June 14, 2002

Mr. Gregg R. Overbeck
Senior Vice President, Nuclear
Arizona Public Service Company
P. O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNIT 2 - REQUEST FOR ADDITION INFORMATION REGARDING POWER UPRATE LICENSE AMENDMENT REQUEST (TAC NO. MB3696)

Dear Mr. Overbeck:

By letter dated December 21, 2001, you requested an amendment to the operating license for Palo Verde Nuclear Generating Station, Unit 2. The amendment supports the proposed steam generator replacement and subsequent power operation at 3990 megawatts-thermal (MWt), an increase of 2.94 percent over the current authorized power level of 3876 MWt.

The Nuclear Regulatory Commission (NRC) staff has determined that additional information is needed in order for it to complete its review and evaluation of your license amendment request. The enclosed questions were e-mailed in May 2002 to, and then discussed with, your staff. Any differences between the enclosed request for additional information and the questions that were e-mailed are editorial, to delete unneeded questions, or to clarify the question. Your staff agreed to submit the responses to the questions in sets by technical branch, or as they are completed, with all of the responses submitted by August 30, 2002. The responses are needed by no later than that time for the NRC staff to complete its review on the schedule you requested. If it is believed that any of this information has already been submitted to the staff, please provide us with a specific reference to the submittals.

Sincerely,

/RA/

Jack Donohew, Senior Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. STN 50-529

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

PROPOSED AMENDMENT FOR

STEAM GENERATOR REPLACEMENT AND POWER UPRATE

PALO VERDE NUCLEAR GENERATING STATION, UNIT 2

DOCKET NO.: 50-529

By letter dated December 21, 2001, Arizona Public Service Company, the licensee for the Palo Verde Nuclear Generating Station, submitted information and proposed Technical Specification changes to increase the Unit 2 rated power level from 3876 megawatts-thermal (MWt) to 3990 MWt. The Nuclear Regulatory Commission (NRC) staff has completed the preliminary review and identified a number of questions for which responses are needed for the staff to complete its review. Attachments 2 and 6 of the application are the License Amendment Request Analysis and the Power Uprate Licensing Report (PURLR), respectively.

The following questions are from the (1) Mechanical and Civil Engineering Branch, (2) Reactor Systems Branch, (3) Materials and Chemical Engineering Branch, (4) Plant Systems Branch, and (5) Probabilistic Safety Assessment Branch.

Mechanical and Civil Engineering Branch:

1. The Nuclear Steam Supply System (NSSS) at Palo Verde Nuclear Generating Station (PVNGS), Unit 2 was approved by the NRC staff via NUREG-0852, “Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System Combustion Engineering Standard Safety Analysis Report (CESSAR) System 80.” The CESSAR describes the design of the reactor coolant system (RCS), its components, and their supports. The CESSAR describes the methodologies used to develop limiting loads and their locations, and also contains interface requirements between the CE-supplied System 80 NSSS and the rest of the plant. The PURLR, Attachment 6 to the application implies that the analyses which support steam generator replacement (SGR) and power uprate (PUR) may be significantly changing the CESSAR methodologies and assumptions for Unit 2.

    a. With respect to RCS stresses, including piping, components, supports, and tributary piping, provide a clear description of the methodologies used for determining the limiting stresses and cumulative usage factors (CUFs) for the SGR/PUR conditions. Describe any changes to the methodologies that were approved as part of CESSAR, and justify the acceptability of any methodology changes for showing compliance with the American Society of Mechanical

Enclosure

Engineers (ASME) Boiler and Pressure Vessel Code (i.e., ASME Code) requirements. Describe any significant changes to the design transients that are used for the structural design of the NSSS. Also, discuss any changes to the interface requirements that resulted from the SGR/PUR.

b. The application indicates that leak before break (LBB) is being utilized for the design of more components than discussed in the CESSAR or in the current licensing basis for Unit 2. Describe and justify the new applications of LBB and/or changes in the postulated break locations. Discuss whether these applications of LBB are based on a generic staff safety evaluation, or whether they are changes to the licensing basis that require specific NRC staff review and approval. Also, provide additional information on the continued applicability of LBB for the SGR/PUR conditions (i.e., evaluate the SGR/PUR condition against the criteria evaluated in Supplement 3 to NUREG-0852, including the margin between the leakage-size crack and the critical-size crack, and the material properties of the replacement steam generators (RSGs), replacement cold leg elbows, and associated field welds).

2. For the RCS piping (including pressurizer (PZR) surge line and tributary piping), components (including reactor vessel (RV), reactor coolant pumps (RCPs), RSGs, and PZR), and supports, provide the calculated maximum stresses and CUFs at the critical locations. Include the ASME Code allowable limits and the ASME Code edition and addenda used in the evaluation of the SRG/PUR conditions. If different from the ASME Code of record, provide a justification.

3. Section 5.3.3.1 of the PURLR indicates that the response spectra for the containment basemat in the vertical direction for the operating basis earthquake is not bounded by the analysis of record. Provide an evaluation of the containment basemat stresses for this condition.

4. For the RV internals provide the maximum calculated stresses and CUFs for the SGR/PUR condition. Include the ASME Code allowable limits used in the evaluation, and the ASME Code edition and addenda. If different from the ASME Code of record, provide a justification.

5. For the control element drive mechanisms (CEDMs), the PURLR describes changes in the methodology for determining stresses and CUFs. Describe the benchmarking of the new methodology, and discuss the new methodology's acceptability for determining stresses and CUFs for the SGR/PUR condition. Provide the maximum calculated stresses and CUFs at the critical locations of the CEDMs for the SGR/PUR condition. Include the ASME Code allowable stresses and the ASME Code edition and addenda used in the evaluation of SRG/PUR. If different from the ASME Code of record, provide a justification.

6. Discuss the potential for flow-induced vibration of the steam generator (SG) tubes due to various mechanisms, including, the fluid-elastic instability, in the RSG at the PUR condition. Describe the analysis methodology, damping value of the tubes, and the computer code used in the analysis. Also provide the results of the predicted vibration levels during the normal operating condition and the worst case transient condition,

including the calculated fluid-elastic instability ratios. Explain whether the above analysis results are applicable to the degraded SG condition and why.

7. Describe any changes to the thermal stratification of the PZR surge line and any changes to the thermal fatigue of the PZR spray nozzle.

