Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

Docket No. 50-460
January 11, 1983
G01-83-0013

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

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

Subject: NUCLEAR PROJECT NO. 1
AUXILIARY SYSTEMS BRANCH
REQUEST FOR ADDITIONAL INFORMATION

Reference: Letter, TM Novak, NRC, to RL Ferguson, Supply System, "Request for Additional Information on Washington Nuclear Project, Unit No. 1", dated November 26, 1982

The reference requested a response to the subject requests for additional information within 30 days.

To respond within the requested period would require that we temporarily stop our efforts of documenting deviations from the Standard Review Plans (SRPs), and even if this effort were stopped we would probably not be able to respond to the 102 questions within 30 days. We have discussed this problem with Mr. Mohan Thadani of your staff and he replied that we should propose a schedule.

We have reviewed the requests for additional information and the status of our SRP effort and propose that we schedule submittal of FSAR Amendment No. 2 for April 1983. The amendment will provide the SRP deviation material completed at the time and responses to about 74 of the 102 questions with schedules provided for responding to the remaining requests.

B001

Because the staff has indicated that review of the FSAR will proceed on a "manpower available" basis, we believe our SRP effort should go forth as rapidly as possible so that it will be of maximum benefit to the staff in conducting its FSAR review.

GD Bouchey

GD Bouchey, Manager
Nuclear Safety & Regulatory Programs

GDB:AGH:caa

cc: M. Thadani, NRC
NS Reynolds, Debevoise & Liberman
EG Ward, B&W
G. Valentenyi, UE&C (8U6)
FDCC (899)
ORM (847)

WNP-2 COMMITMENT(s) MADE TO THE NRC

LETTER NO: DATE:

TO: H R Denton LOCATION: NRC

SUBJECT: WNP-1 Auxiliary Systems Branch Request For Additional Information

REFERENCE(s): Letter TM Novak to RL Ferguson, same subject, dated Nov 26, 1982

COMMITMENT(s) (description)	ACTION
Submit FSAR Amendment No 2 April, 1983 • Respond to 74 of 120 Questions and provide schedule for remainder	A.G. Hosler

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

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Docket No. 50-460

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

Dear Mr. Ferguson:

Subject: Request for Additional Information on Washington Nuclear Project, Unit No. 1

In our letter dated July 16, 1982, regarding the acceptance of your application for operating license for Washington Nuclear Plant Project, Unit No. 1, it was indicated to you that it is the staff's intent to proceed on a "manpower available" basis with review of those portions of the application which parallel other current applications of similar design or with similar features. In accordance with this intent the staff is reviewing the appropriate portions of the Final Safety Analysis Report (FSAR) for Washington Nuclear Project, Unit 1, and is in the process of developing input for the Safety Evaluation Report (SER). In the course of this review, the Auxiliary Systems Branch has identified, in the enclosure, additional information necessary in order to complete the review and prepare an input to the Safety Evaluation Report.

Please supply the additional information within 30 days of the receipt of this letter. Should you have any questions on the attached, contact Mr. Mohan Thadani at (301) 492-8941.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Elinor G. Adensam
for Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

cc: See next page

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WNP

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AUXILIARY SYSTEMS BRANCH
REQUEST FOR ADDITIONAL INFORMATION
WASHINGTON NUCLEAR PROJECT NO. 1
DOCKET NO. 50-460

410.1 (3.4.1) Are the compartments containing safety related equipment watertight? Are the interior doors to these compartments watertight? Are the wall penetrations (both electrical and mechanical) watertight? What is the design pressure for these water seals?

410.2 (3.4.1) In the analysis for internal flooding, no operator action was assumed for 10 minutes. Provide the results of a similar analysis which includes the following additional assumptions:

- No operator action for 20 minutes in the control room, 30 minutes if required outside of the control room.

- All non-seismic Category I piping, tanks, sumps, valves, and equipment fail.

Provide the minimum flood level required to affect each safety related component.

410.3 (3.5.1) Verify that high pressure gas bottles and accumulators were considered as potential missiles in your analysis of internally generated missiles impacting on safety related equipment (both inside and outside containment). If this was not the case, discuss how you will take these potential missile sources into consideration in the design of the plant.

410.4 (3.5.1) Verify that your analysis included the generating or impinging of internally generated missiles on safety related equipment required to achieve and maintain cold shutdown. If this was not the case, discuss the procedure for achieving and maintaining cold shutdown for each identified internally generated missile assuming the failure of all cold shutdown equipment in the path and range of the missile.

410.5 (3.5.1) Verify that any internally generated missile from safety related equipment will not affect the redundant safety related train.

410.6 (3.5.1) Verify that the following potential missile sources inside containment have been included in your evaluation and that safety related equipment has been protected.

