14.1 INTRODUCTION

This chapter presents the results and major assumptions of the safety analysis under which the Palisades Plant is currently licensed. Additional detail for each analyzed event can be found in the references given at the end of each section. Also included for completeness is a list of references for other analyses that were done for Palisades for reasons not directly affecting the licensing basis (eg, Owner's Group work and steam generator plugged tube studies). Where important to the understanding of Plant safety, a summary of these analyses is provided.

14.1.1 BACKGROUND

The current licensed core thermal power level of the Palisades Plant is 2,565.4 MWt. The analyses described in this chapter use the previous licensed power level of 2,530 MWt as an input, with a 2% uncertainty. The uncertainty associated with 2565.4 MWt is 0.5925%. The licensed power level, adjusted for the associated uncertainty, results in the same value, 2580.6 MWt, for both cases.

On December 15, 1973, a major revision of the FSAR was made for a power uprating from 2,200 to 2,650 MWt. Extensive reanalyses were performed in support of the power increase. Most transients were either reanalyzed, or a determination was made that they were bounded by other events. The power increase was not approved by the NRC, but the analyses were bounding for the licensed power level (2,200 MWt), and were used as the reference analysis.

In 1976, the Palisades Plant was reloaded with Exxon[1] fuel. The limiting transients were reanalyzed using Exxon computer codes. The remaining events were not reanalyzed since the FSAR reference cycle analysis was still applicable and enveloped the Exxon analysis.

In July 1977, a topical report on Plant transient analysis of the Palisades reactor at a power uprating of 2,530 MWt was prepared by Exxon (Reference 1). The NRC approved the power increase to 2,530 MWt by license amendment No 31 dated November 1, 1977. The core was comprised of both Exxon and CE fuel elements.

[1] Palisades' original fuel vendor for cycle 1 was Combustion Engineering. Starting with cycle 2, Exxon Nuclear Corporation designed and manufactured all fuel for the reactor. Over the years, Exxon Nuclear has undergone the following company name changes: from Exxon Nuclear Corporation (ENC), to Advanced Nuclear Fuels (ANF) Corporation, to Siemens Nuclear Power (SNP), Siemens Power Corporation (SPC), Framatome ANP Richland, Inc to the present name Areva NP Inc.

A major revision of the Palisades Technical Specifications was approved by the NRC as Amendment 118 dated November 15, 1988. Advanced Nuclear Fuels Corporation performed analyses in support of Palisades operation with up to 29.3% steam generator tube plugging and a modified Reactor Protective System (Reference 2). The modified Reactor Protective System included a variable high-power trip and an improved thermal margin/low pressure trip with axial shape monitoring. Additional analyses were performed in Reference 3 supporting higher assembly peaking factors required for a revised fuel management scheme to reduce reactor vessel neutron flux.

Advanced Nuclear Fuels Corporation performed analyses (Reference 11) in support of Palisades Cycle 9 Operation. The analyses take into account changes that were made upon completion of Cycle 8. These include replacement of the Steam Generators, addition of High Thermal Performance (HTP) fuel assemblies, an increase in radial power peaking (F_r) to accommodate a low radial leakage core, and minor changes in assumed equipment set points and analysis uncertainties.

In support of the Steam Generator Replacement Project, ABB-Combustion Engineering reanalyzed the Steam Generator Tube Rupture (SGTR) event and the Main Steam Line Break (MSLB) containment analysis. The reanalysis was performed because of physical changes associated with the replacement Steam Generators, specifically the lower tube plugging levels, the thinner tube walls, and the steam outlet flow restrictor. For flexibility in performing future analysis, the MSLB containment analysis was subsequently reanalyzed using CPCo's in-house containment analysis code (see Section 14.18.2 for details).

Siemens Nuclear Power Corporation performed a Disposition of Events (Reference 13) to support a change to the Variable High Power Trip (VHPT) reset margin from 10% to 15% relative to the licensing basis for Palisades Cycle 9. A Disposition of Events (References 14 & 15) was also performed in support of Cycle 10 operation, and takes into account changes that were made upon completion of Cycle 9. These changes include insertion of a second full reload of HTP fuel assemblies, an increase in radial power peaking (F_r^A), and inclusion of eight Partial Shielding Assemblies (PSA) in low powered peripheral locations to reduce vessel flux.

In 1993, Siemens Power Corporation analyzed the Loss of Load Event to account for safety relief valve accumulation, which had not been previously accounted for, and to use a setpoint tolerance of 3% for both the primary and the secondary safety valves. For the Loss of Load analysis (Reference 18), an ANF-RELAP model was created and the ANF-RELAP code was used.

In support of Cycle 11 operation, a Disposition of Events was performed (References 16 & 17) that takes into account the insertion of the third full reload of HTP fuel assemblies, an increase in rod and assembly radial power peaking, and reinsertion of eight partial shielding assemblies and sixteen reconstituted reload L assemblies in low powered peripheral locations to reduce vessel fluence.

A Disposition of Events (References 22 and 26) was performed to support Cycle 12 operation. The Disposition of Events supported: 1) the use of the HTP correlation (Reference 21) for Palisades HTP fuel, 2) an Alternate Enrichment Scheme (AES) in the reload P fuel assemblies, 3) modified spacer side plates and guide bar retention slots for reload P assemblies, 4) 16 HTP assemblies with seven stainless steel rods replacing some corner fuel rods, and 5) natural axial blankets on the gadolinia rods.

The Large Break Loss of Coolant Accident (LOCA) was also analyzed (Reference 25) in order to support the Alternate Enrichment Scheme (AES). The AES causes the peak power in the assembly to decrease; however, the powers of the rods surrounding the peak rod are increased. This will cause a decrease in the radiative heat transfer from the peak rod, hence the analysis was reanalyzed.

A Disposition of Events (References 27 and 28) was performed for Cycle 13 to support: 1) modified enrichment pattern for Reload Q, 2) five axial zones in the gadolinia rods, 3) reduced U235 cutback (region of U235 only) in the gadolinia rods, 4) new safety injection pump curves, 5) new containment air coolers, and 6) an increase in the fuel pellet density. These were the major changes.

The Large Break LOCA (LBLOCA) and Small Break LOCA (SBLOCA) events were both reanalyzed due to new safety injection pump curves, new containment air coolers, and increased pellet density to name the major changes. The results from both analyses were acceptable.

A Disposition of Events (References 30 and 32) was performed for Cycle 14 to support major changes including a reduction in assumed PCS flowrates. Other events reanalyzed for Cycle 14 include Large Break LOCA (Reference 34), Small Break LOCA (see Section 14.17.2), Control Rod Ejection (Reference 33), and Main Steam Line Break (Reference 36). The results from all analyses were acceptable.

A Disposition of Events (Reference 7) was performed for Cycle 15 to consider the impact of changes in fuel design and plant operation, including the removal of the F_r^A peaking limit. Events reanalyzed for Cycle 15 include the Loss of Normal Feedwater Flow event, a Bank Drop event resulting from the replacement of one or more safety grade control rod clutch power supplies with non-qualified versions, and the Boron Dilution event. The results from all analyses were acceptable. During Cycle 15, Siemens performed a reanalysis for the large break LOCA event (Reference 37) using new SEM/PWR-98 methodology.