8. Describe the methodology used to evaluate the balance of plant (BOP) piping, components (including pumps, valves, and heat exchangers), and supports. Justify differences from the original design methodology. Also, provide the calculated maximum stresses for the critical BOP piping systems. Include the ASME Code edition and addenda and ASME Code allowable limits. If different from the ASME Code of record, provide a justification.

9. The PUR results in an increase in the main steam flow and the feedwater flow. Discuss the potential for flow-induced vibration in the main steam and feedwater piping and the BOP heaters and heat exchangers following PUR. Also, clarify whether vibration monitoring, consistent with OM-3, will be included in the startup testing program for PUR.

10. The SGR/PUR increases the post-accident containment temperature and pressure. Discuss the effects of the SGR/PUR on the overpressurization of isolated piping segments (reference: Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions”).

11. Confirm whether the SGR/PUR will increase the accident sub-compartment temperature and pressure that affect the design basis for steel and concrete in the containment. If the structural steel and concrete will be affected, provide the design-basis margin and margins after considering increased accident loading due to the SGR/PUR.

12. The acceptability of several secondary system items (i.e., steam traps) relies on an improvement in the steam quality to offset the increase in steam flow associated with PUR. Steam quality is expected to go from 0.25 percent to 0.1 percent as a result of the SGR. Clarify whether the startup testing program for PUR includes a test of the steam quality. Also, clarify whether the 0.25 percent steam quality assumed for current conditions is based on measurements or design numbers, and whether any secondary system items (i.e., steam traps) are close to their operational limits at current conditions.

13. Section 9.1 of the PURLR states that a modification will be made to the main steam isolation valve bypass valve. Describe the modification.

<u>Reactor Systems Branch</u>:

1. Attachment 2, Section 2, of the application: The proposed uprate from 3876 MWt to 3990 MWt will add 114 MWt. The submittal states that correspondingly 55 megawatts electric (MWe) will be added. How will you achieve a 48.2 percent conversion for the 114 MWt while the original thermal efficiency is about 32 percent?

2. Attachment 6, Section 2.1.3: The proposed uprate is based on the anticipated performance of the RSGs. However, the submittal does not state any provisions to verify (during initial operation) that the parameters chosen in the analysis stage are indeed those present in the operation of the plant.

3. Attachment 6, Sections 6.1.2 and 6.1.3: For the loss of coolant accident (LOCA), there is the large break LOCA (LBLOCA) and small break LOCA (SBLOCA). For the LBLOCA and SBLOCA of record, the planar linear heat generation rate (PLHGR) is listed as 13.1 kW/ft for the LBLOCA, and 13.5 kW/ft for the SBLOCA. What caused the difference in PLHGR in the two cases?

   What codes have been used for the LBLOCA and SBLOCA? Where have these codes been reviewed and approved by the NRC staff? For the SBLOCA, have any changes to the code been made and, if yes, what would the effect be on the peak clad temperature and the amount of oxidation? (See also question 32b.)

4. Attachment 6, Section 6.3.0.1, "Methods and Computer Codes": For the sheared reactor coolant pump shaft with loss of power transient; the proposed analyses assume that the operator will manually refill the affected SG. Previous analyses did not assume manual action for this event. Why did the uprate require operator action? How was the operator response time estimated? Does this assumption meet regulatory guidance for operator action for design-basis events?

5. Attachment 6, Section 6.3.0.1, "Methodology and Computer Codes" paragraph on Methods and Assumption Changes: Are the proposed changes in the assumptions within the scope of the approved methodologies?

6. Attachment 6, Section 6.3.0.3, Table 6.3-3, low SG pressure: The numerical values in the table are in conflict with those on Page 4-12, please explain.

7. Attachment 6, Section 6.3.1.3, "Increased Main Steam Flow": Table 6.3-6 indicates an automatic main steam isolation valve (MSIV) closure; however, the description of the transient simulation indicates manual closure of the MSIVs. Please explain this discrepancy.

8. Attachment 6, Section 6.3.1.4.3 and elsewhere, use of CENTS computer code for non-loss of coolant accident (non-LOCA) transient simulation: It is stated that the Unit 2 Amendment 137 qualified the CENTS code for non-LOCA transient analysis. It is also stated that there are limitations in the code. Was the code qualified for plant analysis for operation at 4070 MWt (3990 x 1.02 = 4070) and if yes, how was it accomplished?

9. Attachment 6, Section 6.3.1.6, "Steam System Piping Failures Inside and Outside Containment - Mode 3 Operation" and elsewhere: The computer code HRISE is used for the estimation of the departure from nucleate boiling ratio (DNBR). Has this code been reviewed and approved by the NRC staff? Is the associated hand calculation of the linear heat generation rate at the time of return-to-power part of the approved process?

10. Attachment 6, Section 6.3.1.7.3, “Description of Analysis”: This section identifies the limiting scenario (for “Pre-trip Main Steam Line Break Power Excursions”) as the “...full power event with offsite power available”. However, Table 6.3-21 which lists the parameters used for the analysis indicates a 95 percent power level. Do the conclusions listed in Section 6.3.1.7.6 reflect the full power run, or the 95 percent run, as indicated in Table 6.3-21?

    The DNBR calculation was performed using the CETOP-D and TORC computer codes. Have these codes been reviewed and approved by the NRC staff? How does the analyses using CETOP-D and TORC differ from analyses using the HRISE code elsewhere in this submittal? Discuss why CETOP-D and TORC were used instead of HRISE?

11. Attachment 6, Section 6.3.2.8.2.4 “Input Parameters, Initial Conditions, and Assumptions”: This section discusses the most limiting size break. Has this changed from the previous power level? Discuss how the most limiting break size was determined (e.g., by review of existing data, new analysis, or some other means).

    In the same section, the 4^{th} assumption, the primary to secondary heat transfer was assumed degraded. What is the physical basis for this degradation, and by what amount is the degradation?

    Discuss if the values of the moderator temperature coefficient (MTC) in Tables 6.3-31 and 6.3-33 should be negative?

12. Attachment 6, Section 6.3.3.1.5, Table 6.3-34: At 28.8 sec, the main steam safety valves (MSSVs) begin to cycle. The pressure-time function indicates a wavy pattern as shown in Figure 6.3-141. Discuss, if this is caused by MSSV cycling, is it not likely that the MSSVs will fail. Unlike the power operated relief valves, the MSSVs are not designed for prolonged cycling.

