regulatory Guide 1.70, provide the following information in Table 6.2.4-1:

a. General design criteria or regulatory guide recommendations that have been met or other defined bases for acceptability;
b. Engineered-safety-feature system (yes or no);
c. Through-line leakage classification (dual containments);
d. Type C leakage test (yes or no);
e. Length of pipe from containment to outermost isolation valve (or the maximum length that will not be exceeded);
f. Power source.

440.3 (15.0) Identify all required operator actions for the events analyzed.

440.4 (15.0) Discuss and evaluate the effect of operator errors for the events analyzed.

440.5 (15.7.4) As per Regulatory Guide 1.70, provide the peak linear power density for the assembly discharged.

440.6 (15.7.4) As per Regulatory Guide 1.70, provide the maximum centerline operating fuel temperature for the fuel assembly discharged.

451.0 ACCIDENT EVALUATION - METEOROLOGY

451.1 (2.3.1.2) Per Regulatory Guide 1.70, provide the probable maximum annual frequency of occurrance and time of duration of freezing rain (ice storms). An estimate of the weight of the 100-year return period snowpack and the weight of the 48-hour Probable Maximum Winter Precipitation for the site vicinity should be provided. Further, provide the meteorological data used for evaluating the performance of the ultimate heat sink.

451.2 (2.3.2.1) Provide column headings for data presented in Table 2.3.2-1 so data can be clearly read and interpreted. Also, provide data for all meteorological parameters based on "shorter-term onsite data". Further, for long term monthly and annual summaries, the following categories of data should be referenced or provided in this section:

a. monthly and annual wind roses
b. absolute humidity
c. precipitation - the number of hours with precipitation, rainfall rate distribution and monthly precipitation wind roses with precipitation rate classes.
d. fog - the extremes of duration
e. atmospheric stability
f. mixing height data - including frequency and duration of inversion conditions
g. hourly averages of wind speed and direction
h. hourly averages of atmospheric stability as defined by vertical temperature gradient or other well-documented parameters that have been substantiated by diffusion data.

451.3 (2.3.3) Per Regulatory Guide 1.70, discuss additional sources of meteorological data for consideration in the description of airflow trajectories from the site to a distance of 80KM. Also joint frequency distribution of wind speed and direction by atmospheric stability class are to be provided.

451.4 (2.4.5.1) Per Regulatory Guide 1.70, provide a detailed explanation of how probable maximum meteorological winds are determined.

460.0 EFFLUENT TREATMENT SYSTEMS

460.1 (12.2.1) As per Regulatory Guide 1.70, describe those sources that are contained in equipment of the radioactive waste management systems.

471.0 RADIOLOGICAL ASSESSMENT

471.1 (12.1.1) Indicate whether the guidance provided by Regulatory Guides 1.8, 8.8, and 8.10 has been followed.

471.2 (12.1.2) Indicate whether the guidance provided by Section C.1 of Regulatory Guide 8.8 has been followed.

471.3 (12.1.3) Section 12.1.3 of the FSAR references Section C.4 of Regulatory Guides 8.8 and 8.10 while Regulatory Guide 1.70 references Section C.1. Provide justification for not utilizing Seetion C.1.

471.4 (12.3.1) Figures 12.3.1-1 through 12.3.1-6 are not legible (the drawing print reduction is too small). Provide legible figures. Those figures should contain the following additional information:

a. shield wall thickness
b. traffic patterns
c. location of airborne radioactivity and area radiation monitors
d. personnel and equipment decontamination areas
e. location of the health physics facilities
f. location of the counting room.

471.5 (12.3.1) As per Regulatory Guide 1.70, provide the design basis radiation level in the counting room during normal operation and anticipated operational occurrences.

471.6 (12.3.2.2) Indicate whether the guidance of Regulatory Guide 8.8 has been followed.

471.7 (12.3.4.1) As per Regulatory Guide 1.70, describe the radiation instrumentation that will be used to meet the criticality accident monitoring requirements of 70.24 of 10 CFR part 70 for the storage area for new fuel.

471.8 (12.3.4.1) Indicate whether the guidance of Regulatory Guides 1.21, 8.12, 1.97 and ANSI N13.1-1969 has been followed.

471.9 (12.4) Indicate whether the guidance of Regulatory Guide 8.19 has been followed.

471.10 (12.5.1) Indicate whether the guidance of Regulatory Guides 1.8, 8.2, 8.8, and 8.10 has been followed.

471.11 (12.5.2) Indicate whether the guidance of Regulatory Guides 8.4, 8.8, 8.15, and 1.97 has been followed.

471.12 (12.5.3) Indicate whether the guidance of Regulatory Guides 8.2, 8.9, 8.13, 1.8, 1.16, 1.33 and 1.39 has been followed.

480.0 CONTAINMENT SYSTEMS

480.1 (6.2.1.1.2) Per the requirements of Regulatory Guide 1.70, provide or reference a discussion of the functional capability and frequency of operations of the systems provided to maintain the containment and subcompartment atmospheres within prescribed pressure, temperature, and humidity limits during normal plant operation.

480.2 (6.2.2.2) Per the requirements of Regulatory Guide 1.70, provide or reference a discussion of the following:

a. For the containment heat removal system describe design provisions that facilitate periodic inspection and operability testing of the systems and system components.

b. Identify the codes, standards and guides applied in the design of the containment heat removal systems and system components.

c. Specify the times following postulated accidents that the containment heat removal systems are assumed to be fully operational. Discuss the delay times following receipt of the system actuation signals that are inherent in bringing the systems into service.

d. Describe the qualification tests that have been or will be performed on system components, such as spray nozzles, fan cooler heat exchanger recirculation heat exchangers, pump and fan motors, valves, valve operators, and instrumentation. Discuss the test results. Demonstrat that the environmental test conditions (temperature, pressure, humidit radiation, water pH) are representative of postaccident conditions tha the equipment would be expected to be exposed to. Graphically show the environmental test conditions as a function of time or refer to the section in the SAR where this information can be found.

e. With respect to the fan systems, provide the following additional information:

1. Identify the ductwork and equipment housings that must remain intact following a loss-of-coolant accident;

2. Discuss the design provisions (e.g., pressure relief devices, conservative structural design) that ensure that the ductwork and equipment housings will remain intact; and