1. Reactor vessel

   a. closure head nut

   b. incore instrumentation assembly

2. Steam Generator

   a. primary manway stud and nut

   b. secondary handhole stud and nut

   c. secondary manway stud and nut

3. Pressurizer

   a. safety valve with flange

   b. safety valve flange bolt

   c. relief valve with flange

   d. safety valve from bonnet flange

   e. lower temperature element

   f. manway stud and nut

4. Main coolant piping temperature nozzle with resistance temperature detector

5. Surge and spray piping wells with resistance temperature detector assembly

6. Reactor coolant pump thermowell with resistance temperature detector

7. Shutdown cooling valve stem

8. Reactor coolant pump mounting flange leakoff connections

410.7 (3.5.2) The general arrangement drawings do not indicate tornado missile protection for equipment listed below. Provide detailed drawings of the tornado missile protection for each of the following:

- diesel generator exhausts

- atmospheric dump valve exhausts

- safety relief valve exhausts

- every heating and ventilating system intake and exhaust

- remote air intakes for the control room ventilation system

- spray ponds

- spray trees

410.8 (3.6.1) The figures listed in Table 3.6-2 are not in the FSAR. Provide drawings for all high energy lines outside of containment which include all pipe anchors, break locations, and high stress locations.

410.9 (3.6.1) The construction permit application was tendered on July 16, 1973. The FSAR states that the criteria used for pipe breaks outside of containment ~~is~~ are in the Branch Technical Position ASB 3-1. Based on the date the construction permit application was tendered for WNP-1, BTP ASB 3-1 Paragraph B.4.b requires the applicant to conform to the July 12, 1973 letter from J. F. O'Leary or the position itself. It is not clear in the FSAR which option was selected for WNP-1. Provide a statement of conformance to either the J. F. O'Leary letter or the Branch Technical Position itself.

410.10 (4.6) The FSAR states that the cooling water for the control rod drive mechanisms (CRDMs) is not required to maintain the ability to trip. With the reactor at full power, how long can the CRDMs be without cooling water before the insertion time of the control rod assemblies is affected? Is the loss of cooling water to a CRDM alarmed in the control room? If the time required for operator action is twenty minutes or less (30 minutes if the required action is outside of the control room) from the time of the alarm, provide a discussion of the operating procedure to bring the plant to cold shutdown. Provide drawings of CRDMs showing the details of the mechanisms and layout drawings showing the routing of the cooling water piping and the electrical cables up to and including the containment penetrations.

410.11 (4.6) All control rods need to be tested monthly and after each refueling. Discuss the procedures used to perform each test.

410.12 (4.6) Verify that the soluble poison control is capable of maintaining the core subcritical under conditions of cold shutdown independent of the control rods.

410.13 (5.2.5) In measuring the identified leakage rate, the FSAR states there are two level sensors that monitor the water flow through a weir. One sensor is connected to the plant computer and the second sensor is connected to a level indicator on the MSS Primary Panel. An annunciator is activated when the 1.0 gpm flow rate is exceeded. Verify that the level indicator on the MSS Primary Panel activates the annunciator.

410.14 (5.2.5) The FSAR states that each radioparticulate and radiogas monitor provides a signal to two sets of electronics. One set of electronics provides an output in μCi/cc and the other set uses a digital discriminator set at 2 MEV for radioparticulate and 1 MEV for radiogas monitoring. The FSAR is not clear as to what the sensitivity for the monitors and, ultimately, the readouts are. Verify that the radioparticulate and the radiogas instrumentation detect and indicate a radioactivity of 10^{-9} and 10^{-6} μCi/cc, respectively.

410.15 (9.1.1) Describe what the "optimum credible moderation" condition is. How is this condition different from optimum moderation? What is the K_{eff} for the new fuel storage facility under the optimum moderation conditions?

410.16 (9.1.1) The discussion provided in the FSAR is not clear with respect to the design of the new fuel storage racks. Provide a detailed drawing of the fuel cell connections to the fuel rack structure and the dry empty weight of a fuel cell. Verify that an uplift force equal to the dry empty weight of a fuel cell will cause the failure of the attachments of the cell to the structure. Assuming the fuel cell is removed from the rack when a fuel assembly is being removed, discuss the actions to be taken by the operator and how the rack would be repaired.

410.17 (9.1.2) Verify conformance with the ANS 57.2 standard, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," as specified by NUREG-0800, Standard Review Plan, Section 9.1.2.

410.18 (9.1.2) Verify that the spent fuel pool liner and all gates which separate the spent fuel pool from the cask loading area and the transfer canal are seismic Category I.

