APS

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Generating Station

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Reference: 1. Letter No. 102-04641-CDM/RAB, Dated December 21, 2001, from C. D. Mauldin, APS, to U. S. Nuclear Regulatory Commission, "Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations"

2. Letter, Dated June 14, 2002 from J. N. Donohew, USNRC, to G. R. Overbeck, APS, "Palo Verde Nuclear Generating Station, Unit 2 – Request For Additional Information Regarding Power Uprate License Amendment Request (TAC No. MB3696)"

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS) Unit 2, Docket No. STN 50-529 Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request

In Reference 1, Arizona Public Service Company (APS) submitted a license amendment request to support steam generator replacement and uprated power operations for PVNGS Unit 2. In Reference 2, the NRC provided requests for additional information from the Mechanical and Civil Engineering Branch, the Reactor Systems Branch, the Materials and Chemical Engineering Branch, the Plant Systems Branch and the Probabilistic Safety Assessment Branch.

Attachment 2 to this letter provides written responses to the questions from the Plant Systems Branch. Responses to questions from the remaining branches will be submitted separately.

No commitments are being made to the NRC in this letter.

A001

Should you have any questions, please contact Thomas N. Weber at (623) 393-5764.

Sincerely

Angela K Krainik for David Mauldin

CDM/TNW/RAB

Attachments:

1. Notarized Affidavit
2. Plant Systems Branch Questions and APS Responses

cc:	E. W. Merschoff	(NRC Region IV)
	J. N. Donohew	(NRC Project Manager)
	D. G. Naujock	(NRC Project Manager)
	N. L. Salgado	(PVNGS)
	A. V. Godwin	(ARRA)

Attachment 1

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, Angela K. Krainik, represent that I am Director, Emergency Services, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

Angela K. Krainik

Angela K. Krainik

Sworn To Before Me This 27 Day Of August, 2002.

Nora E. Meador

Notary Public

"OFFICIAL SEAL"
Nora E. Meador
Notary Public-Arizona
Maricopa County
My Commission Expires 4/8/2003

Notary Commission Stamp

Attachment 2

NRC Plant Systems Branch Questions and APS Responses

Plant Systems Branch

Balance-of-Plant Systems

NRC Question 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.

APS Response:

Section 4.2.1.3 reflects the Main Steam Safety Valves (MSSVs) "accumulated ASME rated capacity at 3% accumulation" as 19.53E+06 lb_m/hr. This is the minimum total accumulated capacity and is found by adding the nameplate rated capacity at each valve's set pressure for the 20 valves.

Section 8.8.1 reflects the MSSVs "accumulated maximum actual capacity at 3% accumulation" as 22.56E+06 lb_m/hr (note that this value is in error, the correct value is 22.258E+06 lb_m/hr). This value is the actual total accumulated capacity of the MSSVs and is found by adding the maximum actual capacity at each valve's set pressure for each of the 20 MSSVs.

Both of these numbers are listed in UFSAR Table 10.3-1 (Main Steam Supply System Design Data).

NRC Question 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:

NRC Question 2.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 off load outages for pre-PUR and PUR conditions.

Notes: 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 off load).

APS Response:

The SFP thermal analyses do not determine a peak calculated SFP temperature instead, the thermal analyses utilize the peak allowable SFP temperature as an endpoint to calculate the maximum allowable heat load. The predicted actual heat load is compared to the maximum allowable heat load to verify that adequate heat removal capability remains. This approach ensures that the heat removal capacity is greater than the predicted heat load and that the peak SFP temperature will remain lower than the maximum allowable temperature. UFSAR Table 9.1-2 provides the minimum heat removal capabilities prior to the 3990 MW_t Power Uprate (PUR). Tables 2.a-1 and 2.a-2 reflect a partial listing of UFSAR Table 9.1-2, that have been modified to contain the SFP heat removal capabilities, the number of previously discharged spent fuel assemblies in the SFP, the SFP heat exchanger cooling water inlet temperature, the peak allowable SFP temperature, and the Ultimate Heat Sink (UHS) temperature for each planned and unplanned outage scenario discussed in the UFSAR. Table 2.a-1 summarizes these parameters for the current 3876 MW_t license condition and Table 2.a-2 provides the same information for the proposed 3990 MW_t license condition.

