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

MAR 02 1983

MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Richard H. Vollmer, Director
Division of Engineering

SUBJECT: DIVISION OF ENGINEERING GEOSCIENCE PLAN TO ADDRESS USGS CLARIFICATION RELATING TO SEISMIC DESIGN EARTHQUAKES IN THE EASTERN SEABOARD OF THE UNITED STATES

A plan for our proposed program to address the U. S. Geological Survey's clarification of position relating to seismic design earthquakes in the Eastern Seaboard of the United States is attached (enclosure 1). This plan elaborates on the outline provided as an attachment to a memorandum entitled, "Clarification of U. S. Geological Survey Position Relating to Seismic Design Earthquakes in the Eastern Seaboard of the United States", which was sent from the Executive Director of Operations to the Commissioners on November 19, 1982.

The plan is divided into two parts. Part one is a short term probabilistic assessment utilizing an extensive new seismic hazard study currently being developed by Lawrence Livermore National Laboratory. Part two is a longer term deterministic assessment based primarily on long range ORES research with the possible need for utility sponsored investigations at some locations after an assessment of the long term research results. Additionally, we recommend that an industry sponsored seismic hazard study be solicited.

We estimate that the effort to establish the seismic hazard level for the sites and make appropriate comparisons will take approximately three years to complete, utilizing staff resources of about 2.5-3.0 SY per year, and $300K per year in technical assistance funds. Our preliminary recommendations on which plants, if any, may need further evaluation should be completed in mid-1984. Because of the required research effort, the deterministic element will not be synthesized until 1985.

The proposed program will complement ongoing PRA reviews and the seismic hazard spectra which are developed can also be used for future SEP evaluations. This program, therefore, is basically a continuation, with modification, of our ongoing work. This program does not include resources to complete a reevaluation effort for plants for which design spectra may need to be reevaluated. We recommend that this contingency be considered and included in the operating plan for FY 84. This plan also presupposes that our interim position for licensing reviews (enclosure 2) is found to be acceptable by ACRS and ASLB while we implement this program.

There is evidence to support this assumption in the recent Appeal Board decision on Summer (ALAB-710).

We have also assessed our ability to implement this plan under the existing regulation, Appendix A to 10 CFR Part 100. We have concluded that, although Appendix A itself does not explicitly recognize the use of probabilistic methods, as a minimum they can be used to assist in reaching deterministic judgements (Seabrook Remand, CLI80-33). It is not clear whether they can be used as the primary tool in setting appropriate ground motion levels. Therefore, we recommend that we implement a limited modification or clarification of Appendix A as previously planned in conjunction with ORES as a parallel, yet independent effort, along with the Charleston plan. This modification has been recommended in SECY-79-300 and endorsed by the Siting Policy Task Force in NUREG-0625 and is necessary to reflect the current state of art. This modification will require an additional 1.0 SY per year for 2 years.

We recommend that you consider placing this effort equally under three resource areas - Operating Reactor Licensing Actions or Safety Technology, Systematic Evaluation Program for older operating plants, and Casework for ongoing OL review plants.

This plan has been developed as a result of extensive discussion within the Geosciences Branch, NRR; and discussions with the Earth Sciences Branch, ORES; and the U. S. Geological Survey.

R H Vollmer

Richard H. Vollmer, Director
Division of Engineering

Enclosure:
As stated

cc: w/enclosure

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Recommended Plan
Eastern U. S. Earthquakes

Introduction

On November 18, 1982, the U. S. Geological Survey (USGS) forwarded a letter to the Nuclear Regulatory Commission clarifying their past position with respect to the 1886 Charleston earthquake. The USGS letter states that:

> "Because the geologic and tectonic features of the Charleston region are similar to those in other regions of the eastern seaboard, we conclude that although there is no recent or historical evidence that other regions have experienced strong earthquakes, the historical record is not, of itself, sufficient grounds for ruling out the occurrence in these other regions of strong seismic ground motions similar to those experienced near Charleston in 1886. Although the probability of strong ground motion due to an earthquake in any given year at a particular location in the eastern seaboard may be very low, deterministic and probabilistic evaluations of the seismic hazard should be made for individual sites in the eastern seaboard to establish the seismic engineering parameters for critical facilities."

We have evaluated the USGS clarification of position and have concluded that it can be addressed predominantly through existing programs at NRC with the possibility of additional requests for utility - sponsored work. We recommend that a two part program be implemented which will address both the deterministic and probabilistic elements mentioned by the USGS.

Part 1 of the proposed program is a short term probabilistic assessment of plants in the eastern seaboard. This part of the plan is necessary because many of the current tectonic working hypotheses are not amenable to investigation by deterministic methods in the short term.

Part 2 of the proposed program is a longer term deterministic assessment of the causes of large earthquakes, such as the Charleston earthquake, in the eastern seaboard. Specific areas of relatively high seismicity and tectonic structures are identified which we recommend be addressed through the ORES long range research plan.

Based on our evaluation of the research results, some applicants or licensees may be required to investigate tectonic structures which may not have been previously identified during the licensing procedure.

Part 1 - Probabilistic Assessment

Discussion

The November 18, 1982 letter from the USGS represents not so much a new understanding but rather a more explicit recognition of existing uncertainties with respect to the causative structure and mechanism of the 1886 Charleston earthquake. Many hypotheses have been proposed as to the locale in the eastern seaboard of future Charleston-size earthquakes. Some of these could be very restrictive in location while others would allow this earthquake to recur over very large areas. Presently, none of these hypotheses are definitive and all contain a strong element of speculation.

Traditional deterministic approaches such as that outlined in Section 2.5.2 of the Standard Review Plan are not generally designed to deal with this situation. Probabilistic methods which allow for the consideration of many hypotheses, their associated credibilities, and the explicit incorporation of uncertainty are much better equipped to provide rational frameworks for decision making. The question that needs to be answered is:

> Taking uncertainties into account, have licensing decisions for plants in the eastern seaboard (i.e., in the region affected by the USGS clarified position on the Charleston Earthquake) resulted in acceptable levels of assumed seismic hazard (exposure to earthquake ground motion) at the individual sites?

One means for answering the above question is a probabilistic assessment of seismic hazard at all nuclear power plant sites east of the Rocky Mountains. Since adequate or acceptable levels of seismic hazard have not been explicitly defined in probabilistic terms, it is assumed that the probability of seismic ground motion exceeding design levels implicitly associated with licensing decisions based upon traditional methods in other regions of the U. S. east of the Rocky Mountains is adequate; these other regions include areas such as the Central Stable Region and the Gulf Coastal Plain. The prime tool for carrying out this assessment is an updated version of the Uniform Hazard Methodology developed for the Systematic Evaluation Program by Lawrence Livermore National Laboratory (LLNL) and its subcontractor TERA Corporation. This methodology relies upon the incorporation of diverse expert opinion with regard to the input parameters needed to make probabilistic estimates. As such, it does not rely upon single hypotheses which do not account for existing uncertainties but rather attempts to incorporate the hypotheses and their uncertainties into the computations. Identification of plants (if any) in the eastern seaboard at which the probability of exceeding design-level ground motion is significantly greater than has been assumed at other locations may result in an integrated seismic evaluation and/or engineering rearalysis to assure the plant's ability to withstand a more severe earthquake. This study may also identify selected plants outside of the eastern seaboard whose design levels may be inappropriate, relative to other plants, with respect to the seismic hazard.

In addition, we are also initiating, through a technical assistance contract, a study to better estimate ground motion from a large earthquake the size of the 1886 Charleston event to gain a better understanding of how this ground motion should be represented.

<u>Major Activities - Probabilistic Assessment</u>

The probabilistic assessment portion of the proposed program is divided into the following elements.

1. January thru April 1983 - Continue development of LLNL study including expert opinion surveys on seismic hazard east of the Rocky Mountains. This study (Seismic Hazard Characterization of the Eastern U.S.) is presently underway as a joint effort of NRR and

ORES. No additional resources above those already allocated are needed.

2. May 1983 thru December 1983 - Calculation of seismic hazard spectra by LLNL for all nuclear power plant sites (approximately 75) east of the Rocky Mountains. An estimation of the probability of seismic ground motion exceeding the design level at each site, taking into account specific site conditions, will be completed and provided as a report. An additional 2.0 SY is needed for LLNL and 0.3 SY for NRC effort during this period.

3. September - December 1983 - Comparison of LLNL study with existing probabilistic studies such as Algermissen and others (1982). An additional 0.2 SY is needed for LLNL effort.

4. March 1983 - December 1983 - Sponsorship by the utilities of a probabilistic estimation of seismic hazard for all nuclear power plants east of the Rocky Mountains. This study, while not a requirement, is strongly recommended so as to complement the LLNL study and provide another independent assessment of seismic hazard. An additional 0.1 SY needed for LLNL and 0.1 SY for NRC effort.

5. December 1983-March 1984 - Using LLNL and other studies, the NRC staff will integrate this information and make comparisons of the probability of seismic ground motion exceeding design levels in the eastern seaboard with probabilities calculated at plants in the rest of the Eastern and Central U. S. Comparisons will be made in several ways including comparison by region alone and by region and plant vintage. Plants in the eastern seaboard (if any) that are associated with significantly greater hazard than those elsewhere

will then be identified. Other comparisons may be needed, but will be decided upon after review of initial results. An additional 0.7 SY is needed for NRC effort.

6. April 1984-September 1984 - Assessment of initial conclusions regarding hazard in light of feedback from expert opinion on original input. A final letter report will be issued with a final recommendation on plants which need reevaluation. An additional 0.2 SY needed for LLNL and 0.2 SY for NRC effort.

7. January 1983-December 1983 - Ground motion estimates at different distances and site conditions from a large Charleston type earthquake. Both theoretical and empirical estimates using data from recent earthquakes will be made. This study is presently being initiated through a technical assistance contract with LLNL. No additional resources are required.

Status summary reports of research into probabilistic estimates of seismic hazard funded by ORES will be needed by December 1983 so as to incorporate them into task number 5.

Implementation of Probabilistic Assessment Results

The implementation of results is outlined above in elements 5 and 6.

NRR Staff and Cost Requirements - Probabilistic Assessment

The additional effort required for this portion of the program will be 2.5 SY for LLNL (1.9 in FY 83, 0.6 in FY 84) and 1.3 SY for NRC (0.3 in FY 83, 1.0 in FY 84). This staff effort can be accommodated with the currently available resources in the Geosciences Branch because this program complements ongoing staff activities and may replace other staff activities for individual sites. This program does not include resources to complete the seismic evaluation and/or engineering reanalysis which some plants may require as a result of the probabilistic elements.

Utility-Sponsored Study in Conjunction with the Probabilistic Assessment

A recommended utility-sponsored study is outlined above in element 4.

Schedule - Probabilistic Assessment

The proposed schedule for implementing this plan appears in Table 1.

Part 2 - Deterministic Assessment

Discussion

The deterministic portion of the proposed program is designed to better understand the causes of large earthquakes, such as the Charleston earthquake, in the eastern seaboard. This effort may require some expansion of immediate and long term ORES programs. Increased understanding of the cause of seismicity in the eastern seaboard will allow a reduction in the uncertainty in estimating the seismic hazard for nuclear power plants. The primary problem with seismic hazard characterization in the eastern seaboard is that no causative mechanism for seismicity has been identified to date and no surface offsets due to earthquakes are known. Although there are literally thousands of crustal structures known in the eastern seaboard, which, if they were active, could produce strong earthquakes, none have been demonstrated to have been active during the Quaternary (the last two million years) or proved to be capable. The result is that, to date, there has been no generally accepted association between eastern seismicity and crustal structure.

The overall approach of the deterministic assessment is to study areas of relatively higher seismicity in the eastern seaboard to determine if tectonic features and processes responsible for the seismicity can be identified and correlated. This will be pursued by crustal studies at hypocentral depths to determine if there is any correlation between crustal structures at hypocentral depths and the earthquake hypocenters. The primary tool for determining crustal structure at hypocentral depths will be the use of multi-channel seismic reflection profiles. The primary tools for locating the hypocenters will be the continued monitoring and analysis of earthquakes from the existing microearthquake nets. These nets will have to be maintained and upgraded in order to improve depth locations of hypocenters if there is to be an improved ability to correlate between hypocenters and tectonic structures at depths of up to 25 kilometers.

This research will be contracted and monitored by ORES, and does not represent a radical departure from past programs. Increased coordination between NRR and ORES will be required, however, to better define the problems that are to be resolved in order to improve our understanding of eastern seismicity in the licensing context. This portion of the program is designed to improve our ability to assess the adequacy of the design of nuclear facilities on the eastern seaboard. The result, in part, will be summary reports which will represent the current status of research including a review and synthesis of available data. These results will be used to modify, if necessary, conclusions drawn from the probabilistic studies and identify individual features, if appropriate, for assessment by utilities.

Major Activities - Deterministic Assessment

The deterministic assessment portion of the proposed program is divided into the following elements appropriate to each region listed.

A. Charleston Region

Since the causative mechanism of the Charleston earthquake of 1886 continues to be one of the primary unresolved problems in evaluating seismicity in the eastern seaboard, research in the Charleston area should continue with the goal of testing the various hypotheses as to the cause of the earthquake. In particular, emphasis should be placed on determining if suggested features such as the Ashley River and Woodstock Fault zones constitute the source zones of the Charleston earthquakes.

1. May 1983 - "Workshop on the 1886 Charleston Earthquake and Its Implications for Today" - the U. S. Geological Survey and the scientific community will present a summary and evaluation of the tectonics and seismicity at Charleston.
2. September 1983 - ORES in consultation with the U. S. Geological Survey and the scientific community should have a program in place to test the most likely tectonic hypothesis for seismicity.
3. June 1984 - ORES presents the esults of the program

of testing the highest-weighted hypothesis.

4. January 1985 - ORES presents summary report describing the results of the Charleston work testing the highest-weighted tectonic hypothesis.

B. Ramapo Fault Zone

The Ramapo Fault Zone, a Precambrian fault zone that was intermittently active until the Mesozoic, is the northwestern boundary of the Newark Triassic Basin. Low level seismicity occurs in the area and may be associated with the fault zone, however, the seismicity in the region forms a band 40 kilometers wide. Detailed field work and limited trenching and core drilling suggest that the Ramapo Fault has not been recently reactivated. The purpose of studying the fault is to establish whether there is a causal relationship between Mesozoic or older faults such as the Ramapo Fault and current seismicity in this area by determining the location and geometry of these faults at hypocentral depths.

1. April 1983 - ORES initiates a new evaluation of the Ramapo Fault. The study should include multi-channel seismic reflection profiling and other geophysical techniques such as in-situ stress measurements and geodetic measurements to determine the current state of stress at hypocentral depths.
2. January 1984 - ORES presents preliminary results of the program to date, and plans for the coming year.
3. January 1985 - ORES presents summary report on this aspect of the Ramapo Fault Study including the identification and analysis of any seismic source zones.

C. Central Virginia Seismic Zone

Recent work by earth scientists at Virginia Polytechnic Institute have suggested that there may be a relationship between the seismicity in Central Virginia and the northeast trending thrust faults and decollement of the Piedmont crust of the Appalachian Orogenic Belt. The purpose of this part of the program is to continue evaluation of the relationship between the faults and the earthquakes.

1. April 1983 - ORES presents a plan for undertaking the seismic reflection profiling, and applying other geophysical techniques such as geodetic measurements and in-situ stress measurements.
2. January 1984 - ORES presents the preliminary results or progress to date, and plans for the coming year.
3. January 1985 - ORES presents a summary report on the the Central Virginia Study including the potential identification and analysis of any seismic source zones.

D. Giles County, Virginia

The Giles County Seismic Zone is a northeast trending linear zone of seismicity which apparently is located beneath the decollement and thrust faults associated with the Valley & Ridge Province of the Appalachian Orogenic Belt. It has been suggested that the seismic zone has occurred as a reactivated northeast trending normal fault associated with the opening of the Proto-Atlantic (called the Iapatus) in the late Proterozoic and early Paleozoic (800-500 million years ago).

1. April 1983 - ORES initiates planning for the proposed research.

2. August 1983 - ORES initiates study of the Giles County structure using seismic reflection profiling.
3. April 1984 - ORES presents preliminary results and plans for the coming year.
4. April 1985 - ORES presents summary results of this phase of the research including the potential identification and analysis of any seismic source zones.

E. New England

The research in New England has been underway for several years and will be continued. Increased emphasis should be placed on evaluation of the source mechanism for the New Brunswick and Gaza, N.H. earthquakes, the neotectonics of seismically active areas, and the orientation and magnitude of the stress field in the seismically active areas of the region. An in-situ stress measurement at hypocentral depths will be conducted at Moodus. Depending on the results of the seismic reflection studies described above, additional seismic reflection surveys may be conducted in seismically active areas of New England such as Moodus, Connecticut; New Hampshire; Massena, New York and New Brunswick, Canada.