3. Provide plan and elevation drawings of the containment showing the routing of airflow guidance ductwork.

f. Describe the design features of the recirculation intake structures (sumps). Provide plan and elevation drawings of the structures; show the level of water in the containment following a loss-of-coolant accident in relation to the structures. Compare the design of the recirculation intake structures to the positions in Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems."

g. Specify the mesh size of each stage of screening and the maximum particle size that could be drawn into the recirculation piping. Of the systems that receive or may receive water from the recirculation intake structures under postaccident conditions, identify the system component that places the limiting requirement on the maximum particle size of debris that may be allowed to pass through the intake structure screening and specify the limiting particle size that the component can circulate without impairing system performance. Describe how the screening is attached to the intake structures to preclude the possibilit of debris bypassing the screening.

h. Discuss the potential for the intake structure screening to become clogged with debris; e.g., insulation, in the light of the effective flow area of the screening and approach velocity of the water. Identi, and discuss the kinds of debris that might be developed following a loss-of-coolant accident. Consider the following potential sources of debris:

1. Piping and equipment insulation,
2. Sand plug materials,
3. All structures displaced by accident pressure to provide vent area,
4. Loose insulation in the containment,
5. Debris generated by failure of non-safety-related equipment.

Describe the precautions made to minimize the potential for debris clogging the screens.

i. Discuss the types of insulation used inside the containment and identify where and in what quantities each type is used. List the materials of construction used for the identified insulation and describe the behavior of the insulation during and after a loss-of-coolant accident. Describe the tests performed or reference test reports available to the Commission that determined the behavior of the insulation under simulated LOCA conditions. Describe the methods of attaching the insulation to piping and components.

480.3 (6.2.3.2) The review of this section cannot be completed until after the information identified as "later" in Table 6.2.3-3 is submitted. Either provide this information or provide a schedule for submittal of the information.

630.0 LICENSEE QUALIFICATIONS

630.1 (13.1.1.2) As per Regulatory Guide 1.70, summarize the degree to which the activities described in Sections 13.1.1.1.1 and 13.1.1.1.2 of Regulatory Guide 1.70 have been accomplished. Provide a schedule for completing these activities

630.2 (13.1.2.2) As per Regulatory Guide 1.70, explain the delegation of authority to operating and shift supervisors concerning the authority to issue standing or special orders.

630.3 (13.2.2.1.1) As per Regulatory Guide 1.70, explain the provisions for conducting fire drills during construction and for indoctrination of construction personnel.

630.4 (13.2.2.2.2) As per Regulatory Guide 1.70, provide the means employed to certify that cold license applicants have had extensive actual operating experience.

640.0 PROCEDURES AND TEST REVIEW

640.1 (13.5.1.3) Provide Figure 13.5-1

640.2 (14.2.5) Describe the methods that will be used to ensure retesting following modifications or maintenance during the testing program.

640.3 (14.2.11) Provide the sequential test schedule for testing individual plant structures, systems and components.

640.4 (14.2.11) Provide schedules which show the time allowed for conducting preoperational tests, fuel loading, low power tests, and power ascension tests. Your response should address the compatibility of these schedules with the schedule for hiring and training of test personnel.

810.0 EMERGENCY PLANNING

810.1 (13.3) As per Regulatory Guide 1.70, provide plots showing ground-level doses for stationary individuals. For both whole body and thyroid resulting from the most serious DBA.

810.2 (13.3) Discuss the extent of compliance of your emergency planning with NUREG 0696

810.3 (13.3) As per Regulatory Guide 1.70, provide a map showing all roads available for vehicular evacuation of the exclusion area and environs extending at least 10 miles from the plant.

810.4 (13.3) As per Regulatory Guide 1.70, provide a map showing demographic data in 1-mile increments from the plant to the outer boundary of the proposed low population zone.

Request For Additional Information
Catawba Units 1 and 2
Docket Nos. 50-413/414

In addition to the TMI-related requirements discussed in Section 1.9 of your FSAR, there are other areas in which requirements have been added or modified, or in which staff concerns have been raised in the review of other pending OL applications. A number of these areas are discussed below. In order to expedite the review process for your application, we request that you evaluate these areas and, where appropriate, upgrade your FSAR to include how these requirements are met or how these staff concerns are resolved for your plant. We further request that you submit these changes to the FSAR, in amendment form, within two months from the docketing date.

(1) Environmental Qualification of Safety Related Electrical Equipment- Commission Memorandum and Order of May 23, 1980 defines the current staff requirements for qualification of this equipment. Additional guidance on this matter was provided in a subsequent NRR Order, dated November 26, 1980 (concerning record requirements), Supplements 2 and 3, dated September 30, 1980 and October 24, 1980, respectively, to IE Bulletin No. 79-01B, and a generic letter to all holders of CPs and OLs, dated October 1, 1980.

(2) Seismic Qualification - A staff request for additional information in this review area has been sent to a number of pending OL applicants. A copy of that request is provided as Enclosure 5.

(3) Emergency Preparedness - Guidance on the preparation of emergency plans is presented in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants". The requirements for the emergency response facilities are included in NUREG-0696, "Functional Criteria for Emergency Response Facilities." Further guidance on emergency preparedness is provided in the revised Appendix E to 10 CFR Part 50.

(4) Fire Protection - The current requirements for the fire protection programs are defined in the new Appendix R to 10 CFR Part 50. As further guidance, a copy of a recent staff request for additional information on Catawba is provided as Enclosure 6.

(5) Masonry Walls - The staff concerns regarding this issue and a request for information to assist in its resolution were provided in a generic letter, dated April 21, 1980 to all CP and OL applicants.

(6) Fracture Prevention of Containment Pressure Boundary (GDC 51) - Enclosure 7 provides clarification on how the staff determines compliance with GDC 51.

(7) Initial Test Program Descriptions (Chapter 14) - Staff review of near term OL applications has revealed a number of concerns which are common to pending applications. The nature of these concerns are typically expressed in the questions the staff has raised in its review of the Summer and the San Onofre 2 & 3 applications.

(8) Special Low Power Test Program (Task Action Plan Item I.G.1) - The staff has recently established guidance on this matter for transmittal to all pending and prospective OL applicants. A copy of that guidance is provided as Enclosure 8.

