APS

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U. S. Nuclear Regulatory Commission
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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, "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 Probabilistic Safety Assessment 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,

Paul F Gawley
for C D Mauldin

CDM/TNW/RAB/kg

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)
Resident Inspector (PVNGS)
A. V. Godwin (ARRA)

Attachment 1

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, Paul F. Crawley, represent that I am Director, Nuclear Fuel Management, 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.

Paul F. Crawley

Paul F. Crawley

Sworn To Before Me This 4 Day Of September, 2002.

Nora E. Meador

Notary Public

"OFFICIAL SEAL"
Nora E. Meador
Notary Public-Arizona
Maricopa County
My Commission Expires 4/10/2003

Notary Commission Stamp

Attachment 2

NRC Probabilistic Safety Assessment Branch Questions and APS Responses

Probabilistic Safety Assessment Branch

NRC Question 1:

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

NRC Question 1.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 immediate/y 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 in 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.

APS Response:

Table 6.4-7 of the PURLR and Table 15.6.5-2 of the UFSAR represent the source term available for release from containment and not the source term discharged from the Reactor Coolant System (RCS) to containment. Therefore, no revision to the analysis is required.

The analysis used for evaluating the Large Break Loss of Coolant Accident (LBLOCA) consequences was verified and it was concluded that no re-analysis is required for the PURLR. The analysis submitted assumes that 100% of the noble gas and 50% of the iodine equilibrium core saturation fission product inventory in Table 6.4-1 are immediately released to the containment atmosphere. 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 the guidelines of Regulatory Guide 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors, Revision 2, June 1974. The remaining iodine and the noble gas activity, presented in Table 6.5-1, are assumed to be immediately available for leakage from the containment.

PURLR Section 6.5.1, "Large Break Loss-of-Coolant Accident Source Term," lists assumptions used in the analysis. These include:

1. one hundred percent of the core equilibrium radioactive noble gas inventory is immediately available for leakage from containment,
2. fifty percent of the core equilibrium radioactive iodine inventory is immediately released to the containment atmosphere, and
3. half of the radioactive iodine inventory to the containment atmosphere is plated out onto the internal surfaces of the containment and the other half of the radioactive iodine inventory to the containment atmosphere is available for leakage from the containment.

NRC Question 1.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.

APS Response:

Consequences of a LBLOCA were evaluated with minimum credit for the containment spray as a fission product removal mechanism. The evaluation results provided in the PURLR credit the containment spray system to remove fission products for 386 seconds (6.43 minutes) after actuation of the containment spray system during the injection phase only. No credit is assumed for spray removal of Iodine species beyond approximately 478 seconds (the total elapsed time for the containment spray actuation is 92 seconds and the total elapsed time to reach an Iodine Decontamination Factor (DF) of 6.51 in the containment sprayed regions is 478 seconds). Therefore, no spray lambda data is required for recirculation sprays. For additional information with regard to the models used for these evaluations, refer to Attachment 2 of APS letter to NRC, Letter No. 161-04019-WFC/JRP, Palo Verde Nuclear Generating station (PVNGS) Unit 1, 2, and 3 Technical Specification Amendment Request, Section 3/4 3.6.2.2 and Bases dated June 25,1991, approved by the NRC in a letter dated September 8, 1992.

NRC Question 1.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.

APS Response:

The model used is consistent with Standard Review Plan (SRP) 6.5.2 and NUREG /CR 4697. A detailed description of the derivation has been provided to the NRC in APS letter to NRC, Letter No. 161-04019-WFC/JRP, PVNGS Unit 1, 2, and 3 Technical Specification Amendment Request Section 3/4 3.6.2.2 and Bases dated June 25, 1991, approved by the NRC in letter dated September 8, 1992. The "Spray elemental-iodine decontamination factor coefficients" in PURLR Table 6.4-7, is the "elemental iodine spray removal coefficient (λ_s (min)$^{-1}$)" described in Section 4.2.1 of Attachment 2 to the APS amendment request. The "Elemental-iodine decontamination factor credited for plate out" in PURLR Table 6.4-7, is the "elemental iodine plate-out removal coefficient (λ_w (min)$^{-1}$)" described in Section 4.3.1 of Attachment 2 to the APS amendment request. Both of these coefficients are calculated using the formula provided in SRP 6.5.2.

NRC Question 1.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.

APS Response:

To eliminate the dependency of this analysis on isolation times and control system performance and to provide margin for evaluating control room consequences, a conservative higher value of 72 seconds is used. The actual elapsed time for CIAS resulting from a LBLOCA and generation of SIAS is calculated to be approximately 4 seconds. Total time for purge valves to close (including CIAS response) is 12 seconds (NUREG-0875, Safety Analysis Report related to the operation Palo Verde Nuclear Generating Station, Unit 1,2, and 3, Supplement 7, Section 6.2.4). The total time for control room isolation dampers to close after receipt of CIAS is estimated at 50 seconds. Therefore, the minimum elapsed time for the control room damper to close is 54 seconds. Table 9.9-2 PURLR states the control room isolation time is a conservative 72 seconds. The 72 seconds includes 12 seconds to close the purge valves, for exposure to the control room, and is 18 seconds longer then the minimum control room isolation time.