Table 2.a-1: UFSAR Table 9.1-2, Excerpts, 3876 MW_t License Power					
Scenario	Spent Fuel Decay Heat, Btu/hr	Heat Removal Capability (BTU/hr)	Max. Allowable SFP Temp. (°F)	SFP Hx Inlet Temp. (°F)	UHS Peak Temp (°F)
Normal plant operating condition - 2 trains of fuel pool cooling and cleanup system (PC) cooled by nuclear cooling water system (NC)	12.6E+06 Administrative limit, conservatively based on the initiating condition post design basis accident SFP cooling capability.	9.9E+06/train 19.8E+06 total	125 °F	105 °F (NC)	N/A
Scheduled Normal refueling plant condition - 2 trains of PC cooled by NC, augmented by 1 train of Shutdown Cooling System (SCS)	45.8E+06 Full core offload beginning at 100 hours after shutdown plus 964 assemblies from the 12 previous annual refuelings.	19.8E+06 (PC) 37.6E+06 (SCS) 57.4E+06 total	125 °F	105 °F (NC)	<91 °F
Emergency plant condition during a scheduled refueling - 1 train of PC cooled by Essential Cooling Water (EW), augmented by 1 train of SCS	45.8E+06 Full core offload beginning at 100 hours after shutdown plus 964 assemblies from the 12 previous annual refuelings.	16.4E+06 (PC) 63.6E+06 (SCS) 80.0E+06 total	145 °F [(a)]	103 °F (EW)	<93 °F
Emergency plant condition during fuel transition mode (off load/ reload) - 1 train of PC cooled by EW, augmented by 1 train of SCS	45.8E+06 Full core offload beginning at 100 hours after shutdown plus 964 assemblies from the 12 previous annual refuelings.	16.4E+06 (PC) 32.6E+06 (SCS) 49.0E+06 total	145 °F [(a)]	103 °F (EW)	<93 °F
Emergency core off load during an unscheduled outage - 1 train of PC cooled by NC, augmented by 1 train of SCS	45.8E+06 Full core offload beginning at 100 hours after shutdown plus 1/3 core offload 90 days before the full core offload plus 884 assemblies from the 12 previous annual refuelings.	9.9E+06 (PC) 45.0E+06 (SCS) 54.9E+06 total	125 °F	95 °F (EW)	<91 °F

Note (a): Maximum temperature for the current licensing basis.

Table 2.a-2: UFSAR Table 9.1-2, Excerpts, Corrected for 3990 MW_t License Power					
Scenario	Spent Fuel Decay Heat, Btu/Hr	Heat Removal Capability (Btu/hr)	Max. Allowable SFP Temp. (°F)	SFP Hx Inlet Temp. (°F)	UHS Peak Temp. (°F)
Normal plant condition - 2 trains of PC cooled by NC	12.6E+06 Administrative limit, conservatively based on the initiating condition post design basis accident SFP cooling capability	9.9E+06/train 19.8E+06 total	125 °F	105 °F (NC)	N/A
Scheduled Normal refueling plant condition - 2 trains of PC cooled by NC, augmented by 1 train of SCS	47.0E+06 Full core offload beginning at 100 hours after shutdown plus 964 assemblies from the 12 previous annual refuelings. (b)	19.8E+06 (PC) 37.6E+06 (SCS) 57.4E+06 total	125 °F	105 °F (NC)	<91 °F
Emergency plant condition during a scheduled refueling - 1 train of PC cooled by EW, augmented by 1 train of SCS	47.0E+06 Full core offload beginning at 100 hours after shutdown plus 964 assemblies from the 12 previous annual refuelings. (b)	16.4E+06 (PC) 63.6E+06 (SCS) 80.0E+06 total	145 °F (a)	103 °F (EW)	<93 °F
Emergency plant condition during fuel transition mode (off load/ reload) - 1 train of PC cooled by EW, augmented by 1 train of SCS	47.0E+06 Full core offload beginning at 100 hours after shutdown plus 964 assemblies from the 12 previous annual refuelings. (b)	16.4E+06 (PC) 32.6E+06 (SCS) 49.0E+06 total	145 °F (a)	103 °F (EW)	<93 °F
Emergency core off load during an unscheduled outage - 1 train of PC cooled by NC, augmented by 1 train of SCS	47.0E+06 Full core offload beginning at 100 hours after shutdown plus 1/3 core offload 90 days before the full core offload plus 884 assemblies from the 12 previous annual refuelings. (b)	9.9E+06 (PC) 45.0E+06 (SCS) 54.9E+06 total	125 °F	95 °F (EW)	<91 °F

Note (a): Maximum temperature for the PUR licensing basis.
(b): See Response to Question 2b.

NRC Question 2.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.

APS Response:

The assumptions used in the SFP thermal-hydraulic analysis include:

- "In-Reactor" Hold Time - The full core offload decay heat load is assumed to commence at 100 hours post reactor shutdown. Commencing offload earlier than 100 hours post reactor shutdown is prohibited per Technical Requirements Manual 3.9.100, Decay Time. An average offload rate of 5 assemblies per hour, starting at 100 hours is assumed consistent with UFSAR Section 9.1.3, Table 9.1-2, Footnote a. This parameter is unchanged for the PUR analyses.
- Number of previously discharged spent fuel assemblies – The number of previously discharged spent fuel assemblies assumed in the normal and emergency shutdown full core offload analysis is 964. This is the design maximum heat load during emergency core offload. When combined with the full core offload of 241 assemblies, all 1205 slots available for spent fuel storage are filled. Note that UFSAR Section 9.1.3 has an additional emergency core offload case with 884 assemblies offloaded from 12 previous outages, 1/3 core (assumed as 80 assemblies) offloaded from the previous cycle 90 days ago, plus a full core offload. Results for all cases are presented in Question 2.a, Table 2.a-2. The assumed number of fuel assemblies are unchanged for the PUR analyses.
- SFP heat exchanger cooling water inlet temperature – Refer to the Question 2.a, Table 2.a-2.
- Ultimate heat sink temperature – Refer to the Question 2.a, Table 2.a-2.

NRC Question 2.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.