(9) Preservice and Inservice Inspections - Staff guidance in this review area has been sent to a number of pending OL applicants. A copy of that guidance is provided as Enclosure 9.

(10) Procedures and Training for Station Blackout - In response to a recommendation in a recent decision by the Atomic Safety and Licensing Appeal Board (ALAB-603), to ensure that station blackout events can be accomodated, the staff is requesting licensees and OL Applicants to implement emergency procedures and a training program for station blackout events. A copy of that request is provided as Enclosure 10.

(11) Preservice Inspection and Testing of Snubbers - The staff has recently established requirements to ensure snubber operability which have been transmitted to pending OL applicants. A copy of those requirements is provided as Enclosure 11.

(12) Effects of Containment Coatings and Sump Debris on ECCS and Containment Spray Operation - A copy of the NRC staff concerns on this issue, including a request for additional information which has been sent to a number of OL applicants, is provided as Enclosure 12.

(13) Instrumentation for Detection of Inadequate Core Cooling (TMI Action Item II.F.2 in NUREG-0737) - Discussion of this item should address how core thermocouple readouts are provided in the control room including location and rate of printout (see Part (4) of attachment 1 to Item II.F.2).

(14) Safety - Related Structures, Systems and Components (Q-list) Controlled by the QA Program - Staff requests for additional information regarding this issue have been sent to a number of OL applicants. A recent request regarding the Diablo Canyon is provided as Enclosure 13.

Equipment Qualification Branch
Seismic Qualification Review Team
Request for Additional Information

1. In accordance with the requirements of GDC 2 and 4 all safety-related equipment is required to be designed to withstand the effects of earthquakes and dynamic loads from normal operation, maintenance, testing and postulated accident conditions. GDC 2 further requires that such equipment be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of earthquake loads.

   The criteria to be used by the staff to determine the acceptability of your equipment qualification program for seismic and dynamic loads are IEEE Std. 344-1975 as supplemented by Regulatory Guides 1.100 and 1.92, and Standard Review Plan Sections 3.9.2 and 3.10. State the extent to which the equipment in your plant meets these requirements and the above requirements to combine seismic and dynamic loads. For equipment that does not meet these requirements provide justification for the use of other criteria.

2. Provide a list of all safety-related systems together with a list of all safety-related equipment and support structures associated with each system. The equipment lists should indicate whether the equipment is NSSS supplied or BOP supplied. These lists should include all safety-related mechanical components, electrical, instrumentation, and control equipment, including valve actuators and other appurtenances of active pumps and valves.

3. For each safety-related equipment item, the following information should be provided:

   (1) Method of qualification used:

      a) Analysis or test (indicate the company that prepared the report, the reference report number and date of the publication).

      b) If by test, describe whether it was a single or multi-frequency test and whether input was single axis or multi-axis.

      c) If by analysis, describe whether static or dynamic, single or multiple-axis analysis was used.

      d) Provide natural frequency (or frequencies) of equipment.

   (2) Indicate whether the equipment has met the qualification requirements.

   (3) Indicate whether the equipment is required for:

      a) hot stand-by

      b) cold shutdown

      c) both

      d) neither

(4) Location of equipment, i.e., building, elevation.

(5) Availability for inspection (Is the equipment already installed at the plant site?)

(6) A compilation of the required response spectra (or time history) and corresponding damping for each seismic and dynamic load specified for the equipment together with all other loads considered in the qualification and the method of combining all loads.

4. Identify all equipment that may be effected by vibration fatigue cycle effects and describe the methods and criteria used to qualify this equipment for such loading conditions

5. Describe the results of any in plant tests, such as in situ impedance tests, and any plans for operational tests which will be used to confirm the qualification of any item of equipment.

6. To confirm the extent to which the safety-related equipment meets the requirements of General Design Criterion 2 and 4, the Seismic Qualification Review Team (SQRT) will conduct a plant site review. For selected equipment, SQRT will review the combined required response spectra (RRS) or the combined dynamic response, examine the equipment configuration and mounting, and then determine whether the test or analysis which has been conducted demonstrates compliance with the RRS if the equipment was qualified by test, or the acceptable analytical criteria if qualified by analysis.

The staff requires that a "Qualification Summary of Equipment" as shown on the attached pages be prepared for each selected piece of equipment and submitted to the staff two weeks prior to the plant site visit. The applicant should make available at the plant site for SQRT review all the pertinent documents and reports of the qualification for the selected equipment. After the visit, the applicant should be prepared to submit certain selected documents and reports for further staff review.

Qualification Summary of Equipment

I. Plant Name: ____________________ Type:

1. Utility: ____________________ PWR ____________

2. NSSS: ____________ 3. A/E: ____________ BWR ____________

II. Component Name ____

1. Scope: [ ] NSSS [ ] BOP

2. Model Number: ____________________ Quantity: ____________

3. Vendor: ____

4. If the component is a cabinet or panel, name and model No. of the devices included: ____________________

____

5. Physical Description a. Appearance ____________________

b. Dimensions ____________________

c. Weight ____________________

6. Location: Building: ____________________

Elevation: ____________________

7. Field Mounting Conditions [ ] Bolt (No._____, Size_____)
[ ] Weld (Length_____)
[ ] ____________

8. a. System in which located: ____________________

b. Functional Description: ____________________

c. Is the equipment required for [ ] Hot Standby [ ] Cold Shutdown
[ ] Both [ ] Neither

9. Pertinent Reference Design Specifications: ____________________

____

III. Is Equipment Available for Inspection in the Plant: [ ] Yes [ ] No

IV. Equipment Qualification Method:

[ ] Test [ ] Analysis [ ] Combination of Test and Analysis

Qualification Report*: ______________________________

(No., Title and Date) ______________________________

Company that Prepared Report: ______________________________

Company that Reviewed Report: ______________________________

V. Vibration Input:

1. Loads considered: a. [ ] Seismic only

   b. [ ] Hydrodynamic only

   c. [ ] Combination of (a) and (b)

2. Method of Combining RRS: [ ] Absolute Sum [ ] SRSS [ ] ______________ (other, specify)

3. Required Response Spectra (attach the graphs): ______________________________

4. Damping Corresponding to RRS: OBE ______________ SSE ______________

5. Required Acceleration in Each Direction: [ ] ZPA [ ] Other ______________ (specify)

   OBE S/S = ______________ F/B = ______________ V = ______________
   SSE S/S = ______________ F/B = ______________ V = ______________

6. Were fatigue effects or other vibration loads considered?

   [ ] Yes [ ] No

   If yes, describe loads considered and how they were treated in overall qualification program: ______________________________

   ____

   ____

*NOTE: If more than one report complete items IV thru VII for each report.