410.19 (9.1.2) Provide a discussion and drawings of the spent fuel pool liner leakage detection system.

410.20 (9.1.2) In order to facilitate independent evaluation of the criticality of the spent fuel storage racks please provide the following additional data:

1. The inner or outer (specify which) dimension of the two stainless steel tubes,

2. The stainless steel type,

3. The nominal values of the density of boron carbide and carbon in the poison slabs and the uncertainties therein.

410.21 (9.1.2) In order to facilitate evaluation of the computation methods used please provide the following additional information concerning methods verification:

1. Description of the experiments against which verification was performed with particular emphasis on experiments which contained features relevant to storage rack design (e.g., poison slabs between assemblies, water gaps between assemblies, etc.). Provide information for both the KENO and PDQ code packages.

2. Indicate the extent of the verification performed by the applicant or his contractor (apparently NUS) and that performed by other users. Greatest weight will be given to the former.

410.22 (9.1.3) RSP The applicant has not provided an analysis which conforms to the Standard Review Plan (NUREG-0800) Section 9.1.3 with respect to the cooling time of 150 hours for the last batch into spent fuel storage, the definitions of normal and maximum normal heat loads, and the maximum temperature of 140°F with a single active failure. The applicant used only 14 reloads when 16 reloads can be placed in the spent fuel storage racks based on the equilibrium fuel cycle reload of 85 fuel assemblies indicated in FSAR Table 9.1-2. The applicant is to provide a discussion of this scenario, the revised calculations, and a revised FSAR section.

410.23 (9.1.3) All calculations of the decay heat loads shall be in accordance with the Branch Technical Position ASB 9-2, "Residual Decay Energy for light Water Reactors for Long Term Cooling." Provide the results of revised calculations using the Branch Technical Position.

410.24 (9.1.3) Verify that direct indication of the spent fuel pool temperature and level is available to the operator in the control room.

410.25 (9.1.3) The FSAR states that the low flow alarm in the spent fuel pool cooling system is disabled when the associated cooling pump is off. Provide a list of alarms which would define a loss of a cooling pump to the operator.

410.26 (9.1.4) Describe, discuss, and verify that the maximum potential kinetic energy contained in all objects of less weight than a spent fuel assembly which will be handled over spent fuel in the storage racks will not exceed the effects of the fuel handling accident in Section 15.7.4 of the FSAR.

410.27 (9.1.4) (9.1.5) Verify compliance with the guidelines of ANS 57.2-1976. For each item where the guidelines are not met, identify the item and provide a discussion of the deviation.

410.28 (9.1.4) Verify that the fuel transfer tube gate valve is seismic Category I.

410.29 (9.1.5) Provide the information requested in the generic letter dated December 22, 1980 regarding conformance to the criteria contained in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

410.30 (9.1.4) Provide a list of all load handling systems and identify which systems are used in moving loads weighing more than one fuel assembly and its associated handling tool. In addition, identify those systems which can move any load over spent fuel either in the storage pool or the open reactor vessel.

410.31 (9.2.1) Provide an explanation of why nuclear service water valves are in the component cooling water system as shown in the FSAR Figure 9.2-11.

410.32 (9.2.1) FSAR Figure 9.2.-1A indicates that valve MSW-V75-A is seismic Category I while FSAR Figure 9.2-11 indicates that this valve is non-seismic Category I. Provide a clarification as to whether the valve is seismic Category I or not.

410.33 (9.2.1) FSAR Tables 9.2-3 and 9.2-4 are not clear. Discuss how the shutdown cooling water system's (SCWS) heat load is 108.91 MBTU/hr (Table 9.2-4) while the nuclear service water system removes 214.88 MBTU/hr (Table 9.2.-3) from the SCWS, on a per train basis.

410.34 (9.2.1) According to the FSAR, there are two redundant nuclear service water systems and either system can remove 100% of the plant heat load. Table 9.2-1 in the FSAR specifies a maximum flow requirement for one train with a minimum flow requirement for the other train. Provide an explanation for this discrepancy.

410.35 (9.2.1) The FSAR states that the shutdown cooling water system is required to supply water to the spent fuel pool heat exchanger "a minimum of 1.75 hours" after the loss of the component cooling water system. This statement is ambiguous. Discuss why the spent fuel pool heat exchanger can operate a "minimum of 1.75 hours" without cooling. Include the conditions which are used to define this minimum.

410.36 (9.2.1) Provide a discussion of the inservice inspection and testing program for the nuclear service water and shutdown cooling water systems. Included should be the identification of valves to be monitored, and what functions are performed from the control room and which are local operations. Is sound powered communication with the control room locally available for the operators at the testing station to instruct the test personnel to place the equipment back in service if the equipment was required to be inservice?

