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Docket No.: 50-508

Mr. R. L. Ferguson
Managing Director
Washington Public Power Supply System
Post Office Box 968
3000 George Washington Way
Richland, Washington 99352

Dear Mr. Ferguson:

Subject; Request for Additional Information on WNP-3 Safety Review

Enclosed are requests for additional information related to several areas of the WNP-3 Safety Review. Your responses should be forwarded to the NRC not later than July 12, 1983. After your responses have been reviewed, a draft SER will be prepared to provide a basis for a series of meetings designed to close out open items.

If clarification of these requests for additional information is necessary, the WNP-3 Project Manager, Ms. Annette Vietti, is available to provide any additional information you need. Ms. Vietti's telephone number is (301)492-4449.

Sincerely,

Original signed by:
Thomas M. Novak

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

cc: See next page

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241.12 (SRP 2.5.4) You have referred to Figures 2.5.A.1 through 2.5.A.3a in the FSAR; however, you have not included these Figures in your submittal. Provide these figures for staff review. Also, revise and correct Figures 2.5.A.10 and 2.5.A.49 to show that elevations below 0 (zero) level are negative elevations.

241.13 (SRP 2.5.4) In the FSAR Section 2.5A, you state that the depths of eight static cone penetration tests (CPT) ranged from 71 to 150 ft. However, the test results shown on Figures 2.5.A.6 through 2.5.A.13 are only given to depths of 22 ft. to 46 ft. Revise the figures and/or the FSAR text so that they are consistent. Provide a discussion of the considerations given to these CPT data in your evaluations of rock properties at the site.

241.14 (SRP 2.5.4) In Section 2.5.4.3.1 of the FSAR, you state that the generalized geologic profile AA is presented on Figure 2.5-71. This reference is in error. Please provide the correct reference.

241.15 (SRP 2.5.4) Your boring log for boring A-3, immediately under the WNP-3 containment building foundation, shows very low values for sandstone sample recovery (0, 24, 25, 39 and 49%, and so on) for many samples. The corresponding RQD values are also low. In view of this, provide the following information.

(i) A discussion to justify that zero core recovery for several feet in this boring does not indicate the presence of voids.

(ii) Sufficient details of rock characteristics in the description given on the boring log so that your basis for rock classification (weathered or fresh sandstone etc.) can be reviewed. Did you use 'change in color' as the only basis to differentiate between weathered sandstone and fresh sandstone? If not, then what other factors did you consider in forming your basis for classifying the samples with low recovery as fresh sandstone, (and not weathered sandstone)?

(iii) A detailed and quantitative discussion of the criteria used in concluding that the rock samples tested in the laboratory sufficiently define the in situ rock properties that were used in the design of Seismic Category I foundations. Include discussions of the considerations given to field RQD values and the results of field seismic surveys and laboratory sonic tests in your projection of rock properties.

241.16 (SRP 2.5.4) On large scale plot plans, show and clearly identify the location of foundations for all WNP-3 Category I structures, including Reactor Building, Control Room, Shield Building, Reactor Auxiliary Building, Fuel Handling Building, UHS Cooling Tower Structure, Seismic Category I Tank Structures, (Refueling water storage and condensate storage tanks), Class IE electrical duct Banks, Buried Essential Service Water Pipeline and Vertical Manholes for Reactor Auxiliary Dewatering System and other Category I structures. On the same plot, show the location of the exploration borings and test pits that are in close proximity to these structures. Also provide in tabular form, for each Category I structure, the borings (immediately underneath or in the vicinity of structures) used to define the generalized geologic profile and the borings used to determine static and dynamic engineering properties of rock which were then used to design and analyze the structures and their foundations.

241.17 (SRP 2.5.4) In Section 2.5.4.10.4 of the FSAR, you state that, for static lateral pressure calculation, you have used a coefficient of horizontal pressure of 0.22. Justify your selection of this value and discuss the long term creep effect of the sandstone on lateral pressures. Show how you considered as-built conditions and the effects of adjacent buildings on lateral earth pressures.

241.18 (SRP 2.5.4) You state, in the FSAR, Section 2.5.4.10.4, that you used a dynamic soil-structure-interaction analysis for dynamic lateral pressure calculations. Describe the method used. Provide assumed rock parameters, ground water table assumptions, and describe the seismic input motion used for this analysis. Present your lateral pressure calculation results that you have already obtained and compare them with the results of calculations using a more commonly accepted procedure, e.g. the Seed and Whitman approach. (Seed, H. B. and R. V. Whitman, 1970, "Design of Earth Retaining Structures for Dynamic Loads," Proceedings of the State of the Art papers presented at 1970 Specialty Conference on Lateral Stresses in the Ground and Design of Earth Retaining Structures, American Society of Civil Engineers, June 22-24, 1970, pp 103-147). Substantiate the input data used in both these analyses, and verify that your dynamic finite element lateral pressure results are sufficiently conservative, and loads due to adjacent buildings have been properly taken into account.

241.19 (SRP 2.5.4) The staff finds that your description of the tuff beds given in various subsections of the FSAR Section 2.5.4, is inconsistent. In Subsection 2.5.4.2.1, you state that tuff beds could affect the Category I structures whereas in Subsection 2.5.4.1.4 you imply that tuff beds 1 and 2 are not present near the plant site below the foundation elevation (326 feet) and tuff beds 3 and 4 are located very deep below the foundations and may not affect the plant foundation. Revise these sections to make them consistent. Also give the thickness of the tuff beds in individual borings that are in the proximity of Category I structures, discuss their characteristics and describe the significance of the presence of these layers in your static and dynamic analysis assumptions.

241.20 (SRP 2.5.4) In interpreting the dynamic laboratory test results, you have

(i) divided the measured axial strains by a correction factor of 1.5 for the 13 strain-controlled cyclic triaxial tests on residual soils.

(ii) multiplied the measured peripheral shear strains by a correction factor of 0.7 for the Resonant Column test results on Residual Soils, and

(iii) divided the measured axial strains by a factor of 4 for strain controlled tests on fresh sandstones.

Justify these correction factors. In addition, submit any laboratory test data used for developing strain-dependent modulus and damping curves for weathered sandstone. Provide and justify your basis for assuming the same dynamic properties for weathered sandstone as for fresh sandstone.

241.21 (SRP 2.5.4) Provide the values of soil properties used in the seismic analysis of the buried pipes. Explain your procedure for calculating dynamic axial and bending stresses and provide the seismic input used for this analysis. Verify that you have adequately accounted for the effect of changes in soil properties in your analysis, along with the proper use of intensification factors at bends.

241.22 (SRP 2.5.4) Your summary of the results of field compaction testing does not provide sufficient detail for an adequate review. Present the results of field density and moisture content tests performed in conjunction with quality control of all backfill placement under and adjacent to safety related structures. Present the results, in a format that will allow ready verification of compliance with compaction specifications, for each Category I structure separately (i.e., present separate data for each seismic Category I structure, electrical duct banks, manholes, and pipelines, as appropriate).