VI. If Qualification by Test, then Complete*:

1. [ ] Single Frequency   [ ] Multi-Frequency: [ ] random
   [ ] sine beat
   [ ] ________________
2. [ ] Single Axis   [ ] Multi-Axis
3. No. of Qualification Tests: OBE __________ SSE _________ Other ______________ (specify)
4. Frequency Range: ______________________________
5. Natural Frequencies in Each Direction (Side/Side, Front/Back, Vertical):

   S/S = ________________ F/B = ______________ V = ______________
6. Method of Determining Natural Frequencies

   [ ] Lab Test   [ ] In-Situ Test   [ ] Analysis
7. TRS enveloping RRS using Multi-Frequency Test [ ] Yes (Attach TRS & RRS grap
   [ ] No
8. Input g-level Test: OBE S/S = ________ F/B = ________ V = ________

   SSE S/S = ________ F/B = ________ V = ________
9. Laboratory Mounting:

   1. [ ] Bolt (No. _____, Size _____) [ ] Weld (Length _____) [ ] _____
10. Functional operability verified: [ ] Yes [ ] No [ ] Not Applicable
11. Test Results including modifications made: ____________________________

    ____
12. Other test performed (such as aging or fragility test, including results):

    ____

    ____

*Note: If qualification by a combination of test and analysis also complete Item VII.

VII. If Qualification by Analysis, then complete:

1. Method of Analysis:

   [ ] Static Analysis [ ] Equivalent Static Analysis

   [ ] Dynamic Analysis: [ ] Time-History [ ] Response Spectrum

2. Natural Frequencies in Each Direction (Side/Side, Front/Back, Vertical):

   S/S = ________________ F/B = ________________ V = ________________

3. Model Type: [ ] 3D [ ] 2D [ ] 1D

   [ ] Finite Element [ ] Beam [ ] Closed Form Solution

4. [ ] Computer Codes: ______________________________

   Frequency Range and No. of modes considered: ______________________________

   [ ] Hand Calculations

5. Method of Combining Dynamic Responses: [ ] Absolute Sum [ ] SRSS
   [ ] Other: ________________ (specify)

6. Damping: OBE________ SSE______ Basis for the damping used: ____________

7. Support Considerations in the model: ______________________________

8. Critical Structural Elements:

A. Identification	Location	Governing Load or Response Combination	Seismic Stress	Total Stress	Stress Allowable

B. Max. Critical Deflection	Location	Maximum Allowable Deflection to Assure Functional Operability

ENCLOSURE 6

FIRE PROTECTION REVIEW
CATAWBA 1 & 2
DOCKET NOS. 50-413 AND 50-414

In accordance with section 9.5.1, Branch Technical Position ASB 9.5-1, position C.4.a.(1) of NRC Standard Review Plan and section III.G of new Appendix R to 10 CFR Part 50, it is the staff's position that cabling for redundant safe shutdown systems should be separated by walls having a three-hour fire rating or equivalent protection (see section III.G.2 of Appendix R). That is, cabling required for or associated with the primary method of shutdown, should be physically separated by the equivalent of a three-hour rated fire barrier from cabling required for or associated with the redundant or alternate method of shutdown. To assure that redundant shutdown cable systems and all other cable systems that are associated with the shutdown cable systems are separated from each other so that both are not subject to damage from a single fire hazard, we require the following information for each system needed to bring the plant to a safe shutdown.

1. Provide a table that lists all equipment including instrumentation and vital support system equipment required to achieve and maintain hot and/or cold shutdown. For each equipment listed:

   a. Differentiate between equipment required to achieve and maintain hot shutdown and equipment required to achieve and maintain cold shutdown.

   b. Define each equipment's location by fire area.

   c. Define each equipment's redundant counterpart.

d. Identify each equipment's essential cabling (instrumentation, control, and power). For each cable identified: (1) Describe the cable routing (by fire area) from source to termination, and (2) Identify each fire area location where the cables are separated by less than a wall having a three-hour fire rating from cables for any redundant shutdown system, and

e. List any problem areas identified by item 1.d.(2) above that will be corrected in accordance with Section III.G.3 of Appendix R (i.e., alternate or dedicated shutdown capability).

2. Provide a table that lists Class 1E and Non-Class 1E cables that are associated with the essential safe shutdown systems identified in item 1 above. For each cable listed: (* See note on Page 3).

a. Define the cables' association to the safe shutdown system (common power source, common raceway, separation less than IEEE Standard-384 guidelines, cables for equipment whose spurious operation will adversely affect shutdown systems, etc.),

b. Describe each associated cable routing (by fire area) from source to termination, and

c. Identify each location where the associated cables are separated by less than a wall having a three-hour fire rating from cables required for or associated with any redundant shutdown system.

3. Provide one of the following for each of the circuits identified in item 2.c above:

   (a) The results of an analysis that demonstrates that failure caused by open, ground, or hot short of cables will not affect it's associated shutdown system. * Note *

   (b) Identify each circuit requiring a solution in accordance with section III.G.3 of Appendix R, or

   (c) Identify each circuit meeting or that will be modified to meet the requirements of section III.G.2 of Appendix R (i.e., three-hour wall, 20 feet of clear space with automatic fire suppression, or one-hour barrier with automatic fire suppression).

4. To assure compliance with GDC 19, we require the following information be provided for the control room. If credit is to be taken for an alternate or dedicated shutdown method for other fire areas (as identified by item 1.e or 3.b above) in accordance with section III.G.3 of new Appendix R to 10 CFR Part 50, the following information will also be required for each of these plant areas.

   a. A table that lists all equipment including instrumentation and vital support system equipment that are required by the primary method of achieving and maintaining hot and/or cold shutdown.

