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SUBJECT: Forwards revised Section 7 to Cycle 3 reload analysis rept for plant.

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WILLIAM F. CONWAY
EXECUTIVE VICE PRESIDENT
NUCLEAR

161-02219-WFC/KLMC
August 25, 1989

Docket No. STN 50-528

Document Control Desk
U. S. Nuclear Regulatory Commission
Mail Station P1-37
Washington, D. C. 20555

Reference: Letter to the NRC from D. B. Karner, ANPP, dated January 18, 1989; Subject: Reload Analysis Report for Unit 1, Cycle 3 (161-01620)

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 1
Revised Section 7 of Reload Analysis Report for Unit 1, Cycle 3
File: 89-E-056-026

Attached is a revision to Section 7 to the Unit 1, Cycle 3, Reload Analysis Report. The Unit 1, Cycle 3, Reload Analysis Report was transmitted by the Reference. The revised Section 7 pages should be incorporated into the Reload Analysis Report and the previously transmitted Section 7 pages discarded. The revision contains the reanalysis of the Inadvertant Opening of an Atmospheric Dump Valve with a concurrent Loss of Off-Site Power. The results of this re-analysis are bounded by the reference cycle.

Sincerely,

WFConway

WFC/KLMC/jle

Attachment

cc: J. B. Martin (all w/attachment)
T. L. Chan
T. J. Polich
A. C. Gehr
M. J. Davis

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7.0 NON-LOCA SAFETY ANALYSIS

7.0.1 Introduction

This section presents the results of the Palo Verde Nuclear Generating Station Unit 1 (PVNGS-1), Cycle 3 Non-LOCA safety analyses at 3800 MWt.

The Design Basis Events (DBEs) considered in the safety analyses are listed in Table 7.0-1. These events are categorized into three groups: Moderate Frequency, Infrequent, and Limiting Fault events. For the purpose of this report, the Moderate Frequency and Infrequent Events will be termed Anticipated Operational Occurrences. The DBEs were evaluated with respect to four criteria: Offsite Dose, Reactor Coolant System (RCS) Pressure, Fuel Performance (DNBR and Centerline Melt SAFDLs), and Loss of Shutdown Margin. Tables 7.0-2 through 7.0-5 present the lists of events analyzed for each criterion. All events were re-evaluated to assure that they meet their respective criteria for Cycle 3. The DBEs chosen for analysis for each criterion are the limiting events with respect to that criterion.

7.0.2 Methods of Analysis

The analytical methodology used for PVNGS-1 Cycle 3 is the same as the Cycle 2 (Reference Cycle) methodology (References 7-1, 7-2 and 7-9) unless otherwise stated in the event presentations. Only methodology that has previously been reviewed and approved on the PVNGS dockets (References 7-10 and 7-11), the CESSAR docket (Reference 7-2), or on other dockets is used.

7.0.3 Mathematical Models

The mathematical models and computer codes used in the Cycle 3 Non-LOCA safety analysis are the same as those used in the Reference Cycle analysis (References 7-1, 7-2 and 7-9). Plant response for Non-LOCA Events was simulated using the CESEC III computer code (Reference 7-3). Simulation of the fluid conditions within the hot channel of the reactor core and calculation of DNBR was performed using the CETOP-D computer code described in Reference 7-4.

The TORC computer code was used to simulate the fluid conditions within the reactor core and to calculate fuel pin DNBR for the RCP Shaft Seizure and Sheared Shaft event. The TORC code is described in References 7-6 and 7-7.

The number of fuel pins predicted to experience clad failure is taken as the number of pins which have a CE-1 DNBR value below 1.24. The only exceptions are the Shaft Seizure and Sheared Shaft events for which the statistical convolution method, described in Reference 7-8, was used. Reference 7-8 has been approved by the NRC and has been used in References 7-1, 7-2 and 7-9.

The HERMITE computer code (Reference 7-5) was used to simulate the reactor core for analyses which required more spatial detail than is provided by a point kinetics model. Reference 7-5 has been approved by the NRC and has been used in References 7-1, 7-2 and 7-9. HERMITE was also used to generate input to the CESEC point kinetics model by partially crediting space-time effects so that the CESEC calculation of core power during a reactor scram is conservative relative to HERMITE.

7.0.4 Input Parameters and Analysis Assumptions

Table 7.0-6 summarizes the core parameters assumed in the Cycle 3 transient analysis and compares them to the values used in the Reference Cycle. Specific initial conditions for each event are tabulated in the section of the report summarizing that event. Tech Spec changes are described in Section 10. The effects of these changes were considered for each DBE and were included as appropriate. For some of the DBEs presented, certain initial core parameters were assumed to be more limiting than the actual calculated Cycle 3 values. Such assumptions resulted in more adverse consequences. Events which have credited CPC trip protection have assumed instrument channel response times which are conservative relative to the Cycle 3 Technical Specifications.