NRC Question 1.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.

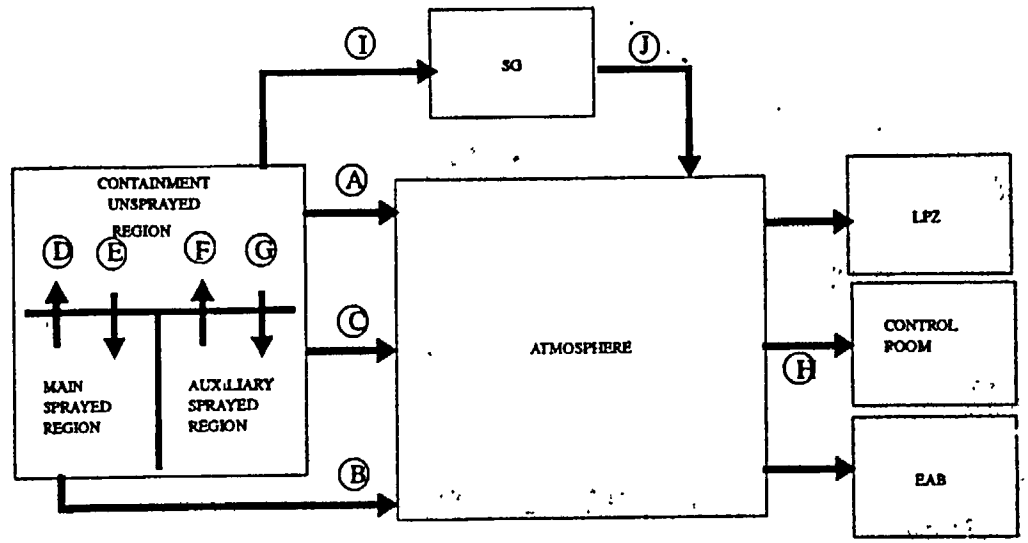
APS Response:

Item 6 of PURLR Table 6.4-7, under the containment model heading, represents the total rate of air flow between the sprayed region and the unsprayed region. The actual value for this flow rate based on containment volume is 8,250 cfm (see UFSAR Table 15.6.5-2, Item 16). Items 7a and 7b of PURLR Table 6.4-7 represent the flow between the two spray regions (main and auxiliary) and the unsprayed region (UFSAR section 6.5.2.3). Path D flow rate is presented by Item 7a (7,582 cfm) and path F flow rate is presented by Item 7b (668 cfm).

Flow path E and G are provided to maintain the mass balance in the model and their flow rates are the same as path D and F. Therefore, Item 6 of PURLR Table 6.4-7 represents the summation of Path D and F or E and G.

Below is a copy of Figure 1, of Attachment 2 to the APS letter dated May 2, 1995 (102-03345-WLS/SAB/GAM),

Figure one : Containment leakage dose model



A- DIRECT UNFILTERED LEAKAGE FRACTION FROM UNSPRAYED REGION

B - DIRECT UNFILTERED LEAKAGE FRACTION FROM MAIN SPRAYED REGION

C - DIRECT UNFILTERED LEAKAGE FRACTION FROM AUXILIARY SPRAYED REGION

D - TRANSFER RATE FROM MAIN SPRAYED REGION TO UNSPRAYED REGION

E - TRANSFER RATE FROM UNSPRAYED REGION TO MAIN SPRAYED REGION

F - TRANSFER RATE FROM AUXILIARY SPRAYED REGION TO UNSPRAYED REGION

G - TRANSFER RATE FROM UNSPRAYED REGION TO AUXILIARY SPRAYED REGION

H - ATMOSPHERIC DISPERSION BETWEEN CONTAINMENT AND OFFSITE AND CONTROL ROOM

I - LEAKAGE RATE FROM CONTAINMENT THROUGH PRIMARY SYSTEM TO SECONDARY

J - DIRECT UNFILTERED LEAKAGE FROM SECONDARY TO ENVIRONMENT

NRC Question 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.

APS Response:

The table below summarizes the flow through the purge valves during the 12 seconds before the isolation of the access purge. The average flow rate to the environment is calculated to be 19,854 cfm.