241.23 (SRP 2.5.4) In your response to Q241.10, you stated that Equation (1) of that response was obtained from "Dynamics of Bases and Foundations" by D. D. Barker McGraw Hill, 1962. We cannot locate this equation in the mentioned reference by Barkan. Please provide copy of relevant pages from your reference where this equation and the basis for its development are given. Also, describe the procedure you used to obtain the shear modulus (G) equal to 330 ksi from Figure 2.5-121; give the corresponding strain level. Discuss your basis for selecting this particular strain value. Justify the statement in your response that the reduction factor, R_d, used to convert the laboratory data to field data, can vary from 5 to 20. What value of R_d did you use, and what is the basis for your selection.

241.24 (SRP 2.5.4) Your response to Question No. 241.9 is inadequate. This response indicated that the basis for selecting a 570 ft depth of rock column for the deconvolution analysis is given in Subsection 3.7.2.4.1. Our review of this subsection reveals that the requested information is not contained in Subsection 3.7.2.4.1.

Note that the WNP-3 site is essentially a rock site. Again, provide your justification for use of de-convolution as well as your basis for selecting a 570 ft deep rock column for de-convolution.

241.25 (SRP 2.5.4) In response to Question No. 241.3 you have indicated that the as-built loading on the containment building foundation is 13.6 k/ft^2, and not 8 k/ft^2 as previously reported in the PSAR. Revise the affected sections of the FSAR to reflect this change and provide the results of your revised calculations for bearing capacity, estimated settlement of the mat, static and dynamic lateral earth pressures and slope stability analysis results under the revised static and dynamic loadings.

281.5 SRP (6.1.2) Indicate the total amount of protective coatings used inside the containment that do not meet the requirements of ANSI N101.2 (1972) and Regulatory Guide 1.54. Evaluate the generation rates vs. time of combustible gases that can be formed from these unqualified organic materials under DBA conditions. Also evaluate the amount (volume) of solid debris that can be formed from these unqualified organic materials under DBA conditions that can reach the containment sump. Provide the technical basis and assumptions used for this evaluation.

281.6 SRP (9.1.3) Regarding the Spent Fuel Pool Cleanup System, provide the following information:

Describe the samples and instrumentation and their frequency of measurement that will be performed to monitor the Spent Fuel Pool water purity and need for ion exchanger resin and filter replacement. State the chemical and radiochemical limits to be used in monitoring the spent fuel pool water and for initiating corrective action. Provide the basis for establishing these limits. Your response should consider variables such as: gross gamma and iodine activity, demineralizer and/or filter differential pressure, demineralizer decontamination factor, pH and crud level.

281.7 SRP (9.3.5) The information you provided on the post accident sampling system (PASS) is inadequate to demonstrate compliance with NUREG-0737, II.B.3. The PASS will be evaluated for compliance with the criteria from NUREG-0737, II.B.3. FSAR Section 9.3.5 partially meets some of these criteria. These eleven items have been copied verbatim from NUREG-0737. System schematics with sufficient information to verify flow paths should be included, consistent with documentation requirements in NUREG-0737, with appropriate discussion so that the reviewer can determine whether the criteria have been met. Further information pertaining to the specific clarifications of NUREG-0737, which will be considered.

Criterion: (1) The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

Clarification: Provide information on sampling(s) and analytical laboratories locations during a discussion of relative elevations, distances and methods for sample transport. Responses to this item should also include a discussion of sample recirculation, sample handling and analytical times to demonstrate that the three-hour time limit will be met (see(6) below relative to radiation exposure). Also describe provisions for sampling during loss of off-site power (i.e. designate an alternative backup power source, not necessarily the vital (Class IE) bus, that can be energized in sufficient time to meet the three-hour sampling and analysis time limit).

Criterion: (2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within three-hour time frame established above, quantification of the following:

(a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and non-volatile isotopes);

(b) hydrogen levels in the containment atmosphere;

(c) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

(d) Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

Clarification: 2 (a) A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation (ALARA). Also a procedure is required for relating radionuclide concentrations to core damage. The procedure should include:

1. Monitoring for short and long lived volatile and non volatile radionuclides such as ^{133}Xe, ^{131}I, ^{137}Cs ^{134}Cs, ^{85}Kr, ^{140}Ba, and ^{88}Kr (See Vol. II, Part 2, pp. 524-527 of Rogovin Report for further information).

2. Provisions to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data and sample location.

2 (b) Show a capability to obtain a grab sample, transport and analyze for hydrogen.

2 (c) Discuss the capabilities to sample and analyze for the accident sample species listed here and in Regulatory Guide 1.97 Rev. 2.

2 (d) Provide a discussion of the reliability and maintenance information to demonstrate that the selected on-line instrument is appropriate for this application. (See (8) and (10) below relative to back-up grab sample capability and instrument range and accuracy).

Criterion: (3) Reactor coolant and containment atmosphere sampling during post accident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

Clarification: System schematics and discussions should clearly demonstrate that post accident sampling, including recirculation, from each sample source is possible without use of an isolated auxiliary system. It should be verified that valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

Criterion: (4) Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

Clarification: Discuss the method whereby total dissolved gas or hydrogen and oxygen can be measured and related to reactor coolant system concentrations. Additionally, if chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is necessary. Verification that dissolved oxygen is <0.1 ppm by measurement of a dissolved hydrogen residual of $\geq$ 10 cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with ALARA, direct monitoring for dissolved oxygen is recommended.

Criterion: (5) The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

Clarification: BWR's on sea or brackish water sites, and plants which use sea or brackfish water in essential heat exchangers (e.g. shutdown cooling) that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hours. All other plants have 96 hours to perform a chlorida analysis. Samples diluted by up to a factor of one thousand are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ____ppm Cl (the licensee should establish this value; the number in tha blank should be no greater than 10.0 ppm Cl) in the reactor coolant system and (2) that dissolved oxygen can be verified at <0.1 ppm, consistent with the guidelines above in clarification no. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

Criterion: (6) The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).

Clarification: Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted man rem exposures based on person-motion for sampling, transport and analysis of all required parameters.

Criterion: (7) The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants).

Clarification: PWR's need to perform boron analysis. The guidelines for BWR's are to have the capability to perform boron analysis but they do not have to do so unless boron was injected.

Criterion: (8) If inline monitoring in used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samplies. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident, and at least one sample per week until the accident condition no longer exists.

Clarification: A capability to obtain both diluted and undiluted backup samples is required. Provisions to flush inline monitors to facilitate access for repair is desirable. If an off-site laboratory is to be relied on for the backup analysis, an explanation of the capability to ship and obtain analysis for one sample per week thereafter until accident condition no longer exists should be provided.

Criterion: (9) The licensee's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1μ Ci/g to 10 Ci/g.

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

Clarification: (9) (a) Provide a discussion of the predicted activity in the samples to be taken and the methods of handling/dilution that will be employed to reduce the activity sufficiently to perform the required analysis. Discuss the range of radionuclide concentration which can be analyzed for, including an assessment of, the amount of overlap between post accident and normal sampling capabilities.

(9) (b) State the predicted background radiation levels in the counting room, including the contribution from samples which are present. Also provide data demonstrating what the background radiation levels and radiation effect will be on a sample being counted to assure an accuracy within a factor of 2.

