Justification for Continued Operation
Steam Generator Tube Rupture Analysis Concerns

JCO-91-02-01

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Revision 1 Summary

Revision 1 to the Steam Generator Tube Rupture Analysis Concerns Justification for Continued Operation increases the administrative control on dose-equivalent Iodine-131 from 0.4 μCi/gm to 0.6 μCi/gm. An evaluation performed subsequent to the initial issue of JCO (91-02-00) determined that an administrative limit of 0.6 μCi/gm will provide the necessary compensatory measure to ensure dose consequences will remain under the 10 CFR Part 100 dose limits of less than 30 Rem during a postulated SGTR + LOP. Change bars are utilized and annotated with a "1" to identify Revision 1 changes. R1

EXECUTIVE SUMMARY

On October 24, 1991, Arizona Public Service Company (APS) internal validation efforts for the upgraded Emergency Operating Procedures (EOPs) identified a concern that early (relative to no operation action for 30 minutes currently assumed) operator actions could result in more steaming during the first 30 minutes of a Steam Generator Tube Rupture (SGTR) than is currently reflected in UFSAR Section 15.6.3.1, and which could result in correspondingly higher offsite doses. This concern was documented in Condition Report/Disposition Request (CRDR) 9-1-0236. Subsequent investigation determined that a Steam Generator Tube Rupture with Loss of Offsite Power (SGTR+LOP), without a single failure, could result in doses higher than those reported for the SGTR+LOP event in CESSAR Section 15.6.3.2. The UFSAR Section 15.6.3 reports dose consequences resulting from a SGTR with offsite power available and the case for a SGTR+LOP with a single failure (SF), but not for a SGTR+LOP.

A review of the licensing submittals and corresponding safety evaluation reports (SERs) for SGTR events determined that, following NRC review of the CESSAR SGTR and SGTR+LOP analyses, Palo Verde Nuclear Generating Station (PVNGS) was required to reanalyze the SGTR+LOP event with a limiting single failure, as documented in UFSAR Section 15.6.3.2. The PVNGS FSAR originally referenced CESSAR for the SGTR and SGTR+LOP events. In 1988, UFSAR Section 15.6.3 was amended to replace the reference to the CESSAR SGTR+LOP with the SGTR+LOP with single failure analysis from UFSAR Appendix 15A, although it is not clear why the SGTR+LOP case was removed. Hence, there exists some confusion as to the PVNGS licensing basis for a SGTR+LOP (without a single failure). In the absence of a clearly defined PVNGS licensing basis for a SGTR+LOP, the Standard Review Plan (SRP) 15.6.3 acceptance criteria is deemed to apply.

An engineering evaluation of the radiological consequences of the postulated SGTR + LOP, incorporating operator actions consistent with both current and upgraded EOPs, was performed. Using the CESSAR assumptions for steam generator flashing and partitioning, and dispersion factors consistent with the UFSAR Section 15.6.3.2 (SGTR+LOP+SF) analysis, the evaluation results are well within the SRP 15.6.3 criteria and 10 CFR Part 100 limits (see Table 2, Case 4 on page 16), though they exceed the corresponding values documented in CESSAR 15.6.3.2. R1

The evaluation was also performed using the more conservative UFSAR assumptions for steam generator flashing and partitioning. By applying the compensatory action of limiting Reactor Coolant System (RCS) dose equivalent Iodine-131 to 0.6 μCi/gm, the offsite doses are in compliance with the 10 CFR Part 100 dose limits and within the acceptance criteria of SRP 15.6.3 (see Table 2, Case 1 on page 16). Hence, continued operation of PVNGS Units 1, 2, and 3 under the current Emergency Operating Procedures is justified. R1

The 0.6 μCi/gm administrative limit is therefore imposed on PVNGS Units 1, 2 and 3 as a compensatory measure based on the attached JCO. The above limit does not supercede the more restrictive limit of 0.2 μCi/gm currently imposed on PVNGS Units 2 and 3 for the Interfacing System Loss of Coolant Accident (ISLOCA) JCO. R1

This compensatory action will remain in effect until it is demonstrated that the FSAR analyses envelope those actions that the operators may take in accordance with the EOPs. This may require changes to the EOPs, possible reanalysis and UFSAR changes.

I. PURPOSE

This Justification for Continued Operation (JCO) demonstrates that 10 CFR Part 100 Limits will not be exceeded should a Steam Generator Tube Rupture with concurrent Loss of Offsite Power (SGTR+LOP) occur, and reactor operators perform necessary actions as allowed in the Emergency Operating Procedures (EOPs) and in accordance with CEN-152 (Reference 1).