Time Period (sec)		Flow Rate	Total Flow
Beginning	End	(cfm)	(ft^3)
0.00	0.01	39,000.00	6.50
0.01	1.00	35,990.00	593.88
1.00	2.00	37,490.00	624.88
2.00	3.00	37,490.00	624.66
3.00	4.00	37,490.00	624.66
4.00	5.00	33,830.00	564.28
5.00	6.00	26,660.00	444.69
6.00	7.00	19,900.00	330.74
7.00	8.00	13,830.00	230.68
8.00	9.00	8,652.00	144.32
9.00	10.00	4,585.00	76.48
10.00	11.00	1,785.00	29.77
11.00	12.00	359.10	5.97
12.00	13.00	5.26E-04	0.00
13.00	14.00	0.00	0.00

NRC Question 1.g:

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

APS Response:

PURLR Table 6.4-1 was prepared in accordance with the guidelines in Regulatory Guide 1.49, Power Levels of Nuclear Power Plants, Revision 1, December 1973. All values presented are calculated at 102 percent of requested license power of 3990 MW_t or 4070 MW_t. The title of this table reflected the requested license power and not the analysis power.

NRC Question 2:

Regarding the SBLOCA analyses (PURLR Section 6.4.6.3.1)

NRC Question 2.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).

APS Response:

The appropriate table is UFSAR Table 15.6.5-1 "Radiological Consequences of Small Break LOCA," with the exception of the section of this table that reflects the consequences. As stated in the PURLR submittal, no specific analysis was conducted for this event for Power Uprate (PUR). A review of parameters concluded that consequences of a small break LOCA (SBLOCA) for PUR remains bounded by the consequences of the LBLOCA that is presented in Table 6.4-8 of the PURLR submittal.

All parameters are assumed the same with the exception of the following:

1. Core inventory increases as result of the power increase. Table 6.4-1 of PURLR provides isotopic composition of this source term. This source term replaces Item A.1 of UFSAR table 15.6.5-1.
2. The one ft^2 RCS break hydraulic evaluation shows that core uncovery would not occur. The current evaluation reflected on page one of UFSAR Table 15.6.5-1 shows core uncovery at 3700 seconds and recovery at 4400 sec.
3. The volume of the RCS increases as a result of the Replacement Steam Generator (RSG) project. The RCS volume will increase from 9.18E+03 ft^3 to 1.03E+04 ft^3.

NRC Question 2.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) 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.

APS Response:

Spray initiation times (CIAS) are provided in UFSAR Table 15.6.5-1. A conservative 33 second penalty is added to these values to account for the start of the diesel and sequencer (note: that this assumes Loss of AC Power (LOP) at the same time as SIAS). The total elapsed time is used in the dose assessment model to start the containment spray and credit removal of iodine species. For example in the case of a 0.05 ft^2 break, the model assumes no containment spray initiation into containment (no

removal of Iodine) until 265 seconds into the event [232 seconds (from page 1 of Table 15.6.5-1) plus 33 seconds (Item 19 of same Table)].

NRC Question 2.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) 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.

APS Response:

SBLOCA (0.05-0.005 ft^2) AOR was based on Regulatory Guide 1.4 Revision 2 and Standard Review Plan section 15.6.5, Appendix A. Application of non-mechanistic models is appropriate as long as sufficient conservatism is included in the analysis.

SBLOCA radiological analysis mimicked the LBLOCA consequences event methodology. This is justified since neither US-AEC-TID14844 nor Regulatory Guide 1.4 restricts the application of the plate-out model to minimum concentrations of radioiodines or reactor power. In addition, iodine species would behave similarly, independent of the type of fuel failure once they are released in the RCS as long as chemical and environmental conditions in the RCS and Containment remain the same.

The AOR is very conservative since it assumes that all fuel in the reactor core would develop clad perforation. Realistically, fuel failure would be significantly less for RCS breaks in this range. Also, at these break sizes the containment pressure remains low. Based on containment response analysis, peak pressure for a 0.05 ft^2 break would be less than one half of LBLOCA. Therefore, the leakage from the containment to the environment would be significantly less than what has been assumed and consequently the offsite doses would be less. Application of mechanistic releases and time dependent plate-out models would result in a small change in dose for the containment leakage pathway. However, all consequences of the SBLOCA would remain bounded by the LBLOCA.

NRC Question 2.d:

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

APS Response:

The same models are used for SBLOCA and LBLOCA. Please refer to responses for large break LOCA with exception of Item 1f. Note that purge isolation times are different for SBLOCA than LBLOCA (refer to Items 10 and 19 of UFSAR Table 15.6.5-1). Average flow rates during purge are:

1. For a break size of 0.01 ft^2, the total discharge prior to purge isolation is approximately 153,000 CF with an average flow rate of 26,500 CFM

2. For a break size of 0.005 ft^2, the containment pressurizes slowly to CIAS setpoint of 3 psia. The total discharge prior to purge isolation is approximately 50,600 CF with an average flow rate of 2200 CFM.