Framatome ANP, Richland, Inc., performed a Disposition of Events (Reference 38) for Cycle 16 to consider the impact of changes in fuel design and plant operation, including the removal of the F_r^A peaking limit. Events reanalyzed for Cycle 16 include the Control Rod Ejection, Loss of Electrical Load, and Boron Dilution events. The LBLOCA event was also reanalyzed for Cycle 16 (Reference 39). The results from all analyses were acceptable.

Framatome ANP performed a Disposition of Events (Reference 5) for Cycle 17 to consider the impact of changes in fuel design and plant operation. Events reanalyzed for Cycle 17 include the Main Steam Line Break (Reference 36) and the Loss of Normal Feedwater Flow (Reference 9) events. Framatome ANP also analyzed the Small Break LOCA (Reference 35) event using its new S-RELAP5 methodology. The results from all analyses were acceptable.

Framatome ANP performed a Disposition of Events (Reference 29) for Cycle 18 to consider the impact of changes in fuel design on plant operation. The Increase in Steam Flow event was evaluated during Cycle 17 to provide transient biases for the statistical setpoint analysis (Reference 29). The statistical setpoint methodology used for Cycle 18 is described in Reference 31.

The 2 wt% Gadolinia bearing rods used in Cycle 18 were evaluated for the Large Break LOCA event.

AREVA (formerly Framatome ANP) performed a Disposition of Events (Reference 47) for Cycle 19 to consider the impact of changes in fuel design on plant operation. The results from all analyses were acceptable.

AREVA performed a Disposition of Events (Reference 49) for Cycle 20 to consider the impact of changes in fuel design on plant operation. The results from all analyses were acceptable.

AREVA performed a Disposition of Events (Reference 50) for Cycle 21 to consider the impact of changes in the fuel design on plant operation. The LBLOCA transient was analyzed (Reference 51) using the Realistic LOCA methodology presented in Reference 52. The SBLOCA transient was analyzed (Reference 53) for the introduction of M5 cladding. The results from all analyses were acceptable.

AREVA performed a Disposition of Events (Reference 55) for Cycle 22 to consider the impact of changes in the fuel design on plant operation. The results from all analyses were acceptable.

AREVA performed a Disposition of Events for Cycle 23 (Reference 56) to consider the impact of changes in the fuel design on plant operation. The results from all analyses were acceptable.

AREVA performed a Disposition of Events (Reference 57) for Cycle 24 to consider the impact of changes in the fuel design on plant operation. The results from all analyses were acceptable.

AREVA performed a Disposition of Events (Reference 58) for Cycle 25 to consider the impact of changes in the reload design on plant operation. The results from all analyses were acceptable.

14.1.2 ANALYSES AT NOMINAL POWER LEVEL OF 2,650 MWT

No current analyses are performed at 2,650 MWt.

14.1.3 ANALYSES PERFORMED AT 2580.6 MWT INCLUDING UNCERTAINTY

The transient analyses for the Palisades Plant were generally performed using the Plant Transient Simulation model for Pressurized Water Reactors (PTSPWR2) (Reference 4). The PTSPWR2 code is a digital computer program developed to describe the behavior of pressurized water reactors subjected to abnormal operating conditions. The model is based on the solution of the basic transient conservation equations for the primary and secondary coolant system, on the transient conduction equation for the fuel rods, and on the point kinetics equation for the core neutronics. The program calculates fluid conditions such as flow, pressure, mass inventory and quality, heat flux in the core, reactor power, and reactivity during the transient. Various control and safety system components are included as necessary to analyze desired transients. The models contained within PTSPWR2 code are described in detail in Reference 4.

More recent transient analysis, including the Small Break LOCA, Main Steam Line Break, Loss of Normal Feedwater Flow, and Increase in Steam Flow have been performed using the S-RELAP5 code (Reference 10). Control Rod Ejection and Loss of Load analyses have been performed using the ANF-RELAP computer code (Reference 19). ANF-RELAP is a Siemens Power Corporation modification of the RELAP5/MOD2 computer code (Reference 20). S-RELAP5 is a Framatome ANP modification of the RELAP5/MOD2 computer code. The RELAP program calculates thermal-hydraulic transients with a complete two-fluid, two-velocity and two-temperature description. A set of six equations (2-mass, 2-momentum, 2-energy) describe the two fluid model. Two-velocity phenomena such as entrainment and slip are calculated by simultaneous solution of separate phasic mass and momentum equations. Interphase friction correlations are flow regime dependent, and there is no reliance on direct empirical correlations for slip velocity, flooding rate, or entrainment fraction.

Core parameters calculated using the PTSPWR2, ANF-RELAP or S-RELAP5 code are used as boundary conditions to the XCOBRA-IIIC thermal-hydraulic code (Reference 8) in order to evaluate the departure from nucleate boiling ratio (DNBR) for a given transient. Depending on the predicted system conditions, either the HTP correlation (Reference 21) or the modified Barnett Critical Heat Flux (CHF) correlation (Reference 6) is used to predict the minimum DNBR.

Per Technical Specifications SR 3.1.4.6, the time from receiving a reactor trip to the point where the control rods are 90% inserted shall be no greater than 2.5 seconds. The trip set points and their associated delay times to scram are given in Table 14.1-3. A rod scram curve used in the transient analysis is shown in Figure 14.1-3.

A summary Disposition of Events for Palisades is given in Table 14.1-4. This table lists each SRP Chapter 15 event, indicates whether the event is analyzed and provides a reference to the bounding event for events not analyzed. The FSAR section containing a summary of the analyzed events is also given in the table. A description of bounded events and the basis for selecting the bounding event is given in the current Disposition of Events Report. Therefore, bounded events are not described in this document.

The calculated primary system and core response for each of the Standard Review Plan Chapter 15 events that are not bounded by another event are presented in Table 14.1-5. For events that are not analyzed for primary system pressurization or MDNBR, the acceptance criteria that are applicable are presented in the table. Table 14.1-6 summarizes the offsite and control room radiological dose consequences and limits for the accidents evaluated in Chapter 14. It is shown that the radiological consequences for Palisades are below the limits specified in 10CFR100.11 (Reference 42), the Standard Review Plan (Reference 43), and General Design Criterion 19 (Reference 44), and/or Regulatory Guide 1.183 (Reference 54), as applicable.

REFERENCES

1. "Plant Transient Analysis of the Palisades Reactor at 2,530 MWt," XN-NF-77-18, Exxon Nuclear Company, July 1977, (0763/0007).

2. "Palisades Modified Reactor Protection System Report: Analysis of Chapter 15 Events," ANF-87-150(NP), Volume 2, Advanced Nuclear Fuels, Inc, June 1988, (C029/1767).