Criterion: (10) Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Clarification: The recommended ranges for the required accident sample analyses are given in Regulatory Guide 1.97, Rev. 2. The necessary accuracy within the recommended ranges are as follows:

- Gross activity, gamma spectrum: measured to estimate core damage, these analyses should be accurate within a factor of two across the entire range.

- Boron: measure to verify shutdown margin.

In general this analysis should be accurate within $\pm 5\%$ of the measured value (i.e. at 6,000 ppm B the tolerance is $\pm$ 300 ppm while at 1,000 ppm B the tolerance is $\pm$ 50ppm). For concentrations below 1,000 ppm the tolerance band should remain at $\pm$ 50 ppm.

- Chloride: measured to determine coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm chloride the analysis should be accurate within $\pm$ 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at $\pm$ 0.05 ppm.

- Hydrogen or Total Gas: monitored to estimate core degradation and corrosion potential of the coolant.

An accuracy of $\pm$ 10% is desirable between 50 and 2000 cc/kg but $\pm$ 20% can be acceptable. For concentration below 50 cc/kg the tolerance remains at $\pm$ 5.0 cc/kg.

- Oxygen: monitored to assess coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm oxygen the analysis should be accurate within $\pm$ 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at $\pm$ 0.05 ppm.

- pH: measured to assess coolant corrosion potential.

Between a pH of 5 to 9, the reading should be accurate within ± 0.3 pH units. For all other ranges $\pm$ 0.5 pH units is acceptable.

To demonstrate that the selected procedures and instrumentation will achieve the above listed accuracies, it is necessary to provide information demonstrating their applicability in the post accident water chemistry and radiation environment. This can be accomplished by performing tests utilizing the standard test matrix provided below or by providing evidence that the selected procedure or instrument has been used successfully in a similar environment.

STANDARD TEST MATRIX
FOR
UNDILUTED REACTOR COOLANT SAMPLES IN A POST-ACCIDENT ENVIRONMENT

Constituient	Nominal Concentration (ppm)	Added as (chemical salt)
I^-	40	Potassium Iodide
Cs+	250	Cesium Nitrate
Ba+2	10	Barium Nitrate
La+3	5	Lanthanum Chloride
Ce+4	5	Ammonium Cerium Nitrate
Cl-	10	
B	2000	Boric Acid
Li+	2	Lithium Hydroxide
NO_3^-	150	
NH_4^+	5	
K+	20	
Gamma Radiation (Induced Field)	10^4 Rad/gm of Reactor Coolant	Adsorbed Dose

NOTES:

1) Instrumentation and procedures which are applicable to diluted samples only, should be tested with an equally diluted chemical test matrix. The induced radiation environment should be adjusted commensurate with the weight of actual reactor coolant in the sample being tested.

2) For PWRs, procedures which may be affected by spray additive chemicals must be tested in both the standard test matrix plus appropriate spray additives. Both procedures (with and without spray additives) are required to be available.

3) For BWRs, if procedures are verified with boron in the test matrix, they do not have to be tested without boron.

STANDARD TEST MATRIX
FOR
UNDILUTED REACTOR COOLANT SAMPLES IN A POST-ACCIDENT ENVIRONMENT

Constituient	Nominal Concentration (ppm)	Added as (chemical salt)
I^-	40	Potassium Iodide
Cs^+	250	Cesium Nitrate
Ba^{+2}	10	Barium Nitrate
La^{+3}	5	Lanthanum Chloride
Ce^{+4}	5	Ammonium Cerium Nitrate
Cl^-	10	
B	2000	Boric Acid
Li^+	2	Lithium Hydroxide
NO_3^-	150	
NH_4^+	5	
K^+	20	
Gamma Radiation (Induced Field)	10^4 Rad/gm of Reactor Coolant	Adsorbed Dose

NOTES:

1) Instrumentation and procedures which are applicable to diluted samples only, should be tested with an equally diluted chemical test matrix. The induced radiation environment should be adjusted commensurate with the weight of actual reactor coolant in the sample being tested.

2) For PWRs, procedures which may be affected by spray additive chemicals must be tested in both the standard test matrix plus appropriate spray additives. Both procedures (with and without spray additives) are required to be available.

3) For BWRs, if procedures are verified with boron in the test matrix, they do not have to be tested without boron.

4) In lieu of conducting tests utilizing the standard test matrix for instruments and procedures, provide evidence that the selected instrument or procedure has been used successfully in a similar environment.

All equipment and procedures which are used for post accident sampling and analyses should be calibrated or tested at a frequency which will ensure, to a high degree of reliability, that it will be available if required. Operators should receive initial and refresher training in post accident sampling, analysis and transport. A minimum frequency for the above efforts is considered to be every six months if indicated by testing. These provisions should be submitted in revised Technical Specifications in accordance with Enclosure 1 of NUREG-0737. The staff will provide model Technical Specifications at a later date.

Criterion: (11) In the design of the post accident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.

Clarification: (11)(a) A description of the provisions which address each of the items in clarification 11.a should be provided. Such items, as heat tracing and purge velocities, should be addressed. To demonstrate that samples are representative of core conditions a discussion of mixing, both short and long term, is needed. If a given sample location can be rendered inaccurate due to the accident (i.e. sampling from a hot or cold leg loop which may have a steam or gas pocket) describe the backup sampling capabilities or address the maximum time that this condition can exist.

BWR's should specifically address samples which are taken from the core shroud area and demonstrate how they are representative of core conditions.

Passive flow restrictors in the sample lines may be replaced by redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close on containment isolation or safety injection signals.

(11)(b) A dedicated sample station filtration system is not required, provided a positive exhaust exists which is subsequently routed through charcoal absorbers and HEPA filters.

281-8 SRP (9.3.2) CESSAR interface requirement 5.1.4.K.3 specifies sampling of RCS during shutdown. In the FSAR Table 1.9-1 you indicate compliance to this requirement in Section 9.3.6.1. We have not found this Section. Indicate correct reference which shows compliance with the interface requirements.

281-9 SRP (9.3.2) CESSAR interface requirement Section 5.4.1.K.10 specifies that the pressurizer steam space sample line shall contain 7/32" x 1" orifice as close to the pressurizer as possible. In FSAR Table 1.9-1 you indicate that this interface requirement is satisfied in FSAR Section 9.3.2.1. Since this is not the case, verify WNP-3 conformance with this interface requirement.

281-10 SRP (9.3.2) CESSAR interface requirement Section 5.1.4.6.4 specifies that sample lines in contact with the reactor coolant, including welds, shall be designed such that the material is compatible with fluid chemistry described in CESSAR Section 9.3.4. In FSAR Table 1.9-1 you indicate that this interface requirement is satisfied in FSAR Section 5.2.3.2.1. We have not found this Section, indicate correct references which shows compliance with the interface requirement.

281-11 SRP (9.3.2) CESSAR interface requirement 5.4.7.1.3.K.1 specifies that the component cooling water pH shall be controlled between 8.3 and 10.5. The water chemistry limits in the FSAR Section 9.2.2.2.2.h for pH are 5.8-9.5. Explain the discrepancy.

