UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. 103 TO FACILITY OPERATING LICENSE NO. NPF-51,

AND AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NOS. 1, 2, AND 3

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

By application dated May 2, 1995, as supplemented by letter dated March 7, 1996, the Arizona Public Service Company (APS or the licensee) requested a modification to License Nos. NPF-41, NPF-51, and NPF-74, respectively for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3. APS submitted this request on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority. The proposed changes would modify the licenses to authorize revision of the Updated Final Safety Analysis Report (UFSAR) to incorporate certain changes. The changes describe a revised large-break loss of coolant accident (LOCA) analysis that addresses a previously unanalyzed release path through the steam generators to the atmosphere.

2.0 BACKGROUND

The Arizona Public Service Company, the licensee for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3, identified a potential release path that had not been previously evaluated for a design-basis large-break LOCA. The licensee calculated the radiological consequences of the large-break LOCA that included the potential release path and compared the results to previous large-break LOCA dose calculations that are contained in the UFSAR. The licensee determined that radioactivity contributions from the potential release path increased the radiological doses, but the total radiological doses from a large-break LOCA remained within the acceptance criteria presented in 10 CFR Part 100 for the exclusion area boundary and low population zone and General Design Criterion 19 for the control room.

In accordance with the provisions of 10 CFR 50.59, the licensee evaluated the changes in their large-break LOCA analysis and determined that it constituted an unreviewed safety question, which required prior NRC review and approval of

an amendment to the license. In letters dated May 2, 1995, and March 7, 1996, the licensee proposed changes to UFSAR Sections 6.2.4, 6.2.6, and 15.6.5, for the PVNGS Units 1, 2, and 3. The proposed changes are to the large-break LOCA dose consequence analysis and include a previously unanalyzed release path to the environment.

3.0 EVALUATION

3.1 Description of the Event

Following a large-break LOCA, steam generator pressure will remain high until pressure is relieved by the operators. The emergency operating procedures instruct the operators to depressurize the steam generators following the reflood stage of a large-break LOCA with the atmospheric dump valves or turbine bypass valves. When the operators perform this action, there is the potential that the steam generator pressure will reduce to a point lower than containment pressure while the steam generators are open to atmosphere and the steam generator tubes are uncovered. When the pressure in the containment is higher than the pressure in the steam generators, there could be flow of the containment atmosphere through the pre-existing cracks in the steam generator tubes into the steam generators. With the steam generators open to atmosphere, that flow could be available to be released into the atmosphere through the open atmospheric dump valves. This release path had not been analyzed previously by the licensee for the large-break LOCA. The licensee's revised calculations assume a single failure of an isolation valve (GDC 57 valve) or a stuck open atmospheric dump valve; therefore, no credit is taken for these valves for containment isolation. The staff agrees with the licensee's conclusion that these valves can continue to be excluded from 10 CFR 50 Appendix J Type C leakage rate testing.

3.2 Leakage Flow Calculation

The licensee has attempted to calculate the flow rate of the potential leakage through the steam generator after a design-basis large-break LOCA by assuming the maximum leak rate permitted by plant technical specifications (TSs) during normal operation and then predicting the maximum leakage that could be attained after the accident, when the steam generator secondary pressures are reduced below the containment pressure. The calculated flow rates were then used by the licensee to determine the radiological consequences of a large-break LOCA.

Because of the complexity of the calculations, the licensee frequently made conservative or bounding assumptions to ensure that the results are conservative. Thus, the results of the leakage flow calculations are not considered accurate or a best estimate of the true flow rate. The plant TSs allow a leak rate of 1 gpm of water (total) through the steam generator tubes. This flow rate occurs from any number of cracks in the steam generator tubes from the primary coolant system to the secondary main steam system. The steam generator tubes are water-filled in the primary system and water-covered in the secondary system during normal operation to create a water-to-water interface. The pressure differential during normal operation is about 1255

psi, whereas after a design-basis large-break LOCA, the maximum pressure differential is 60 psi, which corresponds to the design pressure of containment with the steam generators open to the atmosphere. Due to the pressure difference between containment and the depressurized steam generators following a large-break LOCA, the containment atmosphere or air flows across the leak paths in the steam generator tubes and into the steam generator, which is open to the atmosphere.