1. April 1983 - ORES completes plans for stress measurement at Moodus.
2. August 1983 - Conduct stress measurements at Moodus.
3. April 1984 - ORES presents preliminary results of stress measurements and their relationship to the local seismicity and tectonics.
4. January 1985 - ORES presents summary results of stress measurements and other studies described above.

Implementation of Deterministic Assessment Results

As the results from the deterministic studies become available, they will be evaluated, and, the effect, if any, on operating plants and plants in the Operating License stage of review will be determined. The need for additional evaluations of particular structures by utilities will be assessed as the information becomes available. Two problems will be addressed by the deterministic portion of the program; (1) whether or not the deterministic findings warrant any reassessment of the conclusions drawn from the probabilistic study; and (2) whether there are any particular tectonic structures which are associated with or similar to tectonic structures associated with seismicity which, because of their proximity to individual sites, should be analyzed by the utilities. The above effort will take about two to three years (early 1985) to complete. The impact of this research on nuclear power plants will be determined by the NRC staff with technical assistance contracts, if necessary.

NRR Staff and Cost Requirements - Deterministic Assessment

This effort will require continuous communication among NRR, ORES and the contractors. As research funds are limited and the amount of time is short, careful interaction will be necessary to obtain the information required to allow a resolution of eastern seismicity. It is estimated that one staff year per year for three years will be necessary for NRR to implement this deterministic part of the overall plan. The research effort will be funded by ORES and technical assistance contracts will be funded by NRR. It is estimated that for the deterministic assessment, $200,000 may be required to implement the NRR technical assistance program to determine the impacts of the findings on the nuclear facilities in the eastern U. S.

Utility-Sponsored Studies as Result of the Deterministic Asessment

During FY 1983 no deterministic work by the utilities is currently recommended, beyond that necessary to pursue their normal efforts to continue to assess any hazards identified by them for their sites. After the results of the research are available and if any source zones are identified which have particular importance to specific sites or have impact on the probabilistic program, some utilities may be required to investigate structures in the vicinity of their plants.

Schedule - Deterministic Assessment

The proposed schedule for implementing this plan follows as Table 1. Our ability to meet this proposed schedule may be somewhat optimistic and is contingent on implementing the appropriate contracts. We will be better able to assess this schedule when the work has been initiated.

REFERENCE

Algermissen, S. T., D. M. Perkins, P. C. Thenhaus, S. L. Hanson, and B. L. Bender, 1982, Probabilistic Estimates of Maximum Acceleration and Velocity in Rock in the Contiguous United States, United States Department of Interior, Geological Survey. Open-File Report 82-1033, 99 p.

Calendar Year Schedule for Probabiilstic and Deterministic Seismic Hazard Program

	1983 / 1984 / 1985
Part 1 Short Term	Meet with ACRS to discuss Program (1983); Meet with ACRS to to discuss Preliminary Recommendations (1984)
1. Update LLNL Seismic Hazard Methodology	______+Complete Methodology
2. Calculate Seismic Hazard Spectra for Eastern Sites	+______+Report with Spectra
3, Compare with other available probability studies	+______+Report with Comparisons
4. Initiation of Industry-Sponsored Seismic Hazard Study	+______+Production of Study Results
5..Comparison of Seismic Hazard at Sites	+______+ Letter Report with Preliminary Recommendations ______+Final Recommendations
6. Assessment of Impact of Expert Feedback	+ Initate Feedback ______+ Assess Impact on Previous Results
7. Charleston Ground Motion Study	+ Initiate Tac with LLNL ______+Issue Report

Table 1

Calendar Year Schedule for Probabilistic and Deterministic Seismic Hazard Program

1983 — 1984 — 1985

Part 2 Long Term				
	Meet with ACRS to discuss Program	Meet with ACRS to to discuss Preliminary Recommendations		
1. Charleston Research	Workshop-Interim Synthesis	Progress Report on Hypothesis Testing	Results of Testing	
2. Ramapo Fault Research	Initiate Study	Preliminary Report	Summary Report	
3. Central Va. Research	Initiate Study	Preliminary Results Report	Summary Report	
4. Giles County, Va. Research	RFP	Initiation of Study	Preliminary Results	Summary Report
5. New England Seismotectonic Research	Stress Measurements Plan	Conduct Measurements	Preliminary Results	Summary Report
6. Assessment of Impact of Deterministic Studies on Sites	Preliminary Evaluation of Results of RES	Summary of Source Zones	Summarize Review of Determinsitic Work	

Table 1 (cont'd)

Interim Position on Charleston Earthquake for Licensing Proceeding

The NRR Staff position with respect to the Intensity X 1886 Charleston earthquake has been that, in the context of the tectonic province approach used for licensing nuclear power plants, this earthquake should be restricted to the Charleston vicinity. This position was based, in part, on information provided by the United States Geological Survey (USGS) in a letter dated December 30, 1980 from J. E. Devine to R. E. Jackson (see Summer Safety Evaluation Report). The USGS has been reassessing its position and issued a clarification on November 18, 1982 in a letter from J. E. Devine to R. E. Jackson. As a result of this letter, a preliminary evaluation and outline for NRC action was forwarded to the Commission in a memorandum from W. J. Dircks on November 19, 1982.

The USGS letter states that:

> "Because the geologic and tectonic features of the Charleston region are similar to those in other regions of the eastern seaboard, we conclude that although there is no recent or historical evidence that other regions have experienced strong earthquakes, the historical record is not, of itself, sufficient grounds for ruling out the occurrence in these other regions of strong seismic ground motions similar to those experienced near Charleston in 1886. Although the probability of strong ground motion due to an earthquake in any given year at a particular location in the eastern seaboard may be very low, deterministic and probabilistic evaluations of the seismic hazard should be made for individual sites in the eastern seaboard to establish the seismic engineering parameters for critical facilities."

The USGS clarification represents not so much a new understanding but rather a more explicit recognition of existing uncertainties with respect to the causative structure and mechanism of the 1886 Charleston earthquake. Many hypotheses have been proposed as to the locale in the eastern seaboard of future Charleston-size earthquakes. Some of these could be very restrictive in location while others would allow this earthquake to recur over very large areas. Presently, none of these hypotheses are definitive and all contain a strong element of speculation.

We are addressing this uncertainty in both longer-term deterministic and shorter-term probabilistic programs. The deterministic studies, funded primarily by the Office of Research of the NRC should reduce the uncertainty by better identifying (1) the causal mechanism of the Charleston earthquake and (2) the potential for the occurrence of large earthquakes throughout the eastern seaboard. The probabilistic studies, primarily that being conducted for NRC by Lawrence Livermore National Laboratory (LLNL) will take into account existing uncertainties. They will have as their aim to determine differences, if any, between the probabilities of seismic ground motion exceeding design levels in the eastern seaboard (i.e. as affected by the USGS clarified position on the Charleston earthquake) and the probabilities of seismic ground motion exceeding design levels elsewhere in the central and eastern U. S. Any plants where the probabilities of exceeding design level ground motions are significantly higher than those calculated for other plants in the Central and Eastern U. S. will be identified and evaluated for possible further engineering analysis.

Given the speculative nature of the hypotheses with respect to the recurrence of large Charleston-type earthquakes as a result of our limited scientific knowledge and the generalized low probability associated with such events, we do not see a need for any action for specific sites at this time. It is our position, as it has been in the past, that facilities should be designed to withstand the recurrence of an earthquake the size of the 1886 earthquake in the vicinity of Charleston. At the conclusion of the shorter-term probabilistic program and during the longer-term deterministic studies, we will be assessing the need for a modified position with respect to specific sites.

REQUEST FOR ADDITIONAL INFORMATION
MILLSTONE NUCLEAR POWER STATION UNIT 3
DOCKET NO. 50-423

230.0 GEOSCIENCES BRANCH, SEISMOLOGY SECTION

230.3 (SRP 2.5.2) RSP

On November 18, 1982 the USGS in a letter from James F. Devine, USGS to Robert Jackson, NRC, clarified its position regarding the localization of the seismicity in the vicinity of Charleston, S.C.

The staff is presently evaluating the significance of the USGS clarification regarding the localization of Charelston seismicity. Attached are copies of the staff's interim position on the Charleston Earthquake and our recommended plan to address Eastern U. S. Earthquakes. This position will be included in the Safety Evaluation Report.

230.4 (SRP 2.5.2.2, 2.5.2.3, 2.5.2.4)

In licensing decisions made since approximately 1976, regarding the seismic design basis of nuclear power plants located in New England and the northern Piedmont, the staff has recognized the New England-Piedmont Tectonic Province. On January 9, 1982 a magnitude 5.7 earthquake occurred in south central New Brunswick, Canada, in geologic terrain that is similar to that which characterizes the New England Piedmont Province. As discussed in FSAR sections 2.5.1.1.4 and 2.5.2.2 and shown in figure 2.5.2-10 the northern Appalachian region is sub-divided into a number of tectonic provinces, which is different than the New England-Piedmont Province. With respect to the appropriate choice of tectonic provinces and the effect of the New Brunswick earthquake on the site, two options, either of which would be generally acceptable to the staff, can be chosen to resolve the above difference. We will also review any other approaches that are suggested.

Option A: Assume that the site is located in the New-England Piedmont Tectonic Province and that the m_b = 5.7 New Brunswick earthquake is the maximum historical earthquake as defined in Appendix A 10CFR100 for this province. Calculate a site specific spectra using an m_b = 5.7 as the target magnitude. This can be accomplished by collecting a suite of strong motion response spectra recorded on rock sites, within distances of less than about 25 kilometers, for magnitudes of 5.7 + .50. Three such collections are currently available, although the target magnitudes are in some cases larger than 5.7 or the set of spectra do not include strong motion recordings from recent earthquakes. Two spectra were completed by Lawrence Livermore National Laboratory, the most recent of which was completed for the Seabrook site, and the third spectra was completed by the Tennessee Valley Authority as part of the Sequoyah review. Although one of the above could be used, the staff would recommend that development of a spectra specifically for an m_b = 5.7 using the most recent information that is available.

Option B: Extensive research is under way regarding the New Brunswick earthquake and its relationships to the New England-Piedmont Province. A large portion of this effort has been undertaken as a result of

230.0 GEOSCIENCES BRANCH, SEISMOLOGY SECTION (CONT'D)

reviews of the Seabrook and Maine Yankee sites. We recommend active attention and awareness of these studies. Using information provided by these studies, update and provide a complete discussion regarding the current choice of tectonic provinces. Include as a minimum the following information:

1) A discussion and justification of any association of the Central New Brunswick earthquake sequence with a specific geologic structure or fault within the meaning of Appendix A 10CFR100.

2) A discussion and justification of any province sub-division with respect to the New England Piedmont Tectonic Province.

3) An estimate of the ground motion and response spectra resulting from any province sub-division. Both peaks and spectra should be compared to that of the SSE. We note that the relationship of Murphy and O'Brien (1977) was used to arrive at a peak acceleration of 0.10g from an Modified Mercalli Intensity VII. It has been the staff's position to use the "trend of mean" of the relationship in Trifunac and Brady (1975) coupled with a Regulatory Guide 1.60 response spectrum, when intensity is used to describe the SSE. In addition, in recent OL reviews the staff has requested that the comparison of site specific spectra using the magnitude of the maximum historical earthquake which has not been associated with a fault or structure. It has been the staff's position that a Modified Mercalli VII corresponds to a m_b=5.3 (Nuttli and Herrmann 1978). Rock site specific spectra are available for use, however the staff recommends that you develop a spectra specifically for an m_b=5.3 using as much recent information as possible. The existing rock site specific spectra are discussed in the Wolf Creek (NUREG-0881), Perry (NUREG-0887) and Catawba (NUREG-0954) staff Safety Evaluation Reports.

The staff recommends that a meeting be held to specifically discuss these questions and the discussed options.

230.5
(SRP 2.5.2.1)

Update the FSAR to consider all pertinent seismologic information that have been developed in the region since publication of the FSAR. The most recent published seismologic reference in the FSAR bibliography is 1979. Considerable seismological research has been done in the northeastern U. S. since that time. Evaluate this information and determine whether or not it is significant to the seismological analysis of the site. This update should be completed using Standard Review Plan section 2.5.2 (NUREG-0800 July 1981) and should include as a minimum the following:

REQUEST FOR ADDITIONAL INFORMATION
MILLSTONE NUCLEAR POWER STATION UNIT 3
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230.0 GEOSCIENCES BRANCH, SEISMOLOGY SECTION (CONT'D)

a) Update Table 2.5-2-3 to include all earthquakes having Modified Mercalli intensity greater than IV or magnitude greater than 3 which have been reported to date in an area within 200 miles from the site. Include the seismic data provided by the Northeastern U. S.Seismic Network.

b) Update Table 2.5-2-4 to include all recorded and or felt earthquakes, to date, within a 50 mile radius of the site. Include applicable data provided by the Northeastern U. S. Seismic Network.

c) Provide a complete discussion of the 1981 microearthquake swarm sequence near Moodus, Connecticut and assess the significance of these events with respect to the OBE and SSE.

REQUEST FOR ADDITIONAL INFORMATION

MILLSTONE NUCLEAR POWER PLANT, UNIT 3

DOCKET NO. 50-423

PLANT PERSONNEL TRAINING

630.0 LICENSEE QUALIFICATION BRANCH

630.2 Discuss the program which will provide the training to Reactor Operators and Senior Reactor Operators in the following areas:

(a) Recognition of emergency conditions.

(b) Classification of observed emergency conditions in accordance with the Emergency Classification System.

(c) Notification of emergency to off-site authorities.

(d) Recommendation of protective actions to off-site authorities.

(e) Direction of station staff to take protective actions.

(Ref. NUREG-0800, Sections 13.2.1.I.B.1 and 13.2.1.II.1.b)

630.3 Provide the outlines of the courses, Fundamentals of Nuclear Training and Nuclear Plant Training. (Ref. NUREG-0800, Section 13.2.1.I.B.1)

630.4 With respect to the simulator training, provide the following information:

(a) The details of the program in accordance with the guidelines as specified in the Regulatory Guide 1.149.

(b) Discussion of the certification examination provided to demonstrate the candidate's ability.

(Ref. NUREG-0800, Section 13.2.1.I.B.2 and 12.2.1.II.2)

630.5 Discuss the qualification of the training instructors in the training program and the requalification program administered to the instructors in order to have them remain certified as instructors as specified in Enclosure 1 of H. R. Denton's March 28, 1980 letter to all power reactor applicants and licensees and in Item I.A.2.3 of NUREG-0737.

<u>REQUEST FOR ADDITIONAL INFORMATION</u>

<u>MILLSTONE NUCLEAR POWER STATION, UNIT 3</u>

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630.0 <u>LICENSEE QUALIFICATION BRANCH</u> - continued

630.6 Discuss the certification completed pursuant to Sections 55.10 (a)(6) and 55.33(a)(4) and (5) of 10 CFR Part 55. Provide the title of the individual who will certify the eligibility of individuals for licensing or renewal of license. (Ref. Enclosure 1 of H. R. Denton's March 20, 1980 letter, Section A.3)

630.7 Provide a commitment to comply with the following TMI-related requirements as specified in Item I.A.2.1 of NUREG-0737:

(a) As an operating license applicant, Millstone 3 is not subjected to the one year experience requirements for cold license SRO candidates. However, after one year of station operation, we will require Millstone 3 to comply with the one year experience requirement for hot license SRO applicants.

(b) The requirement for three months on shift experience for control room operators and SRO candidates as an extra person on shift is not required for cold license candidates and, hence, is not applicable to Millstone 3. However, we will require Millstone 3 to comply with this requirement for hot license candidates after three months of station operation.

630.8 Provide a detailed training program for mitigating core damage (as described in Item II.B.4 of NUREG-0737) in accordance with the guidance as specified in Enclosure 3 of H. R. Denton's letter dated March 28, 1980. Provide a listing of those individuals and their qualifications who must participate in the training program and provide a schedule for that training as related to the presently scheduled fuel load date.

630.9 Provide a detailed description of the training program for the Shift Technical Advisor in accordance with the guidance as specified in NUREG-0737, Appendix C.

630.10 With regard to fire brigade training, the program for drills should be revised to include all the guidelines as described in NUREG-0800, Section 13.2.2.II.C.6.A.iii(b).

630.11 Describe the training program provided for individuals (non-licensed operators) permitted to operate systems or equipment independently that could affect the quality of structures, systems, and components important to safety. (Ref. NUREG-0800, Section 13.2.2.IV.2)

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250.0 MATERIALS ENGINEERING BRANCH, Inservice Inspection Section

Review of the FSAR and request for additional information regarding preservice (PSI)/inservice (ISI) inspection program.

250.2 (5.2.4) (6.6) To complete our review, we will require the following information:

(1) A preservice inspection plan.

(2) All requests for relief with a supporting technical justification.

(3) An inservice inspection plan submitted six months after licensing.

250.3 (5.2.4) Plans for preservice and inservice examinations of the reactor pressure vessel welds should address the degree of compliance with Regulatory Guide 1.150, Rev. 1, as required by Generic Letter 83-15, dated March 23, 1983.