281-12 SRP (9.3.4) CESSAR interface requirement 9.3.4.6.P.1 specifies the RWT sizing requirement. Table 6.3-1 in CESSAR references this Table for the requirement in FSAR Table 1.9-1. We have not found Table 6.3-1. Indicate correct reference which specifies this interface requirement.

281-13 SRP (9.3.4) Section 9.3.4 indicates that the design of the CVCS conforms to CE interface requirements as described in CESSAR-F subsection 9.3.4.6.A.1 through item R.1. You have not described how the interface requirements are met. Provide a CESSAR interface evaluation similar to that provided in Palo Verde FSAR section 9.3.4.2.

281.14 SRP (10.3.5) MTEB BTP 5-3 SRP 5.4.21

The information that you have provided is insufficient for us to evaluate the secondary water chemistry control program. Provide a summary of operative procedures to be used for the steam generator secondary water chemistry control and monitoring program, addressing the following:

1. Identify the sampling schedule for the critical chemical and other parameters and the control points or limits for these parameters for each operating mode of the plant, i.e. dry lay-up, cold shutdown, hot standby/shutdown, and power operation.

2. Identify the procedures used to measure the values of the critical parameters, i.e. standard identifiable procedures and/or instruments.

3. Identify the sampling points, considering as a minimum the steam generator blowdown, the hot well discharge, the feedwater, and the demineralizer effluent. We recommend a process flow chart similar to that in EPRI NP-2704-SR "PWR Secondary Water Chemistry Guidelines".

4. State the procedures for recording and management of data, defining corrective actions for various out-of-specification parameters.

5. Identify (a) the authority responsible for interpreting the data and initiating action (b) the sequence and timing of administrative events required to initiate corrective action.

281.15 SRP (10.4.6)

Explain how you prevent resin breakthrough into the steam generators from the full flow demineralizers.

281.16 (9.1.2)

Provide a description of any materials monitoring program for the pool. In particular provide information on the frequency of inspection and type of samples used in the monitoring program.

281.17 SRP (9.1.2)

Provide a list of the Codes and Standards used in the design and fabrication of the spent fuel racks.

321.0
321.1
(SRP 6.5.1)

Table 6.5.1-1 of SRP 6.5.1, providing guidance on minimum instrumentation for ESF atmosphere cleanup systems and Position C.2.g both call for signal, alarm, and recording of system flow rates in the control room. Section 6.5.1.5 of the WNP-3 FSAR does not indicate provisions for flow rate instrumentation on any of the three ESF atmosphere cleanup systems. Provide your justification for not including this instrumentation in the WNP-3 design.

321.2
(SRP 10.4.2, 11.3)

In the last paragraph of Section 10.4.2.1, page 10.4-6, it is stated that an evaluation of the radioactive discharge from the main condenser evacuation system is discussed in Section 11.3, together with the basis. Tables 11.3.3.1 and 11.3.3-3 of the FSAR contain entries for the condenser evacuation system exhaust, however, the discussion of the evaluation does not appear in Section 11.3

Provide a description of the release path for the discharge from the mechanical vacuum pumps of the main condenser evacuation system in the non-radioactive operational mode, including the designation of the normal (non-radioactive) atmospheric release point, together with the height, dimensions, average temperature of exhaust air, and expected range of airflow in either volumetric flow rate or linear velocity at the point of discharge; describe the procedure or mechanism for diverting flow to the reactor auxiliary building exhaust ventilation system upon detection of radioactivity in the mechanical vacuum pump exhaust.

321.3 (SRP 11.1, 11.3) There are several discrepancies in the descriptions of radioactive gaseous effluent release points between the text, tables, and figures in the Environmental Report and the Safety Analysis Report. For example, Table 3.5-8 (ER) identifies six release points, referencing ER Figure 3.5-8 as a pictorial guide. In the FSAR, the six release points are designated "HVAC Vent Stacks #1 through #6", while Table 3.5-8 (ER) gives such titles as "Fuel Handling Building Vent". Figure 3.5-8 (ER) and Figure 1.2-15a (FSAR) show discrepancies in stack height; for example, Vent Stack #4 in Figure 1.2-15a is shown to be 485 feet above mean sea level (msl) while Figure 3.5-8 (ER) shows the same vent at 502.8 feet above msl. No release point is shown for the effluent from the main condenser air ejector during normal operation -- only the alternate is shown for use during periods of known radioactive release. For all radioactive effluent release points, please provide corrected and consistent tables, diagrams, and text showing location, designation or title, release point elevation, shape and inside dimensions of release point cross-section, average effluent temperature, and either exit velocity or volumetric flow rate. The requested data is needed to adequately assess the meteorological dispersion conditions attending gaseous effluent releases.

321.4 (SRP 11.3) Your description of the Gas Analyzer Package, beginning on page 11.3.2 (Amendment 2, 12/82) does not meet the acceptance criteria of SRP 11.3. For systems which are not designed to withstand a hydrogen explosion, Section II.B.5 of SRP 11.3 states "..(gaseous waste management systems) should be provided with dual gas analyzers with automatic control functions to preclude the formation or buildup of explosive mixtures...with dual being defined as two independent gas analyzers continuously operating and providing two independent measurements verifying that hydrogen and/or oxygen are not present in potentially dangerous concentrations... control features to reduce potential for explosion should be automatically initiated... The automatic control features should be as follows... for systems designed to preclude explosions by maintaining either hydrogen or oxygen below 4%, the source of hydrogen or oxygen...should be automatically isolated from the system...(or) injection of diluents to reduce concentrations below the limits specified... If gas analyzers are to be used to sequentially measure several points in a system not designed to withstand a hydrogen explosion, at least one gas analyzer which is continuously on-stream is required...(and) should be at a point common to streams monitored sequentially..."

Your design provisions for one sequential hydrogen analyzer and one sequential oxygen analyzer, with no provisions for automatic control features, do not comply with the minimum acceptance criteria of SRP 11.3. You should provide an additional continuously-operating gas analyzer serving one fixed point, preferably between the waste gas compressor and the on-line gas decay tank. You should additionally provide for one of the automatic control features described in SRP 11.3.

321.5 (SRP 11.5) Section 9.4.2.1(a) specifies periodic radiation monitoring of air sampled from the Fuel Handling Building Exhaust duct. It is our position that such monitoring should be continuous during any time the system is in use. Provide your rationale for periodic monitoring or clarify your statement to provide for continuous monitoring.

321.6 (11.5) Provide information to show conformance with Items II.F.1, Attachments 1 and 2, NUREG-0737.

410.15 (3.4.1) For the auxiliary building or any other building housing safety-related equipment discuss the protection afforded the structure against flooding as a result of ground water.

410.16 (3.5.1.1) Discuss the protection afforded spent fuel from internally generated missiles.

410.17 (3.5.1.1) Provide the results of an analysis which shows that turbine driven pumps will not become a source of missiles or that missiles from the turbine cannot damage safety-related equipment.

410.18 (3.5.1.1) Explain why consideration was not given to the secondary effects of postulated missiles such as ricochet or missiles penetrating reinforced concrete. (FSAR Subsection 3.5.1.1.2.)

410.19 (3.5.1.2) With regards to internally generated missiles (inside containment) verify that a seismic event will not result in gravity missiles which could cause damage to essential systems required to assure a safe shutdown or result in unacceptable releases of radioactivity.