On October 24, 1991, APS internal validation efforts for the upgraded EOPs identified a concern that early (relative to no operator action for 30 minutes as currently assumed) operator actions could result in more steaming during the first 30 minutes of a Steam Generator Tube Rupture (SGTR) than is currently reflected in UFSAR Section 15.6.3.1, and which could result in correspondingly higher off-site doses. This concern was documented in Condition Report/Disposition Request (CRDR) 9-1-0236. Subsequent investigation determined that a SGTR+LOP (without a single failure) could also result in doses higher than previously reported in CESSAR, which may exceed the criteria set forth in Standard Review Plan (SRP) 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)."

II. DISCUSSION

A. Licensing Bases

A review of the licensing submittals and corresponding safety evaluation reports (SERs) for SGTR events determined that, following NRC review of the CESSAR SGTR and SGTR+LOP analyses, PVNGS was required to reanalyze the SGTR+LOP event with a limiting single failure (SF), as currently documented in UFSAR Section 15.6.3.2. The PVNGS FSAR originally referenced CESSAR for the SGTR and SGTR+LOP events. However, the radiological consequences of the SGTR+LOP were not explicitly approved by the NRC for either CESSAR or the UFSAR. CE was required to reanalyze assuming a most limiting single failure, as documented in CESSAR Appendix 15D. Since the original PVNGS design did not include atmospheric dump valve (ADV) block valves, a separate SGTR+LOP analysis was required assuming that one ADV remained stuck open for the duration of the event. The PVNGS SGTR+LOP+SF analysis, as currently documented in UFSAR Section 15.6.3.2 and Appendix 15A, was subsequently accepted by the NRC.

In 1988, UFSAR Section 15.6.3 was amended to replace the reference to the CESSAR SGTR+LOP with the SGTR+LOP+SF analysis from UFSAR Appendix 15A. It is not clear why the SGTR+LOP case was removed from the UFSAR as the NRC did not clearly indicate that the SGTR+LOP+SF was to be used in lieu of the SGTR+LOP case. Since the NRC explicitly accepted the SGTR+LOP+SF analysis, there exists some confusion as to the PVNGS licensing basis for a SGTR+LOP

(without a single failure). In the absence of a clearly defined PVNGS licensing basis for a SGTR+LOP, the SRP 15.6.3 criteria noted below is deemed to apply.

The SRP 15.6.3 criteria for a postulated SGTR, with and without loss of offsite power, is "less than a small fraction of the 10 CFR Part 100 guidelines" (or 30 Rem) for 2 and 8 hour thyroid doses for a postulated event with the equilibrium iodine concentration at full power operation in combination with an assumed event generated iodine spike (GIS), and "within 10 CFR Part 100 guidelines" (or 300 Rem) for 2 and 8 hour thyroid doses for an accident with an assumed pre-existing iodine spike (PIS).

B. Summary of CESSAR SGTR+LOP Postulated Event (CESSAR Section 15.6.3.2)

The CESSAR SGTR+LOP event is a penetration of the barrier between the reactor coolant system and the main steam system which results from the failure of a steam generator U-tube. Integrity of the barrier between the RCS and main steam system is significant from a radiological release standpoint. The radioactivity from the leaking steam generator tube mixes with the shell-side water in the affected steam generator. The loss of offsite power subsequent to the time of reactor trip and turbine/generator trip is assumed in the analysis, since it produces the most adverse effect on the radiological releases due to the direct release of steam to the atmosphere. Under such circumstances the plant would experience a loss of load, normal feedwater flow, forced reactor coolant flow, condenser vacuum, and steam generator blowdown capability. The plant is assumed to operate at full power for a period of approximately 20 minutes before the consequences of the primary-to-secondary leak cause the reactor trip. Thus, during this time period the activity in the affected steam generator increases and could be released by operator action for stabilizing the plant (possible steaming through the atmospheric dump valves (ADVs) as allowed by the EOPs) or by the opening of the main safety valves. As a result of the reactor trip, the turbine/generator trips within one second after the pressurizer low pressure reactor trip signal. As a consequence of the turbine/generator trip, offsite power is assumed to be lost due to grid instability. A three second delay between the time of turbine trip and the time of loss of offsite power, as documented in CESSAR Section 15.6.3.2, is assumed in the analysis. The CESSAR SGTR+LOP event assumes no operator action for 30 minutes following the reactor trip, at which time the affected steam generator is isolated, terminating the release from the affected generator.

C. Summary of Operator Actions and EOPs

The UFSAR SGTR and CESSAR SGTR+LOP analyses assume that operator action is delayed until 30 minutes after initiation of the event. Thereafter, the operators are assumed to isolate the affected steam generator, then begin a controlled cool down and depressurization of the Reactor Coolant System (RCS). Hence, no steaming

through the ADVs occurs in either event until after the affected steam generator is isolated.