250.4 (6.6) Section 6.6.3 of the Millstone 3 Nuclear Power Station FSAR references ASME Code Section XI, Winter 1975 Addenda, and Paragraph IWA-2200 of the Summer 1976 Addenda which has not been referenced in 10 CFR 50.55a(b). Note that 10 CFR 50.55a(b) states that when applying the 1974 ASME Section XI Code Edition, only the Addenda through Summer 1975 may be used.

If Appendix III is used it must be used in conjunction with Summer 1978 Addenda or later Addenda as referenced by 10 CFR 50.55a(b).

When using Appendix III of Section XI for preservice or inservice examination of either ferritic or austenitic piping welds the following should be incorporated:

A. Any crack-like indication, regardless of amplitude, discovered during examination of piping welds or adjacent base metal materials should be recorded and investigated by a Level II or Level III examiner to the extent necessary to determine the shape, identity, and location of the reflector.

250.0 MATERIALS ENGINEERING BRANCH, Inservice Inspection Section - continued

Confirm that high energy fluid system piping between containment isolation valves will receive an augmented examination as follows:

A. Protective measures, structures, and guard pipes should not prevent the access required to conduct the inservice examination specified in the ASME Code, Section XI.

B. For those portions of high energy fluid system piping between containment isolation valves, the extent of inservice examination completed during each inspection interval (ASME Code Section XI) should provide 100% volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping.

C. For those portions of high energy fluid system piping enclosed in guard pipes, inspection ports should be provided in the guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in that portion of the guard pipe passing through the annulus of dual barrier containment structures.

D. For those items requiring ISI, a baseline or preservice examination for establishing the integrity of the original condition is also required by the ASME Code.

Confirm that the augmented examination for high energy system piping is maintained throughout the entire piping system up to the outboard restraint. If the restraint is located at the isolation valve, a classification change at the valve interface is acceptable.

Confirm that welds between outboard containment isolation valves and piping restraints are included in the PSI and ISI program plan as required.

250.5
(5.2.4)
(6.6)
FSAR Sections 5.2.4 and 6.6 state that the "preservice/inservice inspection program has been developed using the criteria of the ASME Code, Section XI, 1974 Edition, Summer 1975 Addenda along with existing construction drawings as they are issued. A PSI/ISI program will be finalized and submitted to the NRC pursuant to 10 CFR Part 50, at which time relief requests will be identified as necessary."

Indicate the anticipated date for submittal of this information and all requests for relief from impractical examinations. The Preservice Inspection Program should include the following information:

250.0 MATERIALS ENGINEERING BRANCH, Inservice Inspection Section - continued

B. The Owner should evaluate and take corrective action for the disposition of any indication investigated and found to be other than geometrical or metallurgical in nature.

In later editions of the ASME Code, Appendix III of Section XI, is specified for ferritic piping welds. If this requirement is not applicable (for example, for austenitic piping welds), ultra sonic examination is required to be conducted in accordance with the applicable requirements of Article 5 of Section V, as amended by IWA-2232. Discuss the criteria for applying Article 5 of Section V, as amended by IWA-2232. Provide a technical justification for any alternatives used such as Section XI, Appendix III, Supplement 7, for austenitic piping welds and discuss the following:

A. All modifications permitted by Supplement 7.

B. Methods of ensuring adequate examination sensitivity over the required examination volume.

C. Methods of qualifying the procedures for examination through the weld (if complete examination is to be considered for examinations conducted with only one side access).

250.6 (6.6) Clarify the statement in the Millstone FSAR Section 6.6, Augmented Inservice Inspection to Protect Against Postulated Piping Failures, which states "Welds in certain portions of high energy fluid system piping will receive supplemental examination."

High energy lines within the "break exclusion" of the containment penetration area, whether encased in guard pipes or not, must receive augmented preservice/inservice examination regardless of the requirements of Section XI of the ASME Code as discussed in SRP 3.6.1 and 3.6.2. However, high energy lines meeting the "modified break exclusion region" criteria need not be subjected to augmented preservice/inservice examination. The "modified break exclusion region" criteria may be applied in those special cases in which guard pipes are necessary, and it has been demonstrated to the satisfaction of the NRC that access to perform an examination is extremely difficult to achieve. In such areas the examination requirements may be eliminated provided the guard pipe is designed for the full dynamic effects of a longitudinal or circumferential break of the enclosed process pipe including jet impingement, pipe whip impact and environmental effects.

If the high energy fluid system piping does not meet the "modified break exclusion region" criteria, submit the required augmented preservice/inservice examination program for this piping.

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250.0 MATERIALS ENGINEERING BRANCH, Inservice Inspection Section - continued

A. For ASME Code Class 1 and 2 components, provide a table similar to IWB-2600 and IWC-2600 confirming that either the entire Section XI preservice examination was performed on the component or relief is requested with a technical justification supporting the request.

B. Where relief is requested for pressure retaining welds in the reactor vessel, identify the specific welds that did not receive a 100% preservice ultrasonic examination and estimate the extent of the examination that was performed.

C. Where relief is requested for piping system welds (Examination Category B-J, C-F, and C-G), provide a list of the specific welds that did not receive a complete Section XI preservice examination including drawing or isometric identification number, system, weld number, and physical configuration; e.g., pipe-to-nozzle weld, etc. Estimate the extent of the preservice examination that was performed. When the volumetric examination was performed from one side of the weld, discuss whether the entire weld volume and the heat affected zone (HAZ) and base metal on the far side of the weld were examined. State the primary reason that a specific examination is impractical; e.g., support or component restricts access, fitting prevents adequate ultrasonic coupling on one side, component-to-component weld prevents ultrasonic examination, etc. Indicate any alternative or supplemental examinations performed and method(s) of fabrication examination.

Detailed guidelines for the preparation and content of relief requests are attached as Appendix A to these questions.

250.7 (5.4.2.2) Confirm that you will comply with NUREG-0452, Rev. 4, Standard Technical Specification, which states that the PSI will be performed on 100% of the length of all tubes. This supersedes the guidance in R.G. 1.83 Rev. 1.

APPENDIX A

GUIDANCE FOR PREPARING REQUESTS FOR RELIEF FROM CERTAIN CODE REQUIREMENTS PURSUANT TO 10 CFR 50.55a(g)

A. Description of Requests for Relief

The guidance in this enclosure is intended to illustrate the type and extent of information that is necessary for "request for relief" of items that cannot be fully examined to the requirements of Section XI of the ASME Code. The preservice/inservice inspection program should identify the examination and pressure testing requirements of the applicable portion of Section XI that are deemed impractical because of the limitation of design, geometry, radiation considerations or materials of construction of the components. The request for relief should provide the information requested in the following section of this appendix for the examinations and pressure tests identified above.

B. Request for Relief from Certain Examination and Testing Requirements

Many requests for relief from examination and/or testing requirements submitted by the Applicants or 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 request(s) submitted with a justification such as "impractical", "inaccessible", or any other categorical basis, require additional information to permit 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 to make a proper evaluation and to adequately document the basis for granting the relief in the Safety Evaluation Report. Subsequent requests for additional information and delays in completing the review can be considerably reduced if this information is provided in the initial relief request submittal.

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

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

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

3. The ASME Code class.

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

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 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:

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

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

3. State the time period for which the request for relief is needed.

Technical justification or data must be submitted 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 examinations and/or tests prescribed in Section XI that not apply to your facility. A statement of "N/A" (not applicable) or "None" will suffice.

C. Request for Relief for Radiation Considerations

Exposures of 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 in the manner described above for inaccessibility and must be subsequently approved by the NRC Staff.

Some of the radiation considerations will only be known at the time of the examination or test. However, from experience at operating facilities, the Applicant or Licensee generally is aware 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 or test.

2. The radiation levels at the examination or test area.

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

4. A proposal for alternate examination techniques.

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

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

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

8. A discussion of the failure consequences of the weld which was not examined.

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451.0 METB-Meteorology Section

451.1 (SRP 2.3.3) Provide a meteorological data tape of onsite measurements in accordance with the attached meteorological data tape format. The data should include at least two consecutive annual cycles including the most recent one-year period as described in Regulatory Guide 1.70, Section 2.3.3.

451.2 (SRP 2.3.3) Describe any proposed supplemental meteorological monitoring, on or near the site, to aid in characterizing effluent transport during the occurrence of sea/land breeze circulations.

460.0 METB-Effuent Treatment Systems Section

460.5 (SRP 6.5.1) Table 6.5.1-1 of SRP 6.5.1, providing guidance on minimum instrumentation of ESF atmospheric cleanup systems, and Position C.2.g of Regulatory Guide 1.52, call for recorded indication of flow rates in the control room. Sections 6.5.1.5, 9.4, 1.8, and 1.9 of the FSAR indicate that no flow rate instrumentation is included and that flow can be estimated by fan curves verified every 18 months. The staff does not consider these fan curves as an acceptable alternative for instrumentation. Provide the necessary flow rate instruments for indication and recording on the four ESF systems (Section 6.5.1), including filtered and unfiltered flow operation.

460.6 (SRP 11.3) Acceptance Criterion II.A.6 and Position C.2.c of Regulatory Guide 1.140 call for remote recorded indication of flow rates in

460.0 METB, EFFLUENT TREATMENT SYSTEMS SECTION (CONT'D)

460.6 (SRP 11.3) normal ventilation systems. Sections 11.3, 9.4, and 1.8 of the FSAR excludes flow rate instrumentation. In order to assure representative monitoring/sampling by the off-line monitors on the normal ventilation systems, the staff recommends that flow rate instrumentation be provided in lieu of fan curves. Provide your justification for not including this instrumentation in the design.

460.7 (SRP 11.1) Acceptance Criteria, Requirement II.a, calls for meeting the Positions in Regulatory Guide 1.110. Section 11A of the FSAR does not provide the cost of borrowed money. Provide the cost of borrowed money expressed in percent that was used in your cost-benefit analysis for Appendix I to 10 CFR 50.

460.8 (SRP 11.1) Acceptance Criteria, Requirement II.b calls for meeting the Positions in Regulatory Guide 1.112. Section 11.3.3 of the FSAR states that the release points are indicated on Figure 1.2-1. Clarify if this should be Figure 1.2-2. Compare Figure 1.2-2 with Figure 2.1-5 of the ER. Provide the information requested in Appendix B to Regulatory Guide 1.112, item 6d, for the height and location relative to adjacent structures. Include the turbine building, warehouse, steam relief vents, and outside tank vents, for example. Provide some detail on how the release lines lead to the stack and the reactor plant vent.

460.9 (SRP 11.3) In our review, Subsection I.1 of SRP 11.3, the staff has located several P&ID differences from the description provided in the FSAR.

460.0 METB, EFFLUENT TREATMENT SYSTEMS SECTION (CONT'D)

460.9 (SRP 11.3) Provide clarification on the following:

a) The flow direction of the reactor plant gaseous waste and the condenser air removal lines to the radioactive waste line on Figure 9.3-5 (Sheet 1) at K-9 disagrees with Figure 10.4-2 (Sheet 2) at K-2.

b) The waste tank inlet and outlet are connected on Figure 9.3-6 (Sheet 1) at K-4 or Figure 11.2-1.

c) The BRS inlet line EM-109A at E-10 is not shown on Figure 9.3-9 (Sheet 1) at B-2.

d) Figure 11.5-1 indicates monitor 3HVR-RE1D. Should this be 3HVR-RE10A and for 10B, as given on Figure 3.8-62 (Sheet 4)?

e) Figure 9.4-3 (Sheet 2) at I-1 indicates ventilation release to Unit No. 2. How is this line monitored prior to gas release at Unit No. 3?

460.10 (SRP 11.3) Acceptance Criterion II.B.6 requires special provisions for radioactive gaseous wastes that have the potential for a hydrogen explosion. Sections 11.3.2.2, 9.3.5.2 and 15.7.2, describing the use of the GWS degassifier on input to the BRS, does not clearly state that hydrogenated reactor coolant can not be collected in the BRS holdup tanks. Your description of the BRS tanks does not indicate provisions, such as diaphragms on the holdup and test tanks, to assure that hydrogen gas can not mix with the air above the liquid with an open vent to the atmosphere.

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460.0 METB, EFFLUENT TREATMENT SYSTEMS SECTION (CONT'D)

460.10 (SRP 11.3)

a) Provide all possible flow directions for the four 3-way valves on the input to the BRS shown on Figure 9.3-4 (Sheet 1).

b) Provide an analysis of 3-way valve and control failure for the valving in a) above.

c) Describe how you plan to mitigate explosive gas mixtures of hydrogen and air in the BRS components and vents.

460.11 (SRP 11.3, BTP-ETSB 11.5)

Branch Technical Position ETSB 11.5 attached to SRP 11.3 provides the staff position for the analysis of the consequences of a failure in the waste gas system. Section 15.7.1 provides an analysis of the waste gas system based on line rupture upstream of the adsorbers. Table 1.9-2 states that an amendment is proposed. Clarify your position or provide a date for the additional information.

460.12 (SRP 6.5.1, 11.3)

Regulatory Guide 1.52 and 1.140 recommend leak testing of dampers used in ESF and non-ESF air filtration systems. In Table 1.8-1, pages 21 and 55, you have taken exception to testing every damper, and propose to test every type of damper. Since leakage is a function of valve size, we recommend that you determine Class B leakage rates for at least one damper of each size and type used in the ESF and non-ESF atmospheric cleanup systems, as an acceptable alternative.

460.13 (SRP 11.4)

SRP 11.4 calls for a description and design bases for solid radioactive waste handling systems. Table 1.8-1, page 17, of the FSAR, states that the charcoal adsorber in filtration trains will be

460.0 METB, EFFLUENT TREATMENT SYSTEMS SECTION (CONT'D)

460.13 (SRP 11.4) replaced using an external vacuum system. Provide the description, P&IDs, bases, and details relative to this equipment and describe the provisions for handling the charcoal adsorber media from the trains to the SWS.

460.14 (SRP 11.4) SRP 11.4 calls for a description and design bases for solid radioactive waste handling systems. Although Table 11.4-1 includes spent resins from the condensate demineralizers, we find no provisions for handling these spent resins (Section 10.4.6.2). Provide the information, design bases and the appropriate P&ID figure(s).

460.15 (SRP 11.3) Acceptance Criteria II.6 of SRP 11.3 calls for meeting the Positions of Regulatory Guides 1.140 and 1.143 for gaseous radwaste systems including normal ventilation subsystems that operate during normal operation, including anticipated operational occurrences. Sections 9.4.8.2.1 and 11.3 do not address the design bases for the condensate demineralizer regeneration and drain system in the Warehouse No. 5. The evaporator feed tanks and the neutralization sumps, for example, vent to the room air and, therefore, do not meet GDC 64. What provisions are included to prevent GWS waste from being drawn into the distillate tank. Provide the information, design bases and consider changes to the P&IDs.

460.16 (SRP 11.5) SRP 11.5 requires additional review of monitoring/sampling instrumentation at the FSAR-OL stage, as indicated by Subsection I.2.

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460.0 METB, EFFLUENT TREATMENT SYSTEMS SECTION (CONT'D)

460.16 (SRP 11.5) In order for the staff to complete a review for Millstone, Unit No. 3, we need the following information for the instrumentation described in Sections 11.5, Tables 11.5-1 and 11.5-2, and the figures.

a) Describe the monitors by type (e.g., in-line, off-line, etc.).

b) Describe the particulate monitor detector (11.5.2.2.9).

c) Specify redundancy, where applicable. Describe range switching.

d) Clarify tag numbers vs. plant name. Clarify use of letters A and B for different monitor detectors and different ranges or redundant monitors.

e) Locate the monitors by P&ID and building layout figure.

f) Locate the sampling points and sampling stations.

g) Describe the actions performed (manual or automatic) by signal from the monitor(s).

h) Describe the calibration laboratory and the sample analysis laboratory (i.e., location, purpose, facilities) for calibrating, repairing and testing monitors or providing radiochemical analysis.

i) Provide a composite figure of the liquid process and effluent monitors listed in Table 11.5-2.

j) Specify the sample locations for iodine adsorber sampling devices per Table 1 of SRP 11.5.

460.17 (SRP 11.4) The Subsection 11.4.2.4 of the FSAR states that a Process Control Program (PCP) will be implemented. Provide a commitment to submit

460.0 METB, EFFLUENT TREATMENT SYSTEMS SECTION (CONT'D)

460.17 (SRP 11.4) the PCP for solidification and packaging of wet solid waste at least six months prior to fuel loading, for review of the Technical Specification 3.11.3.

Provide an acceptable reference, such as a Topical Report, for the moble solidification system shared with Unit Nos. 1 and 2, and the dewatering equipment for spent resins. We need this information to confirm compliance with the BTP ETSB 11-3, attached in SRP 11.4, and to confirm your design bases (11.4.1, item 4) in the FSAR.