410.20 (3.5.1.2) Verify that analyses of potential missile sources inside containment have included the reactor head bolts in addition to the possible missile sources considered in your FSAR.

410.21 (3.5.1.2) Confirm that secondary missiles, if any, generated by impact of the primary missiles inside containment will not cause damage to essential systems required to assure a safe shutdown or result in unacceptable releases of radioactivity.

410.22 (3.5.1.2) Discuss why you do not consider the likelihood of internally generated missiles inside containment caused by postulated failures of rotating equipment.

410.23 (3.5.1.4) The FSAR refers to CESSAR-F subsection 4.2.5.B.1 for compliance with interface requirements imposed by this subsection but this referred subsection does not exist in the standard CESSAR-F. Correct this discrepancy.

410.24 (3.5.1.1 & 3.5.1.2) Provide a discussion of the methodology used as the basis for determining that the safety-related structures, systems, and components are adequately protected against internally generated missiles.

410.25 (3.6.1) Provide typical layout drawings of safety-related areas outside containment showing the routing of high and moderate energy piping systems and their relative position to safety-related equipment and components. These drawings should identify postulated break and crack locations in high and moderate energy lines. Further, provide a table which identifies the means of protection (i.e., pipe whip restraint, jet impingement barrier, separation, floor drainage, etc.) for safety-related equipment from the effects of the postulated high and moderate energy pipe breaks.

410.26 (3.6.1) It is our position that the compartments which house the main steam lines, feedwater lines and the isolation valves be designed to consider the environmental effects (pressure, temperature, humidity) and potential flooding consequences from an assumed crack of one square foot. The essential equipment, including the atmospheric dump valves, main steam isolation valves and feedwater isolation valves and their operators, and the essential auxiliary feedwater pumps and associated equipment should be capabale of operating in the environment resulting from the above crack. Further, if this assumed crack could cause the structural failure of these compartments, then the failure should not jeopardize the safe shutdown of the plant.

We, therefore, request that you submit a subcompartment pressure analysis to confirm that the design of the above compartments conforms to our position as outlined above. When you submit the results of your evaluation, identify the computer codes used, and the assumptions used for mass and energy release rates. The peak pressure and temperatures resulting from the postulated break of a high energy pipe located in these compartments are dependent on the mass and energy flows during the time of the break. Therefore, for the pipe break or crack analyzed, provide the total blowdown time and the mechanism used to terminate or limit the time of blowdown flow so that the environmental effects will not affect safe shutdown of the facility.

Also provide a similar analysis for other compartments outside containment in the vicinity of safety-related structures, systems and components which house high energy lines such as CVCS charging, letdown and steam generator blowdown.

410.27 (3.6.1) The FSAR refers to the interface requirements of CESSAR-F Subsections 5.4.7.1.3.6.1, 5.4.7.1.3.8.1, and 9.3.4.6.C.1, but these references do not exist in the standard CESSAR-F. Refer to the correct subsections.

410.28 (3.6.1) Discuss the capability of individual safety-related systems to mitigate the consequences of pipe breaks assuming a single failure as identified in the criteria of BTP ASB 3-1.

410.29 (5.2.5) Provide the following additional information concerning leakage from the reactor coolant pressure boundary:

(a) Discuss how the reactor drain tank (RDT) can be used to detect leakage of primary coolant to the shutdown cooling system as identified in FSAR Section 5.2.5.1.5.

(b) Describe the means of detection of leakage of primary coolant from the CVCS, reactor coolant pump seals and other radioactive fluid sources to normally non-radioactive systems such as the nuclear cooling water system.

(c) Verify that the containment radioactive gas and air particulate monitor has an accuracy of 1 gpm or better in accordance with the guidelines of Regulatory Guide. 1.45.

410.30 (9.1.1) Verify that the vault housing the new fuel storage racks is not located in the vicinity of any moderate or high energy lines or rotating machinery to insure physical protection for the new fuel from internally generated missiles and the effects of pipe breaks.

410.31 (9.1.2) Figures 9.1.2-1, 9.1.2-2a, and 9.1 2-2b are missing from FSAR. Provide these figures.

410.32 (9.1.2) Verify that the spent fuel pool is not located in the vicinity of any high-energy lines or rotating machinery to ensure physical protection for the fuel from internally generated missiles and the effects of pipe breaks.

410.33 (9.1.3) Is there any portion of the spent fuel pool cooling and cleanup system designed to non-seismic category requirements? If so, verify that failure of the non-seismic Category I portion in an earthquake will not affect the operation of the cooling trains.

410.34 (9.1.4) With regards to the overall heavy load handling systems verify that your design meets the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Phase I and II and provide sufficient information so that we can make an independent evaluation of how the guidelines of NUREG-0612 are met.

410.38 (9.4) Some of the licensees have provided measures for detecting and correcting dust accumulation on safety-related equipment in order to assure their availability on demand. Verify that dust accumulation doesn't pose a problem in this plant.

410.39 (9.4) Describe the effect on the safety function of the essential HVAC systems in the event of a single failure in a fire damper in the ventilation system ducts. It is our position that such a failure not compromise the safety function of the HVAC system.

410.40 (9.4.1) The FSAR states that the HVAC system, with the exception of some ductwork located in office areas and emergency living quarters, is designed to seismic Categoy I requirements. Verify that the control room HVAC air intake chlorine and radiation monitors are seismic Category I.

410.41 (9.4.1) In the event of indication of radioactive contamination of the normal control room intake, the normal ventilation system is automatically shutoff and isolated as the essential control room system is started. However, the control building normal air handling unit or essential ESF switchgear room air handling unit (if operating) would continue to function and circulate potentially contaminated air to other areas of the control building. Describe the measures provided to prevent contamination of vital areas of the control building and still assure a proper environment for operation of essential equipment.

410.42 (9.4.1) Verify that a single failure in any safety-related damper or total failure of all nonsafety-related dampers and ducts in the ESF switchgear, ESF equipment and battery rooms HVAC system will not prevent at least one train of the essential ESF switchgear room HVAC system from performing its safety function.

410.43 (9.4.1) Describe the measures for assuring a proper operating environment for essential control room and ESF switchgear room air handling units when the normal control building HVAC system is not available during emergency conditions.

410.44 (9.4.2) It is our position that those portions of the fuel building ventilation system utilized during the emergency filtration modes be seismic Category I and that the system be designed so that failure of the nonsafety portion of the system will not compromise the operability of the safety-related portion. Verify that your plant complies with the above position.

410.45 (9.4.3) Describe the means provided for isolating the radwaste building ventilation system following a design basis event (such as SSE) in order to prevent the release of potentially radioactive airborne contaminants through building openings.

410.35 (9.2.5) In accordance with Regulatory Guide 1.27, a continuous capability to maintain the plant in a safe shutdown condition for at least 30 days is recommended for the ultimate heat sink. The FSAR refers to an analysis of the 30-day period following a design basis accident but the analysis is inconclusive due to the following missing information:

(a) Table 9.2.5-5 (total heat load from component cooling water system) provides the data up to 25.9722 hours only, not for 30-day period.