NRC Question 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.

APS Response:

Analysis presented in section 15.7.4 of the UFSAR has been performed for a power level of 4070 MW_t. Iodines and short lived isotope concentrations are calculated for an infinite cycle (saturation) using US-AEC-TID14844 methodology. Long lived noble gases such as Kr -85 are calculated at end of equilibrium cycle using ORIGEN-S/SCALE 4.4 Package for bounding future fuel cycle burn-up of 70,000 MWD/MTU at fuel enrichment of 5%. Therefore, the existing analysis in the UFSAR remains applicable for PUR conditions.

NRC Question 4:

With regard to general methodology described in PURLR Section 6.4.0:

NRC Question 4.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.

APS Response:

The base iodine appearance rate was calculated with the conservative assumptions as shown in Table 4.a-1:

Table 4.a-1: Assumptions used in Base Iodine Appearance		
Filtration Efficiency	1.0	most conservative, theoretical upper value
Filtration Flow	150 gpm	beyond charging/letdown capacity of 132 gpm
Primary System Initial Activity	1.0 μCi/gm	Technical Specification limit
I-131 Decay Constant	9.97 E-7 sec^{-1}	

An increase of 500 over this existing appearance rate was assumed for the entire duration of the 2 and 8 hour radiological consequences calculation. The primary system activity resulting from this increase in appearance rate is shown in Figure 4a-1. Figure

4a-2 illustrates RCS activity based upon accident Generated Iodine Spike (GIS), Pre-existing Iodine Spike (PIS), and Technical Specification limits. This figure corresponds to events that experience only primary to secondary tube leakage (e.g., Main Steam Line Break (MSLB), Feedwater Line Break (FWLB), Single Reactor Coolant Pump (RCP) Sheared Shaft/Seized Rotor (SS/SR), Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (IOSGADV) plus LOP) and no significant loss of RCS inventory.