460.18 (SRP 11.3) Describe the hydrogen and oxygen monitor shown on Figure 11.3-2 (Sheet 2) at K-5. Provide the instrument and readout location, range, set point, sensor checks, and calibration. Will the instruments be nonsparking and capable of withstanding a hydrogen explosion as required by Acceptance Criterion 6 of SRP 11.3?

460.19 (SRP 11.5) Section 11.5.2.5 refers to the use of glass sampling bulbs. Experience at many nuclear power plants indicates that it is not prudent to use glass containers for collecting and transferring radioactive samples due to breakage of the container. Justify your position for using glass sampling bulbs.

Enclosure 1

PROPOSED FORMAT FOR HOURLY METEOROLOGICAL
DATA TO BE PLACED ON MAGNETIC TAPE

Use: 9 track tape (7 will be acceptable)

Standard Label which would include

Record Length = 160

Block Size (3200 - fixed block size)

Density (1600 BPI - 800 will be accepted)

Do Not Use: Magnetic tapes with unformatted or spanned records.

At the beginning of each tape, use the first five (5) records (which is the equivalent of ten cards) to give a tape description. Include plant name, and location (latitude, longitude) dates of data, information explaining data contained in the "other" fields if they are used, height of measurements, and any additional information pertinent to identification of the tape. Make sure all five records are included, even if some are blank. Format for the first five records will be 160A1. Meteorological data format is (I6, I2, I3, I4, 25F5.1, F5.2, 3F5.1). Decimal points should not be included when copying data onto the tape.

All data should be given to a tenth of a unit except solar radiation which should be given to a hundreth of a unit. This does not necessarily indicate the accuracy of the data. (e.g. wind direction is usually given to the nearest degree but record it with a zero in the tenth's place. That is 275 degrees would be 275.0 degrees and placed on the tape as 2750.) All nines in any field indicates a lost record (99999). All sevens in a wind direction field indicates calm (77777). If only two levels of data, use the upper & lower levels. If only one level of data, use the upper level.

Enclosure 1

MAGNETIC TAPE
METEOROLOGICAL DATA

LOCATION:

DATE OF DATA RECORD:

I6	Identifier (can be anything)	
I2	Year	
I3	Julian Day	
I4	Hour (on 24 hr clock)	
		ACCURACY
F5.1	Upper Measurements: Level = meters	
F5.1	Wind Direction (degrees)	______
F5.1	Wind Speed (meter/sec)	______
F5.1	Sigma Theta (degrees)	______
F5.1	Ambient Temperature (°C)	______
F5.1	Moisture: ______	______
F5.1	Other: ______	______
F5.1	Intermediate Measurements: Level = meters	
F5.1	Wind Direction (degrees)	______
F5.1	Wind Speed (meters/sec)	______
F5.1	Sigma Theta (degrees)	______
F5.1	Ambient Temperature (°C)	______
F5.1	Moisture: ______	______
F5.1	Other: ______	______

F5.1	Lower Measurements: Level = meters	
F5.1	Wind Direction (degrees)	______
F5.1	Wind Speed (meters/sec)	______
F5.1	Sigma Theta (degrees)	______
F5.1	Ambient Temperature (°C)	______
F5.1	Moisture: ______	______
F5.1	Other: ______	______
F5.1	Temp Diff (Upper-Lower) (°C/100 meters)	______
F5.1	Temp Diff (Upper-Intermediate) (°C/100 meters)	______
F5.1	Temp Diff (Intermediate-Lower) (°C/100 meters)	______
F5.1	Precipitation (mm)	______
F5.2	Solar Radiation ($cal/cm^2/min$)	______
F5.1	Visibility (km)	______
F5.1	Other: ______	______
F5.1	Other: ______	______

Mode:	9-track, 1600bpi, EBCDIC[1]	UNIT = TAPE9, DEN = 3	NT,PE,[illegible]
Internal Labels:	none[2]	LABEL = (,NL)	—
Record Format:	fixed length/blocked	RECFM = FB	RT = F, BT = K
Record Length:	160 characters	LRECL = 160	FL = 160
Blocking:	3200 characters/block	BLKSIZE = 3200	RB = 20

[1] 9-track, 800 bpi, EBCDIC or
7-track, 800 or 556 bpi, BCD
are also acceptable

[2] IBM standard labels are also acceptable

DO NOT USE

Variable length or unformatted records or records that span tape blocks.

e.g. IBM's RECFM = U or VBS
e.g. CDC SCOPE standard tape data format (use the S parameter on the REQUEST to avoid this)