Alternatively, the current emergency procedure directs the operators to stabilize nominal RCS conditions (i.e., temperature at T_{cold}< 568°F and RCS pressure sufficient to maintain subcooling > 28°F) until they enter the appropriate recovery operation procedure. These actions occur after reactor trip and before the SGTR is diagnosed. In a SGTR+LOP, this would necessitate steaming through the ADVs while supplying at least 250 gpm of auxiliary feedwater to at least one generator. In the safety evaluation presented below, actions to stabilize the RCS are assumed to begin at 7 minutes after reactor trip. Upon entering the SGTR recovery operation procedure, the operators are directed to begin a controlled cool down and establish a hot leg temperature of 550°F, prior to isolating the affected steam generator. This cool down is assumed to begin 17 minutes after reactor trip. The cool down rate for this interval is limited to 100°F/hr. If T_{cold} is 570°F, and T_{hot} is approximately 30 degrees higher, then at least 30 minutes would be required to achieve the specified T_{hot} isolation temperature. Operator actions simulated in the "cool down algorithm"[1] required approximately 57 minutes (from the time of rupture) to isolate the affected steam generator.

III. SAFETY EVALUATION

A. Engineering Analysis of Postulated Events Based on Current EOPs for SGTR and SGTR+LOP Events

In response to the above concerns an analysis of a SGTR+LOP at PVNGS was performed using best estimate assumptions for operator actions consistent with the guidance contained in CEN-152 and the corresponding PVNGS EOPs. The SGTR+LOP event effectively bounds the SGTR event, since LOP requires that the atmospheric dump valves or main steam safety valves, rather than the steam bypass control system, be used for steaming.

The objective of this analysis was to determine the radiological consequences for a SGTR+LOP, based on operator actions performed in accordance with CEN-152 guidance, and compare the results to the SRP criteria. For comparison purposes, the analysis was performed using both CESSAR and UFSAR steam generator

[1] The thermal hydraulic response of the NSSS to the SGTR+LOP was simulated using the CESEC-III computer program up to the time the operator takes control of the plant, and a CESEC-III based "cooldown algorithm" thereafter. These algorithms were used in the UFSAR Section 15.6.3.2 analysis.

partitioning[2] and flashing fraction[3] assumptions based on an initial activity of 1.0 μCi/gm. An evaluation was performed using the more conservative assumptions presented in the UFSAR which determined that an initial activity of 0.6 μCi/gm could be achieved while maintaining dose consequences within the allowed 10 CFR Part 100 limits. R1

Other initial conditions and assumptions used for the analysis are summarized in Table 1. These conditions were selected to generate an early reactor trip, similar to that in UFSAR Section 15.6.3.2. An early trip is deemed conservative because it maximizes the interval in which steaming occurs, through either the main steam safety valves (MSSVs) or the ADVs, within the two hour interval.

The radiological consequences were calculated based on the following assumptions and parameters:

1. The accident doses are calculated for two different assumptions: a) assuming a Pre-existing Iodine Spike (PIS) and b) assuming a Generated Iodine Spike (GIS) at the start of the event.

2. A spiking factor of 500 employed for the Generated Iodine Spiking case.

3. The Technical Specification limit of 60 μCi/gm dose equivalent Iodine-131 was used to evaluate the Pre-existing Spiking case.

4. The postulated guillotine rupture of a steam generator tube would not result in additional fuel failure.

5. The atmospheric dispersion factors for the site Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) are based on Table 2.3-31 of PVNGS UFSAR.

The operators are assumed to enter the post-trip emergency procedure and stabilize RCS temperatures by opening one ADV on each generator 7 minutes after trip. It is further assumed that the operators will determine that a SGTR has occurred, enter the SGTR recovery operation procedure, and initiate a controlled cooldown at 17

[2] The definition of partitioning is the ratio of the concentration of a nuclide in the gas phase to that in the liquid phase when the liquid and gas are at equilibrium.

[3] The definition of flashing fraction is that fraction of the leaked liquid that flashes into steam once it enters the affected generator.

minutes after trip.[4] The cooldown rate was assumed to be limited to 90 °F/hr (versus the Technical Specification limit of 100 °F/hr). Following trip, the hot leg temperature would be at least 600°F (e.g., approximately 30°F above T_{cold}). At this rate, at least 30 minutes would be required to establish a hot leg temperature of 550 °F so that the affected steam generator could be isolated (the nominal cooldown rate for this interval of the analysis was approximately 80 °F/hr). These assumptions were made for consistency with both the current and upgraded EOPs. Operator actions at 7 and 17 minutes after trip are deemed conservative based on ANSI N58.8 guidance, and are consistent with expected operator performance based on training experience in the simulator. Additionally, sufficient margin is reserved to account for an increase in subcooling margin requirements associated with the pending upgraded EOPs. R1

A review of the licensing history for PVNGS regarding SGTR events determined that the licensing basis for a PVNGS SGTR+LOP event is unclear. UFSAR Section 15.6.3.1 presents a SGTR analysis using flashing fraction and partitioning assumptions taken from the original CESSAR analysis. These assumptions are consistent with those used in the CESSAR SGTR+LOP. CESSAR assumes that the flashed portion of the leak is discharged from the affected steam generator during steaming, regardless of steam generator level, with a partition factor of 1.0. The unflashed leakage is assumed to mix with the steam generator liquid inventory. No credit is taken for iodine scrubbing from the flashed portion during intervals when level is above the U-tubes.