Figure 4.a-1: Primary System Activity

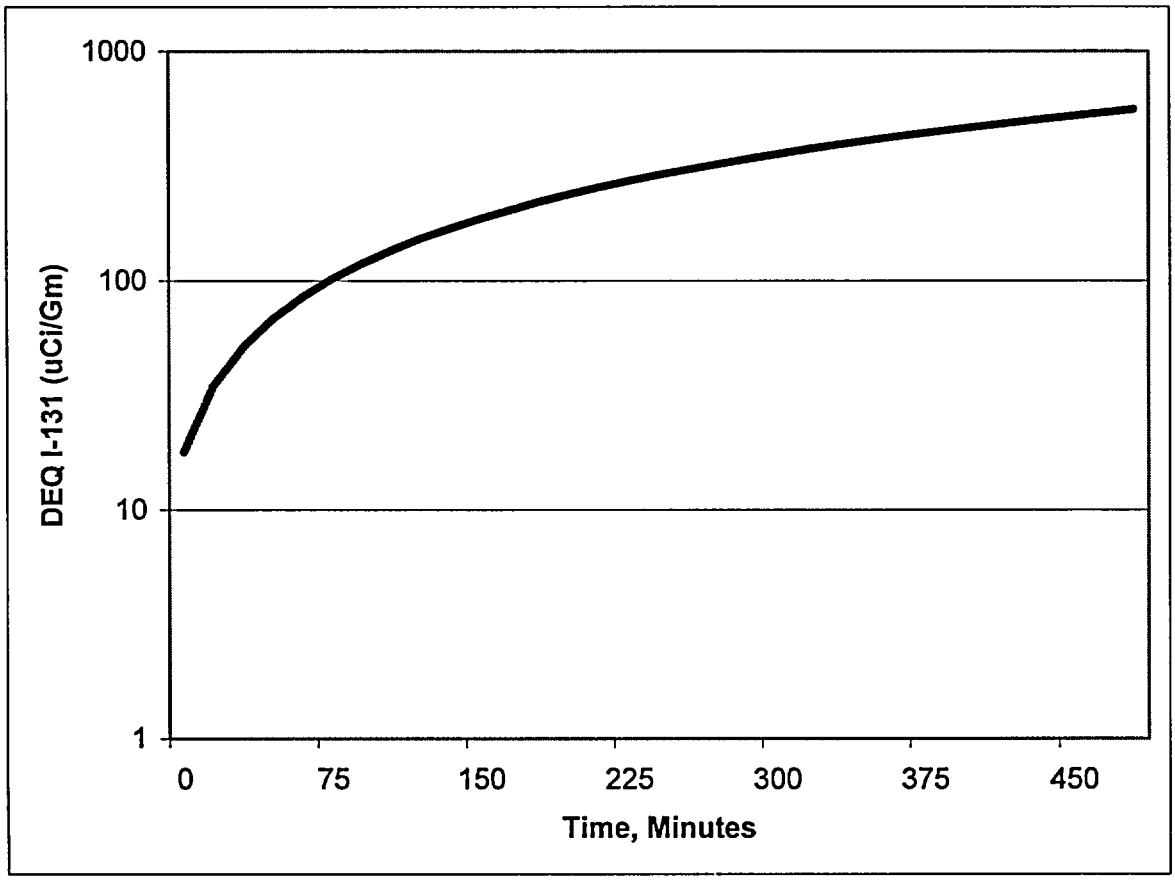
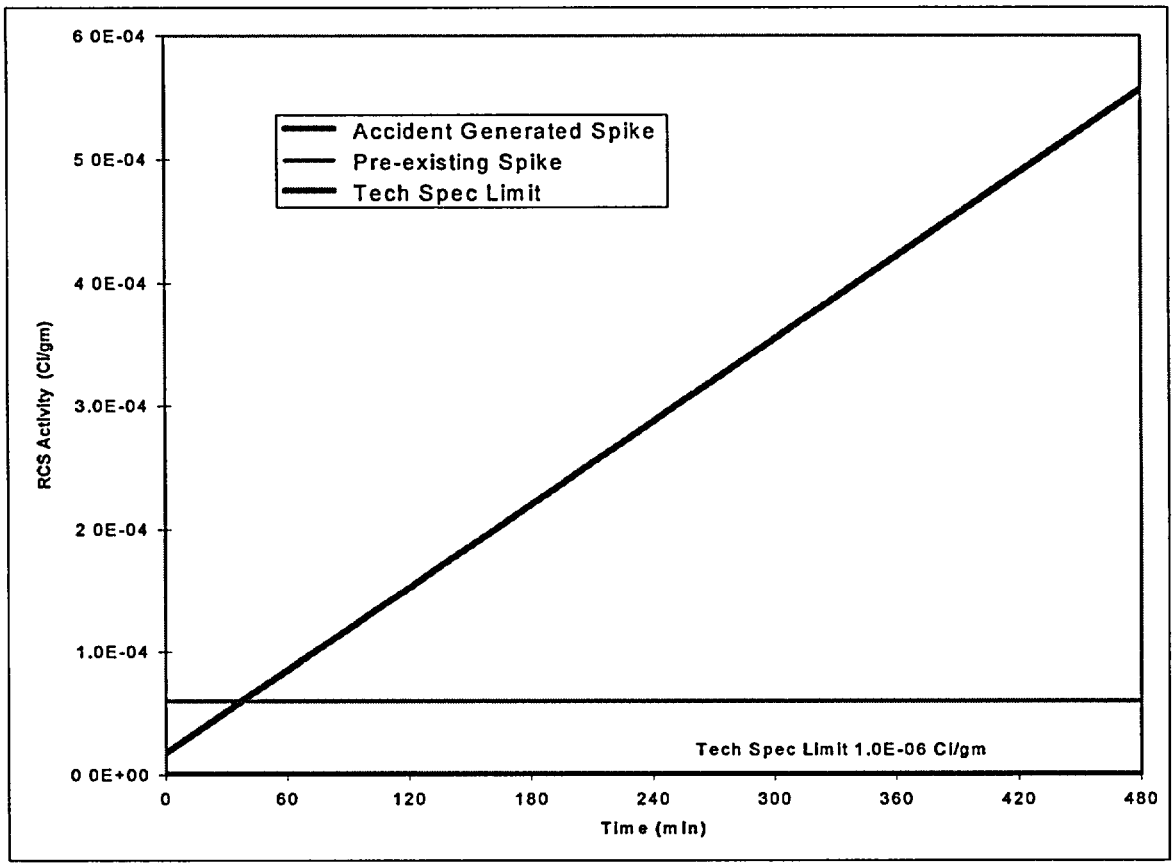


Figure 4.a-2: RCS Activity Based Upon GIS, PIS, and Technical Specification Limits



NRC Question 4.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 15.656; 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.

APS Response:

Item 7 contains an error. For situations where a SG is assumed to dryout due to a stuck open Atmospheric Dump Valve (ADV) (i.e., SS/SR with stuck open ADV, IOSGADV plus single failure), the initial inventory of that SG and the primary to secondary leakage inventory through that SG is assumed to be released directly to the environment with a DF of 1.0. Similar to other events with a "faulted" SG, the IOSGADV plus LOP event (Section 6.4.2.1) states "the affected SG empties and all leakage into that SG is released directly to the atmosphere."

The corrected assumptions from the PURLR, Section 6.4.0, Methodology Used for Radiological Assessment Analyses, page 6-447 should read:

The assumptions used for calculating radiological releases to the atmosphere follow. Any deviation from these assumptions is specified in the event specific discussions.