APS Response:

The most limiting single failure is a Loss of AC Power (LOP). A LOP in conjunction with the mechanical failure of a pump results in the greatest decrease in heat removal capacity. During refueling the Technical Specifications allow for the removal of one emergency diesel. Therefore, a mechanical failure of a pump becomes the more limiting fault. Considering this event/single failure, the remaining heat removal path consisting of 1 train of PC/EW/UHS is adequate to remove the allowable SFP heat load of 12.6E+06 BTU/hr. During outages, LOP in conjunction with the mechanical failure of a pump (Spray Pond system (SP), EW, or PC) would again result in the greatest decrease in heat removal capacity. Considering this event/single failure, the remaining heat removal path consisting of 1 train of PC/EW/UHS augmented by one train of SCS, is adequate to remove the maximum partial or full core offload decay heat load of 47.0E+06 BTU/hr. Refer to the Question 2.a, Table 2.a-2.

NRC Question 2.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?

APS Response:

The SCS is not utilized to maintain the SFP below the design temperature prior to a planned or unplanned full core offload. The PC system is capable of maintaining SFP below the design temperature prior to a planned or unplanned full core offload as described in UFSAR Section 9.1.3.3.1.1. With two trains of PC operating prior to a planned or unplanned full core offload, the SFP will be maintained below 125 °F. With one train of PC operating prior to a planned or unplanned full core offload, the SFP will be maintained below 145 °F.

The SCS may be used to augment the PC system during a planned or unplanned full core offload. One train of SCS in conjunction with one train of PC is sufficient to maintain the SFP below 125 °F during a planned or unplanned full core offload under normal conditions, as described in UFSAR Section 9.1.3.3.1.1. Also, refer to Question 2.a, Table 2.a-2.

NRC Question 2.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.

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

APS Response:

The calculated peak SFP temperature is below 150°F for planned refueling outages under either partial or full-core offload conditions (see response to Question 2.a, Tables 2.a-1 and 2.a-2).

NRC Question 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.

APS Response:

A cycle-specific SFP thermal analysis is not performed. The expected SFP heat load is calculated and verified to be bound by the current analysis. Evaluation of the "in-reactor" fuel cycles parameters show that the limiting required decay time remains less than 100 hours post shutdown. The design and licensing basis SFP thermal analyses considers that full core offload starts at 100 hours following reactor shutdown. Commencing offload earlier than 100 hours post reactor shutdown is prohibited per Technical Requirements Manual 3.9.100, Decay Time. An average offload rate of 5 assemblies per hour, starting at 100 hours is assumed as stated in UFSAR Section 9.1.3, Table 9.1-2 as discussion in Question 2.a.

NRC Question 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:

NRC Question 4.a:

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

APS Response:

Continuous remote monitoring of the SFP temperature is provided in the control room at all times via high temperature alarms. Two SFP Temperature-Hi alarms are provided to notify the plant operators of high SFP temperatures. The SFP high temperature alarm is 125 °F. The SFP temperature Hi-Hi alarm setpoint is 145 °F. These alarms provide early warning to the control room operators so that prompt action may be taken to reduce SFP temperature.

The actual indicated SFP temperature is checked on the Fuel Building local panel during operator rounds each shift.

<u>NRC Question 4.b:</u>

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.

<u>APS Response:</u>

The SFP water temperature alarm setpoints are discussed in the response to Question 4.a. At a high alarm temperature of 125 °F and assuming no pool cooling and a PUR heat load of 47.0E+06 BTU/hr, an operator has greater than 1 hour to respond and mitigate the event. This response time ensures that the temperature limit of 145 °F will not be exceeded.

<u>NRC Question 4.c:</u>

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.

<u>APS Response:</u>

If the SFP high temperature alarm setpoint of 125 °F is reached, a control room alarm window will actuate to alert the operators of a condition. The alarm response procedure directs the Reactor Operator (RO) in the control room to send an Area Operator (AO) to verify the alarm on the local alarm panel. After verifying the alarm, the AO will utilize the local alarm response procedure to address the condition.

If the SFP high-high temperature alarm setpoint of 145 °F is reached, then the control room alarm for "Fuel Pool Temp Hi-Hi" is received. The alarm response procedure directs the RO in the control room to send an AO to verify the alarm on the local alarm panel. After verifying the alarm, the AO will utilize the local alarm response procedure to address the condition. The control room operator may then enter the appropriate alarm response procedure.

For a high or high-high temperature alarm the procedure directs the operator to:

1. Verify the alarm at local panel indicator.
2. Verify the current cooling system alignment.
3. Verify the cooling water supply to the heat exchanger is adequate.
4. If necessary, to place the standby train of PC cooling in operation.

The procedure also provides information to augment SFP cooling with SCS, or to place EW and SP in operation in the event that NC is not available.

Heating, Ventilation, and Air Conditioning (HVAC) Systems

NRC Question 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.

APS Response:

The impact of PUR on the various HVAC systems is documented in an engineering study prepared in support of the licensing submittal. The Auxiliary Building HVAC (HA) and Control Building HVAC (HJ) are the only HVAC systems credited in safety analysis that are impacted by operation at PUR conditions. The worst-case event for heat loads on HA remains LOCA and MSLB. The worst-case event for heat loads on HJ remains any event resulting in a LOP. The Containment Building HVAC (HC) and Turbine Building HVAC (HT) systems are not credited in any safety analysis. New individual design calculations were prepared and/or revised as required to evaluate the heat load impact of the Replacement Steam Generators (RSGs)/PUR on the normal operation of the plant HVAC systems. The HVAC study and associated design calculations are available for review at PVNGS.