13. Attachment 6, Section 6.3.3.4.4, sheared shaft event, “Input Parameters, Initial Conditions, and Assumptions”: Table 6.3-35 includes an MTC value of -0.18E-04 $\Delta\rho/^{\circ}F$. Discuss how this value was selected.

14. Attachment 6, Section 6.3.6.3.2.3.1, “Transient Simulation”: The section credits a 100 sec manual reactor trip to maximize integrated steam flow out of the (assumed) stuck open automatic depressurization valve. However, recent experience with a steam generator tube rupture (SGTR), at Indian Point Unit 2, indicates that the operator was not able to trip the reactor for several minutes. Discuss how the time of this trip is justified in view of this actual operating experience. How would the transient have evolved should a more realistic manual trip time be assumed?

15. Attachment 6, Section 6.3.6.3.2.4, SGTRLOP, [steam generator tube rupture loss of offsite power] “Input Parameters, Initial Conditions, and Assumptions”: As in Section 6.3.6.3.2.3.1 above, an assumption is made for a 100 sec operator trip. Discuss how the time of this trip is justified in view of actual operating experience, and being earlier than the core protection calculator (CPC) action?

16. Attachment 6, Section 6.3.8, "Limiting Infrequent Events": In the selection of the limiting abnormal operating occurrence (AOO) with a single active failure, the loss of offsite power (LOP) from DNBR at the specified acceptable fuel design limit (SAFDL) value was selected. The calculated minimum DNBR was 1.17. However, the sheared shaft transient (Figure 6.3-154) resulted in the same DNBR value of 1.17 but started from normal operation DNBR.

    Discuss would it be reasonable to conclude that the sheared shaft is the limiting AOO, because it would result in a lower DNBR value than 1.17 had it started from the SAFDL DNBR value?

17. Attachment 6, Section 7.5, "Neutron Fluence": The vessel fluence in the analysis of record (AOR) was calculated for a power level of 4200 MWt. Discuss the following: (1) does the AOR satisfy the guidance in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (2) what is the value of the end of life reference temperature for null ductility transition (RT_{NDT}), and (3) in the RT_{NDT} computation process, were any adjustments made on the calculated fluence value based on dosimetry measurements? If yes, for item (3), provide the data for the adjustment.

18. Attachment 6, Section 6.3: For the analyses of various design basis transient events described in the section, the values of the fuel rod gap conductance range from 500 Btu/hr-ft^2-°F for the sheared RCP shaft event, 6,100 Btu/hr-ft^2-°F for the steam bypass control system malfunction event, to 6,984 Btu/hr-ft^2-°F for the CEA ejection event.

    a. Discuss how these various values of fuel rod gap conductance are calculated, and what are the bases for determining which value to use as an initial condition of a particular event.

    b. Does the gap conductance value change or is it held constant during a transient and what is the basis for this?

    c. Provide sensitivity study results which show the effects of the input values of gap conductance on the analysis results of various events.

19. Attachment 6, Sections 7.1 and 7.2: With respect to the impacts of the proposed PUR on the core thermal-hydraulic design and core design, confirm that all parameters and assumptions to be used for analyses described in Sections 7.1 and 7.2 remain within any code limitations or restrictions. Describe the process used to support the conclusions.

20. Attachment 6, Section 7.1: References 7-2 and 7-4 are letters requesting that the NRC review and approve a revised core inlet flow distribution methodology and the specific application of the CETOP-D computer code for Palo Verde Units 1, 2 and 3. For the revised core inlet flow distribution methodology, the safety evaluation report (SER) was written specifically for Unit 1 and included a statement that the licensee plans to submit a generic application addressing a revised maximum departure from nucleate boiling ratio (MDNBR) setpoint for all three units. Has the generic application been submitted,

reviewed, and approved by the NRC? Please provide a reference to the NRC SERs which granted these approvals for Unit 2. Also for the PUR conditions, are all parameters within the restrictions or limitations of this methodology?

21. Attachment 6, Section 7.1: The steady state departure from nucleate boiling (DNB) analysis was performed using the methodology of Reference 7-5, CEN-356(V)-P-A, Revision 01-P-A, “Modified Statistical Combination of Uncertainties,” dated May 1988. The NRC SER for this methodology was written specifically for Palo Verde Unit 1. For clarification, please provide the technical justification for the application of this methodology to Unit 2, or provide a reference to the approved SER for Unit 2. Have any modifications been made to the methodology since implementation on Unit 1? If so, please describe and provide the technical basis to support the change.

22. Attachment 6, Section 7.1: The section provides a TORC computer code calculated 95/95 DNBR limit of 1.34. This value is not consistent with the Updated Final Safety Analysis Report (UFSAR) value of 1.30, or with the Reference 7-5 value of 1.24. Please provide the technical basis for this difference. Also, discuss the impact of this change on the CPC and core operating limit supervisory system overall uncertainty penalty factors.

23. Attachment 6, Section 6.3.4 - Reactivity and Power Distribution Anomalies:

    a. Discuss the methodology used to calculate the reactivity insertion rates used in these analyses and provide the technical basis for the initial power level assumptions for the UCEAW from subcritical and Hot Zero Power.

    b. All reactivity transients assume that no SG tubes are plugged. For each of the transients, please discuss the impact of 10 percent SG tube plugging on the results of these events and compare to the corresponding acceptance criteria. What is the basis for the upper limit of 10 percent SG tube plugging for the PUR conditions?

24. Attachment 6, Section 6.3.4.1.1: For the UCEAW from subcritical event, there is a rather significant difference in the duration of this event when compared to the UFSAR analysis of record. The UFSAR analysis shows this event terminates at approximately 300 seconds after initiation, while the new analyses performed for the PUR terminates at approximately 55 seconds. Discuss the changes in key parameters as a result of the PUR which would explain this difference.

25. Attachment 6, Section 6.3.4.1.1: For the UCEAW from Subcritical event, Figure 6.3-158 shows that the RCS pressure oscillates for approximately 15 seconds following the reactor scram. Discuss the expected magnitude, frequency, and the physical phenomenon causing these oscillations. Oscillations are not evident for any other key parameters for this event.

26. Attachment 6, Section 6.3.4.1.1: For the UCEAW from Hot Zero Power event, the resulting maximum RCS pressure in Table 6.3-40 is not consistent with the results shown in Figure 6.3-164. Explain this discrepancy.