3. "Palisades Cycle 8: Disposition and Analysis of Standard Review Plan Chapter 15 Events," ANF-88-108, Revision 1, Advanced Nuclear Fuels, Inc, September 1988, (3781/1530).

4. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)," XN-NF-74-5(P)(A), Revision 2 and Supplements 1-6, Exxon Nuclear Company, October 1986, (8029/2447), (8029/2171), (8029/2390), (8029/2648), (8029/2515), (8029/2785), and (8029/2633).

5. EMF-2838, Revision 1, Palisades Cycle 17 Safety Analysis Report, Framatome ANP, April 2003.

6. "A Correlation of Rod Bundle Heat Flux for Water in the Pressure Range of 150 to 725 Psia," IN-1412, Idaho Nuclear Corporation, July 1970.

7. EMF-2259, Revision 2, "Palisades Cycle 15 Safety Analysis Report," Siemens Power Corporation, September 2000.

8. "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-NF-75-21(P)(A), Revision 2, Exxon Nuclear Company, January 1986, (2741/1496).

9. EMF-2845, Revision 0, Palisades Loss of Normal Feedwater Analysis, Framatome ANP, January 2003.

10. EMF-2310(P)(A), Revision 0, SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors, Framatome ANP, May 2001.

11. "Palisades Cycle 9: Analysis of Standard Review Plan Chapter 15 Events," ANF-90-078, Advanced Nuclear Fuels, Inc, September 1990, (C677/1283).

12. Deleted

13. "Review and Analysis of SRP Chapter 15 Events for Palisades with 15% Variable High Power Trip Reset," ANF-90-181, Advanced Nuclear Fuels Corporation, November 1990, (C679/2019).

14. "Palisades Cycle 10: Disposition and Analysis of Standard Review Plan Chapter 15 Events," EMF-91-176, Siemens Nuclear Power Corporation, October 1991, (F288/1248).

15. EA-GCP-91-03, Rev 0, "Palisades Cycle 10 Disposition of Events," October 1991, (F029/1528).

16. "Palisades Cycle 11: Disposition and Analysis of Standard Review Plan Chapter 15 Events," EMF-92-178 Revision 3, Siemens Power Corporation, November 1993, (G262/0189).

17. EA-FC-934-01 Rev 2, "Palisades Cycle 11 Disposition of Events Review," November 1993, (F509/0399) and (F586/2069).

18. "Palisades Loss of Load Analysis," EMF-93-086(P), Siemens Power Corporation, April 1993, (F853/0165).

19. "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," ANF-89-151(P)(A), Advanced Nuclear Fuels Corporation, May 1992, (F859/2215).

20. "RELAP5/MOD2 Code Manual," NUREG/CR-4312 Rev 2, Volumes 1 and 2, V.H. Ransom et al, EG&G Idaho, Inc., March 1987.

21. Siemens Power Corporation, "HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel," EMF-92-153 (P)(A) Revision 1, January 2005.

22. Siemens Power Corporation, "Palisades Cycle 12 Disposition and Analysis of Standard Review Plan Chapter 15 Events," April 1995, EMF-95-022, (G846/2018).

23. Deleted

24. Deleted

25. Siemens Power Corporation, "Palisades Large Break LOCA/ECCS Analysis With Increased Radial Peaking and Reduced ECCS Flow," EMF-91-177, October 1991; and Supplement 1, "Palisades Large Break LOCA/ECCS/Analysis With Increased Radial Peaking and Reduced ECCS Flow: Supplement-Alternate Enrichment Scheme," June 1994, (F288/1359) and (G162/0492).

26. EA-TAM-95-02, "Palisades Cycle 12 Disposition of Events Review", July 1995, (G352/0465).

27. Siemens Power Corporation, “Palisades Cycle 13: Disposition and Analysis of Standard Review Plan Chapter 15 Events,” EMF-96-140, November 1996, (4023/0102).

28. EA-TAM-96-06, “Palisades Cycle 13 Disposition of Events Review,” December 1996, (H029/2150).

29. EMF-3097 Revision 0, Palisades Cycle 18 Safety Analysis Report, June 2004.

30. EMF-98-013, Revision 2, "Palisades Cycle 14: Disposition and Analysis of Standard Review Plan Chapter 15 Events," Siemens Power Corporation, January 1999.

31. EMF-1961(P)(A), Revision 0, Siemens Power Corporation, July 2000, “Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors.”

32. EA-GEJ-98-01, Revision 1, "Palisades Cycle 14: Disposition of Events Review," June 1998, (4297/1887).

33. EMF-98-021, Revision 0, "Palisades Control Rod Ejection Analysis," Siemens Power Corporation, April 1998.

34. EMF-98-026, Revision 1, "Palisades Large Break LOCA/ECCS Analysis," Siemens Power Corporation, April 1998.

35. EMF-2889, Revision 0, Palisades Small Break LOCA Analysis, Framatome ANP, January 2003.

36. EMF-2853, Revision 0, Palisades Steam Line Break Analysis, Framatome ANP, January 2003.

37. EMF-2369, Revision 0, “Palisades Nuclear Plant Large Break LOCA/ECCS Analysis Using SEM/PWR-98,” August 2000.

38. EMF-2507, Revision 0, “Palisades Cycle 16 Safety Analysis Report,” Framatome ANP Richland, Inc., March 2001.

39. EMF-2545, Revision 0, “Palisades Large Break LOCA/ECCS Analysis,” Framatome ANP Richland, Inc., March 2001.

40. Deleted

41. EA-TAM-96-02, Revision 3, “Palisades Control Room Habitability Following FSAR Chapter 14 Accidents with Radiological Consequences.”

42. Code of Federal Regulations, Volume 10, Part 100 Code of Federal Regulations, Volume 10 Part 100 Section 11, "Reactor Site Criteria" Section 11 "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

43. NUREG-0800, U.S. NRC, "Standard Review Plan," Revision 2, dated July 1981.

44. Code of Federal Regulations, Volume 10 Part 50 Appendix A, General Design Criterion 19 (GDC 19), "Control Room."

45. Standard Review Plan Chapter 6, Section 4, "Control Room Habitability System."

46. Letter from D.L. Ziemann (NRC) to D.P. Hoffman (CPCo) dated February 5, 1980, Subject: XV-20 - Radiological Consequences of Fuel Damaging Accidents (Inside and Outside Containment), (2614/0667).

47. ANP-2507-001, Palisades Cycle 19 Safety Analysis Report, April 2006.

48. IN-1412, "A Correlation of Rod Bundle Critical Heat Flux for Water in Pressure Range 150 to 725 psia," Idaho Nuclear Corporation, July 1970.

49. ANP-2633-001, Palisades Cycle 20 Safety Analysis Report, October 2007.

50. ANP-2777-000, Palisades Cycle 21 Safety Analysis Report, December 2008.

51. ANP-2750(NP), Revision 1, Palisades Nuclear Plant Realistic Large Break LOCA Summary Report, December 2008.

52. EMF-2103(P)(A), Revision 0, Realistic Large Break LOCA Methodology for Pressurized Water Reactors, Framatome ANP, Inc., April 2003.