* NOTE

Option 3a is considered to be one method of meeting the requirements of Section II.G.3 Appendix R. If option 3a is selected the information requested in items 2a and 2c above should be provided in general terms and the information requested by 2b need not be provided.

b. A table that lists all equipment including instrumentation and vital support system equipment that are required by the alternate, dedicated, or remote method of achieving and maintaining hot and/or cold shutdown.

c. Identify each alternate shutdown equipment listed in item 4.b above with essential cables (instrumentation, control, and power) that are located in the fire area containing the primary shutdown equipment. For each equipment listed provide one of the following:

(1) Detailed electrical schematic drawings that show the essential cables that are duplicated elsewhere and are electrically isolated from the subject fire areas, or

(2) The results of an analysis that demonstrates that failure (open, ground, or hot short) of each cable identified will not affect the capability to achieve and maintain hot or cold shutdown.

d. Provide a table that lists Class 1E and Non-Class 1E cables that are associated with the alternate, dedicated, or remote method of shutdown. For each item listed, identify each associated cable located in the fire area containing the primary shutdown equipment. For each cable so identified provide the results of an analysis that demonstrates that failure (open, ground, or hot short) of the associated cable will not adversely affect the alternate, dedicated, or remote method of shutdown.

5. The residual heat removal system is generally a low pressure system that interfaces with the high pressure primary coolant system. To preclude a LOCA through this interface, we require compliance with the recommendations of Branch Technical Position RSB 5-1. Thus, this interface most likely consists of two redundant and independent motor operated valves with diverse interlocks in accordance with Branch Technical Position ICSB 3. These two motor operated valves and their associated cable may be subject to a single fire hazard. It is our concern that this single fire could cause the two valves to open resulting in a fire-initiated LOCA through the subject high-low pressure system interface. To assure that this interface and other high-low pressure interfaces are adequately protected from the effects of a single fire, we require the following information:

   a. Identify each high-low pressure interface that uses redundant electrically controlled devices (such as two series motor operated valves) to isolate or preclude rupture of any primary coolant boundary.

   b. Identify each device's essential cabling (power and control) and describe the cable routing (by fire area) from source to termination.

   c. Identify each location where the identified cables are separated by less than a wall having a three-hour fire rating from cables for the redundant device.

d. For the areas identified in item 5.c above (if any), provide the bases and justification as to the acceptability of the existing design or any proposed modifications.

Enclosure 7

Fracture Prevention of Containment Pressure Boundary (GDC-51)

GDC-51 requires that under operating, maintenance, testing and postulated accident conditions, (1) the Ferritic materials of the containment pressure boundary behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The Ferritic materials of the containment pressure boundary which are assessed by the staff are those of components such as freestanding containment vessel, equipment hatches, personnel airlocks, primary containment drywell head, heads containment penetration sleeves, proccess pipes, end closure caps and flued heads and penetrating piping systems downstream of penetration process pipes extending to and including the system isolation valves.

The acceptability of these materials within the context of GDC-51 is determined in accordance with the fracture toughness criteria identified for Class 2 materials by the Summer 1977 Addenda to ASME Code Section III.

SUBJECT: TMI-2 TASK ACTION PLAN ITEM I.G.1 - SPECIAL LOW POWER TESTING

NUREG-0694 "TMI Related Requirements for New Operating Licenses", Item I.G.1, requires applicants to perform "a special low power testing program approved by NRC to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training". To comply with this requirement new PWR applicants have committed to a series of natural circulation tests. To date such tests have been performed at the Sequoyah 1, North Anna 2, and Salem 2 facilities. Based on the success of the programs at these plants, the staff has concluded that augmented natural circulation training should be performed for all future PWR operating licenses. This is to be implemented by including descriptions of natural circulation tests in your FSAR (Chapter 14 - Initial Test Program). If they are not already included in your FSAR, the natural circulation tests and associated training should be included, either by modifying existing or adding new test descriptions in accordance with Regulatory Guide 1.70 (Paragraph 14.2.12). The tests should fulfill the following objectives:

Training

Each licensed reactor operator (RO or SRO who performs RO or SRO duties, respectively) should participate in the initiation, maintenance and recovery from natural circulation mode. Operators should be able to recognize when natural circulation has stabilized, and should be able to control saturation margin, RCS pressure, and heat removal rate without exceeding specified operating limits.

Testing

The tests should demonstrate the following plant characteristics: length of time required to stabilize natural circulation, core flow distribution, ability to establish and maintain natural circulation with or without onsite and offsite power, the ability to uniformly borate and cool down to hot shutdown conditions using natural circulation, and subcooling monitor performance.

If these tests have been performed at a comparable prototype plant, they need be repeated only to the extent necessary to accomplish the above training objectives.

Procedure Validation

The tests should make maximum practical use of written plant procedures to validate the completeness and accuracy of the procedures.

The natural circulation tests require a source of actual or simulated decay heat. The tests may be performed during initial startup using nuclear heat to simulate decay heat, or may be performed later in the initial fuel cycle when actual decay heat is adequate to permit meaningful testing. If the test objectives are not compromised, pump heat during forced circulation operation could provide an acceptable source of simulated decay heat (e.g., the Loss-of-Onsite and Offsite A/C Test performed at North Anna 2).

Applicants who perform a natural circulation boron-mixing and cooldown test to demonstrate compliance with Branch Technical Position RSB BTP 5-1 may use that test to accomplish some or all of the above training and testing objectives.

This guidance is provided for all new PWR OL applicants. Regulatory Guide 1.68 and/or the Standard Review Plan will be revised at a future date to include natural circulation testing and the associated training. OL applicants should submit test descriptions in accordance with Regulatory Guide 1.70 Paragraph 14.2.12 as part of their FSAR or an amendment thereto. Detailed test procedures should be made available for NRC review 60 days prior to scheduled test performance (see Regulatory Guide 1.68 Appendix B). When required by 10 CFR 50.59, a safety analysis must be prepared and distributed in accordance with the requirements stated therein.

PRESERVICE INSPECTION PROGRAM REVIEWS FOR OPERATING LICENSES

121.0 MATERIALS ENGINEERING BRANCH

We require that your inspection program for Class 1, 2 and 3 components be in accordance with the revised rules in 10 CFR Part 50, Section 50.55a, paragraph (g). Accordingly, submit the following information:

(1) A preservice inspection plan which is consistent with the required edition of the ASME Code. This inspection plan should include any exceptions you propose to the Code requirements.