Item 5. Primary-to-secondary SG tube leakage of **0.5 gpm/SG** is included in the calculation of activity releases to atmosphere via secondary system release path.

Item 7. SG iodine DFs used for non-LOCA scenarios are:

- empty (dry), DF = **1.0** and
- level maintained, DF = 100.

NRC Question 4.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.

APS Response:

During early stages of the Steam Generator Tube Rupture (SGTR) with a LOP (SGTRLOP and SGTRLOP plus single failure events) an iodine DF of 100 is assumed for release of activity in the condenser. Upon the reactor/turbine trip and subsequent LOP, the condenser becomes unavailable. All subsequent steaming is directly to the atmosphere through secondary valves (e.g., MSSVs and ADVs). Besides the SGTR events, all other events assume that the condenser is unavailable due to a LOP.

NRC Question 4.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.

APS Response:

Based upon event specific steaming and release path, offsite doses were calculated as a function of RCS activity for events that experience fuel failure. The offsite doses presented for these non-LOCA events were based upon the following, back-calculated amount of fuel failure shown in Table 4.d-1. Future core reload designs will be selected to ensure that RCS activity remains below the value assumed in these dose calculations.

Table 4.d-1: Back-Calculated Amount of Fuel Failure	
Event	Amount of Fuel Failure
IOSGADV plus LOP	5.5%
Outside Containment MSLB with LOP	1.0%
Single RCP SS/SR with LOP	17.0%
Control Element Assembly (CEA) Ejection	PURLR Section 6.4.4.1
Limiting Infrequent Anticipated Operational Occurrence (AOO)	10.0%

NRC Question 5:

With regard to the events that involve secondary releases, the NRC staff understands that the licensee used an integrated thermo hydraulic-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:

NRC Question 5.a:

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

APS Response:

Although the primary to secondary SG tube leakage limits are 150 gallons per day per SG per PVNGS Technical Specification 3.4.14.d, leakage assumptions for the radiological consequences calculations are shown in Table 5.a-1. The duration is a 0-8 hour time period for which radiological consequences are calculated.

Table 5.a-1: Leakage/Break Assumptions				
Section	Event	Affected SG Leak Flow (duration)	Unaffected SG Leak Flow (duration)	Affected SG Break Flow (duration)
6.4.1.1	IOSGADV with LOP	0.5 GPM (0-8 hrs)	0.5 GPM (0-8 hrs)	N/A
6.4.1.2	MSLB Outside Containment with LOP	0.5 GPM (0-8 hrs)	0.5 GPM (0-8 hrs)	N/A
6.4.2.1	FWLB with LOP	0.5 GPM (0-8 hrs)	0.5 GPM (0-8 hrs)	N/A
6.4.3.1	Single RCP SS/SR with LOP	0.5 GPM (0-8 hrs)	0.5 GPM (0-8 hrs)	N/A
6.4.4.1	Control Element Assembly (CEA) Ejection with LOP	N/A see PURLR Section 6.4.4.1	N/A see PURLR Section 6.4.4.1	N/A
6.4.6.2.1	SGTR with LOP	Figure 6.3-231	1.0 GPM (0-8 hrs)	Figure 6.3-231
6.4.6.2.2	SGTR with LOP and single failure	Figure 6.3-218	1.0 GPM (0-8 hrs)	Figure 6.3-218
6.4.8	Limiting AOO	0.5 GPM (0-8 hrs)	0.5 GPM (0-8 hrs)	N/A

NRC Question 5.b:

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

APS Response:

See response to Question 5.a.

NRC Question 5.c:

Duration of break flow into affected SG,

APS Response:

See response to Question 5.a.

NRC Question 5.d:

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

APS Response:

The steam mass releases are presented in Table 5.d-1.

Releases from the 'unaffected SG' are based upon the heat loads imposed by decay heat production, removal of energy from the RCS during cooldown, and the heat added by the RCPs. The RCP heat addition is conservatively assumed to be from all four running RCPs throughout the cooldown. Steaming rate is depicted for two different cooldown rates:

1. a maximum cooldown rate of 100 °F/hr which results in a more severe 2 hour dose and
2. a minimum cooldown rate which allows SG radioactivity concentration to build, thereby, resulting in a more severe 8-hour Low Population Zone (LPZ) thyroid dose consequence.

No credit is taken for heat removal capacity of the releases from the 'affected SG'.

The two SGTR scenarios are the exception to the above discussion. For these events, rather than a conservative hand calculation of the heat load, the CENTS code was used to simulate operator control of the NSSS during the cooldown.