(b) Figures 9.2.5-2a through 9.2.5-2d have not been supplied yet.

(c) Table 9.2.5-3 indicates that certain data will be provided later.

410.36 (9.3.1) Concerning the compressed air system, provide the following information:

(a) Describe the means provided to verify that proper instrument air quality will be maintained over the plant life to assure the safety function of the system (i.e., air operated valves will fail in their safe position on loss of instrument air supply). Include the air quality limits which should not be exceeded in order to assure the above safety function.

(b) Verify that a single failure of any air operated valve to assume its fail safe position will not prevent the function of a safety-related system or compromise the ability to safely shutdown.

(c) Identify the testing requirements and frequency of tests for the accumulators and check valves provided within the compressed air system.

(d) Discuss how the safety-related portions of the system meet the requirements of GDC 2 and 4 regarding protection against natural phenomena, missiles, and environmental effects.

(e) Verify that failure of this nonsafety-related system does not affect safety system functions.

410.37 (9.3.3) Regarding internal flooding verify that adequate protection has been provided for safety-related equipment assuming a pipe rupture for all non-seismic piping systems (such as fire protection system and cooling water system) and components (such as tanks) located in safety-related areas. This protection cannot assume credit for non-seismic Category I sump pumps. Your response should include the time required for operator action if necessary to provide protection of essential equipment once indication from the class 1E level switches is given.

410.46 (9.4.4) Figure 9.4.4-1 showing the turbine building ventilation system is missing from the FSAR. Provide the missing figure.

410.47 (10.3) Identify main steam isolation valves (MSIVs) in Figure 10.3-1. In order to prevent blowdown of more than one steam generator, verify that the main steam isolation valves are designed to stop full main steam flow at the maximum design differential pressure in both directions in the event of a main steam line break in one steam line upstream of an MSIV and corresponding single failure (to close) in an MSIV to the other steam generator.

410.48 (10.4.7) The FSAR refers to CE interface requirement of CESSAR-F Subsection 6.1.4.E.2 but this subsection does not exist in the CESSAR-F. Clarify this discrepancy and refer to the correct subsection.

410.49 (10.4.7) The FSAR states that the portion of the main feedwater system piping from the Break Exclusion Area at the Reactor Auxiliary Building to the steam generator feed nozzles are seismic Category I. Does this portion include the main feedwater isolation valves? Identify these valves on appropriate figures.

410.50 (10.4.7) FSAR Section 10.4.7.3.2(b) discusses compliance with CE interface requirements of CESSAR-R Subsection 5.4.4.I.10. This subsection requires maintaining a leak rate of less than 1000 cc/hr under certain main feedwater line break conditions. The FSAR does not cover the above interface requirement. Clarify the discrepancy.

410.51 (10.4.7) There is a discrepancy in FSAR Subsection 10.4.7.6(j) regarding CE interface requirement of CESSAR-F Subsection 5.1.4.M.15. This interface requires 90° elbow facing downward attached to feedwater nozzle whereas the FSAR discusses inservice inspection to meet this subsection requirement. Explain the discrepancies.

410.52 (10.4.7) It is our position that you commit to perform a steam generator/feedwater water hammer test in accordance with the guidance for preheat type steam generators as identified in NUREG/CR-1606, "An Evaluation of Condensation-Induced Water Hammmer in Preheat Steam Generators." The following procedure should be followed:

"Run the plant at approximately 15% of full power by using feedwater through the downcomer nozzle at the lowest feedwater temperature that the plant Standard Operating Procedure (SOP) allows. Switch the feedwater at that temperature from the downcomer nozzle to the economizer nozzle by following the SOP. Observe and record the transient that follows."

410.53
(10.4.9) Provide a response to the staff's March 10, 1980 letter to near-term operating license applicants concerning your EFW system design (TMI-2 Task Action Plan, NUREG-0737, Item II.E.1.1). This response should include the following:

(a) A review of the EFW system design against Standard Review Plan Section 10.4.9, and Branch Technical Position ASB 10-1.

(b) A review of the EFW system design, Technical Specifications and operating procedures against the generic short-term and long-term requirements discussed in the March 10, 1980 letter.

(c) The design basis for the EFW flow requirements and verifications that the EFW system will meet these requirements (refer to Enclosure 2 of the March 10, 1980 letter).

450.1 (SRP 6.5.2) FSAR Section 6.5.2 describes the operation of the containment spray system as a fission product removal system. In this Section, the FSAR states that a nitrogen cover gas is provided as "the driving head for the tank contents to the containment spray lines." This Section also states that the NaOH injection rate is controlled by the flow rate of the containment spray pumps. Describe in greater detail the NaOH injection system including: the mixture weight percent of NaOH, the design overpressure of the nitrogen cover gas, the method by which the pump flow rates control the injection rate (including any operator actions necessary to manipulate the control valves), the time delay expected from automatic initiation of the spray system until the preset minimum flow is established and chemical addition begins, and how a control room operator can determine if the NaOH addition rate will adjust the spray pH within the prescribed pH range.

450.2 (SRP 6.5.2) FSAR Figure 6.5.2 references to Figure 6.2.2-1. Please provide the proper reference as there is no such figure in the FSAR.

450.3 (SRP 6.5.3) FSAR Tables 6.5.2-4 and 15-1 list the containment free volume as 3,404,696 cubic feet while FSAR Tables 6.2.1-3 and 6.5.3-1 list the containment free volume as 3,218,00 cubic feet. Provide the actual design containment free volume and revise the FSAR Tables as appropriate.

450.4 (SRP 6.5.3) FSAR Table 6.5.3-1 identifies the leak rate for the primary containment and specific fractions of the containment leakage to particular pathways. Provide the basis for the specified leakage fractions. The staff also notes that the "conservative" case in Table 6.5.3-1 is non-conservative with respect to offsite consequences because the "anticipated" case contains more unfiltered direct leakage to the environment.

450.5 (SRP's 9.4.2, 9.4.3) The systems descriptions of the safety-related and non-safety related portions of the Fuel Handling Ventilation System, Reactor Auxiliary Building Main Ventilation System and the ECCS Area/Fuel Handling Building Filtered Exhaust Systems are not provided in sufficient detail for us to complete our review. It is not clear how these three systems interact during an accident. Provide one drawing which shows the interconnection of all these systems and a description of the alignment of the isolation dampers in these systems for both normal operation and for the loss-of-coolant and fuel handling accidents.

450.6 (SRP 15.6.3) The operating procedures for responding to a steam generator tube rupture are currently an open issue on CESSAR. Resolution of the bases for analysis of this accident must either be accomplished for CESSAR, or on a plant specific basis for WNP-3. In order to resolve the issue for WNP-3, the following question must be answered. The current operating procedures call for the operators to steam the affected steam generator to prevent overfilling. The present steam generator tube rupture accident evaluation in the FSAR assumed that no releases occur from the affected steam generator after 30 minutes. Describe why the CESSAR FSAR evaluation is bounding for a steam generator tube rupture event in light of the operator action guidance.