(2) An inservice inspection plan submitted within six months of the anticipated date for commercial operation.

This preservice inspection plan will be required to support the safety evaluation report finding regarding your compliance with preservice and inservice inspection requirements. Our determination of your compliance will be based on the edition of Section XI of the ASME Code referenced in your FSAR or later editions of Section XI referenced in the FEDERAL REGISTER that you may elect to apply.

Your response to this item should define the applicable edition(s) and subsections of Section XI of the ASME Code. If any of the examination requirements of the particular edition of Section XI you referenced in the FSAR cannot be met, a request for relief must be submitted, including complete technical justification to support your request.

Detailed guidelines for the preparation and content of the inspection programs to be submitted for staff review and for relief requests are attached as an Appendix to Section 121.0 of our review questions.

APPENDIX TO SECTION 121.0

GUIDANCE FOR PREPARING PRESERVICE AND INSERVICE INSPECTION PROGRAMS AND RELIEF REQUESTS PURSUANT TO 10 CFR 50.55a(g)

A. Description of the Preservice/Inservice Inspection Program

This program should cover the requirements set forth in Section 50.55a(b) and (g) of 10 CFR Part 50; the ASME Boiler and Pressure Vessel Code, Section XI. Subsections IAW, IWB, IWC and IWD; and Standard Review Plans 5.2.4 and 6.6. The guidance provided in this enclosure is intended to illustrate the type and extent of information that should be provided for NRC review. It also describes the information necessary for "request for relief" of items that cannot be fully inspected to the requirements of Section XI of the ASME Code. By utilizing these guidelines, applicants can significantly reduce the need for requests for additional information from the NRC staff.

B. Contents of the Submittal

The information listed below should be included in the submittal:

1. For each facility, include the applicable date for the ASME Code and the appropriate addenda date.

2. The period and interval for which this program is applicable.

3. Provide the proposed codes and addenda to be used for repairs, modifications, additions or alternations to the facility which might be implemented during this inspection period.

4. Indicate the components and lines that you have exempted under the rules of Section XI of the ASME Code. A reference to the applicable paragraph of the code that grants the exemption is necessary. The inspection requirements for exempted components should be stated (e.g., visual inspection during a pressure test).

5. Identify the inspection and pressure testing requirements of the applicable portion of Section XI that are deemed impractical because of the limitations of design, geometry, or materials of construction of the components. Provide the information requested in the following section of this appendix for the inspections and pressure tests identified in Item 4 above.

C. Request for Relief from Certain Inspection and Testing Requirements

It has been the staff's experience that many requests for relief from testing requirements submitted by applicants and licensees have not been supported by adequate descriptive and detailed technical information. This detailed information is necessary to: (1) document the impracticality of the ASME Code requirements within the limitations of design, geometry, and materials of construction of components; and (2) determine whether the use of alternatives will provide an acceptable level of quality and safety.

Relief requests submitted with a justification such as "impractical," "inaccessible," or any other categorical basis, require additional information to permit the staff to make an evaluation of that relief request. The objective of the guidance provided in this section is to illustrate the extent of the information that is required by the NRC staff to make a proper evaluation and to adequately document the basis for granting the relief in the staff's Safety Evaluation Report. The NRC staff believes subsequent requests for additional information and delays in completing the review can be considerably reduced if this information is provided initially in the applicant's submittal.

For each relief request submitted, the following information should be included:

1. An identification of the component(s) and/or the examination requirements for which relief is requested.

2. The number of items associated with the requested relief.

3. The ASME Code class.

4. An identification of the specific ASME Code requirement that has been determined to be impractical.

5. The information to support the determination that the requirement is impractical; i.e., state and explain the basis for requesting relief.

6. An identification of the alternative examinations that are proposed: (a) in lieu of the requirements of Section XI; or (b) to supplement examinations performed partially in compliance with the requirements of Section XI.

7. A description and justification of any changes expected in the overall level of plant safety by performing the proposed alternative examinations in lieu of the examination required by Section XI. If it is not possible to perform alternate examinations, discuss the impact on the overall level of plant quality and safety.

For inservice inspection, provide the following additional information regarding the inspection frequency:

8. State when the request for relief would apply during the inspection period or interval (i.e., whether the request is to defer an examination).

9. State when the proposed alternative examinations will be implemented and performed.

10. State the time period for which the requested relief is needed.

Technical justification or data must be sumitted to support the relief request. Opinions without substantiation that a change will not affect the quality level are unsatisfactory. If the relief is requested for inaccessibility, a detailed description or drawing which depicts the inaccessibility must accompany the request. A relief request is not required for tests prescribed in Section XI that do not apply to your facility. A statement of "N/A" (not applicable) or "None" will suffice.

D. Request for Relief for Radiation Considerations

Exposures of test personnel to radiation to accomplish the examinations prescribed in Section XI of the ASME Code can be an important factor in determining whether, or under what conditions, an examination must be performed. A request for relief must be submitted by the licensee in the manner described above for inaccessibility and must be subsequently approved by the NRC staff.

We recognize that some of the radiation considerations will only be known at the time of the test. However, the licensee generally is aware, from experience at operating facilities, of those areas where relief will be necessary and should submit as a minimum, the following information with the request for relief:

1. The total estimated man-rem exposure involved in the examination.

2. The radiation levels at the test area.

3. Flushing or shielding capabilities which might reduce radiation levels.

4. A proposal for alternate inspection techniques.

5. A discussion of the considerations involved in remote inspections.

6. Similar welds in redundant systems or similar welds in the same systems which can be inspected.

7. The results of preservice inspection and any inservice results for the welds for which the relief is being requested.

8. A discussion for the consequences if the weld which was not examined, did fail.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 25, 1981

TO ALL LICENSEES OF OPERATING NUCLEAR POWER REACTORS AND APPLICANTS FOR OPERATING LICENSES (EXCEPT FOR ST. LUCIE UNIT NOS. 1 & 2)

SUBJECT: EMERGENCY PROCEDURES AND TRAINING FOR STATION BLACKOUT EVENTS (Generic Letter 81-04)

A recent decision by the Atomic Safety and Licensing Appeal Board (ALAB-603) concluded that station blackout (i.e., loss of all offsite and onsite AC power) should be considered a design basis event for St. Lucie Unit No. 2. An amendment to the Construction Permit for St. Lucie Unit No. 2 was subsequently issued on September 18, 1980. The NRC staff is currently assessing station blackout events on a generic basis (Unresolved Safety Issue A-44). The results of this study, which is scheduled to be completed in 1982, will identify the extent to which design provisions should be included to reduce the potential for or consequences of a station blackout event.