27. Attachment 6, Section 6.3.4.2 - UCEAW at Power: The UFSAR states that parametric studies performed for initial condition determinations indicate that minimum DNBR during the CEAW at power is most sensitive to initial core inlet temperature. As such, the UFSAR analysis of record assumed the maximum allowable core inlet temperature (580°F). In the PUR reanalysis for this event, an initial core inlet temperature at the lower limit of the range (548°F) was assumed. Provide a justification for assuming the lower limit of the core inlet temperature range and discuss the impact on DNBR results if the higher temperature is assumed.

28. Attachment 6, Section 6.3.4.3 - Full Length Control Element Assembly (CEA) Drop Event:

    a. Provide the technical basis for assuming the initial core power of 95 percent of rated thermal power. Discuss if this assumption bounds the requested uprate power of 3990 MWt?

    b. Provide the technical basis for reactivity parameter values assumed for this event and listed in Table 6.3-43.

    c. Provide a plot of the acceptance criteria (MDNBR and linear heat generation rate) vs. time for this event.

29. Attachment 6, Section 6.3.4.8 - CEA Ejection Event:

    a. For evaluating the fuel performance and peak RCS pressure cases, the licensee assumed that the CEDM rupture was plugged by the ejected CEA. Because this is an unusual assumption, is there a design feature of this plant that justifies this assumption? Does this assumption provide the limiting results for the fuel performance case? Also, what break size (ft^2) was assumed in the analysis?

    b. The initial SG level assumed in this analysis is outside the range of initial conditions listed in Table 6.3-2. Provide the technical basis for this assumption, and discuss its impact on the results of the analysis.

    c. Discuss the methodology used to determine the ejected CEA worth.

    d. The UFSAR analysis of record for this event assumed a coincident loss of offsite power, while the PUR analyses do not. Provide the technical basis for changing the licensing basis for this case. Discuss the impact that this assumption would have on the PUR results for this event.

    e. For the fuel performance case, provide a table of the initial parameter assumptions, sequence of events, and corresponding results.

30. Section 6.3.4.1 of Attachment 6: The section describes the analysis of uncontrolled CEA withdrawal from a subcritical or low power condition. Sections 6.3.4.1.3.3 and 6.3.4.1.4.3, respectively, describe the results of analyses for the CEA withdrawal from a subcritical and low power conditions, and state that the hot channel minimum DNBR remains above the safety limit, and that the linear heat generation rate (LHGR)

exceeded the safety limit as defined in the Technical Specifications (TSs). This occurs for a short time with a resulting peak fuel temperature well below the limiting fuel centerline temperature for melting fuel. The NRC staff agrees that there is no safety concern as the resulting peak fuel temperature is below the fuel melt limit. However, violation of a TS safety limit even for a very short time is not acceptable, as this is a violation of 10 CFR 50.36, which requires that TS Limiting Safety System Settings be in place to prevent safety limits from being exceeded during analyzed events.

It has recently been identified in a number of TSs of CE designed plants that the peak linear heat rate (PLHR) safety limit is violated for the uncontrolled CEA withdrawal from subcritical or low power conditions. Because the PLHR safety limit is defined as a measure to prevent fuel centerline temperature from reaching the fuel melt temperature, some licensees have resolved this issue by converting from a PLHR safety limit to a fuel centerline temperature safety limit. This issue must be resolved prior to NRC staff approval of the proposed Unit 2 PUR amendment. Discuss how you propose to resolve this issue.

31. Confirm that the generically approved LOCA analysis methodologies used for the Palo Verde uprate LOCA analyses apply specifically to Unit 2, and:

    a. Identify all restrictions placed on the LBLOCA and SBLOCA models and show how they are resolved for this uprate. Specifically, the NRC staff wants to understand the resolution of the SER limitation to 3800 MWt and the process used to justify by extension of the methodology to power levels in excess of 3800 MWt.

    b. Address all other restrictions on, conditions of applicability, or changes to the Palo Verde LBLOCA and SBLOCA methodologies by NRC staff SERs or other report findings (e.g., Part 21 or 10 CFR 50.59 reports, error adjustments, etc) in a table, listing the item in sufficient detail to identify the concern, the date of the SER or other report, and its disposition (e.g., not applicable to Palo Verde, reanalysis reflecting error correction, or Palo Verde is in the applicable class and how the condition is satisfied for Palo Verde).

    c. Show that Unit 2, operating at the uprated power is bounded by the assumptions used in the analyses to support the approval of the generic LOCA methodologies.

    d. Provide a comprehensive statement that Palo Verde and its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as-operated plant values for those parameters, and briefly discuss these processes. (Do not use specific procedures and process components as examples in your response.)

32. What are the calculated peak cladding temperature (PCT) and oxidation values for the LBLOCA and SBLOCA per 10 CFR 50.46(b), for Palo Verde at the uprated power? Will this calculation become the official analysis of record for future reporting under Section 10 CFR 50.46?

33. Discuss the effect on the Unit 2 SBLOCA analyses of the recently discovered coding error in the CEFLASH component of the SBLOCA methodology. Provide both PCT and oxidation before and after results.

34. Discuss the design of the Palo Verde emergency core cooling system (ECCS) switchover from the injection mode to the ECCS sump recirculation mode. What was the decay heat source assumed in the design of the ECCS switchover for the present power? Discuss if the assumed heat source and the timing of the switchover change for the uprated power.

35. Provide a complete description of the long term cooling, boron precipitation model that is used to establish compliance with 50.46(b)(5).

<u>Materials and Chemical Engineering Branch</u>:

1, In Section 5.5 of the PURLR, the licensee stated that RSGs are being designed and analyzed in accordance with the ASME Code for structural acceptability, U-bend fatigue, tube degradation, tube plugging, and repair requirements. Discuss the following:

   a. The structures and components in the replacement SGs that were being analyzed in the structural acceptability analysis. Discuss whether each of the structures and components has satisfied the relevant ASME Code allowable stresses and fatigue usage factors. Discuss specific ASME subsections and equations used in the structural analysis.

   b. The U-bend fatigue analysis and whether the U-bend fatigue calculation satisfies NRC Bulletin 88-02, “Rapidly Propagating Fatigue Cracks in Steam Generator Tube.”

   c. The analysis to satisfy the tube plugging limit of 40 percent tube wall thickness in the plant technical specifications.

   d. The analysis and/or tests to demonstrate the structural integrity of the SG tubes.

   e. How the ASME Code is used to determine tube degradation or repair requirements and how tube degradation and repair requirements are being analyzed.