53. ANP-2712, Revision 0, Palisades Small Break LOCA Analysis with M5 Cladding, AREVA NP, Inc., November 2008.

54. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, July 2000.

55. ANP-2929, Revision 2, Palisades Cycle 22 Safety Analysis Report, October 2010.

56. ANP-3074-001, Palisades Cycle 23 Safety Analysis Report, February 2012.

57. ANP-3218-001, Palisades Cycle 24 Safety Analysis Report, November 2013.

58. ANP-3399-00, Palisades Cycle 25 Safety Analysis Report, June 2015.

14.2 UNCONTROLLED CONTROL ROD WITHDRAWAL

14.2.1 UNCONTROLLED CONTROL ROD BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER START-UP CONDITION

14.2.1.1 Event Description

This event is initiated by the uncontrolled withdrawal of a control rod bank, which results in the insertion of positive reactivity and consequently a power excursion. It could be caused by a malfunction in the rod control systems. The consequences of a single-bank withdrawal in Modes 2 and 3 are considered in this event category. The consequences at rated power and initial operating conditions are considered in Section 14.2.2.

The response of the reactor core to the withdrawal of control banks from subcritical or low power exhibits an apparent delay in responding to the reactivity insertion, followed by an extremely rapid power ramp. This behavior is based on the time constants of the delayed neutrons and the relatively long time that power takes to respond to the reactivity insertion. When the reactivity insertion is rapid enough that the delayed neutron population cannot bring the power up to significant levels and cause a trip before the reactor approaches prompt critical, a very rapid power excursion can result. The power excursion is first mitigated by the Doppler feedback as the fuel temperature begins to rise and ultimately terminated by the RPS. Because the power increase in this event is very rapid, fuel rod surface heat flux lags behind the neutron power and the primary coolant temperature lags behind the fuel rod surface heat flux.

The power transient (as well as the control rod withdrawal) is either prohibited by procedural control or is eventually terminated by the reactor protection system:

1. Nonsafety grade high rate of change of power trip, .0001% to 15% power (no credit taken);

2. Variable overpower trip;

3. Thermal margin/low pressure trip;

4. High pressurizer pressure trip; or

5. High rate of change of power alarms, which initiate Rod Withdrawal Prohibit Action (no credit taken).

6. Zero Power Mode Bypass Reset

7. If the reactor is in hot shutdown or below and one PCS pump has been out of service for more than 12 hours, circuit breakers 42-01 and 42-02 are open (Reference 13) which prevents a rod or bank withdrawal.

Of these trips, the minimum power setting of the variable overpower trip is the only RPS function relied on in Mode 2 to terminate this event. Further protection is provided by the Doppler reactivity feedback in the fuel and by available DNBR margin between the initial operating condition and the DNB thermal limit. In Mode 3, with fewer than four primary coolant pumps in operation, the Zero Power Mode Bypass Reset, coupled with the Low PCS Flow reaction trip, is relied on to terminate the event.

14.2.1.2 Thermal-Hydraulic Analysis

14.2.1.2.1 Analysis Method

The analysis is performed using the PTSPWR2 (Reference 1) and XCOBRA-IIIC (Reference 2) codes. The PTSPWR2 code models the salient system components and calculates reactor power, fuel thermal response, surface heat transport and fluid conditions, including coolant flow rate, temperature and primary pressure. The core boundary conditions are then input into XCOBRA-IIIC to obtain the MDNBR.

14.2.1.2.2 Bounding Event Input

This event is analyzed with four primary coolant pumps operating. The case input and initial conditions bound startup (Mode 2) and hot standby (Mode 3) operating conditions. The lowest initial power yields the maximum margin to trip, and hence maximum time for withdrawal to trip. This yields the largest prompt multiplication and maximizes overshoot past trip. The power used conservatively bounds the possible initial power in startup (Mode 2) and hot standby (Mode 3) operation. Maximum coolant temperature, maximum radial peaking and minimum core flow rate are chosen to minimize DNBR. The biases for core age and the pellet-to-cladding heat transfer coefficient are selected to minimize Doppler feedback.

14.2.1.2.3 Analysis of Results

The event is initiated by a control bank withdrawal which results in the insertion of positive reactivity and consequentially a power excursion. The rapid power increase results in a fuel temperature increase and negative Doppler reactivity feedback which limits the peak power.

The event is terminated when the reactor trips on variable high power. The calculated values for peak pressurizer pressure, MDNBR, and peak LHR are shown in Table 14.1-5.

The responses of key system parameters are plotted in Figures 14.2.1-1 through 14.2.1-6. The sequence of events is given in Table 14.2.1-1. Based on the analyses in Reference 12, this event was found to be non-limiting relative to other anticipated operational occurrences.

14.2.1.3 Radiological Consequences

A radiological consequences analysis is not applicable for this event.

14.2.1.4 Conclusions

The results of the analysis demonstrates that:

- The predicted MDNBR (Table 14.1-5) is greater than the HTP correlation limit presented in Table 14.1-1. This ensures that, with 95% probability and 95% confidence, DNB is not expected to occur and no fuel failures are expected.

- The predicted peak linear heat rate (Table 14.1-5) is less than fuel centerline melt criterion presented in Table 14.1-1.

Therefore, all the applicable acceptance criteria for the event are met.

14.2.2 UNCONTROLLED CONTROL ROD BANK WITHDRAWAL AT POWER

14.2.2.1 Event Description

As with Event 14.2.1, this event is initiated by an uncontrolled withdrawal of a control rod bank. This withdrawal adds positive reactivity to the core which leads to potential power and temperature excursions. This event considers the consequences of control bank withdrawals at rated and operating initial power levels.

The reactor protection trip system is designed and set to preclude penetration of the Specified Acceptable Fuel Design Limits (SAFDLs). Because of the design of this analysis, the thermal margin/low pressure and variable overpower trips are principally challenged.

The thermal margin/low pressure trip function is designed and set to protect against DNB. Principal DNB parameters such as power (the highest auctioned value of either calorimetric or neutronic power), core inlet temperature, and core power distribution are measured. This function is based on the core protection boundaries. Operation within these boundaries assures protection of the SAFDLs.

A broad range of reactivity insertion rates and initial operating conditions are possible. The range of reactivity insertion is from very slow, as would be associated with a gradual boron dilution, and bounded on the fast end of the range by bank withdrawal.

The objective of the analysis is to demonstrate the adequacy of the trip set points to assure meeting the acceptance criteria. To assure this objective, the analysis is performed for a spectrum of reactivity insertion rates and initial power levels. Since neutronic feedback as a function of cycle exposure and design also influences results, these effects are also included in the analysis.

14.2.2.2 Thermal-Hydraulic Analysis

14.2.2.2.1 Analysis Method

The analysis is described in Reference 3 and uses the PTSPWR2 (Reference 1) and XCOBRA-IIIC (Reference 2) codes. The PTSPWR2 code models the salient system components and calculates neutron power, fuel thermal response, and fluid conditions. The fluid conditions and rod surface heat transport at the time of MDNBR are input to the XCOBRA-IIIC code for calculation of the MDNBR. Systems which minimize DNBR are enabled in the analysis.