410.37 (9.2.2) The FSAR provides the following information. Table 9.2-5 states that the shutdown cooling water pumps have a design flow of 13,500 gpm each. Section 9.2.1.2.2.4 states that only one shutdown cooling water system can be cross-connected to supply the essential equipment in

the component cooling water system at at given time. Furthermore, this section goes on to specify that if the 13,500 gpm flow is available, 7,500 gpm is provided to the decay heat removal heat exchanger, 4,500 gpm is provided to the containment spray heat exchanger, and the remainder (1,500 gpm) is provided to the spent fuel pool heat exchanger. The decay heat removal and containment spray heat exchangers are part of the normal shutdown cooling system. The spent fuel pool heat exchanger is part of the component cooling water system. Table 9.2-8 indicates a required flow of 4,905 gpm through the cross-connection during a loss of offsite power event. Considering the loss of the redundant shutdown cooling system as the single failure and the revised decay heat load calculation (Question 410.23), provide an explanation of the apparent discrepancy in the required vs. available flow through the cross-connection.

410.38 (9.2.2) FSAR Table 9.2-8 indicates less cooling water flow requirements for the essential safety related equipment after a LOCA concurrent with a safe shutdown earthquake and the resulting loss of offsite power than required during a loss of offsite power. Provide the information which indicates your conclusion. For example, with the maximum normal spent fuel decay heat load as recalculated in Question 410.23 and maintaining a maximum pool temperature of 154°F as per the FSAR, discuss why a loss of offsite power requires 2,100 to 3,594 gpm coolant flow as compared to the 1,750 gpm required during a LOCA, plus SSE and the resulting loss of offsite power.

410.39 (9.2.2) Describe the component operational degradation (e.g., pump leakage) and the procedures that will be followed to detect and correct these conditions when degradation becomes excessive.

410.40 (9.2.2) In FSAR Table 9.2-9 the total heat load on the component cooling water system during normal plant operation is 127.94 MBTU/hr per cooling loop while Tables 9.2-3 and 9.2-10 specify a cooling capacity for the component cooling water heat exchanger of 57 MBTU/hr. Provide a discussion which explains the apparent discrepancy.

410.41 (9.2.3) Discuss the effects on the spray nozzles and piping of pumping 23,000 gpm from the emergency shutdown water system when the spray system is designed for 21,465 gpm, which represents 107.15% of the spray system design flow.

410.42 (9.2.4) The FSAR states that the balance of plant service water system consists of three pumps. Each pump is rated for 50% of the normal full load cooling water flow requirements for a flow of 20,000 gpm each, as per Table 9.2-18. FSAR Table 9.2-17 indicates a required normal full load cooling water flow requirement of 51,645 gpm total. Provide a discussion of this apparent discrepancy.

410.43 (9.2.5) In FSAR Table 9.2-24 the total integrated heat load to the ultimate heat sink following a loss of offsite power is 34,048 MBTU/hr. Table 9.2-23 indicates a heat load of approximately 58,000 MBTU/hr. Provide a discussion of this apparent discrepancy.

410.44 (9.2.5) In the FSAR Table 9.2-25 the heat load to the ultimate heat sink is for a loss of coolant and loss of offsite power accident. The rate of heat rejection to the ultimate heat sink and the total heat rejected indicated in this table is less than in Table 9.2-23 which is only for a loss of offsite power. The heat load indicated in Table 9.2-25 does not agree with the heat load to the ultimate heat sink indicated in Table 9.2-24 (this table is applicable by reference from Table 9.2-26). Provide a discussion of this apparent discrepancy.

410.45 (9.2.5) RSP Table 9.2-92 in the FSAR indicates that the loss of one spray system in the ultimate heat sink due to a tornado will have no effect on the cooling capability of the pond because of the redundant spray tree system. Unless the spray trees are tornado missile protected, no credit can be taken for any spray tree system or portion of a spray tree system. Provide the results of an analysis as the result of not having any spray trees, due to tornado missiles, and the most limiting single active failure. As an alternate, provide complete tornado missile protection for the spray tree systems.

410.46 (9.2.5) Regulatory Guide 1.27 requires that there be sufficient water in the spray ponds for cooling without makeup. Discuss how you will monitor the buildup of sediment on the floor of the ponds so as to assure availability of the 30-day water supply. Describe how you will clean the spray ponds without losing redundancy or degrading the system. the system.

410.47 (9.2.5) Since WNP-4 has been withdrawn, discuss the effect on the equipment and design of the demineralized water makeup, potable water and sanitary systems at WNP-1.