2. Discuss tests and/or analyses performed to demonstrate the corrosion resistance of Alloy 690 SG tubing under the power PUR uprate conditions.

3. Discuss tests and/or analyses performed to demonstrate leakage integrity of the replacement SG tubes under the PUR conditions.

4. Describe briefly the RSG. For example, provide information on (1) the model SG, (2) the nominal diameter and wall thickness of the tubes, (3) the configuration and

material of the tube support, (4) U-bend support configuration, and (5) designs that would mitigate the potential for tube degradation or internal component degradation.

5. Because the effects of flow accelerated corrosion (FAC) on the degradation of carbon steel components are plant-specific, the NRC staff requests the licensee to provide a predictive analysis methodology that must include the values of the parameters affecting FAC, such as velocity and temperature, and the corresponding changes in component wear rates before and after the PUR. Please include predicted FAC wear rate changes in balance of plant components and those components most susceptible to FAC.

6. The NRC staff requests that the licensee indicate the degree of compliance with NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning." This letter requires that an effective program be implemented to maintain structural integrity of high-energy carbon steel systems. The licensee should describe how this program was modified to account for the PUR. If there is a generic computer code (e.g., CHECWORKS) used in predicting wall thinning by FAC, specify it; however, if the code is plant-specific to Unit 2, provide a description of the code.

7. The PURLR does not discuss the power-uprate-related effects on reactor vessel integrity. Discuss the effect of the PUR on the following for Unit 2: pressurized thermal shock, fluence evaluation, heat-up and cooldown pressure temperature limit curves, low temperature overpressure protection, upper shelf energy, and surveillance capsule withdrawal schedule.

Plant Systems Branch:

Balance-of-Plant Systems

1. In Section 4.2.1.3 "Main Steam Safety Valves" of the PURLR, the licensee stated that the total MSSV capacity is 19.53E+06 lb_m/hr. However, in Section 8.8.1 "Main Steam Safety Valves", it is stated that the total MSSV design relief capacity is 22.56E+06 lb_m/hr. Please clarify the above discrepancy.

2. As a result of plant operations at the proposed PUR level, the decay heat load for any specific fuel discharge scenario will increase. The licensee stated that the maximum allowable spent fuel pool (SFP) decay heat load is administratively controlled so that the heat load in the SFP is less than the available SFP heat removal capability, considering single failure. It was further stated that the SFP heat load is analytically confirmed to be less than the available SFP heat removal capability before the return to power operation following a refueling outage. However, the licensee did not provide the discussion of its SFP cooling evaluations or the administrative control procedures in the PURLR. The licensee is requested to provide the following information for both pre-PUR and PUR conditions:

a. SFP heat loads and the corresponding peak calculated temperatures during planned[1] (normal) refueling outages under partial and full-core offload conditions, and unplanned[2] (abnormal) full-core offload outages for pre-PUR and PUR conditions.

b. Assumptions used in the SFP thermal-hydraulic analysis (i.e., fuel assemblies "in-reactor" hold time, number of the previously discharged spent fuel assemblies (SFAs) in the SFP, SFP heat exchanger cooling water inlet temperatures, ultimate heat sink temperature, etc.) for each scenario.

c. For the planned refueling outages under partial and full-core offload conditions, discuss how the most severe single failure has been identified and accounted for in the SFP thermal-hydraulic analyses. A single failure need not be assumed for the unplanned full-core offload events.

d. Shutdown cooling systems are utilized to maintain the SFP below the design temperature, prior to a planned or unplanned full-core offload event when needed. How many trains of SFP cooling system, and shutdown cooling systems are required to be operable and available for SFP cooling?

e. For the planned refueling outages under partial and full-core offload conditions, if the calculated peak SFP temperature is above 150°F[3], provide the duration during which the SFP temperature is above 150°F and the thermal stress analyses to demonstrate that the SFP structure can withstand the new high temperature.

3. The heat removal capability of the SFP cooling system is a function of the redundant nuclear cooling water system water temperature, and the decay heat load is a function of the SFAs "in-reactor" hold time prior to fuel being discharged from the reactor. The "in-reactor" hold time for offload can be adjusted, as long as the time exceeds the time assumed for the fuel handling accident. A licensee can opt to perform a cycle-specific SFP thermal-hydraulic evaluation prior to every planned offload using the actual conditions at the time of the offload. If a cycle-specific SFP thermal-hydraulic evaluation is performed prior to every planned offload for Unit 2, discuss the provisions established in the plant operating procedures to ensure that the SFP operating temperature limit of 150°F will not be exceeded.

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[1] A planned offload is the offload of fuel assemblies to the SFP for any expected (or planned) reason (e.g. refueling outage).

[2] An unplanned offload is the offload of fuel assemblies to the SFP due to an unforeseen condition (e.g., unexpected shutdown that includes an offload).

[3] Temperature limit specified for concrete in the American Concrete Institute (ACI) Standard 349.

4. Discuss the provisions established or to be established in plant operating procedures to monitor and control the SFP water temperature during core offload events. Information should include:

   a. Frequency of monitoring the SFP water temperature during planned and unplanned core off-load outages.

   b. Information (such as high SFP water temperature alarm setpoint) supporting a determination that there is sufficient time for operators to intervene in order to ensure that the temperature limit of 150°F will not be exceeded.

   c. Compensatory actions (e.g., prohibit fuel handing, align other systems to provide SFP cooling, etc.) to be taken in the event of a high SFP water temperature alarm.

Heating, Ventilation, and Air Conditioning (HVAC) Systems

5. In PURLR Sections 8.10.1.1 "Containment Heating, Ventilation, and Air Conditioning," 8.10.1.3 "Turbine Building Heating, Ventilation, and Air Conditioning System," and 8.10.1.4 "Control Building Heating, Ventilation, and Air Conditioning System," the licensee concluded that the heat loads used for the original plant design remains bounding for the PUR heat loads of 102.94 percent of rated thermal power. Also, in Section 8.10.1.2 "Auxiliary Building Ventilation," the licensee concluded that the auxiliary building HVAC system piping design temperatures, pump motor maximum operating horsepower, electrical equipment, and lighting heat loads are, with one exception, not affected by the PUR, and that the increased reactor power results in an increased post-accident (i.e., LOCA and main steam line break (MSLB)) containment temperature. This affects the transmission of heat loads through the containment wall into the adjacent rooms and results in an increase of heat loads in the adjacent rooms. However, this increase in heat loads remains bounded by the original system design.