7.0.5 Conclusion

All DBEs have been evaluated for PVNGS-1, Cycle 3 to determine whether their results are bounded by the Reference Cycle.

Table 7.0-1

PVNGS Unit 1 Design Basis
Events Considered in the Cycle 3 Safety Analysis

7.1 Increase in Heat Removal by the Secondary System

- 7.1.1 Decrease in Feedwater Temperature
- 7.1.2 Increase in Feedwater Flow
- 7.1.3 Increased Main Steam Flow
- 7.1.4 Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve
- 7.1.5* Steam System Piping Failures

7.2 Decrease in Heat Removal by the Secondary System

- 7.2.1 Loss of External Load
- 7.2.2 Turbine Trip
- 7.2.3 Loss of Condenser Vacuum
- 7.2.4 Loss of Normal AC Power
- 7.2.5 Loss of Normal Feedwater
- 7.2.6* Feedwater System Pipe Breaks

7.3 Decrease in Reactor Coolant Flowrate

- 7.3.1 Total Loss of Forced Reactor Coolant Flow
- 7.3.2* Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

7.4 Reactivity and Power Distribution Anomalies

- 7.4.1 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition
- 7.4.2 Uncontrolled CEA Withdrawal at Power
- 7.4.3 CEA Misoperation Events
- 7.4.4 CVCS Malfunction (Inadvertent Boron Dilution)
- 7.4.5 Startup of an Inactive Reactor Coolant System Pump
- 7.4.6* Control Element Assembly Ejection

7.5 Increase in Reactor Coolant System Inventory

- 7.5.1 CVCS Malfunction
- 7.5.2 Inadvertent Operation of the ECCS During Power Operation

* Categorized as Limiting Fault Events

Table 7.0-1 (continued)

7.6 Decrease in Reactor Coolant System Inventory

- 7.6.1 Pressurizer Pressure Decrease Events
- 7.6.2* Small Primary Line Break Outside Containment
- 7.6.3* Steam Generator Tube Rupture

7.7 Miscellaneous

- 7.7.1 Asymmetric Steam Generator Events

* Categorized as Limiting Fault Events

Table 7.0-2

DBEs Evaluated with Respect to Offsite Dose Criterion

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.1.4	1) Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Bounded by Reference Cycle
7.2.4	2) Loss of Normal AC Power	Bounded by Reference Cycle
	B) Limiting Fault Events	
	1) Steam System Piping Failures:	Bounded by Reference Cycle
7.1.5a	a) Pre-Trip Power Excursions	
7.1.5b	b) Post-Trip Return-to-Power	
7.2.6	2) Feedwater System Pipe Breaks	Bounded by Reference Cycle
7.3.2	3) Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	Bounded by Reference Cycle
7.4.6	4) Control Element Assembly Ejection	Bounded by Reference Cycle
7.6.2	5) Small Primary Line Break Outside Containment	Bounded by Reference Cycle
7.6.3	6) Steam Generator Tube Rupture	Bounded by Reference Cycle

Table 7.0-3

DBEs Evaluated with Respect to RCS Pressure Criterion

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.2.1	1) Loss of External Load	Bounded by Reference Cycle
7.2.2	2) Turbine Trip	Bounded by Reference Cycle
7.2.3	3) Loss of Condenser Vacuum	Bounded by Reference Cycle
7.2.4	4) Loss of Normal AC Power	Bounded by Reference Cycle
7.2.5	5) Loss of Normal Feedwater	Bounded by Reference Cycle
7.4.1	6) Uncontrolled CEA Withdrawal from Subcritical or Low Power Condition	Bounded by Reference Cycle
7.4.2	7) Uncontrolled CEA Withdrawal at Power	Bounded by Reference Cycle
7.5.1	8) CVCS Malfunction	Bounded by Reference Cycle
7.5.2	9) Inadvertent Operation of the ECCS During Power Operation	Bounded by Reference Cycle
	B) Limiting Fault Events	
7.2.6	1) Feedwater System Pipe Breaks	Bounded by Reference Cycle
7.4.6	2) Control Element Assembly Ejection	Bounded by Reference Cycle