410.48 (9.2.6) Specify the seismic categorization of the water treatment building. If the building is not seismic Category I, provide a discussion of the effect of the collapse of the building and the systems within the building on the main steam and feedwater lines and the main steam and feedwater isolation area which shares a common wall with the water treatment building.

410.49 (9.3.1) For the Plant Service Air System, the Instrument Air Supply System, and the Nuclear Instrument Air System, provide the following:

1. A commitment to perform periodic testing of the air system.

2. Verification of conformance with Regulatory Guide 1.80, "Preoperational Testing of Instrument Air Systems."

3. Verification that the air system will provide an air quality as specified in ANSI MC 11.1-1976 (ISA 57.3) or better. FSAR Table 9.3-5 indicates non-compliance with ANSI MC 11.1-1976.

410.50 (9.3.1) Portions of the non-seismic Category I compressed air systems are located inside the General Service Building which is seismic Category I and contains safety related equipment. Verify that the complete failure of the non-seismic Category I compressed air system and its supports will not affect any safety related equipment as the result of a safe shutdown earthquake. As part of your verification, provide general arrangement drawings which show the safety-related equipment and the seismic and non-seismic Category I compressed air piping.

410.51 (9.3.1) Figures 9.3-1 and 9.3-2 in the FSAR do not identify where the compressed air is used. Provide a revised set of drawings which clearly identifies the equipment or component being supplied compressed air. If your response to Question 410.50 shows all compressed air piping, then this question need not be addressed.

410.52 (9.3.1) The FSAR states that the instrument air supply system (IAS) supplies the nuclear instrument air system (IAC) under normal operating conditions. Furthermore, the IAS maintains a header pressure between 80 and 100 psig. Provide a discussion which explains how the IAC header pressure is to be maintained at 100 to 125 psig under normal operating conditions as indicated in the FSAR.

410.53 (9.3.3) Provide a full size riser diagram the radioactive equipment and floor drainage piping system which identifies each potential water source, valves, sumps, and interconnections. Provide a similar diagram the non-radioactive equipment and floor drainage piping system. For each area where there is both radioactive and nonradioactive drains or piping, provide an isometric layout drawing and a discussion of any event where radioactive fluid could enter the non-radioactive drain as the result of excessive flow, plugage of the radioactive drain, or pipe failure.

410.54 (9.3.3) Verify that the containment isolation valves V-22-B, V-23-B, V-72-A and V-73-B are seismic Category I with a Class IE power supply.

410.55 (9.4.1) Provide a physical drawing of the control room remote air intakes and a location drawing. Verify that they are seismic Category I and tornado missile protected.

410.56 (9.4.1) The FSAR states that the electric heaters in the control room HVAC are not Class 1E. In some design basis events, such as an Appendix R fire, offsite power is assumed lost for 72 hours. Assuming such an event occurred during the most severe winter weather, describe how the control room will be maintained at 75°F DB and 45-50% RH.

410.57 (9.4.1) The FSAR Figure 9.4-5 indicates that there are two dampers in series for each intake of the remote air intakes. One damper is supplied power from one Class 1E bus and the second damper is powered by the redundant Class 1E bus. Therefore it appears that the failure of one Class 1E bus will isolate both remote air intakes. If this indeed is correct, it is unacceptable. Provide a discussion and a revised drawing to clarify the arrangement of the dampers and their power sources.

410.58 (9.4.1) Assuming the need to use the remote air intake and a high radiation level in one of the air intakes, describe how that air intake will be purged until the radiation level is within acceptable limits without supplying air to the rest of the control room HVAC system.

410.59 (9.4.1) Provide one or more sections in the FSAR which describe all of the chilled water systems with P&IDs, layout drawings, and seismic qualifications.

410.60 (9.4.2) The FSAR is not clear concerning the continuous minimum flow of 4,000 CFM exhaust from the fuel handling area ventilation system. Verify compliance with Position C.4 of Regulatory Guide 1.13 and the use of Regulatory Guide 1.25 as the minimum potential source of radiation for a fuel handling accident. If the number of fuel pins damaged by the accident evaluated in Question 410.26 is greater than the design basis case specified in Regulatory Guide 1.25, then the design requirements in Regulatory Guide 1.25 should be increased to take into consideration the larger number of the fuel pins damaged as the design requirement for the fuel handling area ventilation.

410.61 (9.4.5) Provide the results of an evaluation of the environmental conditions vs. equipment qualification for safety related equipment serviced by the safeguards area ventilation system for each of the following conditions.

1. A safe shutdown earthquake, with the resulting loss of offsite power, during the design -10°F winter day with the single active failure of one diesel resulting in the loss of one HVAC train.

2. A tornadic event, with the resulting loss of offsite power, during the design 110°F summer day with the single active failure of one diesel resulting in the loss of one HVAC train.