   Because no details were provided to support the above conclusions, for each of the above cited sections (Sections 8.10.1.1, 8. 10.1.2, 8.10.1.3 and 8.10.1.4), provide a worst-case example demonstrating how, based on a review of design basis calculations, the total heat load increases are within the design margin at PUR conditions. State where the comparison with the evaluations at PUR conditions is documented, and would be available to the NRC staff for review upon request.

Containment LOCA Response Analysis (PURLR Section 6.2.2)

6. Some of the initial conditions used for the calculation of the peak containment pressure response to a LOCA are different from those previously defined for the same analysis in the UFSAR. Also, the sump temperature response exhibits a different response when compared to previous UFSAR analyses. Provide the following information:

   a. In UFSAR Table 6.2.1-6, the limiting relative humidity for the LOCA is stated to be 50 percent, while in the submittal, Table 6.2-1, the value is changed to "Zero" percent. Why has this value been changed? Provide an estimate of the portion of the peak pressure change for the PUR analysis, as compared to the current UFSAR analysis, resulting from this revision.

b. In UFSAR Table 6.2.1-6, the RCS expansion multiplier for the LOCA is stated to be 3 percent, while in the submittal, Table 6.2-1, the volumetric expansion multiplier value is changed to 2 percent. Why has this value been changed? Provide an estimate of the portion of the peak pressure change for the PUR analysis, as compared to the current UFSAR analysis, resulting from this revision.

c. Provide a description of the mechanisms which result in the lower sump temperature (see PURLR Figure 6.2-3) after about 20 seconds. Discuss if this is primarily due to the differences in the energy discharged into containment (see Figure 6.2-1)?

Main Steam Line Break Containment Analysis (PURLR Section 6.2.3)

7. The analysis of the MSLB for equipment qualification (EQ) identifies some changes to the model and input assumptions used for the analysis of the peak containment pressure response to a MSLB. In addition to the additional super-heating upon U-tube uncovery and the 8 percent condensate revaporization, are there any other EQ-related assumptions used in the EQ calculation?

Main Steam Line Break Outside Containment Analysis (PURLR Section 6.2.4)

8. A new version of the SGNIII computer program which includes a more detailed modeling of the four main steam lines versus the original analysis that modeled only two main steam lines, the closing of the MSIVs, and the steam flow through the main steam line cross header path following the closure of MSIVs was used for the main steam break outside containment analysis. Provide a description of these models and the verification and validation used to determine that their implementation in SGNIII is correct and yields expected results.

9. The mass and energy calculations for the MSLB were based on SRP 6.2.1, which the NRC staff interprets to mean SRP 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures." In addition to the changes to the computer program and model, the revised analysis also includes a reduction in conservative input values by taking credit for the MTC in determining the time of reactor trip. This appears to be inconsistent with the SRP guidance to maximize the post-accident pressure and temperature. The purpose of the analysis is to demonstrate continued compliance with GDC 4, "Environmental and Dynamic Effects Design Bases." The overall result from implementing these changes is a reduction in the calculated peak temperature in the main steam support structure when compared to the analysis of record (367°F versus 383°F). Provide a breakdown of the importance of each of the changes (model changes and input value changes) on the resulting calculated value.

10. In the submittal, it is stated that "Figure 6.2-13 provides comparison plot, of current versus proposed power, of rate of energy discharge to compartments containing breaks at zero and 4070 MWt (102% of 3990 MWt)." It appears that the figure provides a

comparison of the 0 percent and 102 percent energy discharged at the PUR power and does not provide a comparison of the current versus proposed power. Please clarify. For the limiting case, provide the comparison plot of the energy discharged from the break.

Subcompartment Loads

11. The evaluation of subcompartment loads as described in SRP 6.2.1.2, "Subcompartment Analysis," is not specifically addressed in the submittal. In UFSAR Section 6.2.1.1.1.1, "Containment Structure Accident Conditions," it is stated, "These analyses were performed at 102% of Licensed Power." In UFSAR Table 6.2.1-6, the reactor power is stated to be 3954 MWt. The PUR power level at 102 percent is 4070 MWt. Provide the reanalysis of the subcompartment loadings at the PUR power level to demonstrate continued compliance with GDC 4, "Environmental and Dynamic Effects Design Bases" and GDC 50, "Containment Design Basis." If a reanalysis not thought to be necessary then provide a discussion which supports continued use of the current licensing analyses to confirm continued compliance with GDC 4 and GDC 50.

Minimum Containment Pressure Analysis for
Emergency Core Cooling System Performance Capability Studies

12. The evaluation of the minimum pressure for ECCS performance as described in SRP 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," is not specifically addressed in the submittal. In UFSAR Section 6.2.1.5, "Minimum Containment Pressure Analysis for ECCS Performance Capability Studies," it is stated that, "A minimum containment pressure analysis was completed in the 1995-1996 time frame, to support an 'ECCS break spectrum analysis' for a licensed, rated thermal power of 3876 MWt. This analysis was revised in 2000, when the 'ECCS limiting break reanalysis' utilized more conservative containment heat sink values. See USAR Section 6.3.3." The PUR power level at 102 percent is 4070 MWt. Provide the reanalysis of the minimum pressure for ECCS performance at the PUR power level to demonstrate continued compliance with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." If a reanalysis is not thought to be necessary, then provide a discussion which supports continued use of the current licensing analyses to confirm continued compliance with 10 CFR 50.46.

Containment Heat Removal System (PURLR Section 4.1.5)

13. Describe how the increased predicted peak containment temperature and pressure as a result of the PUR affects the containment spray distribution in the post-accident containment.