The conservatisms inherent in the CESSAR flashing fraction and partitioning assumptions are documented in Reference 2, and utilized in the CESSAR Section 15.6.3.2 (SGTR+LOP) event. While not explicitly accepted on the PVNGS docket, the CESSAR SGTR+LOP analysis was referenced in the PVNGS FSAR prior to 1988. Additionally, CESSAR Appendix 15D (SGTR+LOP+SF) used the same flashing fraction and partitioning assumptions. This methodology was subsequently accepted by the NRC.

UFSAR Section 15.6.3.2 currently presents a SGTR+LOP plus a single failure (SF) of a fully stuck open ADV. Unlike the CESSAR assumptions, the UFSAR flashing fraction and partitioning assumptions state that the entire leak is discharged from the affected steam generator whenever steaming occurs and the steam generator level is below the U-tubes, with a partition factor of 1.0. When the liquid level is above the U-tubes, the UFSAR assumptions are consistent with CESSAR.

[4] This assumption allowed a nominal interval of 10 minutes to diagnose and enter the SGTR recovery operations procedure.

Based on the results obtained from the above analyses, continued operation of PVNGS Units 1, 2 and 3 under the current Emergency Operating Procedures is justified. These results are summarized in Table 2 and are presented in Section B below.

B. Compliance with 10 CFR Part 100 Limits

In the absence of a clearly defined licensing basis for a SGTR+LOP event, an engineering evaluation of the radiological consequences of the event was performed. Using previously approved CESSAR methods, based on Standard Review Plan 15.6.3 with respect to steam generator partitioning and flashing fraction assumptions, the evaluation results are well within the 10 CFR Part 100 limits. Considering the compensatory action of limiting RCS dose equivalent Iodine-131 to 0.6 μCi/gm, the PIS and GIS cases for off-site dose are in compliance with the 10 CFR Part 100 dose limits and within the acceptance criteria of Section 15.6.3 of the SRP. R1

The analysis of a SGTR+LOP using CESSAR flashing and partitioning assumptions results in 13 and 11 Rem for 2 and 8 hour thyroid doses (GIS), and 32 and 11 Rem for 2 and 8 hour thyroid doses (PIS). These doses are well within the specified acceptance criteria.

The analysis of a SGTR+LOP using the more conservative UFSAR assumptions (regarding leak partitioning) presented in UFSAR Section 15.6.3.2, results in 40 and 16 Rem for 2 and 8 hour thyroid doses (GIS), and 200 and 38 Rem for 2 and 8 hour thyroid doses (PIS). Of these results, only the 2 hour thyroid (GIS) dose exceeds the acceptance criteria. The GIS cases were also analyzed using a postulated initial activity of 0.6 μCi/gm. By imposing a limit of 0.6 μCi/gm, the dose consequences of this analysis is reduced to less than 30 Rem and approximately 12 Rem for 2 and 8 hour thyroid doses (GIS). Hence, an administrative limit on initial activity of 0.6 μCi/gm would ensure that a SGTR+LOP, using UFSAR 15.6.3.2 assumptions and operator actions consistent with current EOP guidance, would not exceed the specified acceptance criteria. An administrative limit of 0.6 μCi/gm will be imposed on PVNGS Units 1, 2, and 3 for this JCO, however, this limit does not supercede the more restrictive limit of 0.2 μCi/gm currently imposed on PVNGS Units 2 and 3 for the ISLOCA JCO. R1

C. Compensatory Actions

PVNGS will monitor the primary coolant specific activity as required by Technical Specification 3/4.4.7. If the specific activity of the primary coolant is found to be greater than 0.6 μCi/gm dose equivalent Iodine-131 for more than 48 hours during one continuous time interval, the unit will be in at least hot standby with T_{cold} less than 500°F within 6 hours. This compensatory action will be implemented by changing the dose equivalent Iodine-131 acceptance criteria of surveillance test R1 R1

procedures 74ST-9RC02, "Reactor Coolant System Specific Activity Surveillance Test." However, the above limit does not supercede the more restrictive limit of 0.2 μCi/gm dose equivalent Iodine-131 currently imposed on PVNGS Units 2 and 3 for the ISLOCA JCO. R1

D. Probability of a SGTR Event

Experience with nuclear steam generators indicates that the probability of complete severance of the Inconel vertical U-tubes is remote. The more probable modes of failure result in considerably smaller penetrations of the pressure barrier. They involve the formation of etch pits or small cracks in the U-tubes or cracks in the welds joining the tubes to the tube sheet. The most limiting steam generator tube rupture event is a double-ended rupture of a U-tube at full power conditions.