410.62 (9.4.5) The FSAR states that when the temperature exceeds 104°F, the temperature controller opens a valve to provide the safeguards area ventilation system with chilled water from the chillers for the spent fuel pool area ventilation system. A reference is made to Section 9.4.2 of the FSAR for details. There appears to be no mention in FSAR Section 9.4.2 of this ability and FSAR Figures 9.4-1 and 9.4-7 do not show any such capability. Provide the details which were to be in FSAR Section 9.4.2 and a drawing which shows every HVAC system with all interconnections to support systems and equipment.

410.63 (9.4.6) The FSAR states that the air is drawn by four fans into the control rod drive area. This does not agree with FSAR Figure 9.4-8. Provide either a revised description or a revised figure.

410.64 (9.4.7) The FSAR states that the Primary Auxiliary Area HVAC does not operate during a LOCA/LOOP and that safety related equipment needed during a LOCA/LOOP is serviced by this HVAC system. The FSAR states that sufficient air flow will be provided by manually opening the manual dampers. FSAR Figure 9.4-10 does not show this safety related equipment, the manual dampers to be opened during a LOCA/LOOP, and the emergency air flow via the manual dampers. Provide a revised figure whuch provides this information. Assuming a LOCA/LOOP, 30 minutes after the LOCA before the manual dampers are opened, and an ambient temperature of 110°F, what is the maximum ambient temperature in the room with safety related equipment? To what temperature is this equipment environmentally qualified. Are the manual dampers remote-manually qualified? Are the manual dampers remote-manually operated? If not, verify that the operator can locally open all the manual dampers without passing through any area in which the local environment is being affected by the LOCA.

410.65 (9.4.7) Verify that FSAR Figure 9.4-10 is correct with respect to air flow into and out of each area. For example, the two R.B.C. Evaporation Surface Condenser rooms have 5,800 cfm and 4,000 cfm flowing into the rooms and neither room has any exhaust air flow.

410.66 (9.4.8) The FSAR states that the evaporative coolers in the diesel generator HVAC are not seismic Category I because cooling is not required during a LOCA/LOOP or other emergencies. Assuming a safe shutdown earthquake, the resulting LOOP, and an ambient temperature of 110°F, verify that the room temperature for the diesel generators and associated equipment will be maintained below the equipment qualification temperature or 130°F, which ever is lower. Specify the calculated maximum room temperatures.

410.67 (9.4.9) The FSAR states that the switchgear, battery, and cable spreading room HVAC system "does not provide life support or removal of radioactivity in the area." Where will the life support equipment and radiation protective garments be kept for operator use in the event of an Appendix R fire in the control room? Assuming an Appendix R fire in the control room, verify that no operator action is required for a minimum of 45 minutes in order for the operator to scram the reactor, don radiation protective clothing and air packs prior to entry into the switchgear room.

410.68 (9.4.12) The FSAR specifies portable radiation monitors for monitoring and alarming high radiation level. The radiation monitoring should be fixed with a high radiation annunciation in the control room. Verify that the radiation monitoring will incorporate these features.

410.69 (9.4.12) The FSAR states there is one axial vane return air fan per air handling unit (AHU) train. This is not consistent with FSAR Figure 9.4-3 which shows the two return air fans having a common suction header and common discharge header. The discharge header feeds a single duct which later divides into three ducts, one for each AHU and one exhaust duct. Therefore either return air fan can feed either AHU. Provide a revised system description or a revised figure, as appropriate.

410.70 (9.4.12) Provide a discussion of the design provisions that permit appropriate inservice inspection and testing of the electrical and piping tunnel HVAC system components and a description of the inservice inspection and testing program.

410.71 (9.4.13) Verify that all components in the general services building which are served by the component cooling and auxiliary pump room HVAC system are qualified and can operate without any degradation in performance for a minimum of 72 hours in an environment of 130°F and 100% relative humidity. This includes the HVAC system itself.

410.72 (9.4.14) With respect to the main steam and feedwater isolation area HVAC system, the FSAR specifies the ductwork supports and hangers are designed to seismic Category I requirements in Section 9.4.14.1 and also specifies the ductwork supports and hangers are designed to non-seismic requirements in Section 9.4.14.3. Provide a revision to the appropriate section. If the ductwork support and hangers are not seismic Category I provide a discussion of the effects of this ductwork falling onto the isolation valves due to a safe shutdown earthquake, its associated loss of offsite power and the most limiting single active failure.

410.73 (9.4.15) Provide a description of the inservice inspection and testing program for the spray pond pump house HVAC system.