14. Describe how the spray distribution is considered in the LOCA and MSLB containment analyses.

15. Discuss what criterion is used for an acceptable spray distribution?

16. Describe or provide a reference or describe how the containment spray distribution is determined for Unit 2.

Fire Protection

17. Is the following an accurate description of the fire protection program at PVNGS, Unit 2 for the proposed PUR:

Based on Section 9.6, "Fire Protection Program," of the Power Uprate Licensing Report dated December 21, 2001, the operation of the PVNGS, Unit 2 at the proposed power uprate rated thermal power level of 3990 MWt (a 2.94 percent increase) will not affect the design or operation of the plant’s fire detection systems, fire suppression systems, or fire barrier assemblies installed to satisfy NRC fire protection requirements, or result in an increase in the potential for a radiological release resulting from a fire. Any changes to the plant configuration or combustible loading as a result of modifications necessary to implement the power uprate will be evaluated by the licensee under the plants existing NRC-approved fire protection plan.

The licensee performed a thermal-hydraulic analysis of the important plant process parameters following a fire assuming the power uprate conditions. This analysis indicates that only the operator time constraints related to the time required to deplete the Condensate Storage Tank and the Reactor Makeup Water Tank volumes during plant cool down are affected by the power uprate. The licensee has concluded that the safe shutdown methodology and results identified in the UFSAR are maintained considering the modified operator response times for the power uprate. All other important plant process parameters and time constraints remain unchanged. The licensee has made no other changes to the plant’s hot standby structures, systems, components, or procedures. The licensee has made no changes to the structures, systems, components or procedures necessary to achieve and maintain cold shutdown conditions within 72 hours.

Probabilistic Safety Assessment Branch

1. Regarding the LBLOCA analyses (PURLR Section 6.4.6.3.2, Table 6.4-7)

   a. The fraction of RCS activity released to the containment atmosphere is given as 0.25 for iodines in Table 6.4-7. UFSAR Table 15.6.5-2 Item 11 provides the fraction of core inventory initially in containment atmosphere for iodine as 0.25. UFSAR Section 6.3.3.6, provides:

   “It is assumed that 100% of the noble gas and 50% of the iodine equilibrium core saturation fission product inventory are immediately released to the containment atmosphere (Source Term is calculated using US-AEC-TID14844, Methodology(32)). Of the iodine released to the containment, 50% is assumed to plate out onto the internal surfaces of the containment or adhere to internal components per guidelines of Regulatory Guide 1.4. The remaining iodine and

the noble gas activity are assumed to be immediately available for leakage from the containment."

Section 6.3.3.6 is a correct paraphrasing of TID-14844. Section 6.5.1 of the PURLR contains similar language. The 0.25 values in PURLR Table 6.4-7 and UFSAR Table 15.6.5-2 represent the fraction of core inventory available for release from containment, after credit has been taken for plateout on containment surfaces. This distinction is significant when the fractions are used in conjunction with the assumption of mechanistic plateout as credited in a subsequent section of Table 6.4-7. Licensees may take credit for the deterministic 50 percent plateout assumed in TID-14844, or may credit mechanistic plateout. In Table 6.4-7, PVNGS is, in effect, crediting plateout of the elemental iodine twice. The NRC staff finds this to be unacceptable. The licensee may credit (1) mechanistic plateout when assuming 50 percent of the core inventory is released to the containment, or (2) credit the deterministic plateout to reduce the 50 percent of core inventory released to containment to the value equivalent to a 25 percent release fraction.

Provide a justification for this deviation from regulatory guidance, or correct this assumption, re-perform the analysis, and appropriately update the PURLR and the amendment request.

b. With regard to containment spray as a fission product removal mechanism, the NRC staff's understanding of the UFSAR discussion is that the spray is effective between 90 seconds and 20 minutes and that, although a recirculation mode is identified as being available in the UFSAR, no credit is being taken for recirculation spray removal in the loss-of-coolant accident (LOCA) analysis. If the staff's understanding is incorrect, please provide (1) the assumed onset and duration of the injection and recirculation modes, and (2) spray lambda data for the recirculation sprays.

c. Explain the derivation and use of the parameters, "Spray elemental-iodine decontamination factor coefficients," and "Elemental-iodine decontamination factor credited for plate out," in PURLR Table 6.4-7.

d. In PURLR Table 9.9-2, the time of the safety inject actuation signal (SIAS) and containment isolation actuation signal (CIAS) is given as 12 seconds, which when summed with the assumed 50 second control room isolation delay is shown to equal an isolation delay of 72 seconds. In PURLR Table 6.4-7, the power access purge model isolation time (based on SIAS/CIAS) is given as 12 seconds. The text indicates that this is the estimated time of the SIAS/CIAS signals. It would appear that the 72 second control room isolation time is in error. Discuss this discrepancy.

e. Explain the relationship between the parameters, "transfer rate between sprayed/unsprayed regions" and "air transfer rates between the containment regions," in PURLR Table 6.4-7. Using Figure 1, of Attachment 2 to the licensee's letter dated May 2, 1995 (102-03345-WLS/SAB/GAM), which illustrates the PVNGS containment leakage model, correlate the three parameters in Table 6.4-7 to paths D through G on Figure 1.

f. The NRC staff understands that the licensee used the computer code LOCADOSE to determine the amount of the power access purge model release. Provide a value for average flow rate or total mass release during the 12 second release period.

g. PURLR Table 6.4-1 tabulates the core inventory at 3990 MWt. Discuss whether or not these values include the 2 percent uncertainty penalty as described in Item 3 on page 6-447 of the PURLR.

2. Regarding the SBLOCA analyses (PURLR Section 6.4.6.3.1)

   a. Page 6-458 of the PURLR indicates that analysis parameters have been reviewed for impacts and are consistent with those shown in UFSAR Table 15.6.5-2. The text indicates that values for core gas gap inventories, break size, and RCS mass and volume were changed. The NRC staff notes that UFSAR Table 15.6.5-2 addresses the LBLOCA, while UFSAR Table 15.6.5-1 addresses the SBLOCA. Discuss which table of assumptions is to be used, and identify any changes to the referenced table made to reflect the PURLR and provide the appropriate revised numeric value(s).

   b. Given the NRC staff believes that the proper reference for the SBLOCA should be to UFSAR Table 15.6.5-1 (for Question 2.a above) and that the SBLOCA for Unit 2 has changed because of the power uprate, explain the relationship of Item 19 in UFSAR Table 15.6.5-1 to the break size dependent timings in UFSAR Table 15.6.5-1.

   c. Given the NRC staff believes that the proper reference for the SBLOCA should be to UFSAR Table 15.6.5-1 (for Question 2.a above) and that the SBLOCA for Unit 2 has changed because of the power uprate, provide justification for the iodine fraction release value in Item 12 of UFSAR Table 15.6.5-1. The 50 percent instantaneous plateout assumption of TID-14844 applies to core melt situations. For the SBLOCA, the licensee has assumed only clad perforation.

   d. Respond to Questions 1.a, 1.b, 1.c, 1.e, and 1.f, in the context of the SBLOCA analysis.