Eddy current testing is used to identify tube damage mechanisms such as fretting wear, cracking, pitting, denting and the presence of loose parts. PVNGS Technical Specification 3/4.4.4 requires that the tubes selected for each in-service inspection shall include at least 3% of the total number of tubes in all steam generators and any tube with greater than 40% through-wall damage be removed from service. Typical eddy current inspection sample sizes on PVNGS units have been in excess of 30%.

Based on the confidence in the capabilities and extent of the PVNGS Steam Generator Eddy Current Program, APS uses the results of eddy current testing for the identification of loose parts in the tubesheet region in lieu of performing secondary side visual inspections. The Eddy Current Program includes complete inspections of the tube bundle periphery and tube lane for the presence of loose parts. With the use of multi-frequency eddy current probes, PVNGS has demonstrated the capability of identifying and characterizing foreign objects in the steam generators. The use of eddy current testing has previously identified loose parts resulting in steam generator tube degradation in all three units at PVNGS. Additional efforts beyond the scope of the program demonstrate the ability to identify and characterize loose parts that are either in contact with or in close proximity to steam generator tubes.

Combustion Engineering (ABB-CE) has performed an evaluation for foreign objects or loose parts recently identified in PVNGS Unit 2. The conclusions reached by APS and ABB-CE indicate that the most likely effect of leaving the foreign objects in the PVNGS steam generators is continued wear in their current locations. Therefore, plugging and staking the affected tubes would minimize any impact on continued operation. The presence of wear coupled with eddy current test results, historical information, known steam generator geometry and localized flow profiles, indicate that the objects have found a "preferred" location and will not migrate from their present location.

The potential consequences of further tube damage of active tubes in the event that migration of the objects occurs, due to dislodging or size reduction, was also evaluated. As part of the engineering evaluation, ABB-CE and APS performed an analysis which considers how loose parts have historically and experimentally interacted with steam generator tubing. The analysis included a specific assessment of the configuration, orientation, kinetic energy and contact forces of each object. Although unlikely, it was determined that any through-wall leak that may develop will produce a small detectable stable leak allowing for controlled and timely shutdown in accordance with PVNGS Technical Specification 3.4.5.2.

The leak detection and leak response capabilities at PVNGS are consistent with the recommendations of IE Notice 91-43 (Reference 3). The type of leak that may develop as a result of leaving existing foreign objects in the Unit 2 steam generators, is considered to be small, stable and easily detectable given the current status of the primary-to-secondary leak detection systems and administrative controls at PVNGS. This conclusion is supported by the assessment performed and documented in APS Engineering Study 02-MS-A72, Revision 0. Generic Letter 85-02 (Reference 4), and NUREG 0844 (Reference 5) reported that the worst case scenario postulated as a result of loose part damage is a tube rupture. The potential for a steam generator tube rupture event resulting from tube damage caused by the identified objects is considered to be negligible based on the assessment performed for each object. Thus, the probability or consequences of a SGTR accident are not increased due to the presence of these small foreign objects.

ABB-CE analyzed the potential for a "Ginna" type tube impact/collapse condition. The events at Ginna are considered very specific, in that the tube rupture event required a unique set of conditions and resulting effects. At Ginna, a large object (nearly 3.5 pounds) repeatedly impacted a plugged, unstaked tube eventually causing tube collapse with a loss of tube structural integrity. The unstaked tube severed and in turn became a whip and subsequently ruptured an active tube. Only plugged unstaked tubes are considered susceptible to this event. A plugged tube has an atmospheric internal pressure, thereby providing less resistance to tube collapse due to repeated impact. Staking the tubes will prevent the tubes from whipping should tube degradation continue to a point where severance may occur (NUREG 0844).

At PVNGS all tubes currently affected by the loose parts have been plugged and staked with one exception. Unit 2 Steam Generator 2, tube 51/170, was plugged due to a potential loose part indication with wear, but was not staked. This is contrary to the current APS practice of plugging and staking tube with this kind of indications. This practice was in the formative stage during the very earliest ECTs at PVNGS, and was not executed for this one particular tube. This tube is in the interior of the bundle, the potential loose part and wear indications were above the second support plate on the hot leg side, and the object must be small since none of the adjacent

tubes had a potential loose part indication. Therefore, there is no resemblance to the Ginna situation and the lack of a stake for this plugged tube is acceptable.