410.74 (9.4.15) Verify that all safety related components in the spray pond pumphouse are qualified and can operate without any degradation in performance for a minimum for 72 hours in an environment of 121°F and 100% relative humidity.

410.75 (9.4.16) In accordance with FSAR Section 9.4.12, the safety related portion of the electrical and piping tunnel ventilation system is powered by the Vital Power Network (VPN). The non-safety related portion of this system is powered by the Auxiliary Power Network (APN). Verify that the seismic Category I Mechanical Equipment Room HVAC, which is safety-related, is powered by the APN as indicated in FSAR Section 9.4.16.3.1. If this safety related system is powered by the APN, provide a discussion and drawings which explain the interconnection between the Class 1E portion of the APN and the non-Class 1E portion of the APN. In addition, provide a list of all HVAC components which are powered by the APN as to which are powered from the Class 1E APN and which are powered from the non-Class 1E APN.

410.76 (9.4.16) Provide a discussion of the design provisions that permit appropriate inservice inspection and testing of the mechanical equipment room HVAC system components and a description of the inservice inspection and testing program.

410.77 (9.4.17) The drawings provided in the FSAR, such as Figure 9.4-19, do not adequately show the tornado missile protection for the ventilation intakes. It appears that the missile protection is not high enough to prevent entrance of tornado missiles. Provide physical drawings of each air intake and exhaust which shows the exact placement of missile shields, ducts, openings, and materials of construction.

410.78 (9.4.17) The FSAR states that there are pneumatic operators to manually reopen the tornado valves. Verify that the air supply to these valves is from the Nuclear Instrument Air System.

410.79 (9.4.17) According to FSAR Figure 9.4-20, the exhaust ducts for the ventilation systems which service the control, switchgear, battery, and cable spreading rooms, the electrical and piping tunnels, nuclear instrument air system, makeup (charging) system, and the component cooling system all pass through a single duct chase with fire dampers at each wall penetration. Assuming an Appendix R fire in the duct chase, which results in closure of all these exhaust systems, during a summer day with the ambient temperature of 110°F, provide a discussion of the effects on the ability to safely control the plant.

410.80 (9.4.17) According to the FSAR figures, there appears to be only one radiation monitor for all of the exhaust systems. Verify this is correct. If there is only one radiation monitor, discuss how this meets the single failure criterion. If there is more than one radiation monitor, provide revised drawings.

410.81 (9.4.17) According to the FSAR figures, the radiation monitoring of the exhausts is performed only in the exhaust plenum. Provide a discussion of the information available to the operator in the control room and of the procedure the operator is to follow upon receipt of a high radiation alarm from the exhaust plenum. Include in this discussion, information on how the exhaust plenum is isolated, air is rerouted for processing prior to being released, and for each ventilation system exhaust duct how the operator will identify which system is releasing radioactivity. According to the FSAR figures, each ventilation system must exhaust some air. As part of the discussion include the design features which will permit rerouting of all exhaust air flow to remove the radioactive particles prior to exhausting into the exhaust plenum.

410.82 (10.3) (10.4.7) RSP The staff requires the main steam and main feedwater isolation valves to fail-closed, not fail-as-is as indicated on FSAR Figure 10.4.1. Verify that these valves will fail in the closed position.

410.83 (10.3) Table 10.1-3 referenced in Table 10.3-1 is not in the FSAR. Provide Table 10.1-3.

410.84 (10.3) Verify that the air operated modulating and on-off dump valves and the main steam isolation valves are supplied by the Nuclear Instrumentation Air System.

410.85 (10.3) Table 10.1-1 in the FSAR specifies the stretch capacity of the steam generators to be 17.16 MLBS/hr. FSAR Table 10.3-1 specifies the guaranteed maximum capacity to be 16,431,353 lbs/hr. Both of these steam rates exceed the total capacity of the main steam isolation valves, which is 16.4 MLBS/hr. as per FSAR Table 10.3-1. As the result of the main steam isolation valves being a flow restriction, verify that excessive steam flow through the valves will not result in additional wear of the valve internals such as to 1) prevent closure of the valve or 2) increase the leakage rate.

410.86 (10.4.5) Provide a detailed evaluation of the effects of a postulated failure of the expansion joint in the circulating water system assuming a double-ended guillotine break. Two specific cases should be addressed 1) the effects when the designed automatic and/or operator actions are taken and 2) the effects when no actions are taken and the pumps continue to operate until the water level reaches equilibrium in all affected areas. The evaluation should include a discussion of the following considerations:

1. The capability of detecting a failure and the means of uniquely distinguishing this type of failure to the operator.

2. Provide a rate of rise and total height of the water for each affected area until equilibrium is reached.

3. For each potentially flooded space, provide a discussion and drawing of the protective barrier provided for all safety-related systems that could be affected in the event of flooding. Include in your discussion the consideration given to passageways, pipe chases, and/or the cableways connecting the flooded spaces to the spaces containing safety-related systems or components outside the turbine building. Discuss the effect of the flood water on all potentially submerged safety-related electrical systems and components.