The sequence of events is generally the same throughout the event spectrum, differing only in which trip is challenged, ie,

1. Reactivity is inserted.
2. Nuclear power increases
3. Thermal power increases
4. Primary temperature increases
5. Reactor trips on thermal margin/low pressure or variable overpower. No engineered safeguard features are challenged.

14.2.2.2.2 Bounding Event Input

The analysis evaluates the consequences of uncontrolled control rod bank withdrawal from 91.5% of rated power. A spectrum of reactivity insertion rates were evaluated in order to bound events ranging from a slow dilution of the primary system boron concentration to the fastest allowed control bank withdrawals. Specifically, the analysis encompasses reactivity insertion rates from 1.0×10^{-6} to $5.0 \times 10^{-5} \Delta\rho$/sec. Middle of cycle (MOC) kinetics are bounded in the analysis by considering conservatively bounding beginning of cycle (BOC) and end of cycle (EOC) kinetics, along with a comprehensive range of reactivity insertion rates. The range of insertion rates was conservatively calculated based on control rod worth and withdrawal speed.

14.2.2.2.3 Analysis of Results

The Uncontrolled Bank Withdrawal transient was analyzed from 91.5% of rated power. The limiting uncontrolled control rod bank withdrawal at 91.5% power and BOC kinetics occurred at an insertion rate of 2.25×10^{-5} $\Delta\rho$/sec.

For the limiting withdrawal rate, the TM/LP trip provided the DNB protection. The VHP trip was not reached. The VHP maximum trip setpoint of 106.5% set the starting power of 91.5% for the limiting event. This starting power is more limiting than the 100% power since at this power (essentially 90% of rated), the axial power shape allowed by the T_{inlet} LCO is significantly more limiting with regard to DNB than the axial power shape allowed for 100% power and the radial peaking associated with 91.5% power is higher.

In the cases where the TM/LP intervenes the reduced peaking will cause the event to be less limiting. For withdrawal rates near the limiting rate, including the withdrawal rates in which the VHP intervenes, the MDNBR occurs prior to the trip and would not be impacted by the increased VHP trip power.

The sequence of events for the Uncontrolled Bank Withdrawal transient is given in Table 14.2.2-1 from Reference 9. The transient response of key system variables is given in Figures 14.2.2-1 through 14.2.2-9.

The MDNBR was evaluated for the insertion rate that yielded the lowest MDNBR. The most limiting MDNBR corresponded to a reactivity insertion rate of 2.25×10^{-5} $\Delta\rho$/second initiated from 91.5% power. The maximum peak pellet LHR occurs in the case which uses BOC kinetics. The bounding MDNBR and peak LHR for this event for the current reload cycle are given in Table 14.1-5.

14.2.2.3 Radiological Consequences

A radiological consequences analysis is not applicable for this event.

14.2.2.4 Conclusions

The results of the analysis demonstrates that:

- The predicted MDNBR (Table 14.1-5) is greater than the HTP correlation limit presented in Table 14.1-1. This ensures that, with 95% probability and 95% confidence, DNB is not expected to occur and no fuel failures are expected.

- The predicted peak linear heat rate (Table 14.1-5) is less than fuel centerline melt criterion presented in Table 14.1-1.

Therefore, all the applicable acceptance criteria for the event are met.

14.2.3 SINGLE CONTROL ROD WITHDRAWAL

14.2.3.1 Event Description

The rod withdrawal event is initiated by an electrical or mechanical failure in the Rod Control System that causes the inadvertent withdrawal of a single control rod. A rod is withdrawn from the reactor core causing an insertion of positive reactivity which results in a power excursion transient. The movement of a single rod out of sequence from the rest of the bank results in a local increase in the radial power-peaking factor.

The combination of these two factors results in a challenge to DNB margin. The system response is essentially the same as that occurring in the Uncontrolled Bank Withdrawal event at power (Event 14.2.2).

Acceptable outcomes for this event rely only on the Reactor Protective System or on the Technical Specifications limiting the conditions of operation.

14.2.3.2 Thermal-Hydraulics Analysis

14.2.3.2.1 Analysis Method

In this event the radial redistribution of power in the core can result in radial peaking factors in excess of those allowed by the Core Operating Limits Report (COLR). The analyses are performed by coupling a conservative power peak to transient response and DNB calculations. The power peak associated with the event is characterized through an augmentation factor which relates the maximum power peak to the steady-state power peak. The steady-state power distributions and augmentation factors are calculated with the PRISM (Reference 5) reactor simulator. The conservatively biased core boundary conditions are then combined in an XCOBRA-IIIC (Reference 2) calculation with a radial augmentation peaking factor calculated to bound the possible single-rod withdrawal radial power redistribution. The analysis is described in Reference 3.

14.2.3.2.2 Bounding Event Input

Increased radial peaking factors impact DNBR for this event. Radial peaking augmentation factors for the single-control rod withdrawal events are calculated at full power for different exposure conditions. The core boundary conditions of average heat flux, temperature, pressure and flow are selected to conservatively bound the consequences of this event at 91.5% of rated power. The bank withdrawal analysis (Event 14.2.2) considers reactivity insertion rates down to 1.0×10^{-6} $\Delta\rho$/s which is representative of a single rod. The boundary conditions used in the calculation of MDNBR are obtained from the limiting transient response from the Uncontrolled Bank Withdrawal at Power (Event 14.2.2). Those conservatively biased core boundary conditions are then combined in an XCOBRA-IIIC calculation with a radial augmentation peaking factor calculated to bound the possible single rod withdrawal radial power redistribution.

14.2.3.2.3 Analysis of Results

The transient response for the Single Control Rod Withdrawal Event is the same as that for the Uncontrolled Control Rod Bank Withdrawal of Power Event (Event 14.2.2).

14.2.3.3 Radiological Consequences

A radiological consequences is not applicable for this event.

14.2.3.4 Conclusions

The results of the analysis demonstrates that:

- The predicted MDNBR (Table 14.1-5) is greater than the HTP correlation limit presented in Table 14.1-1. This ensures that, with 95% probability and 95% confidence, DNB is not expected to occur and no fuel failures are expected.
- The predicted peak linear heat rate (Table 14.1-5) is less than fuel centerline melt criterion presented in Table 14.1-1.

Therefore, all the applicable acceptance criteria for the event are met.

REFERENCES

1. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," XN-NF-74-5(P)(A), Revision 2 and Supplements 1-6, Exxon Nuclear Company, October 1986.

2. "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-NF-75-21(P)(A), Revision 2, Exxon Nuclear Company, January 1986.

3. ANF-84-73, Revision 5, Appendix B(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Siemens Power Corporation, July 1990.

4. Deleted

5. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWR's Volume 1 - Methodology Description, Volume 2 - Benchmarking Results," Siemens Power Corporation, January 1997.