450.7 (SRP 15.6.5) The containment pathway fractions given in FSAR Table 15-1 are not consistent with the fractions presented in FSAR Table 6.5.3-1. Resolve this apparent discrepancy and modify the FSAR as necessary. Please identify your intent to include techincal specifications for all assumed containment leakage pathway fractions.

450.8 (SRP 15.6.5) The information presented in FSAR Chapter 15.6.5, Table 15-1 and Appendix 15I is not sufficient for us to complete our review of the potential radiological consequences following the postulated loss-of-coolant accident. Table 15-1 does not contain: 1) the recirculation and exhaust flow rates for the Shield Building Ventilation System used in the analysis, 2) the exhaust flow rate for the Controlled Ventilation Area System, and 3) the potential sources of ECCS leakage and the estimated leak rates. Provide this information.

450.9 (SRP 15.6.5) On Page 15I-6 of Appendix 15I of the FSAR it is stated that the post-accident containment leakage is limited by technical specifications to 0.2 percent of the containment volume per day for the first 36 hours, and then 50% of this value for the duration of the accident. Table 15-1 lists the containment leak rate as 0.5 percent per day for the first 24 hours and then 50% of this value for the duration of the accident. Identify the proper set of containment leakage assumptions used in calculating the EAB LOCA dose of 250 Rem (as given in FSAR Table 15-3) and modify the FSAR as necessary. (Note that FSAR Table 6.5.3-1 also prescribes a containment leak rate of 0.5 percent per day).

450.10 (SRP 15.7.4) Standard Review Plan Section 15.7.4 requires an evaluation of the offsite consequences following a fuel handling accident inside containment. Provide this analysis, including all the assumptions used, and describe the method of detection and the response times of the ventilation systems. Provide a drawing which identifies the location of the monitors used and the exhaust locations for the ventilation system with respect to the refueling pool.

450.11 (SRP 15.7.4) Provide a drawing which identifies the locations of the redundant radiation monitors above the spent fuel pool as well as the exhaust intakes for the fuel handling building ventilation system with respect to the spent fuel pool.

450.12 Because WNP-3 intends to reference CESSAR for certain accidents, demonstrate how the interface requirements for CESSAR are met.

451-3
SRP 2.3.3

Provide the dates when the onsite meteorological measurements were ceased and will be restarted.

451-4
SRP 2.3.3

If precipitation measurements have continued between 1980 and the present, have any rainfall amounts exceeded previous onsite measured amounts for 24 hours, monthly or annual totals provided in Tables 2.3-86, 2.3-87, and 2.3-88? Provide the amounts and identify the corresponding dates, months and year.

451-5
SRP 2.3.3

As a result of differences in the height and location information for the meteorological tower in Sections 2.3.3.1 and 11.3.1 of the FSAR, provide the correct height and location of the onsite meteorological tower.

451-6
SRP 2.3.4

Provide the value of "A", the smallest vertical plane cross-sectional area of the reactor building used for calculating short term relative concentration (X/Q) values.

451-7
SRP 2.3.1

In accordance with Regulatory Guide 1.70 Section 2.3.1.2, provide an estimate of the weight of the 100-year return period snowpack and the weight of the 48-hour Probable Maximum Winter Precipitation for the site vicinity. Using these estimates provide the weight of snow and ice on the roof of each safety-related structure.

471.10 NUREG 800, Standard Review Plan, lists Reg. Guide 8.8 as an acceptable means of meeting the requirements of 10 CFR 20, 20.1(c). In accordance with Regulatory Guide 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 NUREG 800, Standard Review Plan, lists several Regulatory Guides and NUREGS as programs acceptable to meet the Regulations. Several of these Reg. Guides and NUREGS have been referenced in your FSAR as having been "used as guidance" or as "the technical basis." You should indicate if the guidance in the Regulatory Guides and NUREGs listed below were fully implemented. If not, the particular guidance not followed should be specified and an alternative control described.

. Regulatory Guide 1.8 as it applies to personnel qualifications in Section 12.1.2.

. Regulatory Guide 1.140 as it applies to ventilation design features in Section 12.3.3.3.

. Regulatory Guide 8.2 as it applies to instrumentation in Section 12.3.4.

. Regulatory Guide 8.8 as referenced in section 12.4.1.1.

. Regulatory Guide 8.8 and Regulatory Guide 1.97 where they apply to Health Physics instrumentation selection in section 12.5.2.2.

. Regulatory Guide 8.4, Regulatory Guide 8.8 and Regulatory Guide 8.14 as they apply to selection of personnel monitoring instruments in section 12.5.2.4.

. NUREG-0041 as it applies to respiratory protection devices in Section 12.5.2.4.

. Regulatory Guide 8.9, Regulatory Guide 8.20 and Regulatory Guide 8.26 as they apply to your bioassay program in section 12.5.3.4.2.

471.12 10 CFR 20.103(b)(1) specifies that the licensee shall use process or other engineering controls to the extent practicable to limit the concentrations of airborne radioactive material to below the specified in Appendix B, Table 1, column 1 for any room, enclosure, or operating area; or when averaged over the number of hours in any week during which individuals are in the area, exceeds 25% of the amounts specified in Appendix B, Table 1, column 1. Table 12.2.2.-2 of the FSAR lists the fraction of Maximum Permissible Concentration in air for "those areas normally occupied by operating personnel" as .52 MPC for the Fuel Handling Building and 184 MPC for the containment building. In addition, Table 12.2.2-3 lists

43 areas with airborne concentrations greater than MPC. Describe why additional process or engineering controls are not used to lower these concentrations in each of these areas as required by 10 CFR 20.103(b)(1). (SRP reference, section 12.3-12.4.II.3)

471.13 As specified in Section 12.1.1 of Regulatory Guide 1.70, you should describe the management policy related to ensuring that occupational radiation exposures are ALARA. Describe related activities to be conducted by the management individuals having responsibility for radiation protection and the policy of maintaining occupational exposures ALARA. In particular, identify those individuals responsible for ensuring effective control in the major ares listed in Section 12.1.1.1 of WNP-3 FSAR.

471.14 The dose breakdown for refueling work given in table 12.4-8, adds up to 48% more dose than that given as the total in Table 12.4-2. You should resolve this descrepancy.

471.15 Section 12.5.3.1.2 of WNP-3 FSAR provides for a general exemption of health physics personnel or personnel escorted by health physics personnel. This is not in compliance with the 10 CFR 20.203 requirement of maintaining "positive control over each individual entry" to a high radiation area, and should be deleted from the FSAR. (SRP reference, section 12.3-12.4.II.5)

471.16 As per the requirements of NUREG-800, 12.3-12.4 Facility Design Features, describe the local audible and visible alarming radiation monitors that alert personnel if the lead shot bags, provided for shielding the fuel transfer tube access way, are removed during fuel transfer operations.

471.17 Section 12.5.2.2.4 of the FSAR commits to testing pocket ion chambers in accordance with Regulatory Guide 8.4, February 26, 1983, with the exception of a +20 to -10% vice a $\pm$ 10% tolerance limit. Provide the technical basis for widening the acceptable tolerance limits stated in Regulatory Guide 8.4.