Table 7.0-4

DBEs Evaluated with Respect to Fuel Performance

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.1.1	1) Decrease in Feedwater Temperature	Bounded by Reference Cycle
7.1.2	2) Increase in Feedwater flow	Bounded by Reference Cycle
7.1.3	3) Increased Main Steam Flow	Bounded by Reference Cycle
7.1.4	4) Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Bounded by Reference Cycle
7.3.1	5) Total Loss of Forced Reactor Coolant Flow	Bounded by Reference Cycle
7.4.1	6) Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition	Bounded by Reference Cycle
7.4.2	7) Uncontrolled CEA Withdrawal at Power	Bounded by Reference Cycle
7.4.3	8) CEA Misoperation Events	Bounded by Reference Cycle
7.6.1	9) Pressurizer Pressure Decrease Events	Bounded by Reference Cycle
7.7.1	10) Asymmetric Steam Generator Events	Bounded by Reference Cycle
	B) Limiting Fault Events	
	1) Steam System Piping Failures:	Bounded by Reference Cycle
7.1.5a	a) Pre-Trip Power Excursions	
7.1.5b	b) Post-Trip Return to Power	
7.3.2	2) Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	Bounded by Reference Cycle
7.4.6	3) Control Element Assembly Ejection	Bounded by Reference Cycle

Table 7.0-5

DBEs Evaluated with Respect to Shutdown Margin Criterion

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.1.4	1) Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Bounded by Reference Cycle
7.4.4	2) CVCS Malfunction (Inadvertent Boron Dilution)	Bounded by Reference Cycle
7.4.5	3) Startup of an Inactive Reactor Coolant System Pump	Bounded by Reference Cycle
	B) Limiting Fault Events	
	1) Steam System Piping Failures:	Bounded by Reference Cycle
7.1.5b	a) Post-Trip Return-to-Power	

Table 7.0-6

PVNGS Unit 1, Cycle 3
Core Parameters Input to Safety Analyses

Safety Parameters	Units	Reference Cycle Value	Cycle 3 Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	3898	3898
Core Inlet Steady State Temperature	^oF	560 to 570 (90% power and above) 550 to 572 (below 90% power)	560 to 570 (90% power and above) 550 to 572 (below 90% power)
Steady State RCS Pressure	psia	2000 - 2325	2000 - 2325
Minimum Guaranteed Delivered Volumetric Flow Rate	gpm	423,320	423,320
Axial Shape Index LCO Band Assumed for All Powers	ASI Units	-0.3 to +0.3	-0.3 to +0.3
Maximum CEA Insertion at Full Power	% Insertion of Lead Bank	28	28
	% Insertion of Part-Length	25	25
Maximum Initial Linear Heat Rate	KW/ft	13.5	13.5
Steady State Linear Heat Rate for Fuel Center Line Melt	KW/ft	21.0	21.0
CEA Drop Time from Removal of Power to Holding Coils to 90% Insertion	sec	4.0	4.0
Minimum DNBR			
CE-1 (SAFDL)		1.24	1.24
Macbeth (Fuel failure limit for post-trip SLB with LOAC - References 7-5 and 7-6)		1.30	1.30

Table 7.0-6 (continued)

Safety Parameters	Units	Reference Cycle Value	Cycle 3 Value
Moderator Temperature Coefficient	10^{-4} $\Delta\rho/^{o}F$	Figure 7.0-1	Figure 7.0-1
Shutdown Margin (Value Assumed in Limiting Hot Zero Power SLB)	$\%\Delta\rho$	-6.5	-6.5

7.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

7.1.1 Decrease in Feedwater Temperature

The results are bounded by the Reference Cycle.

7.1.2 Increase in Feedwater Flow

The results are bounded by the Reference Cycle.

7.1.3 Increased Main Steam Flow

The results are bounded by the Reference Cycle.

7.1.4 Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve

The results are bounded by the Reference Cycle.

7.1.5 Steam System Piping Failures

7.1.5a Steam System Piping Failures: Inside and Outside Containment Pre-Trip Power Excursions

The results are bounded by the Reference Cycle.

7.1.5b Steam System Piping Failures: Post-Trip Return to Power

The results are bounded by the Reference Cycle

7.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

7.2.1 Loss of External Load

The results are bounded by the Reference Cycle.

7.2.2 Turbine Trip

The results are bounded by the Reference Cycle.

7.2.3 Loss of Condenser Vacuum

The results are bounded by the Reference Cycle.

7.2.4 Loss of Normal AC Power

The results are bounded by the Reference Cycle.

7.2.5 Loss of Normal Feedwater

The results are bounded by the Reference Cycle.

7.2.6 Feedwater System Pipe Breaks

The results are bounded by the Reference Cycle.

7.3 DECREASE IN REACTOR COOLANT FLOWRATE

7.3.1 Loss of Forced Reactor Coolant

The results are bounded by the Reference Cycle.

7.3.2 Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

The results are bounded by the Reference Cycle.

7.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

7.4.1 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition

The results are bounded by the Reference Cycle.

7.4.2 Uncontrolled CEA Withdrawal at Power

The results are bounded by the Reference Cycle.

7.4.3 CEA Misoperation Event

The results are bounded by the Reference Cycle.

7.4.5 Startup of an Inactive Reactor Coolant Pump Event

The results are bounded by the Reference Cycle.

7.4.6 Control Element Assembly Ejection

The results are bounded by the Reference Cycle.