3. PURLR Section 6.4.7.3 addresses fuel handling accidents. This section states that the source term used is the same as that assumed for the existing analysis in the UFSAR. Given the increase in core flux associated with the power uprate, explain how the pre-uprate source term can reflect the post-uprate source term.

4. With regard to general methodology described in PURLR Section 6.4.0:

   a. Item 6 addresses iodine spiking and identifies a coincident iodine spiking multiplier of 500. Please provide a tabulation of the iodine appearance rates to which this multiplier is applied. Discuss the assumed duration of the iodine spike.

b. Item 7 addresses steam generator iodine decontamination factors (DFs) and states that a DF of 10 is assumed for empty (dry) steam generator (SG). This assumption does not appear to be consistent with the accident-specific discussions in the UFSAR (e.g., Paragraph 8 on page 15.6-56; Table 15.3.4-5) or page 6-456 of the PURLR. This assumption is also contrary to regulatory guidance. Explain the basis for this assumption.

c. Item 8 states that the condenser generates an iodine DF=100 for non-LOCA analyses. Please identify the specific accidents to which this assumption applies. If this assumption is used in an analysis that assumes concurrent loss of offsite power (results in loss of condenser vacuum), please provide a basis for this assumption as it represents a deviation from regulatory guidance.

d. Item 14 states that the licensee back calculated from dose acceptance criteria to determine the maximum amount of fuel damage that could occur and not exceed the acceptance criteria. Discuss which events this assumption applies.

5. With regard to the events that involve secondary releases, the NRC staff understands that the licensee used an integrated thermohydraulic-radioactivity transport model. The staff located some data in the accident-specific tables and discussions in the UFSAR and PURLR. However, there are questions regarding the applicability of some of these data. The staff also notes Item 15 of Section 6.4.0 of the PURLR. However, it is not clear whether these data apply to all SGs or only to the SGs used for cooldown. For data that vary by time, conservative values for appropriate time step increments or a plot vs time would be helpful. In order to perform confirmatory evaluations, the staff needs to have the following information for all associated events:

a. Primary-to-secondary leakage or break mass flow to the affected SG,

b. Primary-to-secondary leakage or break mass flow to the unaffected SG,

c. Duration of break flow into affected SG,

d. Steam mass releases (0-2, 2-8 hour) from the affected SG,

e. Steam mass releases (0-2, 2-8 hour) from the unaffected SG,

f. Duration (and onset if not T=0) of affected SG release(s),

g. Initial RCS mass,

h. Steam generator liquid and steam mass per SG (at program or emergency operating procedure (EOP) level),

i. For applicable accidents, the break flash fraction, and

j. For applicable accidents, periods of assumed tube uncovery.

6. PURLR Section 6.4.1.2 addresses the potential consequences of a main steam line break (MSLB) outside containment concurrent with a loss-of-offsite power (LOOP). This section states that the event at full power is bounding; however, it appears from the UFSAR that the zero power case was deemed more limiting (i.e., it is the only case that has dose results). Provide an explanation for what appears to be a change in assumption.

7. PURLR Section 6.4.3.1 addresses the potential consequences of a single reactor coolant pump (RCP) rotor seizure concurrent with a LOOP and identifies the shaft seizure case as being limiting. UFSAR Section 15.3.3.3.2 provides that radiological consequences due to steam release from the secondary system would be less than the consequences of the shaft shear event. PURLR Table 9.9-2 identifies the shaft shear case (with stuck open atmospheric depressurization valve (ADV)) as being limiting with regard to control room dose. Provide the basis for the change in the limiting case.

8. PURLR Section 6.4.4.1 addresses the potential consequences of a control element assembly (CEA) ejection. The text references UFSAR Table 15.4.8-6 as containing the applicable assumptions. Provide the following information:

    a. Confirm that item D.4 in Table 15.4.8-6 provides the full core inventory and not gap inventory as stated.

    b. Item F.2.a in Table 15.4.8-6 credits a retention of 50 percent iodine in the engineered safety feature sumps. Since Item F.5 credits a flash fraction of 10 percent from the recirculated sump water (i.e., a retention of 90 percent), explain the basis for assuming a 50 percent retention in the sump.

9. PURLR Section 9.9 addresses control room habitability. Provide the following information for control room habitability:

    a. Section 9.9.3.1 provides a table of control room occupancy factors and breathing rates as a function of time following the event. The NRC staff believes these breathing rates are not appropriate for control room assessments. Section IV.3 of the Murphy-Campe report, referenced by NRC SRP Section 6.4, specifies a breathing rate of 3.47E-4 m^3/sec. TID-14844 explains that this breathing rate of 3.47E-4 was based on 50 percent of the daily air consumption occurring during the active 8 hours and the remaining half during inactive or resting periods. This leads to the breathing rate of 3.47E-4 m^3/sec for the active 8 hours and 1.75E-4 m^3/sec for the next 16 hours. After the first day, the average daily inhalation rate of 2.32E-4 m^3/sec is used. However, these breathing rates only apply to persons offsite. Control room personnel are not inactive or resting while on shift. Therefore, the breathing rate of 3.47E-4 m^3/sec should be used for estimating control room dose. In this regard, note that it is the purpose of the occupancy factor to reflect the operator's periods away from the control room when they may be inactive or resting. Assuming a breathing rate of 2.32E-4 m^3/sec and an occupancy factor of 0.4 is effectively double crediting periods of rest.

    The NRC staff considers this to be a technical error that will lead to non-conservative results. Discuss the discrepancy with the above breathing rates provided by the staff. The staff believes that this deficiency should be corrected, and the updated results should be submitted to revise the PURLR.

b. In Table 9.9-2, the control room isolation column for the RCP sheared shaft and SGTR accidents contains the entry “Not applicable release is via secondary.” Explain this entry, including the means of control room isolation and the associated time delays for these events.

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