Auxiliary Building HVAC

The HA system is minimally impacted by the PUR as documented in revisions to the design basis calculations. The increases in post-LOCA and MSLB temperatures have nominally increased the transmission heat loads through the containment wall to the Auxiliary Feedwater (AFW) pump and electrical penetration rooms. The increase in temperature remains below the 104 °F maximum design temperature. This increase in transmission heat load is evaluated in the design basis for the HA and Essential Chilled Water (EC) system calculations. The increased heat load remains well within the design capabilities of the HA and EC systems. The existing HA system design basis calculations bound operation at PUR conditions for the piping and equipment heat loads.

Containment Building HVAC

A design calculation was performed to compare the normal heat load contribution of the RSGs to that of the Original Steam Generators (OSGs). This calculation considered the increased RSG surface area and the improved reflective metal insulation system. Based on the results of this calculation, the RSG's contribution to the total containment heat load is predicted to be less than that contributed by the OSGs.

The heat load from insulated piping and components in the containment building is analyzed in an existing design basis calculation for the bulk containment volume. The piping and equipment heat loads considered in this calculation are based on the original plant Reactor Coolant System (RCS) T_{hot} of 621 °F, which bounds the PUR predicted T_{hot} (618.9 °F with 10% tubes plugged) temperatures. Equipment motor heat is based on the motor rated brake horsepower, and remains bounding for operation at PUR conditions. As a result, the heat loads considered in the existing HC system design basis calculation for containment remain conservative with respect to the predicted PUR heat loads.

The RCS T_{hot} PUR conditions do not affect the non-safety related reactor cavity cooling subsystem and the Control Element Drive Mechanisms (CEDM) cooling units as their original design is based on the original licensed T_{hot} of 621 °F.

In summary, the RSG/PUR conditions are bounded by the existing HC system design.

Control Building HVAC

The control building heat loads evaluated in the HJ system design basis calculations are not impacted and remain bounding for operation at PUR conditions.

Turbine Building HVAC

The turbine building heat loads evaluated in the design basis calculations remain bounding for operation at PUR conditions. These calculations are based on the original plant piping design temperatures that considered the OSG's higher secondary operating pressure and temperature.

Containment LOCA Response Analysis (PURLR Section 6.2.2)

NRC Question 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:

NRC Question 6.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.

APS Response:

The limiting relative humidity has been changed to provide conservative peak pressure results. Consistent with Standard Review Plan (SRP) guidelines, the most limiting initial condition was selected for the PUR submittal. A higher initial relative humidity within containment results in a lower pressure profile and a lower peak value. The additional mass of water vapor in the containment atmosphere acts as a heat sink. A comparison of COPATTA computer runs verify a reduction of 0.37 psi in peak pressure if a value of 50% relative humidity is used. Therefore, assuming 0% relative humidity maximizes the containment pressure response and is conservative.

NRC Question 6.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.

APS Response:

UFSAR Table 6.2.1-6 is in error. The RCS expansion multiplier for the existing Analysis of Record (AOR) is 2%. This error will be corrected in a future UFSAR update.

NRC Question 6.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)?

APS Response:

A review of the models for the existing licensing bases (3876 MW_t) and the PUR analysis (3990 MW_t) indicate that the reason for the lower sump temperature in the PUR analysis case is due to the time when the RCS blowdown phase ends and when level is re-established in the Reactor Vessel (RV). The blowdown phase ends earlier in the PUR case, which allows Safety Injection (SI) flow into the RV to occur earlier. As a result, the RV level is re-established earlier and SI flows begins to spill out of the break earlier in the PUR analysis. Table 6.c-1 provides critical times for the existing and PURLR analysis.

Table 6.c-1: Comparison of Elapse Times		
Parameter, units	Existing Analysis	PURLR
Time of End of Blowdown, sec	24.0	17.1
Spillage to the Sump during Blowdown	1 Safety Injection Tank (SIT)	1 SIT
SI Pump Flow Initiates, sec	24.0	17.1
Time at which Annulus is Full, sec (start of direct SI flow spillage to sump)	30.8	21.0

The spillage to the sump begins at 21 seconds for the PUR analysis whereas the same spillage begins at 30.8 seconds in the existing analysis. Thus, more “cool ” water [water from refueling water tank at 120 °F] is expelled to the sump at an earlier time in the PUR case, resulting in a cooler sump water temperature.

Main Steam Line Break Containment Analysis (PURLR Section 6.2.3)

NRC Question 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?

APS Response:

There are no additional EQ related changes/assumptions.

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

NRC Question 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.

APS Response:

The SGNIII model was changed from a single node header to a two-node header. The change to SGNIII was conducted under the provisions of the Westinghouse Quality Assurance Program. This change provides the analyst with the ability to use one flow resistance (fL/D) and flow area through the cross header prior to the turbine stop valves closing and a different set of values following the closure of the turbine stop valves. This change provides better modeling of the steam header crossover path in that it provides a steaming path to the break following the closure of the turbine stop valves and prior to the Main Steam Isolation Valves (MSIVs) closing. The model change verification and validation is documented in a calculation performed and owned by Westinghouse. The code response was compared to existing data and found to yield the expected results.

NRC Question 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 General Design Criteria (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.

APS Response:

Methods used are consistent with NUREG-0800 Section 3.6.1 and Branch Technical Position (BTP) SPLB 3-1. Both inside and outside containment EQ analyses consider the full range of MTCs allowed by the Technical Specification LCOs.