471.18 Regulatory Guide 8.27 "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants", specifies refresher training should "occur annually, as a minimum." section 12.5.3.5 specifies training "each two years" for persons permanently assigned to nuclear facilities. Provide the basis for this deviation from the criteria of Regulatory Guide 8.27.

471.19 From the resumé listed in Appendix 13B of WNP-3 FSAR, your Radiation Protection Manager does not meet the criteria of Regulatory Guide 1.8. Provide additional information outlining the RPM's experience, particularly that which applies to his radiation protection work in an actual nuclear power station. Also, your Radiation Protection Manager back-up coverage is not discussed. The qualifications of the individual who will act as RPM in the RPM's absence (e.g., while on vacation) should be described in the FSAR. It is our position that the temporary replacement 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 on-site (in accordance with the December 1979 draft of ANSI 3.1). Describe your plan to meet these qualifications for your RPM back-up.

471.20 Based on information contained in NUREG-0731,"Criteria for Utility Management and Technical Competence," it is our position that the radiation protection group should be a separate organization from the chemistry group. Section 12.5.1.2 of your FSAR indicates your radiation protection and chemistry technicians are combined. These technicians report via two technical staffs to the RPM. Chemistry and radiation protection are two separate specialties therefore, a qualified technician must meet the work experience requirement (4.5.2 of ANSI N18.1 - 1971) for each individual. Also, it is our position (based on NUREG-0731) that the chemistry staff report to a Technical Manager other than the RPM. Your FSAR should be revised to outline how your planned radiation protection program reflects these positions.

471.21 As per the Standard Review Plan (NUREG-0800) Section 12.3.-12.4.II. 4.b.1, describe how your airborne radioactivity monitoring system will detect ten MPC-hours of radioactivity (particulate, iodine, and noble gases) from any compartment which has a possibility of containing airborne radioactivity and which normally may be occupied by personnel.

471.22 Provide the information requested in II.F.1.(3) and III.D.3.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements."

471.23 As per the Standard Review Plan, NUREG-0800, commit to the implementation, or provide a description of your alternative approach, for the following:

- . Regulatory Guide 1.97, as it applies to providing radiation monitoring instrumentation following an accident.
- . Regulatory Guide 8.12 and ANSI N16.2-1969, as they relate to the requirements for a criticality accident alarm system.
- . Regulatory Guide 8.19 as it relates to your method of performing assessments of collective occupational radiation dose as part of the ongoing design review process so that exposures will be ALARA.
- . ANSI/ANS-HPSSC-6.8.1-1981 as it relates to the criteria for locating fixed continuous area gamma radiation monitors, and for design features and ranges of measurement.

. Stone and Webster Topical Report, RP-8, 1974, as it relates to methods of analysis employed in determine shielding requirement.

. Regulatory Guide 8.14 as it relates to use of personnel neutron dosimeters where exposure to neutrons occur.

471.24 A portion of the information provided on Figures 12A-1 through 12A-14 is not readable. Provide legible copies of these fiqures.

471.25 In accordance with Regulatory Guide 1.70 "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", Rev. 3, Section 12.3.1, it is our position that FSAR should include plant layouts showing shield wall thicknesses. The shield thickness of major radioactive equipment can be provided in a separate table. Section 12.3.2 of your FSAR should be revised to comply with Section 12.3.1 of Regulatory Guide 1.70.

630.3 Provide 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.

630.4 Provide a 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.

630.5 Discuss the program which will provide the training to Reactor Operators and Senior Reactor Operators in the following areas: SRP reference 13.2.1, I.B.I, II.1.b.

(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.

630.6 Discuss the certifications completed pursuant to Sections 55.10(a)(6) and 55.33a(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. Enclosure I of H. R. Denton's March 28, 1980 letter Section A.3.

630.7 Provide the duration (approximate number of weeks in full time attendance) of the following courses: SRP reference 13.2.1, I.B.1

(a) Academic Fundamentals

(b) Plant Systems - Classroom

(c) Operating Practice

(d) Control Room Operating Experience

630.8 Discuss the difference in the training programs for individuals who will be seeking licenses prior to criticality pursuant to Section 55.25 of 10 CFR Part 55 based on the extent of previous nuclear power plant experience. Experience groups should include the following: SRP reference 13.2.1, I.8.7

(a) Individuals with no previous experience.

(b) Individuals who have had nuclear experience at facilities not subject to licensing.

(c) Individuals who hold, or have held, licenses for comparable facilities.

630.9 Provide the length of the course in weeks for each of the following courses: SRP reference 13.2.1, 2.B.1.

(a) NSSS Lecture Series

(b) Balance of Plant Systems

(c) Senior Operators and Shift Managers

630.10 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, WNP-3 is not subject to the 1-year experience requirements for cold license SRO candidates. However, after 1 year of station operation, we will require WNP-3 to comply with the 1-year experience requirement for hot license SRO applicants.

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

(c) The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license; 80% overall and 70% each category.

630.11 With regard to the fire brigade training program, provide additional discussion of the drills and records in accordance with the guidance outlined in the NUREG-0800, Standard Review Plan Section 13.2.2, II.6.a(iii) and (iv).

630.12 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.

The Chemical Engineering Branch has reviewed FSAR Sections 3.8.1 Concrete Containment, 3.8.2 Steel Containment System, 4.5.1 Control Rod Drive Materials, 4.5.2 Reactor Internals, 5.2.3 Materials Selection, Fabrication and Processing, 6.1.1 Engineered Safety Features Materials, 9.2.1 Service Water System, 9.2.2 Component Cooling Systems for Reactor Auxiliaries, and 10.3.6 Main Steam and Feedwater System Materials, for Materials Compatibility and Corrosion Aspects. Section 3.8.2 is not applicable since a steel lined reinforced concrete containment is being used as described in Section 3.8.1. We are providing the following evaluation in accordance with our secondary review responsibility which includes environmental compatibility.

The materials of construction exposed to the reactor coolant, secondary coolant and containment sprays are compatible with the expected environment as proven by extensive testing and satisfactory performance, and conform to ASME Boiler and Pressure Vessel Code, Sections II, III and IX. General corrosion of all materials is expected to be negligible except for carbon and low alloy steels. Where these materials are exposed to the secondary coolant, general corrosion is expected to be negligible; where these materials might be exposed to leaking primary coolant, their behavior can be readily observed as part of the inservice visual and/or nondestructive inspection program performed to ensure their integrity for subsequent service.

The external nonmetallic insulation to be used on austenitic stainless steel components conforms with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." The onsite cleaning and cleanliness controls during fabrication and erection conform to the regulatory positions of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants." The use of materials of proven performance in service and the conformance with the recommendations of the stated regulatory guides and codes constitutes an acceptable basis for satisfying the requirements of NRC General Design Criterion 4, "Environmental Design," in regard to the compatibility of materials and components with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

The selection and use of the materials further satisfies the requirements of GDC 14, "Reactor Coolant Pressure Boundary" as it relates to the design having an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Service water and component cooling systems are designed to conform to General Design Criteria 2, 4, 5, 44, 45 and 46. The major portion of the system, which is continually in use, is monitored and observed by shift personnel to ensure continued safe operation.

Based upon our evaluation above, we conclude that the materials used in the above systems are acceptable from the corrosion point of view.