A review of all tubes in the vicinity of the objects recently found in Unit 2 (considering potential migration routes) revealed no plugged, unstaked tubes. Similar conclusions have been reached in previous evaluations of foreign objects in Units 1 and 3.

A probabilistic risk assessment was performed for a SGTR+LOP using generic industry data for a SGTR and loss of offsite power. The results indicate that the annual probability of a SGTR+LOP is 4.3 E-6. This is considered a low probability for non-core melt sequences and the potential risks imposed on public health and safety is therefore judged to be minimal.

E. Supplemental Conservatisms

There exists many other conservative factors that, if treated realistically, would result in substantially lower dose consequences. These factors are summarized below, and are provided on a supplemental basis:

1. Atmospheric dilution factors - The results noted above assume conservative atmospheric dilution factors of 3.1×10^{-4} sec/m^3 for the exclusion area boundary (EAB) and 5.1×10^{-5} sec/m^3 for the low population zone (LPZ).[5] The realistic values for these factors are 4.3×10^{-5} sec/m^3 (EAB) and 4.9×10^{-6} sec/m^3 (LPZ).[6] A similar evaluation of dose consequences based on the realistic values would reduce dose consequences to approximately 14% (5.5 Rem versus the calculated value of 40 Rem for EAB) and 10% (1.6 Rem versus the calculated value of 16 Rem for LPZ).

2. Auxiliary Feedwater - The analyses noted above assume auxiliary feedwater rates of no more than 650 gpm/pump. This value is based on the Technical Specifications,[7] and is benchmarked to a steam generator pressure of 1270 psia. During intervals when the steam generators are being steamed through the ADVs, steam generator pressure will be well below this pressure. Hence, the auxiliary

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5 UFSAR Table 2.3-31. Determined in accordance with Reg. Guide 1.145, August 1979.

6 UFSAR Table 2.3-31. Based on the overall 50th percentile 2-hour and the maximum annual average dilution factors.

7 Based on the pending Tech. Spec. change, which is more conservative than current Tech. Spec. value of 750 gpm.

feedwater pumps would be capable of a faster feed rate during this interval. The increased feed rate would allow the operators to establish steam generator liquid levels above the U-tubes more rapidly, thereby reducing dose.

3. Cooldown Rate - The analyses noted above assume a maximum allowable cooldown rate of 90 °F/hr, such that a 50 °F cooldown requires at least 33 minutes. Following isolation of the affected steam generator, a cooldown rate of approximately 50 °F/hr is assumed until 2 hours into the transient. Thereafter, a slower cooldown rate is assumed, such that a total of 8 hours is needed to achieve shutdown cooling entry conditions. Allowable cooldown rates would actually result in reaching shutdown cooling entry conditions within 6 hours, which would reduce the analyzed dose.

4. The CESEC code and "cool down algorithm" utilize a single node, equilibrium steam generator model. This model produces conservative results in predicting a positive primary to secondary leak for the duration of the transient, accompanied by periodic steaming of the affected steam generator to prevent overfilling. More realistic modeling of the steam generator, as used in the CEPAC and CENTS codes, indicates that an essentially equilibrium pressure in the RCS and affected steam generator can be established during the cooldown portion of the transient, and that operator action may be needed to prevent reverse leakage. Hence, the dose consequences predicted due to periodic steaming of the affected steam generator to prevent overfill are overly conservative.

5. Diagnosis of the SGTR accident is facilitated by radiation monitors which initiate alarms and inform the operator of abnormal activity levels. These radiation monitors are located in the condenser air removal pumps, steam generator blowdown lines, and turbine and auxiliary building ventilation ducts and stacks. No credit is taken for operator diagnosis of the event prior to 17 minutes into the event.

IV. CONCLUSION AND JUSTIFICATION FOR CONTINUED OPERATION

In the absence of a clearly defined licensing basis for a SGTR+LOP event, an engineering evaluation of the radiological consequences of the event was performed. Using previously approved CESSAR methods, based on Standard Review Plan 15.6.3 with respect to steam generator partitioning and flashing fraction assumptions, the evaluation results are well within the 10 CFR Part 100 limits. Considering the compensatory action of limiting RCS dose equivalent Iodine-131 to 0.6 μCi/gm, the | RI

PIS and GIS cases for off-site dose are in compliance with the 10 CFR Part 100 dose limits and within the acceptance criteria of Section 15.6.3 of the SRP. This administrative limit will be imposed on PVNGS Units 1, 2 and 3 for this JCO. However, this limit does not supercede the more restrictive limit of 0.2 μCi/gm currently imposed on PVNGS Units 2 and 3 for the ISLOCA JCO. R1