For inside containment EQ case, the most negative MTC was used. MSLBs inside containment typically will trip the reactor quickly on high containment pressure so the choice of MTC does not affect the results of a MSLB inside containment.

For outside containment EQ, the full power case is non-limiting. The high power trip terminates this case. The least negative MTC delays the high power trip, leading to the worst consequences for this case.

The limiting scenario for the outside containment EQ is the no load (0%) case. A sensitivity analysis led to the value of the MTC that was presented in the license submittal. This value leads to a coincident reactor trip signal, based on high reactor power and low steam generator pressure. More negative MTCs lead to a faster reactor trip on high reactor power. Less negative MTCs lead to a faster reactor trip on low SG pressure. Identifying the MTC that results in coincident high power and low SG pressure trips is consistent with SRP guidance to maximize the post-accident pressure and temperature. This logic identifies the longest time it will take for the reactor trip to actuate.

The sensitivity analysis described above for the no load outside containment MSLBs EQ analysis is the primary reason for the lower containment temperature compared to the Analysis of Record (AOR). The AOR zero load case used a MTC more negative than that allowed by the Technical Specifications. This overly conservative MTC led to unrealistically high steam pressures during the event, thus preventing the actuation of the low steam generator pressure trip, and resulting in containment temperatures which are higher than those achievable with an MTC allowed by the Technical Specification LCOs.

NRC Question 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 MW_t (102% of 3990 MW_t)." 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.

APS Response:

The text of the PURLR submittal requires clarification, the first paragraph of Section 6.2.4.4, page 6-30, should read:

> The mass and energy (M&E) release for a 1 ft^2 MSLB outside of the containment was calculated using the SGNIII code. The results were used to calculate the potential effects of a MSLB on essential equipment outside of containment using code PCFLUD (Reference 6-43). Figure 6.2-12 provides a comparison plot, of current versus proposed power, of mass release rate versus time. Figure 6.2-13 provides a comparison plot, of rate of energy discharge to compartments containing breaks at 0% and 102% of requested power level. Pressure and temperature profiles for the most limiting compartment are presented in Figure 6.2-14. Table 6.2-7 provides a comparison of the results of a 1 ft^2 SLB outside containment. As it can be deduced, the change in code modeling results in more realistic pressure and temperature profiles. The peak temperature and pressure for the event remain bounded by the existing analysis.

Figures 6.2-13 and 6.2-14 titles and captions have been revised and are provided in Figures 10-1 and 10-2. Figure 10-3 provides a comparison plot of the limiting (no load) condition energy discharge for current and PUR conditions.

Figure 10-1: Revised Figure 6.2-13
MSLB Outside Containment for Equipment Qualification at 3990 MW_t