7.5 INCREASE IN REACTOR COOLANT SYSTEM INVENTORY

7.5.1 CVCS Malfunction

The results are bounded by the Reference Cycle.

7.5.2 Inadvertent Operation of the ECCS During Power Operation

The results are bounded by the Reference Cycle.

7.6 DECREASE IN REACTOR COOLANT SYSTEM INVENTORY

7.6.1 Pressurizer Pressure Decrease Events

The results are bounded by the Reference Cycle.

7.6.2 Small Primary Line Pipe Break Outside Containment

The results are bounded by the Reference Cycle.

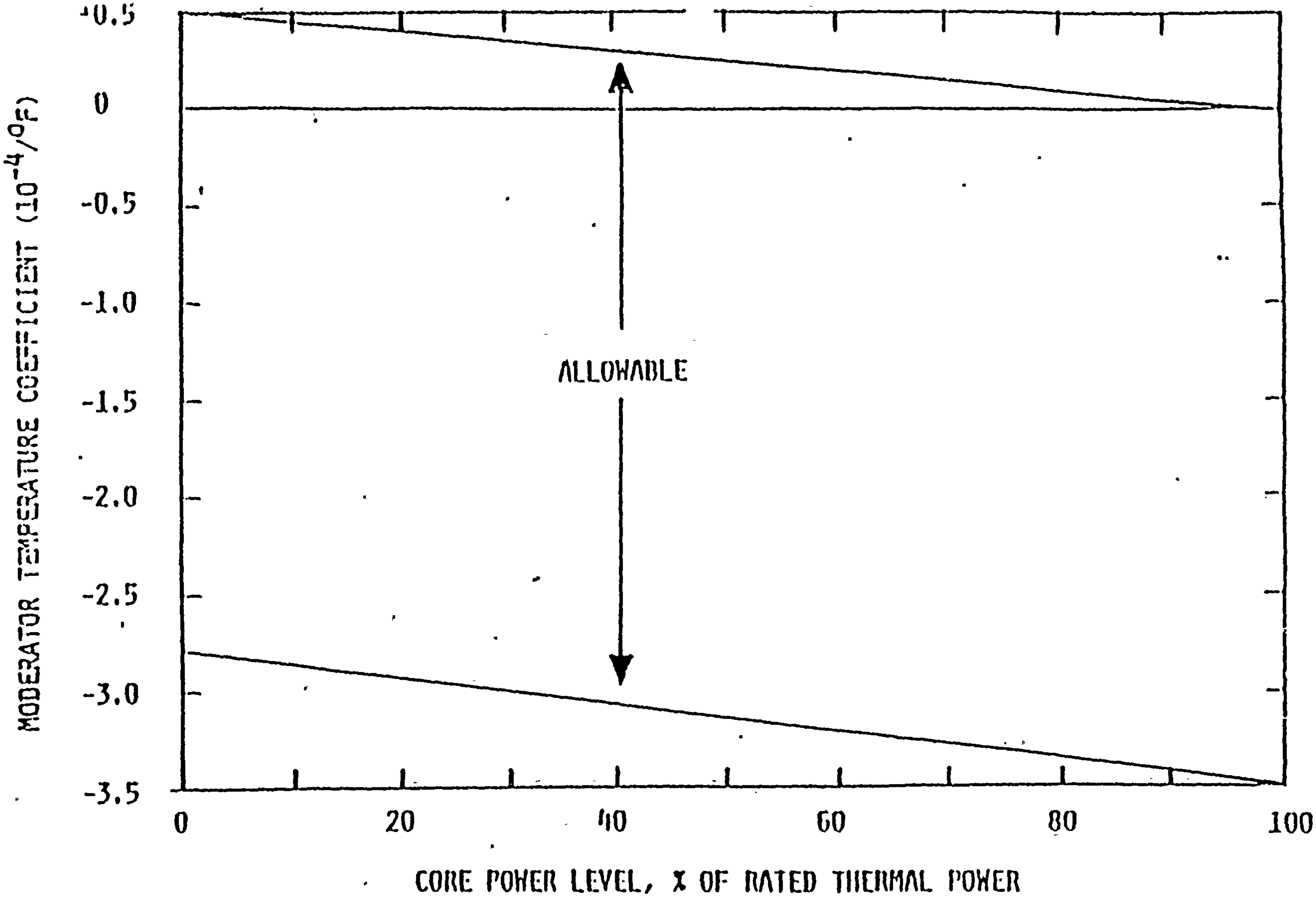
7.6.3 Steam Generator Tube Rupture

The results are bounded by the Reference Cycle.

7.7 MISCELLANEOUS

7.7.1 Asymmetric Steam Generator Events

The results are bounded by the Reference Cycle.



Palo Verde Nuclear Generating Station

ALLOWABLE MTC MODES 1 AND 2

Figure 7.0-1