6. Deleted

7. Deleted

8. Deleted

9. ANF-90-181, "Review and Analysis of SRP Chapter 15 Events for Palisades with a 15% Variable High Power Trip Reset," Siemens Power Corporation, November 1990.

10. Deleted

11. Deleted

12. Siemens Power Corporation, "Palisades Cycle13: Disposition and Analysis of Standard Review Plan Chapter 15 Events," EMF-96-140, November 1996.

13. General Operating Procedure GOP-8, "Power Reduction and Plant Shutdown to Mode 2 or Mode 3 ≥525°F."

14.3 BORON DILUTION

14.3.1 EVENT DESCRIPTION

The Chemical and Volume Control System regulates both the chemistry and the quantity of coolant in the Primary Coolant System (PCS). Changes in boron concentration in the PCS are a part of normal Plant operation, compensating for long-term reactivity effects such as fuel burnup, xenon transients and Plant cooldown.

Boron dilution is a manual operation, conducted under strict administrative control and in accordance with detailed operating procedures which specify permissible limits on rate and magnitude of any increment of boron dilution. Because of the procedures involved and the alarms and indications provided, the probability of a sustained erroneous dilution is very small. Administrative procedures will protect against protracted operator neglect to add boron to compensate for reactivity change induced by post-shutdown cooldown or xenon decay.

The operation of the primary makeup water transfer pumps provides the normal supply of makeup water to the PCS via charging pumps. Inadvertent dilution can be readily terminated by isolating the unborated water source.

During normal operation, concentrated boric acid solution is blended with primary makeup water to the approximate concentration present in the reactor coolant and is introduced into the volume control tank discharge header. This process is a manual operation. A malfunction in this system (such as failure of the boric acid pumps to start or of the boric acid control valve to open) while the operator fails to observe the indicator for correct flow, could initiate a boron dilution incident.

Boron concentration in the PCS can be decreased by controlled feed and bleed operation or by using the deborating demineralizer. (The deborating demineralizer can be used for removal of boron when the primary coolant boron concentration is below 50 ppm.)

To add primary makeup water for a planned boron dilution, the makeup controller mode selector switch is set to “A” (Automatic) and the primary makeup water batch quantity selector is set to the desired quantity. The makeup stop valve is then opened and the controller START button is pushed to initiate flow. When the specified amount has been injected, the primary makeup water control valve is closed automatically. Failure of the valve to close could, on the occasion of very low pressurizer level, result in the introduction of unborated water at the maximum capacity of all three charging pumps (143 gpm), if three pumps are available. The analysis assumes a bounding charging pump flowrate of 143 gpm.

14.3.2 THERMAL-HYDRAULIC ANALYSIS

14.3.2.1 Analysis Method

A summary of the two mathematical approaches used to model the dilution of the PCS is presented in Section 5.7.6 of Reference 1. The uniform mixing model is used for most plant operating conditions and assumes that perfect mixing occurs. The "wave front/slug" flow model is used only when the primary coolant pumps are not in operation and assumes imperfect mixing.

Four different plant conditions are evaluated for the Boron Dilution event to ensure all phases of Plant operation are bounded. Those conditions are:

1. Refueling (Mode 6)

2. Prior to Startup (Modes 3 - 5)

3. Power Operation and Startup (Modes 1 and 2)

4. Failure to Add Boron After Shutdown

The dilution of primary system boron adds positive reactivity to the core and can lead to the erosion of shutdown margin for a subcritical initial condition, or a slow power excursion during power operation.

In the event of a boron dilution transient in Modes 1 - 5, some of the following indications and alarm functions are available to alert the operator:

1. Volume control tank level indication and the high and low level alarms;

2. Letdown diverter valve indication;

3. Charging flow indication;

4. Wide range logarithmic nuclear instrumentation.

Following detection of the event, operator action must be taken to terminate the dilution and to restore the required shutdown margin. The boron dilution analysis must demonstrate that the shutdown margin required by Technical Specifications is sufficient to allow at least 15 minutes for the operator to both recognize and terminate a dilution event from all modes of operation except for Refueling (Mode 6). The time allowed while in Refueling (Mode 6) is 30 minutes.

14.3.2.2 Bounding Event Input

The key parameters affecting the boron dilution time-to-criticality are:

1. The volume of the PCS coolant
2. The PCS charging flowrate
3. The PCS charging boron concentration
4. The PCS boron concentration at event initiation versus operating mode
5. The PCS critical boron concentration versus operating mode

14.3.2.3 Analysis of Results

14.3.2.3.1 Dilution During Refueling (Mode 6)

For dilution to occur during refueling by primary makeup water, it is necessary to have at least one makeup water transfer pump operating, one charging pump operating, and the Primary Make-up Water Make-up Control Valve open. None of these conditions are required for refueling and they would be in violation of operating procedures. Nevertheless, such a dilution incident has been analyzed. The following cases were analyzed:

A. Refueling - Case 1

The following plant configuration was analyzed:

1. No Primary Coolant Pumps (PCPs) in operation when the boron dilution event occurs.
2. The Shutdown Cooling (SDC) system is in operation with a minimum flow rate of 1000 gpm.
3. All three charging pumps are assumed to be in operation (143 gpm).
4. The PCS is only partially filled (3300 ft^3).
5. The Shutdown Margin (SDM) requirement is ≥ 5% assuming all rods inserted (ARI) minus the most reactive stuck rod (MRR).

Based on the plant configuration detailed above, the time-to criticality for this case was shown to meet applicable acceptance criteria.

B. Refueling - Case 2

The following plant configuration was analyzed:

1. No PCPs in operation when the boron dilution event occurs.

2. The SDC system is in operation with a minimum flow rate of 1000 gpm.

3. All three charging pumps are assumed to be in operation (143 gpm).

4. A total PCS volume of 7921 ft^3 was assumed for the analysis. This volume does not credit the volumes of the pressurizer, pressurizer surge line, upper vessel head, and downcomer to upper head bypass flow pass. A total steam generator tube plugging of 15% is also assumed.

5. The SDM requirement is $\geq$ 5% assuming all rods out (ARO).

Based on the plant configuration detailed above, the time-to criticality for this case was shown to meet applicable acceptance criteria.

14.3.2.3.2 Dilution During Cold Shutdown (Mode 5)

In Mode 5, the Shutdown Cooling System is in operation. Primary coolant pumps may or may not be in operation. If one or more coolant pumps are in operation, it is assumed that the dilution flow undergoes perfect mixing with the primary coolant. If no coolant pumps are in operation, the dilution flow does not perfectly mix with the primary coolant and forms a moving front of diluted water that flows toward the core. This scenario is modeled with a dilution front model.

The following cases were analyzed:

A. Cold Shutdown - Case 1

The following plant configuration was analyzed:

1. No PCPs are in operation when boron dilution event occurs.

2. All three charging pumps are assumed to be in operation (143 gpm).

3. A total PCS volume of 7921 ft^3 was assumed for the analysis. This volume does not credit the volumes of the pressurizer, pressurizer surge line, upper vessel head, and downcomer to upper head bypass flow pass. A total steam generator tube plugging of 15% is also assumed.