However, the Board has recommended that more immediate measures be taken to ensure that station blackout events can be accommodated while task A-44 is being conducted. Although we believe that, qualitatively, there appears to be sufficient time available following a station blackout event to restore AC power, we are not sure if licensees have adequately prepared their operators to act during a station blackout event.

Consequently, we request that you review your current plant operations to determine your capability to mitigate a station blackout event and promptly implement, as necessary, emergency procedures and a training program for station blackout events. Your review of procedures and training should consider, but not be limited to:

a. The actions necessary and equipment available to maintain the reactor coolant inventory and heat removal with only DC power available, including consideration of the unavailability of auxiliary systems such as ventilation and component cooling.

b. The estimated time available to restore AC power and its basis.

c. The actions for restoring offsite AC power in the event of a loss of the grid.

d. The actions for restoring offsite AC power when its loss is due to postulated onsite equipment failures.

e. The actions necessary to restore emergency onsite AC power. The actions required to restart diesel generators should include consideration of loading sequence and the unavailability of AC power.

f. Consideration of the availability of emergency lighting, and any actions required to provide such lighting, in equipment areas where operator or maintenance actions may be necessary.

g. Precautions to prevent equipment damage during the return to normal operating conditions following restoration of AC power. For example, the limitations and operating sequence requirements which must be followed to restart the reactor coolant pumps following an extended loss of seal injection water should be considered in the recovery procedures.

The annual requalification training program should consider the emergency procedures and include simulator exercises involving the postulated loss of all AC power with decay heat removal being accomplished by natural circulation and the steam-driven auxiliary feedwater system for PWR plants, and by the steam-driven RCIC and/or HPCI and the safety-relief valves in BWR plants.

We conclude that the actions described above should be completed as soon as they reasonably can be (i.e., within 6 months). In addition, so that we may determine whether your license should be amended to incorporate this requirement, you are requested, pursuant to §50.54(f), to furnish within ninty (90) days of receipt of this letter, an assessment of your existing or planned facility procedures and training programs with respect to the matters described above. Please refer to this letter in your response. In the event that completion within 6 months can not be met, please propose a revised date and justification for the delay.

This request for information was approved by GAO under a blanket clearance number R0072 which expires November 30, 1983. Comments on burden and duplication may be directed to the U.S. General Accounting Office, Regulatory Reports Review, Room 5106, 441 G Street, NW., Washington, D.C. 20548.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

PRESERVICE INSPECTION AND TESTING OF SNUBBERS

TO ALL APPLICANTS:

Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety related systems and components, it is requested that maintenance records for snubbers be documented as follows:

Pre-service Examination

A pre-service examination should be made on all snubbers listed in tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9 This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a mimimum verify the following:

(1) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(2) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifictions.

(3) Snubbers are not seized, frozen or jammed.

(4) Adequate swing clearance is provided to allow snubber movement.

(5) If applicable, fluid is to the recommended level and is not leaking from the snubber system.

(6) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of items 1,4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250° F should be verified as follows:

(a) During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

(b) For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

(c) Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepencies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

REQUEST FOR ADDITIONAL INFORMATION

Containment Sump and its effect on long term cooling following a LOCA

During our reviews of license applications we have identified concerns related to the containment sump design and its effect on long term cooling following a Loss of Coolant Accident (LOCA).

These concerns are related to (1) creation of debris which could potentially block the sump screens and flow passages in the ECCS and the core, (2) inadequate NPSH of the pumps taking suction from the containment sump, (3) air entrainment from streams of water or steam which can cause loss of adequate NPSH, (4) formation of vortices which can cause loss of adequate NPSH, air entrainment and suction of floating debris into the ECCS and (5) inadequate emergency procedures and operator training to enable a correct response to these problems. Preoperational recirculation tests performed by utilities have consistently identified the need for plant modifications.

The NRC has begun a generic program to resolve this issue. However, more immediate actions are required to assure greater reliability of safety system operation. We therefore require you take the following actions to provide additional assurance that long term cooling of the reactor core can be achieved and maintained following a postulated LOCA.

1. Establish a procedure to perform an inspection of the containment, and the containment sump area in particular, to identify any materials which have the potential for becoming debris capable of blocking the containment sump when required for recirculation of coolant water. Typically, these materials consist of: plastic bags, step-off pads, health physics instrumentation, welding equipment, scaffolding, metal chips and screws, portable

inspection lights, unsecured wood, construction materials and tools as well as other miscellaneous loose equipment. "As licensed" cleanliness should be assured prior to each startup.

This inspection shall be performed at the end of each shutdown as socn as practical before containment isolation.

2. Institute an inspection program according to the requirements of Regulatory Guide 1.82, item 14. This item addresses inspection of the containment sump components including screens and intake structures.

3. Develop and implement procedures for the operator which address both a possible vortexing problem (with consequent pump cavitation) and sump blockage due to debris. These procedures should address all likely scenarios and should list all instrumentation available to the operator (and its location) to aid in detecting problems which may arise, indications the operator should look for, and operator actions to mitigate these problems.

4. Pipe breaks, drain flow and channeling of spray flow released below or impinging on the containment water surface in the area of the sump can cause a variety of problems; for example, air entrainment, cavitation and vortex formation.

   Describe any changes you plan to make to reduce vortical flow in the neighborhood of the sump. Ideally, flow should approach uniformly from all directions.

5. Evaluate the extent to which the containment sump(s) in your plant meet the requirements for each of the items previously identified; namely

debris, inadequate NPSH, air entrainment, vortex formation, and operator actions.

The following additional guidance is provided for performing this evaluation.

(1) Refer to the recommendations in Regulatory Guide 1.82 (Section C) which may be of assistance in performing this evaluation.