Table 5.d-1: Secondary System Steam Mass Releases

Section	Event	Affected SG[1] (lb_m)	Unaffected SG		Duration of Affected SG Release
			0-2 hrs (lb_m)	0-8 hrs (lb_m)	
6.4.1.1	IOSGADV with LOP	180,000	1,000,000	2,550,000	0-30 min
6.4.1.2	MSLB Outside Containment with LOP	300,000	1,000,000	2,550,000	0-8 hrs
6.4.2.1	FWLB with LOP	300,000	1,000,000	2,550,000	0-8 hrs
6.4.3.1	Single RCP SS/SR with LOP	180,000	1,000,000	2,550,000	0-90 min (see Section 6.4.3.1)
6.4.4.1	CEA Ejection with LOP	N/A not calculated, see Section 6.4.4.1			
6.4.6.2.1	SGTR with LOP	See Figures 6.3-238 and 6.3-237			
6.4.6.2.2	SGTR with LOP and single failure	See Figure 6.3-222			
6.4.8	Limiting AOO	N/A	1,000,000	2,550,000	N/A

Note 1: For events with an affected SG, sensitivity studies are conducted on both a large SG mass inventory and a small SG mass inventory. The limiting case was presented in the PURLR Section 6.4. See also response to Question 5.h.

NRC Question 5.e:

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

APS Response:

See response to Question 5.d.

NRC Question 5.f:

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

APS Response:

See response to Question 5.d.

NRC Question 5.g:

Initial RCS mass,

APS Response:

A minimum initial RCS mass of 560,000 lb_m was assumed in the radiological consequences calculation. A range of initial RCS liquid inventory from 560,000 lb_m to 625,000 lb_m was evaluated in sensitivity studies to bound allowable plant operations. A low RCS mass will result in the greatest increase in the specific activity of the primary coolant, above the assumed initial Technical Specification activity, for an assumed number of fuel rods experiencing cladding failure.

NRC Question 5.h:

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

APS Response:

In the radiological consequences calculations, a sensitivity study for the range of SG liquid inventory was conducted. Depending on the sensitivity study result, a bounding high (300,000 lb_m) or bounding low (180,000 lb_m) SG inventory is used. The low end of the range was selected to envelope an average SG liquid mass inventory in the unaffected SG over the duration of the transient. The average value for SG liquid inventory is determined based on the transient response (i.e., emergency feedwater start and cut-off, oscillation of SG level after the trip, power level, etc.), and Emergency Operation Procedure (EOP) guidelines that instruct the operators to maintain a certain level. The high end of the range was selected to envelope high SG liquid levels determined by the CENTS model for PUR conditions that were developed over a range of initial conditions.

The two SGTR scenarios are the exception where time-dependent SG inventory is used. For those events, PURLR Figures 6.3-221 and 6.3-234 provide SG liquid mass inventory as a function of time as used in the dose calculations.

NRC Question 5.i:

For applicable accidents, the break flash fraction, and

APS Response:

Figures 6.3-220 and 6.3-233 of the PURLR present the flashing fractions for the SGTR scenarios.

For the other non-LOCA events, for the first 30 minutes of the transient SG level oscillations occur, an explicit flashing fraction was not calculated.

NRC Question 5.j:

For applicable accidents, periods of assumed tube uncovery.

APS Response:

For applicable accidents, Table 5.j-1 presents the periods of assumed tube uncovery time:

Table 5.j-1: Accident Time Periods for Tube Uncovery		
	Affected SG	Unaffected SG
IOSGADV plus LOP	0 – 30 minutes	maintains level
MSLB Outside Containment with LOP	0 – 8 hours	maintains level
FWLB with LOP	0 – 8 hours	0 – 30 minutes
Single RCP SS/SR with LOP	0 – 90 minutes	maintains level
CEA Ejection	See Section 6.4.4.1	
SGTRLOP	See Section 6.4.6.2.1	
SGTRLOPSF	See Section 6.4.6.2.2	
Limiting Infrequent AOO	SG level maintained	

NRC Question 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.

APS Response:

Per PVNGS UFSAR Section 15.1.5.4, the maximum potential for radiological releases due to fuel failure occurs for full power steam line breaks outside containment. This conclusion remains valid for PUR operation. The description of the event in the UFSAR is from the original (Cycle 1) analysis. That analysis used CESEC code and Original Steam Generators (OSGs). The MSLB events were reanalyzed using CENTS and the

RSGs. Sensitivity studies were performed to determine the most limiting case with the new code and plant parameters. Thus, some differences exist between the original MSLB analyses results and initial conditions and the PUR analyses.

The MSLB cases consist of two categories:

(1) Pre-Trip power excursion and
(2) Post-Trip Return-to-Power (R-t-P).

The pre-trip power excursion cases focus on the initial power increase and challenge to the Departure from Nuclear Boiling Ratio (DNBR) to the Specified Acceptable Fuel Design Limits (SAFDL). The post-trip R-t-P cases focus on maintaining shutdown margin. Note that a minimal R-t-P may result in an approach to DNBR and fuel melt SAFDLs. Therefore, both initial power excursion and R-t-P cases are analyzed for various initial and transient parameters. For PVNGS, currently and for PUR, the fuel failure is postulated to occur during the initial power excursion (i.e., Pre-Trip MSLB). Due to the least amount of reserved initial thermal margin to the DNBR SAFDL, the Hot Full Power (HFP) Pre-Trip MSLB scenario is the most limiting case for PUR.