4. The SDC system is in operation with a minimum flow rate of 2810 gpm.

5. The SDM requirement is $\geq$ 2% assuming ARI-MRR.

Based on the plant configuration detailed above, the time-to-criticality for this case was shown to meet applicable acceptance criteria.

B. Cold Shutdown - Case 2

The following plant configuration was analyzed:

1. No PCPs are in operation when the boron dilution event occurs.

2. The SDC system is in operation with a minimum flow rate of 2810 gpm.

3. All three charging pumps are assumed to be in operation (143 gpm).

4. The PCS loops are only partially filled (3300 ft^3).

5. The SDM requirement is $\geq$ 3.5% assuming ARI-MRR.

Based on the plant configuration detailed above, the time-to-criticality for this case was shown to meet applicable acceptance criteria.

C. Cold Shutdown - Case 3

The following plant configuration was analyzed:

1. No PCPs are in operation when the boron dilution event occurs.

2. The SDC system is in operation with a minimum flow rate of 650 gpm.

3. Only one charging pump is assumed to be in operation (60 gpm).

4. The SDM requirement is ≥ 3.5% assuming ARI-MRR.

5. The PCS loops are only partially filled (3300 ft^3).

Based on the plant configuration detailed above, the time-to-criticality for this case was shown to meet applicable acceptance criteria.

14.3.2.3.3 Dilution During Hot Shutdown (Mode 4)

In Mode 4, the PCS could be in any one of the following configurations:

1. PCPs in operation, SDC secured

2. Both PCPs and SDC in operation

3. PCPs secured, SDC secured

If one or more coolant pumps are in operation, it is assumed that the dilution flow undergoes perfect mixing with the primary coolant. If no coolant pumps are in operation, the dilution flow does not perfectly mix with the primary coolant and forms a moving front of diluted water that flows toward the core. This scenario is modeled with a dilution front model.

The following cases were analyzed:

A. Hot Shutdown - Case 1

The following plant configuration was analyzed:

1. At least one PCP is in operation when the boron dilution event occurs. It is assumed that the injected unborated water is perfectly mixed with the primary coolant.

2. All three charging pumps are assumed to be in operation (143 gpm).

3. A total PCS volume of 7921 ft^3 was assumed for the analysis. This volume does not credit the volumes of the pressurizer, pressurizer surge line, upper vessel head, and downcomer to upper head bypass flow pass. A total steam generator tube plugging of 15% is also assumed.

4. The SDM requirement is ≥ 2% assuming ARI-MRR.

Based on the plant configuration detailed above, the time-to-criticality for this case was shown to meet applicable acceptance criteria.

B. Hot Shutdown - Case 2

The following plant configuration was analyzed:

1. No PCPs are in operation when the boron dilution event occurs.

2. The SDC system is in operation with a minimum flow rate of 2810 gpm.

3. All three charging pumps are assumed to be in operation (143 gpm).

4. A total PCS volume of 7921 ft^3 was assumed for the analysis. This volume does not credit the volumes of the pressurizer, pressurizer surge line, upper vessel head, and downcomer to upper head bypass flow pass. A total steam generator tube plugging of 15% is also assumed.

5. The SDM requirement is ≥ 2% assuming ARI-MRR.

Based on the plant configuration detailed above, the time-to-criticality for this case was shown to meet applicable acceptance criteria.

14.3.2.3.4 Dilution during Hot Standby (Mode 3)

Mode 3 operation is a transition mode. The following cases were analyzed:

1. At least one PCP is in operation when the boron dilution event occurs. It is assumed that the injected unborated water is perfectly mixed with the primary coolant.

2. All three charging pumps are assumed to be in operation (143 gpm).

3. A total PCS volume of 7921 ft^3 was assumed for the analysis. This volume does not credit the volumes of the pressurizer, pressurizer surge line, upper vessel head, and downcomer to upper head bypass flow pass. A total steam generator tube plugging of 15% is also assumed.

4. The SDM requirement is ≥ 2% assuming ARI-MRR.

Based on the plant configuration detailed above, the time-to-criticality for this case was shown to meet applicable acceptance criteria.

14.3.2.3.5 Dilution During Power Operation and Startup (Modes 1 and 2)

During dilution at Startup (Mode 2), the operating staff will be monitoring the nuclear instrument readings. An abnormal change in the readings of these instruments will inform the operator that dilution is occurring. The operator will have further indication of the process from volume control tank level and from operation of the letdown diverter valve. Further, should the makeup controller fail to close the makeup stop valve, the operator has visual indication of makeup water flow and of makeup water transfer pump operation.

In any case, should continued dilution occur, the reactivity insertion rate would be less than that considered for uncontrolled rod/rod bank withdrawals. The reactor protection provided for the rod withdrawal incident will also provide protection for the boron dilution incident. Inadvertent injection of primary makeup water into the PCS while the reactor is at power would result in a reactivity addition initially causing a slow rise in power, temperature and, possibly, pressure. In view of the large number of available alarms and indications, it is considered that there is ample time and information available to the operator for identification of the incident and for stopping the makeup water injection and to initiate boration. If the operator takes no corrective action, the power, temperature and pressure would rise. However, this transient would be terminated by either the thermal margin/low pressure or variable overpower trip. Assuming a minimum shutdown worth of 2%Δp, the operator would have $\geq$15 minutes to terminate the dilution following reactor trip.

14.3.2.3.6 Failure to Add Boron to Compensate for Reactivity Changes after Shutdown

Administrative procedures require that boron levels be set and checked before cooldown is initiated. The unlikely event of a failure to add boron before cooldown to compensate for reactivity increases due to cooldown or xenon concentration reduction would result in a loss of shutdown margin and a return to criticality. The maximum cooldown rate is 100°F per hour. Assuming the end of cycle moderator temperature coefficient of reactivity at hot standby with all rods in, the maximum rate of reactivity addition during cooldown from hot standby would be 3.5×10^{-2} $\Delta\rho$/h. The maximum rate of xenon concentration reduction occurs approximately 10 hours after shutdown from full power operation and is conservatively assumed to be equivalent to a reactivity change of 0.01 $\Delta\rho$/h.

The reactivity addition rate due to the reduction of xenon concentration would not normally coincide with cooldown. However, with the combined effect of temperature reduction and xenon reduction at the maximum rate, it would require more than 15 minutes (Reference 2) for the reactor to go critical, assuming a minimum 2% shutdown margin.

14.3.3 RADIOLOGICAL CONSEQUENCES

Not required for this event

14.3.4 CONCLUSIONS

The results show that there is adequate time for the operator to manually terminate the source of dilution flow. The operator can then initiate reboration to recover the shutdown margin. Boron dilution during power operation is bounded by the analyses presented in Section 14.2. However, the results presented here demonstrate that there is adequate time for the operator to manually terminate the source of dilution flow following reactor trip.