(2) Provide a drawing showing the location of the drain sump relative to the containment sumps.

(3) Provide the following information with your evaluation of debris:

(a) Provide the size of openings in the fine screens and compare this with the minimum dimensions in the pumps which take suction from the sump (or torus), the minimum dimension in any spray nozzles and in the fuel assemblies in the reactor core or any other line in the recirculation flow path whose size is comparable to or smaller than the sump screen mesh size in order to show that no flow blockage will occur at any point past the screen.

(b) Estimate the extent to which debris could block the trash rack or screens (50 percent limit). If a blockage problem is identified, describe the corrective actions you plan to take (replace insulation, enlarge cages, etc.).

(c) For each type of thermal insulation used in the containment, provide the following information:

(i) type of material including composition and density,

(ii) manufacturer and brand name,

(iii) method of attachment,

(iv) location and quantity in containment of each type,

(v) an estimate of the tendency of each type to form particles small enough to pass through the fine screen in the suction lines.

(d) Estimate what the effect of these insulation particles would be on the operability and performance of all pumps used for recirculation cooling. Address effects on pump seals and bearings.

REQUEST FOR ADDITIONAL INFORMATION

Diablo Canyon Units 1 & 2

260.17 Section 17.1.2.2 of the standard format (Regulatory Guide 1.70) requires the identification of safety-related structures, systems, and components (Q-list) controlled by the QA program. You are requested to supplement and clarify the Diablo Canyon Q-list in Table 3.2-4 of the FSAR in accordance with the following:

a. The following items do not appear on the Q-list (FSAR Table 3.2-4). Add the appropriate items to the Q-list and provide a commitment that the remaining items are subject to the pertinent requirements of the FSAR operational quality assurance program or justify not doing so.

1. Safety-related masonry walls (see IE Bulletin No. 80-11).

2. Breakwaters.

3. Leak detection system (see FSAR Section 3.5).

4. Missile barriers which protect safety-related items.

5. Onsite power system (Class 1E).

   a) Electrical penetrations of containment - Non-vital including primary and backup fault current protective devices.

   b) Raceway fire stops and seals.

   c) Emergency light battery packs.

6. Radiation monitoring (fixed and portable).

7. Radioactivity monitoring (fixed and portable).

8. Radioactivity sampling (air, surfaces, liquids).

9. Radioactive contamination measurement and analysis.

10. Personnel monitoring internal (e.g., whole body counter) and external (e.g., TLD system).

11. Instrument storage, calibration, and maintenance.

12. Decontamination (facilities, personnel, and equipment).

13. Respiratory protection, including testing.

14. Contamination control.

15. Radiation shielding.

16. Meteorological data collection programs.

17. Expendable and consumable items necessary for the functional performance of safety-related structures, systems, and components (i.e., weld rod, fuel oil, boric acid, snubber oil, etc.).

18. Measuring and test equipment used for safety-related structures, systems, and components.

19. Ground slope east of building complex.

20. Firewater storage reservoir ponds.

21. Hydrogen recombiner, including piping and valves.

22. Containment pressure indication system.

23. Containment water level indication systems.

24. Containment hydrogen indication system.

25. Valve operators for safety-related valves.

26. Motors for safety-related pumps.

b. The following items from the Q-list (FSAR Table 3.2-4) need expansion and/or clarification as noted. Revise the list as indicated or justify not doing so.

1. Portions of the turbine generator building (sheet 4) which enclose the emergency diesel-generator units and ancillary systems as well as other safety-related components should be under the controls of the operational QA program.

2. New fuel storage racks (sheet 3) should be under the controls of the operational QA program.

3. Intake structure and conduit (sheet 5) should be under the controls of the operational QA program.

4. Containment structure sump, sump screen, and vortex suppression should be under the controls of the operational QA program.

5. Reactor cavity sump pump (sheet 18) should be under the controls of the operational QA program.

6. Clarify that the primary system PORV, safety valves, and PORV block valves and their actuators are included under "Reactor Coolant Systems Valves," (sheet 25).

7. Clarify that the main steamline safety valves and steamline PORVs and their actuators are included under "Valves for the Above (Main Steam Piping-SG to MSIV) Portion of System" (sheet 23).

8. Identify the safety-related instrumentation and control systems to the same scope and level of detail as provided in Chapter 7 of the FSAR.

9. The 250V DC Motor Control Center SD 121 (sheet 36) should be under the controls of the operational QA program.

10. Circulating water conduits (sheet 5) should be under the controls of the operational QA program.

c. Enclosure 2 of NUREG-0737, "Clarification of TMI Action Plan Requirements" (November 1980) identified numerous items that are safety-related and appropriate for OL application and therefore should be on the Q-list. These items are listed below. Add the appropriate items to the Q-list and provide a commitment that the remaining items are subject to the pertinent requirements of the FSAR operational quality assurance program or justify not doing so.

	NUREG-0737 (Enclosure 2) Clarification Item
1) Plant-safety-parameter display console.	I.D.2
2) Reactor coolant system vents.	II.B.1
3) Plant shielding.	II.B.2
4) Post accident sampling capabilities.	II.B.3
5) Valve position indication.	II.D.3
6) Auxiliary feedwater system.	II.E.1.1
7) Auxiliary feedwater system initiation and flow.	II.E.1.2
8) Emergency power for pressurizer heaters.	II.E.3.1
9) Dedicated hydrogen penetrations.	II.E.4.1
10) Containment isolation dependability.	II.E.4.2
11) Accident monitoring instrumentation.	II.F.1
12) Instrumentation for detection of inadequate core-cooling.	II.F.2
13) Power supplies for pressurizer relief valves, block valves, and level indicators.	II.G.1
14) Automatic PORV isolation.	II.K.3(1)
15) Automatic trip of reactor coolant pumps.	II.K.3(5)
16) PID controller.	II.K.3(9)
17) Anticipatory reactor trip on turbine trip.	II.K.3(12)
18) Power on pump seals.	II.K.3(25)
19) Emergency plans.	III.A.1.1/III.A.2
20) Emergency support facilities.	III.A.1.2
21) Inplant I_2 radiation monitoring.	III.D.3.3
22) Control-room habitability.	III.D.3.4