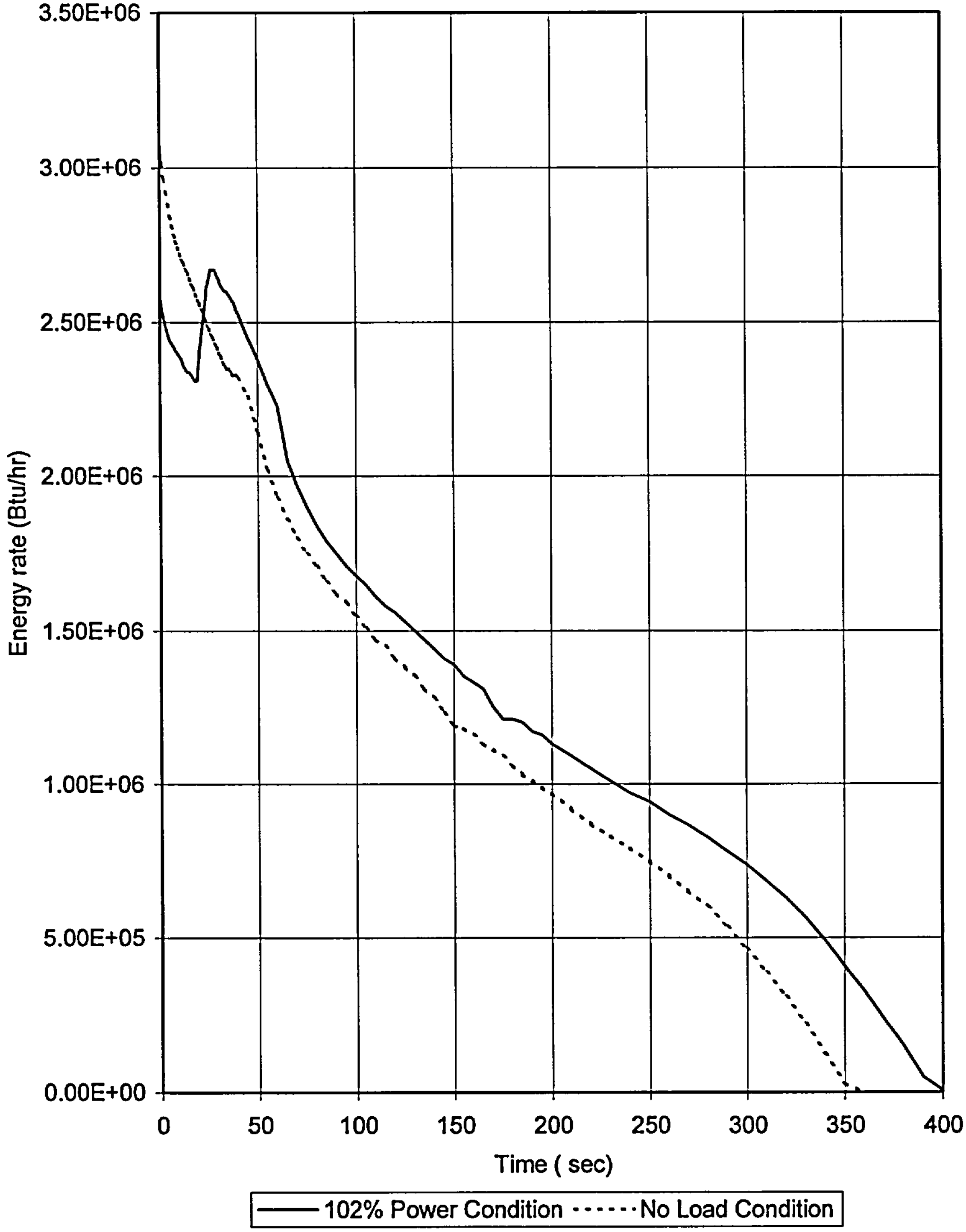


Figure 10-2: Revised Figure 6.2-14
MSLB Outside Containment Pressure and Temperature Profile for EQ at 3990 MW_t

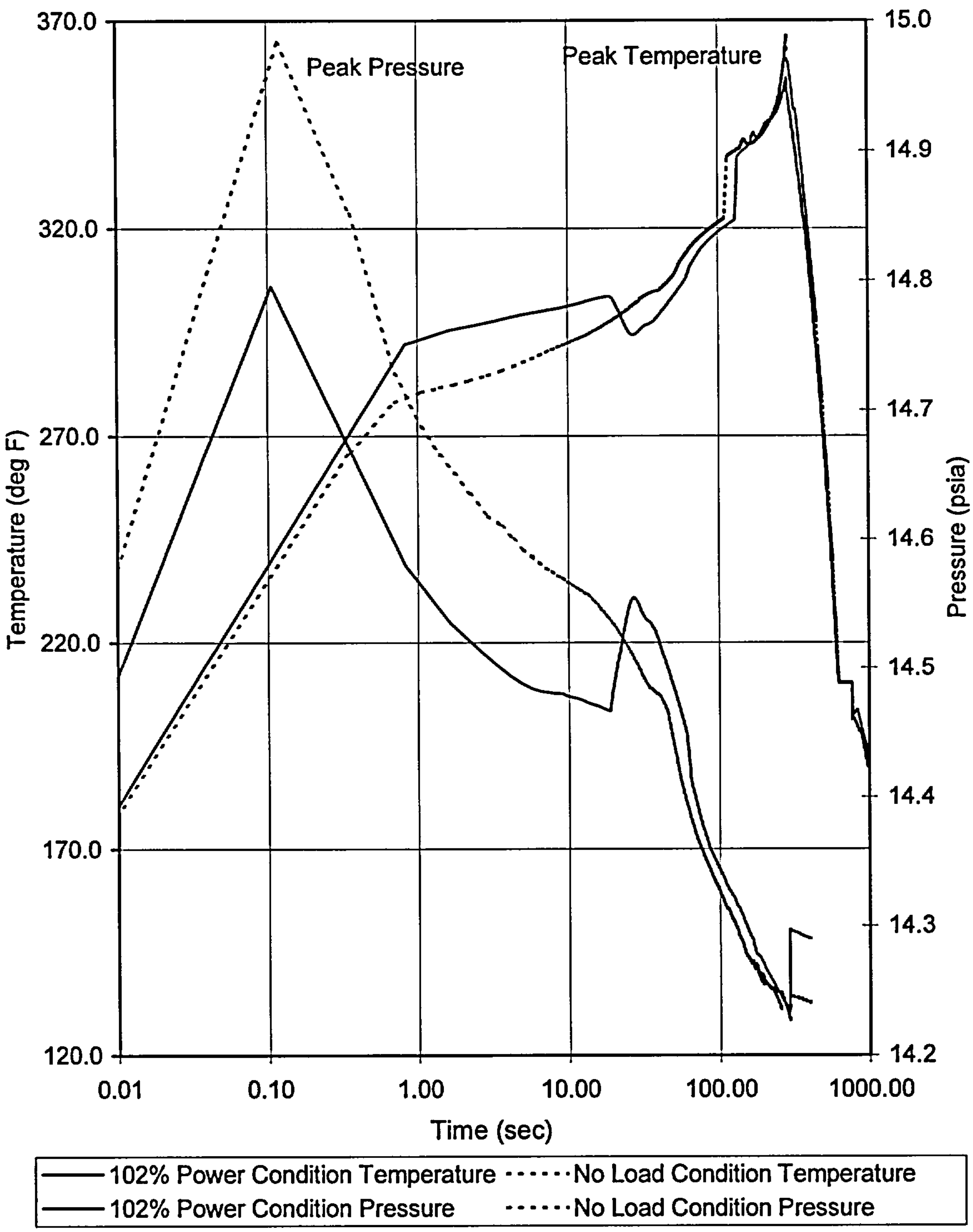
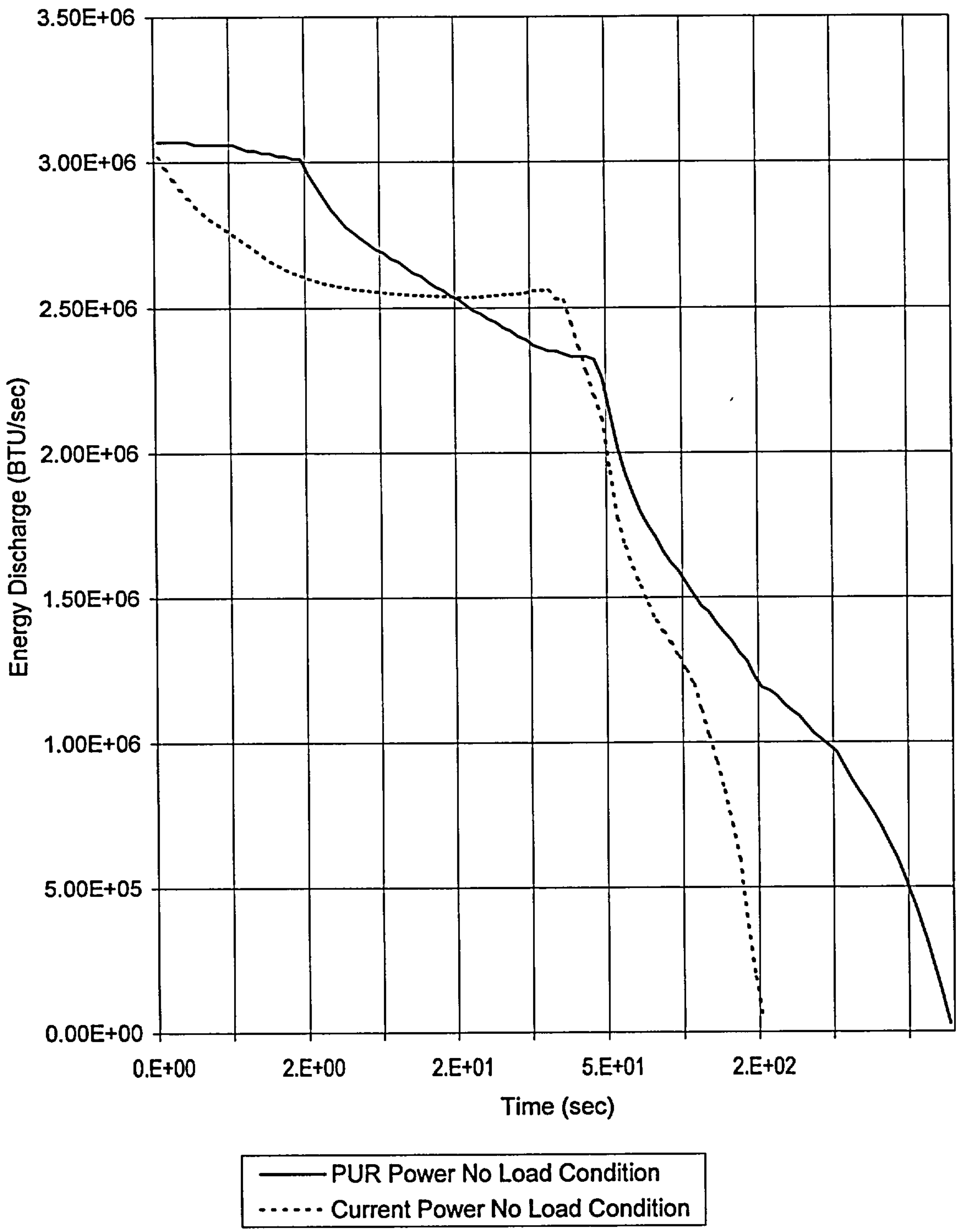