Enclosure 2

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[redacted] NUCLEAR POWER FACILITY - METEOROLOGICAL DATA TAPE          DATA PERIOD : JULY 1   REC     1, LENGTH  3200
974-JUNE 1975    /////      TAPE CREATED MARCH 1977           LOCATION: LAT [redacted] DEG [redacted] MIN [redacted] SEC N     LO
NG [redacted] DEG [redacted] MIN [redacted] SEC W               WIND DIRECTION: 150 FT AND 33 FT     DATA TO NEAREST 1.0 DEG
REE        WIND SPEED: 150FT AND 33FT     DATA TO NEAREST 0.1 METER/SEC
DELTA T (150-33FT):    DATA TO NEAREST 0.1 DEGREE C/100 METERS                    TEMPERATURE: 150 AND
 33 FT     DATA TO NEAREST 0.1 DEGREE C                        THIS IS A COPY OF THE ORIGINAL IV? TAPE
WITH THE FORMAT CHANGED AND THE UNITS     CHANGED AS FOLLOWS: DEGREES F TO DEGREES C ; MPH TO METER
S/SEC ; FT TO METERS; OTHER FIELDS: WIND DIRECTION PERSISTENCE (%)
      74182 100  457 1980   22            22             73                                          100 1700    0
      214              0   19                                             74182 200  457 1800   20          221
      1000                                        100 1900    0            212         0   23
                            74182 300  457 1900   29         215           1000
      100 1700    0          201         0   34                                              74182 400  457
1890   35         211        1000                                         100 1700    9       195          0
  44                                              74182 500  457 1900    37          205      960
                             100 1600    9        191         0   41
      74182 600  457 1950    43         197       1000                                        100 1700    9
      186          0   31                                                 74182 700  457 2050   16        194
      1000                                        100 1800    9            187         0   20
                            74182 800  457 2100    34         204          960
      100 1800    4          201         0    9                                              74182 900  457
2140   31         218        860                                          100 2000   11       220          0
  -6                                             741821000  457 2170    38          245      830
                             100 2000   22        249         0  -11
      741821100  457 2420    51         258       710                                         100 2300   11
      262          0  -11                                                 741821200  457 2340   53        260
      750                                         100 2300   36            271         0  -12
                            741821300  457 2440    50         276          590
      100 2100   36          280         0  -12                                              741821400  457
2210   44         281        660                                          100 2100   27       205          0
 -12                                             741821500  457 1970    44          287      550
                             100 1900   22        291         0  -11
                                                                                             100 1700   22  REC     2, LENGTH  3200
      741821600  457 1930    42         288       730
      292          0   -9                                                 741821700  457 1910   45        292
      960                                         100 1800   22            295         0   -9
                            741821800  457 1930    42         291          960
      100 1800   18          292         0   -5
1740   32         286        960                                          100 1700
  -3                                             741822000  457 1800    29
                             100 1600    4        276         0    2
      741822100  457 1770    30         266       1000
      260          0   16                                                 741822200
      930                                         100 1700    4            248
                            741822300  457 1700    35         248          1000
      100 1500    4          239         0   25
1760   30         241        960                                          100 1700
  22                                             74183 100  457 1820    33
                             100 1600    4        228         0   20
      74183 200  457 1790    33         230       1000
      221          0   26                                                 74183 300
      940                                         100 1600    4            215
                            74183 400  457 1830    26         222          1000
      100 1500    4          214         0   20
1820   24         219        980                                          100 1500
  17                                             74183 600  457 1780    25
                             100 1400    4        208         0   25
      74183 700  457 1810    26         218       940
      205          0   23                                                 74183 800
      780                                         100 1600    9            214
```

The first data record gives the following:

Year: 1974
Day: 182
Time: 0100

Height: 45.7 meters
Wind Direction: 198 degrees
Wind Speed: 2.9 m/sec
Ambient Temperature: 22.5°C
Wind Direction Persistence: 73%

Height: 10.0 meters
Wind Direction: 170 degrees
Wind Speed: 0.4 m/sec
Ambient Temperature: 21.8°C
Wind Direction Persistence: 0%

Temperature Difference (Upper-Lower): 1.3°C/100 meters

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430.0 POWER SYSTEMS BRANCH, ELECTRICAL

430.3 (SRP 8.1) Criterion 50 of Appendix A to 10 CFR 50, IEEE Standard 485, Regulatory Guide 1.63 and branch technical positions ICSB 4, PSB-1 and PSB-2 have not been identified in Table 8.1-2 of the FSAR; thus, a positive statement as to compliance with these criteria and staff guidelines has not been provided in the FSAR. Provide a statement of compliance and justify areas of noncompliance.

430.4 (SRP 8.2) There are four transmission lines between Millstone and Hunts Brook Junction that follow a common right-of-way. It is the staff position that no other transmission lines cross over these four lines and that the four lines be physically separate and independent so that no single event such as a tower falling or line breaking will be able to simultaneously affect all circuits in such a way that none of the four circuits can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded. Line cross overs and physical separation of these four transmission lines has not been described in the FSAR. Provide the description and justify areas of noncompliance with the above staff position.

430.5 (SRP 8.2) The Millstone design provides two immediate access offsite circuits between the switchyard and the 4.16 kv Class 1E busses. It is the staff position that these two circuits be physically separate and independent such that no single event can simultaneously affect both circuits in such a way that neither can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded. The physical separation and independence of these two circuits has not been described or analyzed in the FSAR. Provide the description and analysis and justify areas of noncompliance with the above staff position. The analysis should include separation and independence of control and protective relaying circuits as well as the power circuits.

430.6 (SRP 8.2) The Millstone design arrangement provides two immediate access offsite circuits. One of these circuits utilizes a generator circuit breaker to isolate the turbine generator from the main and normal station service transformers. Other facilities that utilize generator circuit breakers have been required to perform verification testing. Provide a verification test program with results to demonstrate the breaker's ability to perform its intended function during steady-state operation, power system transients, and major faults. (See additional guidelines for this question in Attachment 1.)

430.0 POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

430.7 (SRP 8.2)

a. It is the staff position that the Millstone grid stability analysis must show that loss of the largest single supply to the grid does not result in the complete loss of preferred power. The analysis should consider the loss, through a single event, of the largest capacity being supplied to the grid, removal of the largest load from the grid, or loss of the most critical transmission line. The combined capacity of Millstone Units 1, 2, and 3 is to be supplied to the grid through the common Millstone switchyard. The combined capacity of the three units appears to be the largest capacity being supplied to the grid and should be considered in the Millstone grid stability analysis. Provide the results of the grid stability analysis when simultaneous loss of the combined capacity of Units 1, 2, and 3 is considered and justify areas of noncompliance with the above staff position.

b. There are four transmission circuits that connect the Millstone switchyard to the grid system. The four circuits are routed on two tower lines - two circuits per tower line. Section 8.1.3 of the FSAR indicates that a simultaneous failure of either of the two tower lines with only one circuit in service on the other tower line, may result in instability of Millstone generation. The applicant, in order to prevent instability, has installed a protection scheme to automatically reduce generator output at Millstone Unit 3. Describe the protection scheme.

430.8 (SRP 8.2 & 8.3.1) Each of the 4.16 kV Class 1E busses at Millstone is supplied power from preferred offsite and standby onsite circuits. It is the staff position that these circuits should not have common failure modes. Physical separation and independence of these circuits has not been described or analysed in the FSAR. Provide a description and analysis in accordance with Section 5.2.1(5) of IEEE Standard 308-1974.

430.9 (SRP 8.3.1, Appendix 8A) Section 8.3.1.1.4 of the FSAR indicates that a degraded voltage scheme with two-out-of-four logic is provided on each of the 4.16 kV Class 1E buses. Provide reference to electric schematic drawings that describe the degraded voltage scheme and provide a description, with voltage and time setpoints, to indicate how the Millstone design complies with the guidelines of position 1 of branch technical position PSB-1 (NUREG-0800 Appendix 8A) and provide justification for any deviations.

430.10 (SRP 8.3.1, Appendix 8A) As stated in Section 8.3.1.1.3 of the FSAR, the emergency generator load sequences (EGLS) has the capability to automatically reset during a sustained low voltage condition on the essential bus. It is the staff concern that this capability may unnecessarily

430.C POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

delay the connection of the required mitigating loads within the times allowed by the accident analysis. Address the staff concern, describe the design of the EGLS for automatic reset during sustained low voltage conditions, and describe how the design meets position 2 of branch technical position PSB-1 (NUREG-0800, Appendix 8A) and justify areas of noncompliance with position 2.

430.11 (SRP 8.3.1, Appendix 8A) The voltage levels at the safety-related loads should be optimized for the maximum and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power sources. Perform a voltage analysis and verification by actual measurement in accordance with the guidelines of positions 3 and 4 of branch technical position PSB-1 (NUREG-0800, Appendix 8A). Provide the voltage at the terminals of each Class 1E load as determined by analysis and by actual measurement for all modes of plant operation. Verify that all Class 1E loads will operate at or within design voltage limits under all condition of operation. Where terminal voltage determined by analysis is not adequate to meet the design voltage rating of the equipment, provide justification.

430.12 (SRP 8.3.1) Provide the results of a reliability analysis for the solid state load sequencer that demonstrates that overall reliability or capability of the onsite power system to supply power to safety loads on demand has not been significantly reduced by the use of solid state load sequencers.

430.13 (SRP 8.3.1, Appendix 8A) Section 8.3.1.1.3 of the FSAR indicates that diesel generator protective relaying is bypassed under accident condition in accordance with branch technical position ICSB 17. Provide drawing reference numbers that describe the design of the bypass circuitry, the 2-out-of-3 logic circuitry, and relaying that is not bypassed under accident conditions.

430.14 (SRP 8.3.1) Section 8.1.7 of the FSAR indicates that the diesel generator voltage (prior to connection of the first load block) may drop below the 75 percent minimum level permitted by position 4 of Regulatory Guide 1.9 (Revision 2). Provide justification for this exception to Regulatory Guide 1.9 and correct inconsistency between statements of compliance found in Sections 1.8 and 8.3.1.2.6 of the FSAR.

430.15 (SRP 8.3.1) Section 1.8 of the FSAR indicates that the Millstone design does not comply with position C11 of Regulatory Guide 1.9 Revision 2. Describe and justify the areas of noncompliance.

430.16 (SRP 8.3.1) Section 1.8 of the FSAR indicates that the Millstone design does not comply with position C2(a)4 of Regulatory Guide 1.108. The applicant has implied that the diesel generator load shedding test will be conducted using the 2000 hour rating for rejection of the single largest load and the continuous rating for complete loss of load. Justify use of continuous versus 2000 hour rating for complete loss of load.

430.0 POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

430.17 (SRP 8.3.1) Section 1.8 of the FSAR indicates that the Millstone design does not comply with positions C2(a)2 of Regulatory Guide 1.108. The capability of the diesel generator to accept the design accident load sequence is to be demonstrated under conditions as close to design as possible. Provide clarification and justification.

430.18 (SRP 8.3.1) Section 1.8 of the FSAR indicates that the Millstone design does not comply with position C2(a)3 of Regultory Guide 1.108. It appears that the full load carrying capability of the diesel generator may not be tested for the 2000 hour rating. Justify not testing at the 2000 hour rating and define the 2000 hour rating of the diesel generator at Millstone 3.

430.19 (SRP 8.3.1) Section 6.4.2 of IEEE Standard 387-1977 requires, in part, that the load acceptance test consider the potential effects on load acceptance after prolonged no load or light load operation of the diesel generator. Provide the results of load acceptance tests or analysis that demonstrates the capability of the diesel generator to accept the design accident load sequence after prolonged no load operation. This capability should be demonstrated over the full range of ambient air temperatures that may exist at the diesel engine air intake. If this capability cannot be demonstrated for minimum ambient air temperature, conditions, describe design provision that will assure an acceptable engine air intake temperature during no load operation.

430.20 SRP 8.3.1) In accordance with Section 5.6.2.2(1) of IEEE Standard 387-1977, Section 5 of IEEE Standard 338-1977, and position C2a(8) of Regulatory Guide 1.108, it is the staff position that the diesel generator, when in the test mode and parallel with the offsite power system, be capable of responding to an accident signal. Describe how the Millstone design meets the staff position and justify areas of noncompliance.

430.21 (SRP 8.3.1) The FSAR does not provide a complete description of how design criteria, testing, and analysis is being implemented for the onsite power system diesel generator at Millstone. Provide the description that as a minimum addresses each section of IEEE Standard 387-1977 as supplemented by Regulatory Guide 1.9 and each section of Regulatory Guide 1.108.

430.22 (SRP 8.3.1 Appendix 8A) Section 8.3.1.1.3 of the FSAR describes the surveillance instrumentation provided to monitor the status of the diesel generator. Expand the FSAR to describe how the Millstone design complies with the guidelines of branch technical position PSB-2 (NUREG-0800 Appendix 8A) and provide justification for any deviations.

430.0 POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

430.23 (SRP 8.3.1, 8.3.2) The FSAR does not provide a complete description of how physical separation of Class 1E systems will be accomplished at Millstone Unit 3. Provide a description that as a minimum addresses each section of IEEE Standard 384-1974 as augmented by Regulatory Guide 1.75. The description should include but not be limited to the following items.

1. Separation between redundant Class 1E and between Class 1E and non-Class 1E cables located in cabinets and control switchboards.
2. Separation of actuated Class 1E equipment.
3. Separation of sensors and sensor to process connections.
4. Separation of cable entrances to control switchboards.
5. Identification of cables inside cabinets.
6. Separation between Class 1E conduit and non-Class 1E cable trays.
7. Separtion between redundant and between Class 1E and non-Class 1E terminations
8. Compatibility with mechanical systems

430.24 (SRP 8.3.1, 8.3.2) You imply (by taking exception to position C5 of Regulatory Guide 1.75) that the Millstone design for cable separation does not meet the minimum separation distances specified in IEEE Standard 384-1974. Identify each circuit and the location where it does not meet the minimum separation distance. Provide an analysis for each location identified that demonstrates the adequacy of the lesser separation.

430.25 (SRP 8.3.1, 8.3.2) In Section 1.8 of the FSAR, you identified the following exception or clarification to position C4 of Regulatory Guide 1.75:

"Associated circuits are identified by the same color code as the Class 1E circuit with which they are associated. This color code exists up to and including an isolation device."

Position C4 of Regulatory Guide 1.75 requires that associated circuits (up to and including an isolation device) be subject to all requirements placed on Class 1E circuits unless it can be demonstrated that the absence of such requirements cannot significantly reduce the availability of Class 1E circuits. The applicants clarification or exception implies that associated circuits meet only the color code requirement versus all requirements of Class 1E circuits. Provide justification for the implied exception to position C4.

430.0 POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

430.26 (SRP 8.3.1, 8.3.2) In Section 1.8 of the FSAR, you identified an exception to position C10 of Regulatory Guide 1.75. Class 1E cables are to be marked at intervals not exceeding 15 feet versus 5 feet as required by the Regulatory Guide. In justification of the exception you documented that the 5 foot requirement is a typographical error in Regulatory Guide 1.75 which has been confirmed by the NRC. The staff does not consider the 5 foot requirement to be a typographical error. A 5 foot maximum marking distance is considered necessary to facilitate easy visual verification that the cable installation is in conformance with separation criteria. Justify noncompliance with the 5 foot marking requirement for cables located in raceways as well as inside panels or cabinets.

430.27 (SRP 8.3.1, 8.3.2) In section 1.8 of the FSAR, you imply taking exception to Section 5.1.4 of IEEE Standard 384-1974. Where plant arrangements preclude maintaining the minimum separation distance, you state that redundant circuits will be routed in non-solid raceways. Solid versus non-solid raceways are required by Section 5.1.4 of IEEE 384. Provide clarification and justification for noncompliance.

430.28 (SRP 8.3.1, 8.3.2) In Section 1.8 of the FSAR, you imply taking exception to position C12 of Regulatory Guide 1.75. Position C12 indicates that (1) power supply feeders to instrument and control room distribution panels installed in enclosed raceways should not be considered acceptable, (2) traversing power circuits separated from other circuits in the cable spreading area by a minimum distance of 3 feet and barriers should not be considered acceptable, and (3) traversing power circuits routed in imbedded conduit which in effect removes them from the cable spreading area should be considered acceptable.

Power circuits that traverse the cable spreading area at Millstone are installed in enclosed raceways (rigid steel conduit). In accordance with position C12 of Regulatiry Guide 1.75, the routing should not be considered acceptable. Justify the adequacy of the proposed routing in steel conduit.

430.29 (SRP 8.3.1, 8.3.2) In Section 1.8 of the FSAR, you imply taking exception to position C8 of Regulatory Guide 1.75. Redundant Class 1E cables or Class 1E and non-Class 1E cables are to be routed separately within the same confined space such as the cable tunnel at Millstone that is effectively unventilated. It is the staff concern that routing of cables along opposite sides of a confined space may not provide adequate separation. Identify all confined spaces at Millstone that are effectively unventilated and have Class 1E cables routed in the space. Describe the separation and justify the adequacy of the separation between redundant Class 1E circuits and between Class 1E and non-Class 1E circuits.

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430.0 POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

430.30 (SRP 8.3.1, 8.3.2)

In Section 1.8 of the FSAR, you take exception to position C16 of Regulatory Guide 1.75 and Section 5.6.2 of IEEE Standard 384-1974. Minimum separation between redundant Class 1E cables or between Class 1E and non-Class 1E cables is identified to be 1 inch versus 6 inches inside control switchboards and instrument cabinets. Provide the analysis that demonstrates the adequacy of 1 inch minimum separation.

430.31 (SRP 8.3.1, 8.3.2)

A third or spare charging pump may be connected to either Class 1E bus 34C or 34D. Describe the interlocks that preclude two charging pumps from being powered from the same Class 1E bus and preclude redundant buses from being tied together. Provide a similar description for the third or spare reactor plant component cooling pump.

430.32 (SRP 8.3.1, 8.3.2)

You state in Section 9.5.4.3 of the FSAR, in part, that one fuel oil transfer pump on each fuel oil storage tank is arranged to allow transfer from the A electrical bus to the B electrical bus, or visa versa, by means of a 480-volt, seismically qualified Class 1E transfer switch manually operated under administrative control. It appears that the Millstone design includes provision for manually transferring loads between redundant Class 1E divisions other than those described in Chapter 8 of the FSAR.

It is the staff position that the designs of each interconnection should prevent a single failure or inadvertent closure of one interconnecting device from compromising division independence. An acceptable design includes a minimum of two series connected disconnect devices that are physically separated, interlocked, administratively kept normally open, and annunciated in the control room upon closure. Identify all interconnections between redundant distribution systems; describe how each interconnection meets the above staff position; and justify areas of noncompliance.

430.33 (SRP 8.3.1, 8.3.2)

The electrical 4.16 kV one line diagrams (No. 12179-1F-1K, 1L, 1M, and 1N) included in Section 1.7 of the FSAR indicate that there is an interconnection between Millstone Unit 2 and Unit 3 and between redundant buses 34C and 34D. Figure 8.1-3 and Section 8.0 of the FSAR indicate no interconnection between Units 2 and 3 or between buses 34C and 34D. Provide clarification and justify the interconnection.

430.34 (SRP 8.3.1, 8.3.2)