To extrapolate the containment atmospheric flow rate through the leaking steam generators, the licensee established a set of five calculations with five unknowns and solved the equations using a range of friction loss coefficients. First, the steam generator tube leak area was calculated as a function of the friction loss coefficient using the TS leakage rate with the normal operating pressure using the energy equations. The ideal gas law relationships were then used to calculate the flow of the containment atmosphere (a compressible fluid) through the leak path.

To simplify the equations to the point where there were only five unknowns, the licensee made some bounding assumptions. The friction loss coefficient was unknown, so the licensee plotted the loss coefficient versus the resulting containment atmosphere flow rate through the tubes and chose a limiting loss coefficient. For the flow rate used in the licensee's dose assessment, the corresponding loss coefficient was 1700. This value is high, and at that point on the plot, an increase in loss coefficient does not result in a significant increase in flow rate because the curve is essentially asymptotic. Although the loss coefficient for the crack is unknown, the value chosen is clearly conservative. To make the calculations simpler, the licensee chose to characterize the post-accident containment atmosphere flow as fully choked flow with sonic velocity at the exit. The equations were simplified because the Mach number at the exit equals one in this instance. Although the equations were simplified, it is not expected that the flow is fully choked; rather, it is expected that the velocity would be less than Mach one. However, assuming fully choked flow is conservative.

Conservative inputs to the calculations were also chosen. The specific heat for air was chosen to characterize the post-accident containment atmosphere. This was conservative because the use of steam or an air steam mixture, as would be expected in containment after an accident, would have yielded less limiting results. The TS leak rate was used for the normal operating primary to secondary leakage rather than the lower administrative leak rate. The containment pressure used for the calculations was the design limit rather than the peak calculated containment pressure. The maximum peak containment pressure calculated for a large-break LOCA is less than the design pressure of 60 psi. If the peak containment pressure was used, the results would have been lower.

In performing this calculation, the licensee used fundamental engineering principles and equations. The modeling assumptions were chosen conservatively and the inputs to these calculations were also conservatively chosen. As a result, the licensee's calculated flow rate (0.9 SCFM) should reasonably bound the true flow rate that could occur after a large-break LOCA and are

appropriate to use in assessing the offsite dose consequences. Although the staff did not independently recalculate the flow rates, the staff finds the approach acceptable for this application.

3.3 Radiological Consequences Analysis

Previous radiological consequences analysis of a large-break LOCA did not consider a release path from the primary system to the environment through a depressurized secondary system. The licensee subsequently revised the large-break LOCA dose calculation to include the assumption of a release path to the environment through the secondary system. The licensee calculated radiological doses and compared the results to previous large-break LOCA doses and the acceptance criteria presented in 10 CFR Part 100 for the exclusion area boundary and low population zone and General Design Criterion 19 for the control room. The licensee stated in its applications that calculated doses are below the acceptance criteria presented in 10 CFR 100 and General Design Criterion 19.

The staff has verified the licensee's dose calculations by performing an independent radiological consequence analysis that accounts for the increase in radioactivity released to the environment from a new release path through the depressurized secondary system. The staff calculated radiological doses to the thyroid, which are more limiting than whole body doses in terms of compliance with acceptance criteria, based on the results of previous staff LOCA analyses for PVNGS and information contained in the licensee's submittals. For the large-break LOCA dose calculation, the staff accounted for the increased release of radioactivity (0.9 SCFM) by assuming a fifty percent increase in the containment leak rate (1.8 SCFM) previously used in Supplement No. 5 of NUREG-0857, "Safety Evaluation Report Related to the Operation of Palo Verde Nuclear Generating Station, Units 1, 2, and 3," November 1983, and the staff's SE supporting Amendment Nos. 64, 50, and 37 to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74, respectively, dated September 8, 1992. The results of the large-break LOCA dose calculation indicate that a fifty percent increase in radioactivity release through a new pathway are within the acceptance criteria presented in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (SRP) Sections 6.4, 15.6.5, and Appendices A and B. The revised assumptions used to calculate the large-break LOCA doses are listed in Table 1 and the results are listed in Table 2.