In summary, the above analysis and actions demonstrate that: (1) if a SGTR+LOP were to occur and operator actions were taken in accordance with the current EOPs, the limits on dose consequences would not be exceeded (2) the compensatory action of prohibiting continued operation with equilibrium RCS activity levels above the administrative limit of 0.6 μCi/gm dose equivalent I-131, and an iodine spiking factor of 500, results in radiological consequences that are less than a small fraction of the 10 CFR Part 100 limits for the event Generated Iodine Spiking cases and within the 10 CFR Part 100 limits for the Pre-existing Iodine Spiking cases. Thus, until final determination on improvements to emergency operating procedures, and evaluation of the SGTR events are completed, continued operation does not adversely affect the health and safety of the public. R1

V. POTENTIAL CORRECTIVE ACTIONS

During EOP validation of the SGTR recovery operations procedure, several ways of reducing the dose consequences of this event were identified, as summarized below. These options may be evaluated and, if feasible, will be incorporated into the upgraded EOPs, which are due to be implemented during the third quarter of 1992. Those changes indicated by "*" were incorporated into the current EOPs in January, 1992. R1

*A. During post-trip recovery actions, establish steam generator level between 72 - 80 % wide-range (WR), in lieu of the 35 - 80 %WR currently specified. This will reduce the time required to cover the U-tubes should subsequent diagnosis indicate that a SGTR is in progress.

*B. Prior to entering the SGTR recovery operations procedure, the operators maintain nominal RCS temperatures by a combination of steaming and feeding. If sufficient feedwater is available, maintain RCS nominal conditions by increasing the feed rate, rather than by steaming, until the liquid level is above the top of the U-tubes.

*C. If sufficient feedwater flow is available, initiate cool down by increasing the rate of feedwater injection, rather than by steaming through the ADVs, until the liquid level in the steam generators is above the top of the U-tubes and the affected steam generator is isolated. Thereafter, cool down by steaming and feeding the intact steam generator.

D. Engineering evaluations have been performed to support a more rapid, initial cooldown of the RCS to allow for earlier isolation of the affected steam generator. At the start of the cooldown phase of the transient, after diagnosis of a SGTR, the hot leg temperature is ~600 °F. An initial cooldown of approximately 50 °F would be needed to isolate the affected steam generator. This could be achieved by cooling at a rate of 210 °F/hr for 20 minutes, to be followed by a 40 minute soak time before additional cooling occurs. The faster cooldown rate would allow the affected steam generator to be isolated within a cooldown interval of 15 minutes, with a corresponding reduction in dose consequences. Once the steam generator isolation criteria is achieved and the affected steam generator isolated, T_{hot} would be held constant for a sufficient interval to ensure that the Tech. Spec. cooldown limit of 100 °F/hr is not exceeded.

E. During the cooldown phase of a SGTR+LOP transient, and following steam generator isolation, the affected steam generator may need to be steamed periodically to avoid overfilling. The steam generator level indication for initiating level reduction is currently 80 %WR. A reduction in dose can be achieved by using a steam generator level indication of 85 % narrow-range (NR) for this purpose. The NR indication would provide greater accuracy, and would allow additional leakage to be retained in the affected steam generator, in lieu of steaming.

VI. FUTURE ACTIONS TO REASSESS UFSAR 15.6.3 EVENTS

A. SGTR+LOP

The doses obtained for the SGTR+LOP, using the CESSAR assumptions, indicate that the CESSAR SGTR+LOP analysis may not be bounding for PVNGS because the assumed operator actions listed below may not be consistent with CEN-152 and the corresponding EOPs:

1. Operators are currently directed to steam both generators prior to identifying and isolating the affected steam generator, in order to avoid repeatedly lifting the MSSVs. During this interval, RCS temperatures are maintained at nominal values, and auxiliary feedwater is used to restore steam generator levels between 35 and 80 %WR.

2. Operators are directed to diagnose the event as a possible SGTR before entering the SGTR recovery operations procedure and commencing a cool down. During this interval, operators would continue to steam both steam generators.

3. Operators are required to cool the RCS to 550 °F before isolating the affected steam generator. The cool down rate is currently limited to 100 °F/hr. Typically, a cooldown of ~70 °F is needed to establish isolation criteria.

Due to these actions, more steaming may occur through the ADVs prior to restoring steam generator levels above the U-tubes than is currently reflected in the CESSAR SGTR+LOP. Hence, the CESSAR SGTR+LOP analysis may not be not bounding for PVNGS.