Section 8.3.1.1.4(8) of the FSAR indicates that piping subject to freezing or boron precipitation are electrically heat traced. Two heat tracing circuits are provided for each pipe one heat trace circuit is connected to Class 1E division A while the other circuit is connected through an isolation transformer to division B. Provide a description and justification of the physical and electrical independence between the two heat tracing circuits and between the redundant Class 1E divisions.

430.0 POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

430.35 (SRP 8.3.1, 8.3.2) In Section 8.3.2.1.1 of the FSAR you state that battery charger 5 is powered from a Class 1E emergency bus, furnishes dc power to nonsafety loads, and meets all the requirements of an isolation device. Provide test results and/or analysis that demonstrates that any failure or combination of failure or malfunction in the nonsafety circuits will not cause unacceptable influence on Class 1E circuits. In addition, define the requirements for this isolation device.

430.36 (SRP 8.3.1, 8.3.2) In Section 1.8 of the FSAR, you take exception to position C1 of Regulatory Guide 1.75. Interrupting devices, actuated only by fault current, are used as isolation devices. It is the staff position that non-essential circuits (powered from Class 1E buses) be either disconnected by an accident signal or connected to the Class 1E bus through two series connected and coordinated interrupting devices actuated by fault current. Identify and describe each non-Class 1E or nonessential circuit that is to be isolated from Class 1E circuits by an interrupting device actuated only by fault current and that is in noncompliance with the above staff position. In order to justify noncompliance with the staff position, provide the test or analysis that demonstrates that each non-Class 1E circuit identified will not cause unacceptable influence on Class 1E circuits.

430.37 (SRP 8.3.1, 8.3.2) In Section 8.3.2.1.1 of the FSAR you state that nonsafety 480 volt stub bus 32-3T (that supplies power to a number of nonsafety dc loads located in a nonseismic building) is powered from a Class 1E bus and is automatically shed upon loss of offsite power. It is the staff position that this stub bus should also be automatically shed upon an accident signal. Provide justification for noncompliance with this staff position.

430.38 (SRP 8.3.1, 8.3.2) Non-Class 1E NSSS loads are connected to the Class 1E 120V vital ac buses through transformers. You have stated that these transformers are qualified as isolation devices. Provide test results and/or analysis that demonstrates that any failure or combination of failures (including hot short) in the nonsafety circuits will not cause unacceptable influence on any Class 1E circuits. In addition, provide a description of the non-Class 1E load with respect to its size and the capacity and capability of the Class 1E system to supply the non-Class 1E load.

430.39 (SRP 8.3.1, 8.3.2) In Section 8.3.2.1.2.1 of the FSAR you state that the battery charger is sized based on normal bus loads. Position C.1.b of Regulatory Guide 1.32 on the other hand, requires that the battery charger be sized based on the largest combined demands of the various steady-state loads. Provide clarification and description as to how the Millstone design meets position C1b of Regulatory Guide 1.32. The clarification and description should address but not be limited to the following items: size of each load identified in Table 8.3-4, type of load (normal, continuous, transients or momentary),

430.0 POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

identification of non-Class 1E loads, and the inconsistencies between Tables 8.3-4 and 8.3-5 and between Table 8.3-4 and the 125V dc one line diagrams for battery buses 301A-1 and 301B-1 presented in Section 1.7 of the FSAR.

430.40 (SRP 8.3.1, 8.3.2) Recent operating experience has shown that an incompatibility between the battery rack and the battery may cause cracking of the battery case. The cracking may be caused in part by the improper support at the battery stress points (the plate support bridge). Describe the relationship between the plant support bridge and the battery rack supports and how the seismic qualification test program encompasses the subject stress-related aging of the battery case.

430.41 (SRP 8.3.2) Loads connected to the dc bus may be subject to voltage variations from 90 to 143 volts due to battery discharge and equalizing charge. It is the staff position that dc loads be designed and qualified to to operate when subject to these voltage variations. Describe the extent of compliance of the Millstone design to this position and justify any areas of noncompliance.

430.42 (SRP 8.3.2) Describe the extent to which the recommendations of IEEE Standards 338, 450, 484 and 485 and Regulatory Guides 1.118, 1.128, and 1.129 have been followed in regard to testing, maintenance, installation, and sizing of Class 1E batteries and dc systems.

430.43 (SRP 8.3.2) The specific requirements for dc power system monitoring derive from the generic requirements in Section 5.3.2(4), 5.3.3(5), and 5.3.4(5) of IEEE 308-1974, and in RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." In summary, these general requirements state that the dc system (batteries, distribution systems, and chargers) shall be monitored to the extent that it is shown to be ready to perform its intended function.

It is the staff position that the following indications and alarms of the Class 1E direct current power system status shall be provided in the control room:

- battery float charge (ammeter)
- battery circuit output current (ammeter)
- battery charger output current (ammeter)
- dc bus voltage (voltmeter)
- battery discharge alarm
- dc bus overvoltage alarm
- dc system ground alarm
- battery disconnect open alarm
- battery charger disconnect open alarm
- battery charger failure alarm (one alarm for a number of abnormal conditions which are usually indicated locally)

430.0 POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

The staff has concluded that the above-cited monitoring, augmented by the periodic test and surveillance requirements that are included in the Technical Specifications, provide reasonable assurance that the Class 1E dc power system is ready to perform its intended safety function.

Describe the extent to which the above staff position is followed and justify areas of noncompliance.

430.44 (SRP 8.3.2) In Section 8.3.2.1 of the FSAR you state that power will be available to dc system loads for at least two hours in the event of loss of all ac power. After 2 hours you have assumed that ac power is either restored or that the emergency generators are available to energize the battery chargers. Based on the staff's review of recent applications, this period for restoration of ac power appears to be too short. Provide the basis and operational experience data for the assumption that ac power can be restored within two hours.

Emergency procedures and training requirements for station blackout events are described in generic letter 81-04. Provide a statement of compliance with these generic requirements.

430.45 (SRP 8.3.1, 8.3.2) Provide a description as to how the onsite Class 1E power system meets the guidelines of IEEE Standard 338 and Regulatory Guide 1.118. Identify and justify deviations.

430.46 (SRP 8.3.1, 8.3.2) In Section 8.3.1.1.4 (items 2 and 4) of the FSAR you indicate that primary and backup containment electrical penetration protection is provided only where the available fault-current exceeds the current-carrying capabilities of penetration conductors for loads connnected to safety related buses that are not qualified to the containment accident environment. This design for containment electrical penetration protection does not meet the guidelines of position 1 of Regulatory Guide 1.63. Position 1 requires: a) primary and backup protection where maximum available fault-current exceeds the current-carrying capability of the penetration versus capability of the conductors and b) all conductors, that pass through containment electric penetrations, to have primary and backup protection versus only those that are connected to safety related buses and loads that are not qualified to the containment accident environment.

a. Provide justification for noncompliance with the guidelines of position 1 of Regulatory Guide 1.63.

b. Describe how the Millstone design complies with each of the guidelines of IEEE Standard 317-1976 as augmented by Regulatory Guide 1.63 and provide justification for any deviations.

c. Provide coordinated fault-current versus time curves for each representative type cable that penetrates primary containment.

430.0 POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

For each cable, the curves must show the relationship of the fault carrying capability between the electric penetrations, the primary overcurrent protective device, and the backup overcurrent protective device.

d. Provide the test report with results that substantiates the capability of the electrical penetration to withstand the total range of time versus fault current without seal failure for worst case environmental conditions.

430.47 (SRP 8.3.1, 8.3.2) In Section 1.8 of the FSAR you provided clarifications as to how the guidelines of Regulatory Guide 1.63 are to be implemented in the Millstone design for protection of containment electrical penetrations. The clarifications state that overcurrent protective devices are not required to comply with criteria listed in IEEE 279 (except Section 4.2) and need not be Class 1E or seismically qualified. Position 1 of Regulatory Guide 1.63, on the otherhand, states that overcurrent protective devices should conform to the criteria of IEEE 279. The proposed Millstone design does not meet the guidelines of position 1 of Regulatory Guide 1.63. Provide justification for noncompliance.

430.48 (SRP 8.3.1, 8.3.2) Describe how the Millstone design complies with the guidelines of NUREG-0737 items II.E.3.1 and II.G.1 and justify areas of noncompliance.

430.49 (SRP 8.3.1, 8.3.2) In Section 8.3.1.4.1 of the FSAR, you define design criteria for independence and availability of Class 1E systems. The definition includes the statement that "separation of equipment is maintained to prevent loss of redundant features for single events and accidents." Similarly, in Section 8.3.1.1.2 of the FSAR, you state that redundant Class 1E buses are physically and electrically separated so that any credible event which might affect one bus will not jeopardize proper operation of the other bus.

The above statements imply that, with sufficient separation, only one of the redundant Class 1E divisions need be protected from the effects of any single event or accident. Such a design does not meet the protection requirements of GDC 2 and 4, the single failure requirement of GDC 17, or the guidelines of IEEE Standard 308-1974. Define all credible events, accidents or design basis events and describe how each Class 1E power system component is designed and qualified to withstand (or is protected from) the effects of each defined credible event. Defined credible events should include but not be limited to: Design basis events listed in Table 1 of IEEE Standard 308-1974 and failures of non-Class 1E or nonseismic Category I structures, systems, or components. Where separation is used to prevent loss of redundant features from any single event or accidents, justify noncompliance with the requirements of GDC 2, 4 and 17.

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430.50 (SRP 8.3.1, 8.3.2) In item 5 of Section 8.3.1.4.2 of the FSAR you state, in part, that Class 1E cables of only one train will be installed in potential missile-producing areas or adequate missile protection will be provided when Class 1E cables of redundant trains are installed in missile-producing areas. Based on this statement, it appears that Class 1E equipment and cables are not protected from the effects of accident generated missiles. Identify each Class 1E equipment and cable not protected from the potential effects of missiles. For each cable or equipment identified, provide the results of an analysis that demonstrates that the number of circuits and equipment remaining is sufficient so that the protective functions required can be accomplished assuming a single failure.

430.51 (SRP 8.3.1, 8.3.2) Identify all electrical equipment, both safety and nonsafety, that may become submerged as a result of a LOCA. For all such equipment that is not designed and qualified for service in such an environment provide analysis to determine the following:

1. The safety significance of the failure of this electrical equipment (e.g. spurious actuation or loss of actuation function) as a result of flooding.
2. The effects on Class 1E power sources serving this equipment as a result of such submergence; and
3. Any proposed design changes resulting from this analysis.

430.52 (SRP 8.3.1, 8.3.2) Provide additional information regarding the power sources supplied to the RHR isolation valves. The staff's position is that a single failure of a power supply should not prevent isolation of the RHR when RCS pressure exceeds the design pressure of the RHR system. Additionally, loss of a single power supply should not result in the inability to initiate at least one 100 percent RHR train.

430.53 (SRP 8.3.1, 8.3.2) In Section 8.3.1.1.2(1) of the FSAR, you state that the Class 1E switchgear rooms contain automatic fire protection systems. Provide indication in Section 8.0 of the FSAR that Class 1E equipment in all areas of the plant are either protected from automatic fire protection effluent or designed and qualified to operate in the environment that may be caused by the effluent.

430.54 (SRP 8.3.1, 8.3.2) In Sections 8.3.1.1.2 and 8.3.1.1.3 of the FSAR, you state that controls for the diesel generator and Class 1E circuit breakers are located in the control room and at remote locations. Describe the electrical independence between these two controls.

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430.0 POWER SYSTEMS BRANCH, ELECTRICAL (CONT'D)

430.55 (SRP 8.3.1, 8.3.2)

Section 8.3.1.1.4 of the FSAR implies that there may be safety and nonsafety related electrical equipment connected to Class 1E power supplies and located inside of containment that are not designed and qualified to the containment accident environment. For all such equipment provide analysis to determine the following:

1. The safety significance of the failure of this equipment (e.g. spurious actuation or loss of activation function).

2. The effects on Class 1E power sources serving this equipment.

3. Any proposed design changes resulting from this analysis.

ATTACHMENT 1
ADDITIONAL GUIDELINES FOR QUESTION 430.6
Generator Circuit Breakers/Load Break Switches

A. Background

Generator circuit breakers have been used in recent nuclear generating station designs (McGuire, Catawba) as a means of providing immediate access of the onsite ac power systems to the offsite circuits by isolating the unit generator from the main step-up and unit auxiliary transformers and allowing backfeeding of power through these circuits to the onsite ac power system. Generator load break switches can be used as a means of providing access to the offsite circuits as described above, but only on a delayed basis. Since this is a new design feature, the staff made the use of generator circuit breakers and load break switches a generic item no. B-53. In the case of McGuire and Catawba, an expert consultant was retained to evaluate the generator circuit breaker verification testing program and its results. These requirements are formalization of the results of that extensive work. Also requirements for the load break switches are incorporated, as these devices have some common functional requirements as generator breakers as described above.

The staff has made a determination that only devices which have the capability of interrupting the system maximum available fault current i.e., circuit breakers will be approved as a means of isolating the unit generators from the offsite power system in order to provide immediate access in accordance with GDC 17. This is necessary because a non fault-current interrupting device i.e., load break switch must delay its trip for electrical faults until the switchyard circuit breakers have interrupted the current. Following opening of the load break switch, the switchyard circuit breakers must then be reclosed to establish offsite power to the unit. A generator circuit breaker, however, could interrupt the fault current and isolate the unit generator at the same time, maintaining continuous power to the onsite ac power system.

B. Specific Guidelines

1. Only devices which have maximum fault current interrupting capability i.e. circuit breakers can be used to isolate the unit generator from the offsite and onsite ac power systems in order to provide immediate access for the onsite ac power system to the offsite source. Generator load break switches can only be used for isolating the unit generator for the purpose of providing a delayed access offsite source.

2. Generator circuit breakers should be designed to perform their intended function during steady-state operation, power system transients and major faults; tests should be performed on the circuit breaker to verify these capabilities. As a minimum, the following performance tests and capabilities should be demonstrated:

a. Dielectric Tests

The circuit breaker should be given dielectric strength tests in accordance with the requirements and ratings contained in the applicable ANSI C37 series standards.

b. Load Current Switching

For applications which use only one generator circuit breaker, the circuit breaker should be cycled through 40 load interruption operations (a lesser number requires suitable justification) at a current equal to the normal full load continuous current rating of the circuit breaker. For applications which utilize two generator circuit breakers in a parallel circuit, the circuit breaker should be given 40 load interruption operations (a lesser number requires suitable justification) at a current equal to twice the normal full load continuous current rating of the circuit breakers. The procedures and acceptance criteria utilized for this test should be based upon those given in ANSI C37.06 and C37.09.

c. Fault Current Interrupting Capability

The circuit breaker should have, as a minimum, the capability of interrupting the maximum assymetrical and symmetrical fault current available at the instant of primary arcing contact separation. This current should be calculated by assuming a bolted three phase fault at a point on the system which causes the maximum amount of fault current flowing through the generator circuit breaker. The fault current interrupting capability (short circuit current rating) of the circuit breaker should be demonstrated by performing a series of tests similar to those called for in ANSI C37.04 and C37.09. The tests should include close/open (CO) operations and should be performed at the circuit breaker minimum rated air pressure and control voltage and with a rate of rise of recovery voltage not less than the following rated value.

d. Maximum Rate of Rise of Recovery Voltage

The rated maximum rate of rise of recovery voltage (RRRV) of the circuit breaker should not be less than the maximum RRRV imposed on the breaker in the circuit in which it is used.

e. Short-Time Current Carrying Capability

The circuit breaker should have the capability of carrying a fault current for the length of time that the fault exists assuming failure of the primary protective device to clear it. The fault current chosen should be that due to a fault on the system at a point which causes the largest I^2t heating of the circuit breaker. The short-time current carrying capability should be demonstrated with a current carrying test.

f. Momentary Current Carrying Capability

The circuit breaker should have the capability of carrying the maximum crest value of current calculated for the worst case bolted three phase fault on the system. This capability should be demonstrated by test.

g. Transformer Magnetizing Current Interruption

The circuit breaker interruption of an unloaded station main and/or auxiliary transformer magnetizing current should not generate excessively high surge voltages which could damage the connected bus and transformer insulation. This should be verified by test.

h. Thermal Capability

The thermal capability of the circuit breaker should be demonstrated by a test at its continuous current rating. The test should be in accordance with the requirements and ratings contained in ANSI C37.04 and C37.09. For applications which use two generator circuit breakers in parallel, a test should be conducted to determine the time to reach the maximum permissable temperature on the most limiting component of the breaker when going from the rated continuous current to twice rated continuous current.

i. Mechanical Operation Test

A sufficient number of no-load mechanical operations should be performed by the circuit breaker to provide a reasonable indication of its mechanical reliability and life. The demonstrated life should be adequate for the plant life expectancy.

3. The availability of offsite power to the onsite loads for designs utilizing generator circuit breakers should be no less than comparable designs which utilize separate offsite power transformers to supply offsite power to the station loads. In this regard the trip selectivity between the generator circuit breakers and the switchyard high voltage generator circuit breakers should insure against unnecessary tripping of the switchyard generator circuit breakers during abnormal events in order to maintain offsite power to the station loads.

4. Load break switches should be designed to perform their

intended function during steady-state operation, power system transients and major faults. Except for item 2.C, the switches should have the same capabilities as defined in position 2 for generator circuit breakers. In addition the symmetrical interrupting capability of the load break switch should be at least equal to the maximum identified peak loading capability of the station generator.

REFERENCES

1. Safety Evaluation Report related to operation of McGuire Nuclear Station, Units 1 and 2 NUREG-0442 dated March 1978.

2. FSAR McGuire Nuclear Station Docket 50-369/370

3. FSAR Catawba Nuclear Station Docket 50-413/414

4. ANSI Standard C37.04, Rating Structure for AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis

5. ANSI Standard C37.06, Preferred Ratings and Related Required Capabilities for AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis

6. ANSI Standard C37.09, Test Procedure for AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis

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640.0 PROCEDURES AND SYSTEMS REVIEW BRANCH

640.1 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Regulatory Guide 1.70) specifies that Section 14.2.7 of an FSAR include a list of Regulatory Guides that are applicable to the facility's initial test program and include justification for any exceptions to Regulatory Positions stated in those Guides.

The enclosed list of applicable Regulatory Guides shows the specific revisions which should be addressed in Section 14.2.7 of the FSAR. Section 14.2.7 should be amended as necessary.

1.18	Rev. 1
1.20	Rev. 2
1.37	Rev. 0
1.41	Rev. 0
1.52	Rev. 2
1.68	Rev. 2
1.68.2	Rev. 1
1.68.3	Rev. 0
1.72	Rev. 1
1.79	Rev. 1
1.95	Rev. 1
1.108	Rev. 1
1.116	Rev. 0-R
1.140	Rev. 1

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640.0 PROCEDURES AND SYSTEMS REVIEW BRANCH - continued

640.1 (cont.) Regulatory Guides which need not be addressed in 14.2.7:

1.30 Listed in Standard Review Plan, but has been withdrawn.

1.80 Listed in Standard Review Plan, but has been withdrawn effective April 20, 1982, in favor of Regulatory Guide 1.68.3.

1.128 Listed in Standard Review Plan, but none of the plants are required to conform since their construction permit docket dates were all before the implementation date of December 1, 1977.

1.139 Listed in Standard Review Plan, but none of the plants are required to conform since their construction permit docket dates were all before the implementation date of January 1, 1978.