Figure 10-3: MSLB Outside Containment for Equipment Qualification Comparison at No Load Conditions Current Power and PUR



Subcompartment Loads

NRC Question 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 MW_t. The PUR power level at 102 percent is 4070 MW_t. 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 is 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.

APS Response:

The short-term LOCA-related mass and energy releases are used as input to the subcompartment analyses that are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high energy line pipe rupture within that subcompartment. The subcompartments evaluated include the SG compartment, the reactor cavity region, and the pressurizer compartment. For the SG compartment and the reactor cavity region, the fact that PVNGS Unit 2 is approved to use Leak-Before-Break (LBB) methodology (Per Docket Nos: 50-529/530, November 29, 1985) was used to qualitatively demonstrate that any changes associated with operation at PUR conditions are offset by the LBB benefit of using smaller RCS nozzle breaks. Therefore, the current licensing bases for these compartments remains bounding.

The pressurizer subcompartment analysis assumes a double-ended guillotine break of the pressurizer surge line. The energy release rates were calculated for the proposed PUR condition and compared with the original plant design conditions. The original plant design energy release rates continue to bound the PUR energy release rates by approximately 10%. Since the analysis performed for the initial plant licensing determined the pressurizer subcompartment was adequate, and the energy release rates for the PUR condition remain bounded by the original energy release rates, the pressurizer subcompartment remains structurally acceptable.

The calculated peak internal containment pressure, which does not credit LBB, increases to 57.85 psig as a result of the PUR and RSGs, but remains bounded by the containment internal design pressure of 60.0 psig. Therefore, the containment will continue to meet its original design criteria.