ACTION - NUCLEAR FUEL MANAGEMENT & NUCLEAR LICENSING Determine the PVNGS licensing basis for the SGTR+LOP analysis, and the need to incorporate operator actions, consistent with CEN-152 guidance, into the analysis. A PVNGS specific SGTR+LOP analysis is currently under development and should be completed by December, 1992. R1

B. SGTR+LOP+SF

The doses obtained for the SGTR+LOP, using the UFSAR flashing fraction and partitioning assumptions, indicate that the loss of a single auxiliary feedwater pump may be a more limiting single failure than the fully stuck open ADV analysis presented in UFSAR Section 15.6.3.2. The determination of a limiting single failure is a function of the assumptions used for leak partitioning. If the CESSAR assumptions are applied, the current single failure (i.e., a fully stuck open ADV) analysis is expected to remain bounding. If the UFSAR Section 15.6.3.2 assumptions are used, the dose consequences of the event are significantly affected by the interval in which steaming from the affected steam generator occurs while the U-tubes are uncovered. The loss of an auxiliary feedwater pump, would increase the duration of this interval. The limiting SGTR+LOP+SF was originally determined in CESSAR, and incorporated in the PVNGS FSAR. The more conservative assumptions given in UFSAR Section 15.6.3.2 were applied after the fact to maximize the dose consequences of a single failure consisting of a fully stuck open ADV without block valves.

ACTION - NUCLEAR FUEL MANAGEMENT & NUCLEAR LICENSING Determine the PVNGS licensing basis for the SGTR+LOP+SF analysis, and the need to incorporate operator actions, consistent with CEN-152 guidance, into the analysis. In addition, a re-evaluation of the limiting single failure will be performed. If a more limiting single failure is identified, the SGTR+LOP+SF will be reanalyzed. R1

VII. REFERENCES

1) CEN-152, Combustion Engineering Emergency Procedure Guidelines, Rev. 2&3.

2) Supplemental Safety Evaluation Report Regarding System 80, from C.L. Miller (NRC) to A.E. Scherer (CE) dated August 4, 1989.

3) NRC IE Notice 91-43, Recent Incidents Involving Rapid Increases in Primary-to-Secondary Leak Rate, July 5, 1991.

4) NRC Generic Letter 85-02, Staff Recommended Actions Stemming from NRC Integrated Program of the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity, dated April 17, 1985.

5) NUREG 0844, NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5, Regarding Steam Generator Tube Integrity, September 1988.

Table 1
Initial Conditions and Assumptions for PVNGS SGTR+LOP Reanalysis

Parameter	SGTR+LOP Reanalysis Assumptions	Corresponding UFSAR Section 15.6.3.2 Values
Initial core power level, MWt	3,876	3,876
Core inlet coolant Temp, °F[8]	570	570
RCS mass flow rate, 10^6 lbm/hr	155.8	155[10]
Reactor coolant system pressure, psia	2100[9]	2100[10]
Steam generator pressure, psia	1070	1126[10]
Initial pressurizer liquid volume, ft^3	460	460
Available AFW pump flow, gpm/pump for SG pressure at 1270 psia.	650	750

Additional Assumptions

a. The SGTR+LOP reanalysis assumes an initial SG liquid level of 36.5 ft. above the tube sheet. This assumption corresponds to ~ 46%NR (actual), and allows for 4% instrument uncertainty on SG level during normal operating conditions.

b. Two auxiliary feedwater pumps are assumed to be available. This is consistent with UFSAR Section 15.6.3.2.

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[8] Maximum allowable temperature to maximize flashing.

[9] RCS Pressure of 2100 psia selected to result in early trip, consistent with UFSAR Section 15.6.3.2, to maximize steaming through the MSSVs and ADVs during first two hours of the transient.

[10] Taken from CESSAR 15D, Table 15D-4.

Table 2
Results of PVNGS SGTR+LOP Analysis

Case Description		Offsite Doses, Rem GIS*	Offsite Doses, Rem PIS**
Case 1:	(0.6μCi/gm initial RCS activity. All leaking fluids released when tubes are uncovered)***.		
	0-2 Hr thyroid, exclusion area boundary	<30	200
	0-8 Hr thyroid, low population zone	12	38
Case 2	(0.4μCi/gm initial RCS activity. Only flashed portion of leak is released when tubes are uncovered).		
	0-2 Hr thyroid, exclusion area boundary	5	32
	0-8 Hr thyroid, low population zone	5	11
Case 3:	(1.0μCi/gm initial RCS activity. All leaking fluid is released when tubes are uncovered).		
	0-2 Hr thyroid, exclusion area boundary	40	200
	0-8 Hr thyroid, low population zone	16	38
Case 4:	(1.0μCi/gm initial RCS activity. Only flashed portion of leak is released when tubes are uncovered).		
	0-2 Hr thyroid, exclusion area boundary	13	32
	0-8 Hr thyroid, low population zone	11	11

R1

* Generated Iodine Spike. Spiking factor is 500.
** Pre-existing Iodine Spike, Initial Activity is 60μCi/gm
*** Accounts for increased subcooling margin requirements for upgraded EOP's and use of pressurizer venting in lieu of auxiliary spray for depressurization

R1