1.9 Referenced in Regulatory Guide 1.68, but diesel testing is covered in 1.108 and this guide does not directly bear on the initial plant test program.

1.104 Referenced in Regulatory Guide 1.68, but withdrawn August 16, 1979.

NOTE: All other Regulatory Guides referenced in Regulatory Guide 1.68 or listed in the Standard Review Plan are contained in the Applicable Regulatory Guide List.

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260.0 QUALITY ASSURANCE BRANCH

260.1 (1A5)* Describe the criteria for determining the size of the QA organization(s) including the inspection staff(s).

260.2 (1A5 & 17.2.1)** Figure 1.6 of NU-QA-1 identifies onsite and offsite organizational elements. The figure (per letter of 8/17/82) appears to reflect more the construction phase than the operations phase. Update the figure to include functions such as inspection, test, instrumentation and control, operations, maintenance for the operations phase of Millstone 3. The figure shows both Operations QA and Station Superintendent Millstone as being offsite. Clarify. Identify which positions on the figure are in the NUSCO organization and which positions are in the NUPOC organization.

260.3 (1B1) Section 1.3 of NU-QA-1 describes responsibilities for the Millstone 3 QA program. It states that overall responsibility is assigned to the Director-Nuclear Engineering and Operations Services. This seems to conflict with the Policy Statement which says that "corporate authority is delegated to the NUSCO Manager, Quality Assurance for the preparation and administration of the Quality Assurance Program." Clarify. Assigning the overall responsibility for the QA program to the Director-Nuclear Engineering and Operations Services is acceptable if this position has no other duties or responsibilities unrelated to QA that would prevent his full attention to QA matters and has responsibility for approval of the QA Manual(s). Clarify in the FSAR whether this is indeed the case.

260.4 (General) Clearly differentiate between the authority, duties, and responsibilities of Operations QA organization under the Manager Quality Assurance and the Quality Assurance Supervisor Millstone under the Station Superintendent Millstone.

*Alphanumeric designations in parentheses refer to the acceptance criteria in Section 17.1 of the Standard Review Plan (NUREG-0800, July 1981)

**Numerical designations in parentheses refer to acceptance criteria in Section 17.2 of the Standard Review Plan (NUREG-0800, July 1981).

260.0 QUALITY ASSURANCE BRANCH (CONT'D)

260.5 (1B2) Identify (by organization) the individuals responsible for the verification of conformance to requirements and discuss their independence from the "doers," their training, and their qualification requirements.

260.6 (1B3&4 The last paragraph of 1.4.3.2c of NU-QA-1 enumerates 4 responsibilities of the Millstone Quality Assurance Supervisor and his staff. Identify who (by position title) has the same responsibilities in the Manager, Quality Assurance's organization. Describe how these actions are accomplished.

260.7 (1B5) The last sentence of the introduction of NU-QA-1 states the NUPOC and NUSCO interrelationship shown in Figure 1.6 "indicates how conflicts are resolved." This is not clear. Describe provisions for the resolution of disputes involving quality arising from a difference of opinion between QA personnel and other department (engineering, procurement, manaufacturing, operating, etc.) personnel.

260.8 (1B6) Describe those provisions which assure that designated QA individuals are involved in day-to-day plant activities important to safety (i.e., the QA organization routinely attends and participates in daily plant work schedule and status meetings to assure they are kept abreast of day-to-day work assignments throughout the plant and that there is adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments).

260.9 (1C1) Describe provisions which assure that implementation of the QA program, procedures, and instructions is mandatory as stated in section 2.2.5 of NU-QA-1.

260.10 (1C2) Appendix B of NU-QA-1 lists the qualification requirements for the NUSCO Manager, Quality Assurance. The staff is interested in the qualification requirements for the position that retains overall authority and responsibility for the QA program (Ref. question 260.3). The qualifications for this position should be established in a position description which includes the following prerequisites:

a. Management experience through assignments to responsible positions.
b. Knowledge of QA regulations, policies, practices, and standards.
c. Experience working in QA or related activity in reactor design, construction, or operation or in a similar high technological industry.

260.0 QUALITY ASSURANCE BRANCH (CONT'D)

The qualifications for this position should be at least equivalent to those described in Section 4.4.5 of ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel," as endorsed by the regulatory positions in Regulatory Guide 1.8. Provide such a commitment or propose an alternative for staff evaluation.

260.11 (1C3) Identify by position title the person responsible for the onsite QA program. Clarify that this position is free from non-QA duties and can thus give full attention to assuring that the QA program at the plant site is being effectively implemented and that it has appropriate organizational position, responsibilities, and authority to exercise proper control over the QA program.

260.12 (2A1c) Describe how the QA program will be applied to the development, control, and use of computer programs.

260.13 (2B3) Appendices D and F address commitments to regulations, regulatory guides, and ANSI standards. For Millstone 3, add:

a. Appendix A to 10 CFR Part 50, General Design Criteria 1
b. 10 CFR Part 50, Section 50.54, Condition of Licenses
c. 10 CFR Part 50, Section 50.55a, Codes and Standards
d. Regulatory Guide 1.26, Rev. 3 (2/76)
e. Regulatory Guide 1.29, Rev. 3 (9/78)
f. BTP ASB 9.5-1 (Fire Protection)

and delete:

a. Regulatory Guide 1.70
b. Draft Regulatory Guide concerning preoperational and startup test programs.

For systems, components, and structures covered by the ASME Code Section III (Classes 1, 2 and 3) describe measures which assure that the QA code requirements are supplemented by the specific guidance addressed in the regulatory positions of the applicable Regulatory Guides.

Clarify that the QA organization and the necessary technical organizations determine and identify the extent QA controls are to be applied to specific structures, systems, and components. This effort involves applying a defined graded approach to certain structures, systems, and components in accordance with their importance to safety and affects such disciplines as design, procurement, document control, inspection tests, special processes, records, audits, and others described in 10 CFR Part 50, Appendix B.

260.0 QUALITY ASSURANCE BRANCH (CONT'D)

260.14 (2C1) Sections 1.2.5.1 and 2.2.6 of NU-QA-1 describe an annual Management QA Review under the sponsorship of the Senior VP - Nuclear Engineering and Operations. The involvement of the Senior VP - Nuclear Engineering and Operations should also include frequent contact with the QA program through reports, meetings, and/or audits. Provide such a commitment or describe an alternative for staff evaluation.

260.15 (2D) Describe provisions which assure that:

a. Documentation of formal training and qualification programs includes the objective, content of the program, attendees, and date of attendance.
b. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.
c. Certificates of qualification clearly delineate the specific functions personnel are qualifed to perform and the criteria used to qualify personnel in each function.

260.16 (17.2.2-2) Describe provisions for assuring the QA program for operations is implemented at least 90 days prior to fuel loading.

260.17 (17.2.2-3) Provide confirmation to commit to continued implementation of the PSAR QA program for the remaining design and construction activities and the preoperational test program or provide an acceptable alternative.

260.18 (3D) Describe internal and external design interface controls, procedures, and lines of communication among participating design organizations and across technical disciplines for the review, approval, release, distribution, and revision of documents involving design interfaces to assure structures, systems, and components are compatible geometrically, functionally, and with processes and environment.

260.19 (3E1) Describe provisions which assure a documented check to verify the dimensional accuracy and completeness of design drawings and specifications.

260.20 (3E2) Describe provisions which assure that design drawings and specifications are reviewed by the QA organization to verify that the documents are prepared, reviewed, and approved in accordance with company procedures and that the documents contain the necessary QA requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results.

260.0 QUALITY ASSURANCE BRANCH (CONT'D)

260.21 (3E4b) Describe provisions which assure that design verification, if other than by qualification testing of a prototype or lead production unit, is completed prior to release for procurement, manufacturing, construction or to another organization for use in other design activities. In those cases where this timing cannot be met, the design verification may be deferred, providing that the justification for this action is documented and the unverified portion of the design output document and all design output documents, based on the unverified data, are appropriately identified and controlled. Site activities associated with a design or design change should not proceed without verification past the point where the installation would become irreversible (i.e., require extensive demolition and rework). In all cases, the design verification should be complete prior to relying upon the component, system, or structure to perform its function.

260.22 (3E4c) Describe provisions which assure that procedural control is established for design documents that reflect the commitments of the SAR and that this control differentiates between documents that receive formal design verification by interdisciplinary or multi-organizational teams and those which can be reviewed by a single individual (a signature and date is acceptable documentation for personnel certification). Design documents subject to procedural control include, but are not limited to, specifications, calculations, computer programs, system descriptions, SAR when used as a design document, and drawings including flow diagrams, electrical single line diagrams, structural systems for major facilities, site arrangements, and equipment locations. Specialized reviews should be used when uniqueness or special design considerations warrant.

260.23 (3E4d) Describe provisions which identify the responsibilities of design verifiers, the areas and features to be verified, the pertinent considerations to be verified, and the extent of documentation.

260.24 (3E5) When design verification is by test, describe procedures which assure the following:

a. Procedures provide criteria that specify when verification should be by test.
b. Prototype, component, or feature testing is performed as early as possible prior to installation of plant equipment, or prior to the point when the installation would become irreversible.
c. Verification by test is performed under conditions that simulate the most adverse design conditions as determined by analysis.

260.25 (3E6) Clarify that procedures are established to assure that verified computer codes are certified for use and that their use is specified.

260.0 QUALITY ASSURANCE BRANCH (CONT'D)

260.26 (17.2.3-2) Describe measures which assure that responsible plant personnel, on a timely basis, are made aware of design changes/modifications which may affect the performance of their duties.

260.27 (4B1) Describe organizational responsibilities for bid evaluation. Include the involvement of the QA organization(s).

260.28 (6A1) Expand the list of documents in section 6.2.1 of NU-QA-1 to include documents related to computer codes and as-built documentation.

260.29 (6A2) Describe provisions which assure that the QA organization, or an individual other than the person who generated the document but qualified in quality assurance, reviews and concurs with the documents listed in section 6.2.1 of NU-QA-1 with regards to QA-related aspects.

260.30 (6B1) Describe provisions which assure that obsolete or superseded documents are removed and replaced by applicable revisions in work areas in a timely manner.

260.31 (6C1) Describe provisions which assure the preparation of as-built drawings and related documentation in a timely manner to accurately reflect the actual plant.

260.32 (17.2.6-2) Describe provisions which assure that maintenance, modification, and inspection procedures are reviewed by qualified personnel knowledgeable in QA disciplines (normally the QA organization) to determine:

a. The need for inspection, identification of inspection personnel, and documentation of inspection results.
b. That the necessary inspection requirements, methods, and acceptance criteria have been identified.

Identify the organizational element(s) responsible for this review.

260.33 (7A4) The fifth paragraph of section 4.2.1 of NU-QA-1 addresses the procurement of spare and replacement parts. Describe provisions which assure that such procurements are subject to the pertinent provisions of the latest QA program controls.

260.34 (7B4) Describe measures which assure that, for commercial "off-the-shelf" items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verification requirements shall be established and described to provide the necessary assurance by NUSCO/NUPOC of an acceptable item.

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260.0 QUALITY ASSURANCE BRANCH (CONT'D)

260.35 (7B5) The last paragraph of section 7.2.1 of NU-QA-1 addresses the independent verification of Certificates of Conformance. Clarify that these verifications are documented.

260.36 (10A) Describe provisions which assure that procedures provide criteria for determining the accuracy requirements of inspection equipment and for determining when inspections are required. Clarify NUSCO/NUPOC QA's involvement in the preparation of inspection plans and schedules as discussed in section 10.2.2 of NU-QA-1.

260.37 (10B1 & 17.2.10.2) Clarify in section 10.2.3 of NU-QA-1 whether inspections are performed by NUSCO/NUPOC QA personnel or by personnel outside the QA organization. If inspections are made by personnel outside the QA organization, describe provisions which assure (in addition to commitments in section 10.2.3) that these personnel do not report directly to the immediate supervisors who are responsible for the activity being inspected. If the individuals performing inspections are not part of the QA organization, the inspection procedures, personnel qualification criteria, and independence from undue pressure such as cost and schedule should be reviewed and found acceptable by the QA organization prior to the initiation of the activity. If inspections associated with normal operations of the plant (such as routine maintenance, surveillance, and tests) are performed by individuals other than those who performed or directly supervised the work, but are within the same group, describe measures which assure the following controls are also met:

a. The quality of the work can be demonstrated through a functional test when the activity involves breaching a pressure retaining barrier.
b. The qualification criteria for inspection personnel are reviewed and found acceptable by the QA organization prior to initiating the inspection.

260.38 (10C1g) Clarify in section 10.2.4 of NU-QA-1 that inspection procedures and checklists specify the necessary measuring and test equipment and the required accuracy of this equipment.

260.39 (10C3) Describe provisions which assure that inspection results are evaluated for acceptability and indicate what organization is responsible for this activity.

260.40 (11A1) Describe provisions which assure that test procedures provide criteria for determining the accuracy requirements of test equipment and for determining when a test is required.

260.0 QUALITY ASSURANCE BRANCH (CONT'D)

260.41 (12.2) Describe QA and other organizations' responsibilities for establishing, implementing, and assuring effectiveness of the calibration program.

260.42 (12.3) Identify the organization(s) responsible to establish, review, and concur with procedures for calibration, maintenance, and control of M&TE. Clarify that NDE equipment and instrumentation used during the operations phase of Millstone 3 will be controlled in accordance with section 12.0 of NU-QA-1.

206.43 (12.6) Identify the management authorized to accept calibration of M&TE against standards that have an accuracy of less than four times the accuracy of the equipment being calibrated.

260.44 (12.7) Identify the management authorized to allow the use of calibrating standards with the same accuracy as standards being calibrated.

260.45 (12.8) Describe provisions which assure that reference and transfer standards are traceable to nationally recognized standards and that, where national standards do not exist, provisions are established to document the basis for calibration.

260.46 (17.2.13.2) Describe provisions for the storage of chemicals, reagents (including control of shelf life), lubricants, and other consumable materials.

260.47 (14.3) Describe provisions to control altering the sequence of required tests, inspections, and other operations important to safety. Such actions should be subject to the same controls as the original review and approval.

260.48 (14.4) Identify the organization responsible for documenting the status of nonconforming, inoperative, or malfunctioning structures, systems, and components.

260.49 (15.1) Expand section 15 of NU-QA-1 to include nonconforming services (including computer codes).

260.50 (15.3) Describe provisions to assure that nonconformances are corrected or resolved prior to the initiation of the preoperational testing of the item.

260.51 (15.5) Describe provisions which assure that nonconformance reports are periodically analyzed by the QA organization to show quality trends, and the significant results are reported to upper management for review and assessment.

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260.0 QUALITY ASSURANCE BRANCH (CONT'D)

260.52 (16.1) Describe provisions which assure that NUSCO QA reviews and documents concurrence with corrective action procedures.

260.53 (16.2) The first paragraph of section 16.2.1 of NU-QA-1 indicates that conditions adverse to quality are evaluated to determine the need for corrective action. Clarify that the procedures which require such evaluation also require documentation of this evaluation. Also describe provisions which assure QA concurrence of the adequacy of the corrective action.

260.54 (16.3) Describe provisions which assure that followup action is taken by the QA organization to verify proper implementation of corrective action and to close out the corrective action in a timely manner.

260.55 (17.4) Describe the facilities for the storage of Millstone 3 records.

260.56 (18A4) Expand the list of areas audited in the second paragraph of section 18.2.1 of NU-QA-1 to include:

a. Corrective action, calibration, and nonconformance control systems.
b. SAR commitments.
c. Activities associated with computer codes.

260.57 (FP) Section 3.4 and Appendix A of the Millstone 3 "Fire Protection Evaluation Report" address quality assurance. It is not clear that the pertinent provisions of NU-QA-1 will be applied to the fire protection program (and related hardware) during the operation phase of Millstone 3. Clarify.

471.0 RADIOLOGICAL ASSESSMENT BRANCH, RADIATION PROTECTION SECTION (CONT'D)

471.10 Section 12.1.II of the Standard Review Plan (SRP), NUREG-0800, Rev. 2, lists Regulatory Guide 8.8 as an acceptable means of meeting the requirements of 10 CFR 20.1(c). In accordance with R.G. 8.8 Section C.1.d(2), describe the radiation protection aspects of decommissioning which you have included in your plant design to ensure that exposures to workers, during decommissioning, will be ALARA.

471.11 In accordance with the acceptancecriteria of Section 12.2 of the SRP, NUREG-0800, your FSAR indicates that borated silicon shields are employed in the annular region for neutron streaming. Provide a description of this shield that includes shield thickness, boron loading and the source strength that your design is based on.

471.12 Section 121.2 of your FSAR states that after a point by point comparison of your plant design with provisions in R.G. 8.8 "in nearly all cases, the plant designed was determined to be ALARA". Provide a listing of which specific criteria are not in accordance with R.G. 8.8. Also, since R.G. 8.8 is referenced in Section 12.1 Acceptance Criteria as a means of demonstrating to the staff that occupational radiation exposures will be ALARA, provide the basis for concluding that those aspects of plant design not in conformance with R.G. 8.8 are acceptable.

471.13 Acceptance criteria for Section 12.3 of the SRP, NUREG-0800, include meeting the criteria of Sections II.B.2 of NUREG-0737, Clarification of TMI Action Plan Requirements. In accordance with II.B.2 provide the analysis that demonstrates vital system operation and occupancy of the TSC.

470.0 RADIOLOGICAL ASSESSMENT BRANCH, RADIATION PROTECTION SECTION (CONT'D)

471.14 In accordance with Section 12.5.1.B.3 of the SRP, NUREG-0800, provide a description of the radiation protection facilities. This description should be complete enough to demonstrate how such facilities and services will allow both male and female workers to receive the necessary protection. Include a description of such features as separate locker rooms, shower rooms, decontamination area and dress out areas.

471.15 In accordance with Section 12.3-12.4.I.1 of the SRP, NUREG-0800 and Section 12.3.1 of R.G. 1.70 "Standard Format and Content of Safety Analysis Report for Nuclear Plants", Rev. 3, it is our position that the FSAR should include plant layouts showing shield wall thickness around each major radiation source. The shield wall thickness of major radioactive equipment can be provided in a separate table. Section 12.3.2 of your FSAR should be revised accordingly to provide this information.

471.16 Provide the information specified in Section 12.3.II.1 "Facility Design Features" of the SRP, NUREG-0800, as it refers to the stringent access control around the spent fuel transfer tube. Include in your discussion any access routes to the spent fuel transfer tube, type of marking on accessible poritions of transfer tube, and a description of the alarming radiation monitors employed if other than permanent shielding around the transfer tube is used.

471.17 In accordance with Section 12.2.I.1 of the SRP provide the radioactive source geometry parameters missing from tables 12.2-2 and 12.2-4.

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470.0 RADIOLOGICAL ASSESSMENT BRANCH, RADIATION PROTECTION SECTION (CONT'D)

471.18 10 CFR 20.103(b)(1) states that the licensee shall use process or engineering controls, to the extent practicable, to limit the concentrations of radioactive materials in the air, to below 25% of the concentrations given in Column 1, Table I of 10 CFR 20 Appendix B (as specified in 20.203(d)(1)(ii). Section 12.3.3.1.1 of your FSAR indicates that your ventilation system is designed only to maintain concentrations given in Column 1, Table I of Appendix B. Discuss steps taken to up grade your ventilation system so as to meet the requirements of 10 CFR 20.103(b)(1). Also, Section 12.3.3.1 of your FSAR states that "airflow... is normally from areas of lower to higher potential airborne contamination." Describe the exceptions to this good ventilation practice (which is also required in 12.3.II.3 of the SRP).