4. No credit shall be taken for isolation valve closure unless these valves are designed to safety grade requirements.

5. No credit shall be taken for any doors or barriers which are not watertight, including wall penetrations for electrical conduit or piping.

6. If credit for protection of other areas is taken by virtue of the distance from the turbine building, the topographic layout should be provided which clearly defines the path of the water leaving the turbine building and the limits of water path.

410.87 (104.5) Provide a complete FSAR section of the Plant Makeup Water System with all subsections as provided for the other systems, such as the safety evaluation section.

410.88 (10.4.7) State how Branch Technical Position ASB 10-2, "Design Guidance for Water Hammers in Steam Generators with Top Feedring Designs" is met. Discuss the design features to minimize water hammer and the confirmatory tests to be performed. While the Branch Technical Position addresses a specific type of steam generator, our concern is that following any design basis event, the required auxiliary feedwater system flow might result in damage, due to water hammer, to the auxiliary or main feedwater system as well as to the steam generator.

410.89 (10.4.7) (10.3) (10.4.9) Provide a discussion and drawings of the hydraulic control system(s) for the main steam and main feedwater isolation valves and the auxiliary feedwater steam inlet valves. Include in the discussion the type of fluid, the systems seismic classification, quality group class, the motive power source for the hydraulic fluid along with its seismic qualification, quality group class and whether the power source is AC or DC and its IEEE Class.

410.90 (10.4.7) Verify that the electrical power for the electro-hydraulic controllers for the main feedwater isolation valves is seismic Category I, IEEE Class 1E and that both valves on the same feedwater line are not fed from the same 1E source.

410.91 (10.4.7) Provide a discussion of the design provisions in the condensate and feedwater system which provide the capability to detect and control leakage from the system.

410.92 (10.4.9) Provide a response to our April 24, 1980 generic letter concerning the auxiliary feedwater system.

410.93 (10.4.9) RSP The FSAR does not specify the amount of water required for the auxiliary feedwater system. It is the staff's position that the auxiliary feedwater system must have sufficient water capacity for four (4) hours of operation at hot standby prior to initiation of cooldown, as indicated in Branch Technical Position RSB 5-1. Modify the FSAR to include the required quantity of water for the four hours at hot standby and the cooling down period until initiation of the decay heat removal system.

410.94 (10.4.9) Discuss whether the auxiliary feedwater pump capacities listed in Table 10.4-20 of the FSAR (600 gpm for each motor driven and 1,325 gpm for the turbine driven pumps) include allowance for wear.

410.95 (10.4.9) Describe the design provisions which pronibit the reserved quantity of auxiliary feedwater (330,000 gallons) from being used by any other system. Does this reserve volume include the unusable water volume in the bottom of the tank?

410.96 (10.4.9) The FSAR indicates an alternate water source for the auxiliary feedwater system is the nuclear service water tank. Provide a revised FSAR Figure 10.4-6 which shows the connection between the auxiliary feedwater system and the nuclear service water system.

410.97 (10.4.9) In the FSAR, repeated use of the phrase "hot standby for an extended period" is made. Define this phrase in hours.

410.98 (10.4.9) State the dedicated water volume for the auxiliary feedwater system in the condensate storage tank and in the nuclear service water tank. Provide the design provisions for preventing this dedicated water volume from being used by other systems.

410.99 (10.4.9) The FSAR is unclear with respect to connecting a motor driven auxiliary feedwater pump to feed the steam generator to which it is not normally aligned. FSAR Figure 9.4-6 indicates the alternate flow path can be opened by an air operated remote manual valve. However the normal flow path cannot be isolated except by local manual operation after unlocking the valve. Verify whether this is correct.

410.100 (10.4.9) Discuss how a low water level in the demineralized water tank indicates excessive system leakage when the auxiliary feedwater system is being used. Include how the tank level identifies the location of the leakage, as indicated in the FSAR.

410.101 (10.4.9) The FSAR states, with respect to the auxiliary feedwater pumps, the failure of one subsystem is not expected to affect the operation of the other subsystem. Verify that the phrase "not expected to affect" means that it will not affect the other subsystem. If this is not the meaning, provide a discussion of each failure which would or might be expected to affect another subsystem of the auxiliary feedwater system.

410.102 (10.4.9) Provide a discussion of the design provisions that permit inservice inspection and testing of the system components and a description of the inservice inspection and testing program.