NRC Question 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.

APS Response:

The Single RCP SS with LOP event and Single RCP Locked Rotor (SR) with LOP event have identical steaming and release paths. In addition, both of these events assume that a single ADV sticks-open. The difference in both on-site and off-site doses would be based on fuel performance. Although very similar, the SS event yields slightly higher fuel failure. As result of higher fuel failure (and higher associated RCS activity), the Single RCP SS with LOP event is limiting. This is stated in the PURLR Sections 6.3.3.3 and 6.3.3.4.5. This is consistent with the current PVNGS AOR. The title of Section 6.4.3.1 is incorrect and should read: "Section 6.4.3.1 – Reactor Coolant Pump Shaft Break with Loss of Offsite Power"

NRC Question 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:

NRC Question 8.a:

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

APS Response:

UFSAR Table 15.4.8-6 Item D.4 provides the full core inventory at 102% of licensed power. The UFSAR will be corrected.

NRC Question 8.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.

APS Response:

The model used for estimating the radiological consequences of Emergency Safety Feature (ESF) leakage to the Auxiliary Building post Safety Injection (SI) Recirculation Actuation Single (RAS) mimics SRP 15.6.5, Appendix B (NUREG 75/087). Application of the model is conservative.

The source term used is conservative since

a) all iodine available in the gap of fuels experiencing Departure from Nucleate Boiling (DNB) is released into the containment for release to the environment and
b) 50% of the fuel gap inventory is released from the fuel and assumed to be retained in RCS liquid and spilled to the sump.

This means 150% of the source term is available for release. Although simplistic, the model provides a conservative approach for evaluation of this path way.

Selection of the flashing fraction is also based on SRP 15.6.5, Appendix B, Section II. Water accumulated in the sump would have an average bulk temperature below boiling (30 day release). In addition, sump water is buffered with trisodium phosphate (TSP) and a partition factor of greater than 1000 is expected for sump chemistry.

The contribution from this pathway is shown in UFSAR Table 15.4.8-8. Total dose from this pathway is negligible compared with the containment, the purge leakage and secondary release pathways (contribution is less than 0.07% for Exclusion Area Boundary (EAB) and less than 0.3% for LPZ). This means that changes in the model have an insignificant impact on final radiological conclusions reached.

NRC Question 9:

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

NRC Question 9.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 person's 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.

APS Response:

Table 9-1 provides a dose summary of the control room habitability analysis using a constant breathing rate of 3.47E-4 m^3/sec. Table 9-1 replaces Table 9.9-2 of the PURLR. These analyses use a 10 scfm in-leakage for ingress/egress for a total of 63 scfm unfiltered in-leakage from all sources. In conclusion, the radiological consequences to the control room personnel are within the requirements of 10 CFR Part 50, Appendix A, GDC 19 for all design bases events.

NRC Question 9.b:

In Table 9.9-2, the control room isolation column for the RCP SS 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.

APS Response:

Refer to Table 9-1 for SGTR and RCP Shear Shaft events control room isolation times. Table 9-1 replaces Table 9.9-2 of PURLR.

Table 9-1: Summary of Control Room Radiological Assessment at 62 scfm Total In-Leakage to Containment					
Event	Condition of Fuel During the Accident	Release Duration (hr)	Control room isolation time/signal	Whole Body (rem)	Limiting Organ Dose (thyroid) (rem)
LOCA	100% Fuel melt (1)	720	12 sec SIAS/CIAS + 50 sec for control room to close = 72 sec (conservatively)	2.03E+00	18
CEA Ejection	17% Fuel experiences DNB and maximum peaking factor of 1.77	720 (2)	69 Sec SIAS/CIAS + 50 sec for control room to close = 119 sec	8.34E-01	27 (3)
RCP SS with single failure of an ADV	17% Fuel experiences DNB and maximum peaking factor of 1.72	2.3 (4)	351 sec SIAS + 50 second for control room to close = 401 sec	5.93E-01	14
SGTR with single failure of ADV	No Failed Fuel/PIS	8 (4)	245 sec SIAS + 50 second for control room to close = 295 sec	4.55E-01	18

Notes: (1) Regulatory Guide 1.4 model
(2) Secondary discharge is terminated at initiation of Shutdown Cooling (SDC) approximately .
(3) Based on a DF = 100 for secondary side during the event.
(4) Time to reach SDC. Control room doses are calculated for 720 hr , to include effect of residual radioactivity build up.