471.19 In accordance with the criteria of 12.3.II.4.b of the SRP, NUREG-0800, indicate how your airborne radioactivity monitoring system will detect ten MPC-hours of particulate and radio-iodines from any compartment, normally occupied, which may contain this airborne contamination. Also, in accordance with 12.3.I.4.c, describe your procedures for locating suspected high activity areas.

470.0 RADIOLOGICAL ASSESSMENT BRANCH, RADIATION PROTECTION SECTION (CONT'D)

471.20 In accordance with the criteria in 12.3.4 of the SRP, you should describe the details of your fixed area radiation and continuous airborne radioactivity monitoring instrumentation, as enumerated in 12.3.4 of R.G. 1.70. This should include a description of auxilliary or emergency power supplies; sensitivity, accuracy, precision and alarm setpoints of the instruments; calibration methods and frequencies and also the criteria used for selecting the location for these instruments (include selection criteria for locating portable CAMs).

471.21 Section 12.3.II.4 of the SRP states that an acceptable area radiation monitoring system must meet the provisions in R.G. 1.97, Rev. 2. Table 1 of R.G. 1.97, Rev. 2 specifies two High Range Containment Radiation monitors with a range of from 1 R/hr to 10^7 R/hr. Describe your means of implementing this provision. The placement of these monitors in containment, the accuracy and sensitivity of the monitoring employed, should all be discussed.

471.22 In accordance with 12.3-12.4.5.b of the SRP, describe any additional dose reducing measures, if any, taken as a result of the dose assessment provided in Chapter 12.4 of your FSAR.

471.23 Based on the criteria in C.1.e of R.G. 8.10, the RPM should have sufficient authority to enforce safe plant operation. It is the Staff's position that to ensure the RPM's ability to communicate promptly with an appropriate level of management about haulting an operation he deems unsafe, he should report directly to the Station Superintendent. Your organization description in the FSAR should be revised to reflect this position.

470.0 RADIOLOGICAL ASSESSMENT BRANCH, RADIATION PROTECTION SECTION (CONT'D)

471.24 Indicate (and provide the resume's for) the individuals named as RPM and his backup. In accordance with 12.5.II.A of the SRP, the RPM should meet the qualification specifications of R.G. 1.8. Also it is the staff's position that the individual who will act as RPM in the RPM's absence (e.g., while on vacation), should have at least a B.S. degree in science or engineering, 2 years experience in radiation protection, 1 year of which should be nuclear power plant experience, 6 months of which should be onsite. In addition your FSAR should be changed to address the qualification for health physics technicians as specified in ANSI N18.1.

471.25 Section 12.5.2 of your FSAR states that "areas in the RCA above 10 mRem per hour will have a locked or guarded barrier." The regulations require a locked or guarded barrier at 100 mRem per hour. If this is a typographical error, your FSAR should be changed accordingly.

471.26 Section 12.5.3 of your FSAR states that "special control techniques will be used to minimize airborne exposure arising from special work projects." In accordance with the requirements of Section 12.5.II.C of the SRP, describe what these special control techniques are. Describe your technique for obtaining breathing zone air samples. What conditions require special air sampling and what are your reporting practices for airborne contamination surveys? In addition, although Section 12.3.3.1 states that design and expected airborne concentrations are given in Section 12.4, there is no reference to airborne concentrations in that section. In accordance with Section 12.2.I.2 of the SRP describe the radioactive material sources used for design of personnel protective

470.0 RADIOLOGICAL ASSESSMENT BRANCH, RADIATION PROTECTION SECTION (CONT'D)

measures and for dose assessment. This description should include a tabulation of the calculated concentrations of radioactive material, by nuclide, expected during normal operation, anticipated operational occurrences and accident conditions for equipment cubicals, corridors, and operating areas of the plant. Include the models and parameters used as a basis for these concentrations.

471.27 Section 12.5.II of the SRP lists R.G. 8.13, R.G. 8.27, R.G. 8.29 and NUREG-0731, as acceptable guidance for establishing a radiation protection training program. Your FSAR should be changed to describe your compliance with these references.

471.28 In accordance with the acceptable criteria of Section 12.5 of the SRP, NUREG-0800, describe your plants capability of meeting the iodine sampling criteria listed as item III.D.3.3 of NUREG-0737, clarification of TMI Action Plan Requirements.

471.29 Section 12.5.IIC of the SRP notes that compliance with R.G. 1.33, "Quality Assurance Program Requirements" is an acceptable means to show compliance with the requirements of 10 CFR 50, Appendix B. In Section 12.5.3 of your FSAR, you state that quality assurance inspections will be performed on the radiation protection procedures identified in R.G. 1.33, Rev. 2. Since the listing of procedures in R.G. 1.33, Rev. 2 is not intended to be an all inclusive listing, you should change your FSAR to include all radiation protection procedures (including those such as emergency procedures, and instrument storage, calibration and maintenance procedures, not specifically listed in R.G. 1.33, Rev. 2).

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220.0 STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH, STRUCTURAL ENGINEERING SECTION

220.09 (SRP 3.3.2) FSAR Section 3.3.2.3 states that the metal siding and roofing of the service, turbine, waste disposal, containment enclosure buildings, and portions of the fuel building are assumed to blow off under tornado wind load, but the structural steel framing of these structures is designed to withstand tornado wind loads. Was it assumed that the metal siding and roofing remained intact to the structural steel framing in calculating the tornado wind loads acted on the framing?

220.10 (SRP 3.5.3) Is there any concrete barrier whose thickness is less than that shown in Table 1 of SRP Section 3.5.3? If yes, please identify and justify them.

220.11 (SRP 3.5.3) FSAR Section 3.5.3.1 states that the barriers are designed so that the calculated ductility ratio of the barriers for any load combination is less than the maximum allowable ductility ratio. However, the term "ductility ratio" was never defined. Please provide a definition for it and use a numerical example of concrete barriers to show how "the calculated ductility ratio" was computed.

220.12 (SRP 3.5.3) The last sentence of FSAR Section 3.5.3.1 states that if a concrete barrier is not required to carry other loads during and after impact, the maximum allowable ductility is limited to correspond to a rebar elongation of 5 percent. Is "the maximum allowable ductility" the same as "the maximum allowable ductility ratio" mentioned in the last question? If not, provide a new definition for it. Ordinarily, the ductility ratio of a reinforced concrete section is defined as the ratio of the calculated curvature at failure to the calculated curvature at yield. A curvature calculation involves both the steel reinforcing strain and concrete compressive strain. Therefore, please explain why ductility is measured by rebar strain alone and not concrete compressive strain in the FSAR. Should that be the case, can the text be rephrased as "the maximum allowable rebar elongation is limited to 5 percent" to not involve the undefined word "ductility"? There are several places in Section 3.5.3.1 where the words "ductility" and "ductility ratio" have been misplaced as interchangeable. Please correct them.

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220.0 STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH, STRUCTURAL ENGINEERING SECTION (CONT'D)

220.13 (SRP 3.5.3) FSAR Section 3.5.3.1 states that for beam-column members, where the compressive load is equal to or less than one-third of that which could produce balanced conditions, the allowable ductility is 10. Should the word "ductility" be replaced with "ductility ratio"? If not, please explain the meaning of "the allowable ductility is 10". We presume that your "balanced conditions" mean the same as "balanced strain conditions" as defined in Section 10.3.2 of ACI 318-77 Code. Notice that the balanced strain conditions of the ACI Code exist at a cross section, but not structural members as implied in your description. Therefore, please use an example to illustrate how you actually applied this requirement to beam-column members.

220.14 (SRP 3.5.3) FSAR Section 3.5.3.1 states that for beam-column members, where the design is controlled by compression, the allowable ductility is 1.3. Please define "the allowable ductility" mathematically, and provide technical reasons to justify the number 1.3.

220.15 (SRP 3.5.3) The first paragraph that describes the overall barrier response of the FSAR Section 3.5.3.3 is difficult to read. Please rephrase it. Complete the sentence that reads "For beams, walls, and slabs where flexural controls design, the permissible ductility ratio is based on". Also, use an example to illustrate how the flexural strength is determined from an ultimate strength theory with the limitations on ductility, as stated.

220.16 (SRP 3.7.1) FSAR Section 3.7B.1.1 states that Regulatory Guide 1.60 spectra are not used. Does this mean that "site specific spectra" have been developed for this plant? If so, have these site specific spectra been reviewed and approved by the Geosciences Branch of NRC?

220.17 (SRP 3.7.1) FSAR Section 3.7B.1.3 states that the values of the percentage of critical damping used in the analysis of Seismic Category I structures, systems, and components depends on the seismic input motion used in the analysis. Please explain this in detail.

220.18 (SRP 3.7.1) The format and some of the percent of critical damping values of Table 3.7B-1 are different from that of Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." Either revise the contents in Table 3.7B-1 to comply with Regulatory Guide 1.61 or provide justifications for the deviations, as stated in SRP Section 3.7.1 II 2.

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220.0 STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH, STRUCTURAL ENGINEERING SECTION (CONT'D)

220.19 (SRP 3.7.2) SRP Section 3.7.2 II 4 "Soil-Structure Interaction" requires that two different ways of modeling the supporting soil media of a soil-structure interaction system be considered: half-space and finite boundary, and then envelop response results in structures for the use of designing Seismic Category I structures, systems, and components. Please revise your FSAR to comply with these requirements.

220.20 (SRP 3.7.2) FSAR Section 3.7B 2.9 states that floor response spectra for the cracked and uncracked cases were enveloped. Provide the criteria that was used to determine the cracked or uncracked cases.

220.21 (SRP 3.7.2) SRP Section 3.7.2 II 11 requires that an additional seismicity effect based on a consideration of ± 5% of the maximum building dimension at the level under consideration shall be assumed to account for accidental torsion. Since FSAR Section 3.7B.2.11 has not addressed this requirement, please revise the FSAR to comply with this requirement.

220.22 (SRP 3.7.2) Provide response spectra that correspond to the time histories used at the foundation level of the containment structure shown in Figure 3.7B-9. Also, provide the values of horizontal, vertical, rocking and torsional subgrade springs and subgrade damping values (both translational and rotational). Explain why there are far-coupling situations among M_1, M_2, and M_5 and between M_1 and M_3 in Figure 3.7B-9.

220.23 (SRP 3.7.2) Provide values of M's, K's, and damping of the structure and values of rock springs and damping in Figure 3.7B-10.

220.24 (SRP 3.7.2) Provide response spectra that correspond to the time histories used at the bedrock (EL. - 14'-0") in Figures 3.7B-11 and 3.7B-12.

220.25 (SRP 3.7.2) FSAR Section 3.7B.2.2.1 mentioned "the amplified response spectra (ARS)". Please define the new term and explain how do they differ form "design response spectra" and "floor response spectra"?

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220.0 STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH, STRUCTURAL ENGINEERING SECTION (CONT'D)

220.26 (SRP 3.8.1) FSAR Section 3.8.1.3.1 states that the allowable compressive stress in concrete is 0.45fc'. However, the ASME Section III, Division II Code allowable for primary membrance is only 0.30fc'. Please justify the deviation.

220.27 (SRP 3.8.1) FSAR Section 3.8.1.3.1 states that structural members subjected to test prerssure, temperature, or wind, when combined with other forces, are designed for the allowable stresses increased by 33 percent. Notice that SRP Section 3.8.1.II 5a allows the increase only for temperature, not others, and you should revise the FSAR to comply with it.

220.28 (SRP 3.8.1) The contanment load combinations and factors in FSAR Section 3.8.1.3.1 are not identical to that of SRP Section 3.8.1.II 3. You should either revise the FSAR to comply with SRP requirements or list the deviations and justify them.

220.29 (SRP 3.8.1) The purposes of the descriptions in FSAR Section (SRP 3.8.1) 3.8.1.3.2 are not clear. You should rephrase them so that the purposes can be understood. Specifically, Tables 3.8-1 and 3.8-2 are not in compliance with the requirements of SRP Section 3.8.1 II 4i. You should either revise the FSAR to comply with SRP requirements or list the deviations and justify them.

220.30 (SRP 3.8.1) Does the method used for tangential shear design in FSAR Section 3.8.1.4.1 satisfy the requirements of SRP Section 3.8.1 II f "Tangential Shear"? You should rephrase the FSAR to indicate that if it does, or justify the deviations if it does not.

220.31 (SRP 3.8.1) Provide temperature profiles that were used for containment (SRP 3.8.1) thermal analysis for both operating and accident conditions.

220.32 (SRP 3.8.1) FSAR Section 3.8.1.5.1 states that design of the containment equals or exceeds ACI 318-71 requirements for serviceability. Since the ACI 318 Code is for conventional building structures, not for containment structures, we fail to see the connection between containment serviceability and conventional building serviceability. Please list the requirements for serviceability and explain the connection.

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220.0 **STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH, STRUCTURAL ENGINEERING SECTION (CONT'D)**

220.33 (SRP 3.8.1) SRP Section 3.8.1 II 4 requires that an analysis should be performed to determine the ultimate capacity of the containment. Please revise the FSAR to comply with this requirement.

220.34 (SRP 3.8.3) Have the requirements of ACI 318-71 Section 11.6 "Special Provisions for Walls" been adopted as acceptance criteria in FSAR Section 3.8.3.5? If not, provide reasons to justify it.

220.35 (SRP 3.8.4) FSAR Section 3.8.4.1 states that non-safety related partitions are solid concrete block. Were these concrete block partitions designed to withstand seismic loads so that they will not collapse?

220.36 (SRP 3.8.4) The loads and loading combinations in FSAR Section 3.8.4.4 are not identical to that of SRP Section 3.8.4 II 3. You should either revise the FSAR to comply with SRP requirements or list the deviations and justify them.

220.37 (SRP 3.8.4) Identify the allowable stresses for steel structures in FSAR 3.8.4.4 which deviate from that of SRP Section 3.8.4 II 5 and justify them.

220.38 (SRP 3.8.4) Are any safety-related masonry walls in the plant? If yes, revise the FSAR to comply with the requirements in Appendix A to SRP Section 3.8.4.

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241.0 STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH, GEOTECHNICAL ENGINEERING SECTION

Q241.1 2.5.4.2.4 and 2.5.4.5.1 (SRP 2.5.4)

Subsurface Profiles

You have stated in Section 2.5.4 of the FSAR that some sections of the circulating water discharge tunnel and service water intake lines are founded on soil. Please identify those sections on a site plan and provide cross-sections along soil supported sections of discharge tunnel and intake lines. On those cross-sections, subsurface profiles disclosed by exploratory borings and foundation excavation should be presented. If compacted fill was used to raise the foundation levels, the extent of the fill should be identified and the results of the field moisture and density tests should be provided.

Q241.2 2.5.4.5.1 (SRP 2.5.4)

Rock Failures

Rock failures resulting from blasting during excavation have been reported in the FSAR. Please provide additional information to identify the locations and extent of those failures. Cross-sections showing the high angle jointing should be provided.

Q241.3 2.5.4.5.2 (SRP 2.5.4)

Structural Backfill

You have stated in the FSAR that the structural backfill was used to support the control building and diesel generator enclosure building. Provide plan views and cross-sections showing the extent of the fill placed and the subsurface conditions. Identify the source of the backfill material used for these buildings and provide the backfill compaction test results obtained during construction.

Q241.4 2.5.4.7 3.7.1.4 and 3.7.2.4 (SRP 2.5.4)

Soil-Structure Interaction

You have stated that soil-structure interaction analyses were performed for control building and emergency generator enclosure. Identify the subsurface layers and their material properties used in your analyses. Provide and discuss the results of your analyses.

Q241.5 2.5.4.7 (SRP 2.5.4)

Concrete Encasement

You have reported that the service water intake pipes between the circulating and service water pumphouse and main plant area are embedded in a rectangular concrete encasement. Provide the sectional profiles and details about the concrete encasement. Also, provide the results of compaction tests performed along the intake pipes.

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241.0 STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH, GEOTECHNICAL ENGINEERING SECTION (CONT'D)

Q241.6 Sliding Stability
2.5.4.7
(SRP 2.5.4) You state that the service water encasement has been analyzed for sliding stability due to seismic loading. Provide the details of the analysis and identify the cross-section used in your analysis.

Q241.7 Dynamic Response Analysis of Beach Sand
2.5.4.8.3
(SRP 2.5.4) Based on FSAR Figure 2.5.4-35, the thickness of beach sands varies from a few feet to about 50 feet and the thickness of basal till is also variable. In view of this variation of profile, justify that the one dimensional computer program SHAKE is suitable for analyzing the dynamic response of the shorefront sand deposits. Identify the location of the idealized profile used in the analyses. Substantiate the assumption that the groundwater is 10 feet below the ground surface using field monitoring results.

2.5.4.8.3
2.5.4.8.4
and
2.5.4.9
(SRP 2.5.4) Provide the bases for not using artificial time history conforming to R.G. 1.60 design response spectra in your dynamic response analyses of beach sands.

Q241.8 Liquefaction Analysis of Beach Sand
2.5.4.8.3
(SRP 2.5.4) Discuss the assumptions used in your liquefaction analysis of beach sand about the possible pore pressure build-up and the potential strength reduction under seismic shaking. Also, discuss the after-earthquake effects, slope instability and lateral movements, on beach sand.

Q241.9 Equivalent Stress Cycles
2.5.4.8.3
(SRP 2.5.4) Provide the bases for assuming that the irregular shear stress time history of the SSE can be represented by five uniform cycles of loading.

Q241.10 Liquefaction Potential
2.5.4.8.3
(SRP 2.5.4) You have stated that the standard penetration resistance data have been used in assessing liquefaction potential. Since the standard penetration resistance depends on many factors, i.e., drill rig type, hammer type, the fall height, the number of turns around the cathead and operator characteristics, provide those information and other relevant information associated with the data cited in the FSAR. Discuss the effects of these parameters on your SPT test data and their influence on liquefaction potential evaluation results.

241.0	STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH, GEOTECHNICAL ENGINEERING SECTION (CONT'D)
Q241.11	Bedrock Profile
2.5.4.8.4 (SRP 2.5.4)	In the FSAR Section 2.5.4.8, you state that in the vicinity of the ventilation stack north of Millstone 1, bedrock drops sharply to a trough. Identify the location of this bedrock trough on a plot plan and provide the subsurface profiles of the trough and the overlying soils. Information pertinent to the disclosing of the bedrock trough, such as exploratory boring and/or trenching, should be identified and discussed.
Q241.12	Dynamic Response Analysis of Ablation Till
2.5.4.8.4 (SRP 2.5.4)	Identify the location where the idealized profile was obtained for the dynamic response analyses of Ablation Till and justify the groundwater level assumption.
Q241.13	Bearing Capacity
2.5.4.10 (SRP 2.5.4)	Provide the results of your ultimate bearing capacity calculations shown in Table 2.5.4-14 of the FSAR and demonstrate that the minimum safety factor for the allowable bearing capacity exceeds 3. Also provide bearing capacity evaluation for the discharge tunnel and pumphouse.
Q241.14	Rock Bearing Capacity
2.5.4.10 (SRP 2.5.4)	Provide the bases for allowing the bedrock bearing load as high as 200 KSF.
Q241.15	Settlement Records
2.5.4.10 and 2.5.4.13 (SRP 2.5.4)	Provide settlement monitoring records for the control, fuel, waste disposal, and emergency diesel generator enclosure buildings.
Q241.16	Lateral Earth Pressure
2.5.4.10.3 (SRP 2.5.4)	Provide the design values of the lateral earth pressures used in the design of rigid, unyielding, foundation walls.
Q241.17	Ring Beam
2.5.4.12 2.5.4.13 and 2.5.5.1 (SRP 2.5.4)	You have stated that a reinforced concrete ring beam was placed in the annular space between the excavation face and the containment exterior wall to stabilize the wedges. Provide the criteria and bases used for the ring beam design.

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241.0	STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH, GEOTECHNICAL ENGINEERING SECTION (CONT'D)
Q241.18	Dynamic Slope Stability
2.5.5.1 and 2.5.5.2 (SRP 2.5.5)	You state that the computer program LEASE II was used to analyze the stability of the shoreline slope and an undrained shear strength of Ø equals 20 degrees and C equals 1200 psf was used for beach sands. Provide the bases for justifying the use of this shear strength for beach sands. Provide a summary of all dynamic soil properties used in the dynamic analyses. Justify the groundwater conditions used in the analyses. Discuss the possibility of strength loss of beach sands resulting from earthquake shaking and post-earthquake failure potential.
Q241.19	Final Moisture Contents
Appendix F Section 3.2 (SRP 2.5.5)	Provide final moisture contents of the soil samples used for cyclic triaxial, and resonant column tests (Table 2 & 3).
Q241.20	Grain Size Tests
Appendix F Section 3.3 (SRP 2.5.5)	You have presented Lee & Fitton and Kishida's liquefaction envelopes on Figure 2.5.4-30 of the FSAR. Compare the results of your grain size test data on beach sands with these envelopes. Provide the results of your comparison on a figure and discuss these results as they affect the liquefaction potential evaluation of beach sands.

ENCLOSURE 2

LIST OF QUESTIONS IN ENCLOSURE 1

BRANCH	QUESTION NOS.	TOTAL
ASB	410.7-410.31	25
CMEB - Fire Protection	280.2-280.27	26
CMEB - Chemical Engineering	281.3-281.13	11
CPB - Physics	491.1	1
GIB	730.1	1
GSB - Seismology	230.3-230.5	3
GSB - Geology	231.1-231.3	3
EHEB - Hydrology	240.1-240.9	9
LQB	630.2-630.11	10
METB - Meteorology	451.1-451.2	2
METB - Effluent Treatment	460.5-460.19	15
MTEB - ISI	250.2-250.7	6
PSB - Electrical	430.3-430.55	53
PTRB	640.1	1
QAB	260.1-260.57	57
RAB	471.10-471.29	20
SGEB - Structural	220.9-220.38	30
SGEB - Geotechnical	241.1-241.20	20