3.4 Conclusion

The staff has reviewed the licensee's analyses and verified the licensee's dose calculations by performing an independent radiological consequence analysis. We conclude that the radiological consequences of a large-break LOCA with a new release pathway that increases radioactivity releases by fifty percent of the containment leak rate are within the acceptance criteria presented in SRP 6.4, 15.6.5, and Appendices A and B. The staff also concludes that the new release pathway is credible and that the licensee's calculated flow rates through the new release pathway are conservative and appropriate for assessing the radiological consequences of a large-break LOCA.

Therefore, the staff finds the licensee's revised large-break LOCA radiological consequence analysis acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arizona State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on March 14, 1997 (62 FR 12255).

Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of the amendments will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachments: 1. Table 1
2. Table 2

Principal Contributors: Anthony Huffert/PERB
Christopher Jackson/SRXB
James Pulsipher/SCSB

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TABLE 1

INPUT PARAMETERS FOR PALO VERDE UNITS 1, 2, AND 3 EVALUATION OF A LARGE BREAK LOSS-OF-COOLANT ACCIDENT

Power level, Mwt	3954
Fraction of core inventory available for leakage, %	
Iodines	25
Noble Gases	100
Initial iodine composition in containment, %	
Elemental	91
Organic	4
Particulate	5
Primary Containment volumes, ft^3	
Main sprayed	2.27×10^6
Auxiliary sprayed	0.20×10^6
Unsprayed	0.15×10^6
Primary containment leak rate, %/day	
0-24 hours after accident	0.15*
After 24 hours	0.075*
Containment spray iodine removal efficiencies, hr^{-1}	
Elemental (main sprayed region)	20
(auxiliary sprayed region)	10.3
Organic	0
Particulate (main sprayed region)	0.34
(auxiliary sprayed region)	0.11
Decontamination factor	
Elemental iodine	6.51
Particulate iodine	50
ECCS leak rate, cc/hr	1500**
Containment sump volume, ft^3	56,532

* includes 50% increased leak rate to account for new release path
** based on licensee's TMI Action Plan III.D.1.1 leakage reduction program

TABLE 1

INPUT PARAMETERS FOR PALO VERDE UNITS 1, 2, AND 3
EVALUATION OF A LARGE BREAK LOSS-OF-COOLANT ACCIDENT
(continued)

Atmospheric dispersion factors		sec/m^3
Exclusion area boundary	(0-2 hrs)	3.10×10^{-4}
Low population zone	(0-8 hrs)	5.10×10^{-5}
	(8-24 hrs)	3.80×10^{-5}
	(1-4 days)	2.00×10^{-5}
	(4-30 days)	8.30×10^{-6}
Control room	(0-8 hrs)	2.19×10^{-3}
	(8-24 hrs)	1.29×10^{-3}
	(1-4 days)	5.04×10^{-4}
	(4-30 days)	1.45×10^{-4}
Control room parameters		
Volume (ft^3)		161,000
Makeup flow (cfm)		1,000
Makeup and recirculation flow (cfm)		25,740
Makeup and recirculation filter efficiency (%)		
elemental, organic iodines		95
particulate iodine		99
Unfiltered inleakage (cfm)		10
Occupancy factor	(0-24 hrs)	1.0
	(1-4 days)	0.6
	(4-30 days)	0.4

TABLE 2

CALCULATED THYROID DOSES FOR PALO VERDE UNIT 1, 2, AND 3
LOSS-OF-COOLANT ACCIDENT

	DOSE (rem)		
LOCATION	Containment Leakage	ESF Leakage	Total
EAB	191.1	0.2	191.3*
LPZ	217.5	0.6	218.1*
Control Room	20.4	0.1	20.5**

*   NUREG-0800 Acceptance Criterion = 300 rem thyroid
**   NUREG-0800 Acceptance Criterion = 30 rem thyroid