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

NRC Question 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 MW_t. 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 MW_t. 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.

APS Response:

As stated in Section 6.1.1 of the PURLR, the existing Emergency Core Cooling System (ECCS) performance analysis was performed for a core power of 4070 MW_t (i.e., a rated core power of 3990 MW_t plus a 2% power measurement uncertainty). A rated core power of 3990 MW_t, which is a 5% increase in the original licensed power of 3800 MW_t, was first used in the ECCS performance analysis performed for the 2% stretch power license amendment request (Letter 102-03578-WLS/AKK/GAM, W.L. Stewart (APS) to Document Control Desk (NRC), "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN-50-528/529/530, Proposed Amendments to Facility Operating Licenses and to Technical Specifications and Various Bases, Related to Power Uprate," January 5, 1996) and is the rated core power that has been used in all subsequent ECCS performance analyses. The stretch power ECCS performance analysis, which is the 1995-1996 analysis noted in the above question, was performed for a 5% increase in rated core power. The fact that the stretch power analysis was performed for a 5% increase in power was not described in the 2% stretch power license amendment, nor is it described in the PVNGS UFSAR Section 6.3.3 description of the Large Break LOCA (LBLOCA) ECCS performance analysis.

Consistent with the core power used in the LBLOCA ECCS performance analysis in UFSAR Section 6.3.3, the minimum containment pressure analysis in UFSAR Section 6.2.1.5 also used a core power of 4070 MW_t. This is true for the minimum containment pressure analysis for both the "ECCS break spectrum analysis" and the "ECCS limiting break reanalysis." Like the description of the LBLOCA ECCS performance analysis in UFSAR Section 6.3.3, this fact is not currently described in UFSAR Section 6.2.1.5. Since the minimum containment pressure analysis described in UFSAR Section 6.2.1.5 was performed at the uprated core power of 4070 MW_t, reanalysis of the minimum pressure for ECCS performance to demonstrate continued compliance with 10 CFR 50.46 was not required.

Containment Heat Removal System (PURLR Section 4.1.5)

NRC Question 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.

APS Response:

Containment spray distribution is defined as the combination of spray nozzle configuration (location and orientation), spray pattern (cross-sectional area), and droplet size. The physical configuration of the system remains unchanged. The spray patterns and associated droplet size of the individual nozzles are determined by the pressure differential across the nozzle orifice.

The system flow rate will not be reduced and the corresponding individual nozzle flow rate will be unchanged. To ensure that the spray characteristics are not adversely affected, the applicable containment spray system design calculations have been revised to account for the increased maximum containment pressure. To compensate for the increase in containment pressure, a corresponding small increase in the minimum containment spray pump performance is required. The increase in pump performance was achieved by reducing the total system head loss by installing annubar style flow measuring devices in place of the original flow measuring orifices. Containment spray system pump surveillance testing requirements will be revised accordingly.

NRC Question 14:

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

APS Response:

APS used the COPATTA computer code to assess containment atmospheric pressure and temperature conditions following a postulated design basis accident (LOCA and MSLB). This code does not consider specific spray distribution effects. Rather, the methodology assumes a well mixed atmosphere (see the response to Question 15) and computes the total energy removal rate based on total system spray flow and on a global thermal efficiency factor due to the spray action.

The topical report for the COPATTA code states that the thermal efficiency of the spray flow is correlated to the containment steam/air mass ratio produced by an assumed spray droplet diameter. The assumed droplet size used in the COPATTA code is significantly larger than the mean droplet diameter for the spray nozzles installed at

PVNGS. The response to Question 13 states that the minimum Containment Spray System (CSS) flow rate remains unchanged and that there is no consequential effect on the spray characteristics (pattern and droplet size). As a result, there is no effect on the thermal efficiency predicted by the COPATTA code. Since the thermal efficiency and total system flow rates are not changed, there is no effect on the containment response relative to spray distribution for the PUR case.

NRC Question 15:

Discuss what criterion is used for an acceptable spray distribution?

APS Response:

The CSS was designed in accordance with ANSI/ANS 56.5-1979 (PWR and BWR Containment Spray System Design Criteria) and ensures that the spray distribution promotes adequate mixing of the containment atmosphere. ANSI/ANS 56.5-1979 considers adequate mixing to occur when 90% of the operating deck is directly sprayed from the main spray headers.

In addition, auxiliary spray headers are provided below the operating deck to ensure mixing of those regions inaccessible to the direct main spray. Adequate mixing for the auxiliary spray regions is determined by the relative spray flow rate to volume ratio (spray flux) for that region. The spray flow for these regions is designed to be equal to or greater than that for the main spray region. UFSAR Section 6.5.2.2 states that the spray flow will be the same in the auxiliary regions as those sprayed by the primary (main) spray headers to assure a well mixed atmosphere.

As discussed in the responses to Questions 13 and 14, the total CSS flow rate remains unchanged. Therefore there is no impact on the spray distribution in either the main spray or auxiliary spray regions. Accordingly, the containment atmosphere continues to be adequately mixed.

NRC Question 16:

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

APS Response:

The current AOR, as described in UFSAR Section 6.5.2, remains applicable to Unit 2 at PUR conditions.

Plant Systems Branch

Fire Protection

NRC Question 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 MW_t (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.

APS Response:

The above statement is accurate with regard to the fire protection program for the PVNGS Unit 2 PUR.