REFERENCES

1. EMF-2310 (P)(A), Revision 1, Framatome ANP, Inc., May 2004, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors.”

2. ANP-3218-001, Palisades Cycle 24 Safety Analysis Report, November 2013.

14.4 CONTROL ROD DROP

14.4.1 DROPPED ROD EVENT

14.4.1.1 Event Description

The dropped rod and dropped bank events are initiated by a de-energized control rod drive mechanism or by a malfunction associated with a control rod bank. The dropped rod events are classified as Moderate Frequency events.

In the dropped rod or dropped bank events, the reactor power initially drops in response to the insertion of negative reactivity. This results in reduction of the moderator temperature due to a mismatch between core power being generated and secondary system load demand. The core power redistributes due to the local power effect of the dropped rod or bank. If no RPS trip occurs, the reactor power will return to the initial level due to the positive reactivity inserted by the combined effects of a negative moderator temperature coefficient and a reduced moderator temperature. The moderator temperature will not decrease below the temperature necessary to return the core to initial power because at that temperature, the core power and secondary system load demand are equalized, removing the driving force for further moderator cooldown. The rod and bank drop events challenge the DNBR SAFDL because of the increased radial peaking and the potential return to initial power.

The original design of the Palisades Plant included a turbine runback upon detection of a dropped rod. Later analysis showed that at the beginning of the cycle, in manual mode, turbine runback could have unacceptable effects on reactor performance. Thus, the turbine runback feature has been disabled and is no longer used in response to a dropped rod (Page 52 of Reference 1).

14.4.1.2 Thermal-Hydraulic Analysis

14.4.1.2.1 Analysis Methods

The analysis of rod drop events is described in Reference 7 and uses the PRISM (Reference 2), XCOBRA-IIIC (Reference 4) and PTSPWR2 (Reference 3) codes. In this event the radial redistribution of power in the core can result in radial peaking factors in excess of Core Operating Limits Report (COLR) limits. The analyses are performed by coupling a conservative power peak to transient response and DNB calculations. The power peak associated with the event is characterized through an augmentation factor which relates the maximum power peak to the steady-state power peak. The steady-state power distributions and augmentation factors are calculated with the PRISM reactor simulator. A power augmentation factor is included in the XCOBRA-IIIC MDNBR calculations to account for radial power redistribution effects typical of the event.

Simulation of the system transient for rod drops has not been performed since Cycle 9. Because the secondary system load demand remains constant through the event, the moderator will continue to cool down until moderator feedback is sufficient to restore power to its initial level. At that point, the moderator temperature stabilizes because no mismatch between core power production and secondary system load demand exists. The transient thus results in a new steady-state condition characterized by a power level equal to the initial power and a core coolant temperature substantially reduced from the initial condition value. The DNBR is conservatively evaluated with an XCOBRA-IIIC calculation using the initial condition power, coolant temperature and flow at a reduced pressure. The redistribution of the radial peaking factor is incorporated as noted above.

14.4.1.2.2 Bounding Event Input

Reference 5 is the bounding transient analysis.

A conservative radial peaking augmentation factor was applied in the MDNBR calculation for this event. This power peaking augmentation factor is used to account for radial power redistribution effects typical of the event.

14.4.1.2.3 Analysis of Results

The bounding MDNBR and the peak LHR for this event for the current fuel cycle are given in Table 14.1-5.

14.4.1.3 Radiological Consequences

Not required for this event.

14.4.1.4 Conclusions

The results of the analysis demonstrates that:

- The predicted MDNBR (Table 14.1-5) is greater than the HTP correlation limit presented in Table 14.1-1. This ensures that, with 95% probability and 95% confidence, DNB is not expected to occur and no fuel failures are expected.

- The predicted peak linear heat rate (Table 14.1-5) is less than fuel centerline melt criterion presented in Table 14.1-1.

Therefore, all the applicable acceptance criteria for the event are met.

14.4.2 DROPPED BANK EVENT

14.4.2.1 Event Description

The dropped bank event is distinguished from the dropped rod event by the greater magnitude of augmentation factors. The power initially drops due to the insertion of negative reactivity from the dropped bank. This, in turn produces a power mismatch between the primary and secondary sides. In the presence of negative moderator feedback, the core power increases to match the steam generator load.

14.4.2.2 Thermal-Hydraulic Analysis

14.4.2.2.1 Analysis Methods

The analysis of the dropped bank event is described in Reference 7 and uses the PRISM (Reference 2), XCOBRA-IIIC (Reference 4) and PTSPWR2 (Reference 3) codes. In this event the radial redistribution of power in the core can result in radial peaking factors in excess of COLR limits. The analyses are performed by coupling a conservative power peak to transient response and DNB calculations. The power peak associated with the event is characterized through an augmentation factor which relates the maximum power peak to the steady-state power peak. The steady-state power distributions and augmentation factors are calculated with the PRISM reactor simulator. A power augmentation factor is included in the XCOBRA-IIIC MDNBR calculations to account for radial power redistribution effects typical of the event.

14.4.2.2.2 Bounding Event Input

The Dropped Bank Event is bounded by the Dropped Rod Event in one of two ways (Reference 5):

1. If the dropped bank worth is sufficient enough to cause a reactor trip on a thermal margin/low pressure signal, the reactor will not return to full power. The decreased PCS temperature and the reduced core power will outweigh the higher radial peaking associated with a dropped bank.

2. If the reactor does not trip it will eventually return to rated power, but at a much lower moderator temperature. The magnitude of PCS cooldown required for the core to return to power outweighs the higher peaking.

If one, two, three, or all four safety grade control rod clutch power supplies are replaced with non-qualified versions, a scenario has been identified where approximately half of the control rods could be dropped into the reactor core if a safety grade clutch power supply fails and the parallel non-safety grade power supply is also assumed to fail. In response to the dropped control rods, the reactor power rapidly decreases and then plateaus. The PCS temperature and pressurizer pressure, in turn, also decrease. When the PCS conditions reach the thermal margin/low pressure trip setpoint, the reactor is scrammed and the reactor power decreases to a decay heat level. Because there is no return to power, the "half-scram" analysis is bounded by the single bank drop event (Reference 6).

14.4.2.2.3 Analysis of Results

The consequences of a Dropped Bank Event are bounded by those of a Dropped Rod Event.

14.4.2.3 Radiological Consequences

Not required for this event.

14.4.2.4 Conclusions

The results of the analysis demonstrates that:

- The predicted MDNBR (Table 14.1-5) is greater than the HTP correlation limit presented in Table 14.1-1. This ensures that, with 95% probability and 95% confidence, DNB is not expected to occur and no fuel failures are expected.

- The predicted peak linear heat rate (Table 14.1-5) is less than fuel centerline melt criterion presented in Table 14.1-1.

Therefore, all the applicable acceptance criteria for the event are met.

REFERENCES

1. "Palisades - SEP Topics - Design Basis Events," SEP Topic XV-8, Section 4.6.3, D M Crutchfield letter to D P Hoffman, dated November 3, 1981, (2643/1537).

2. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs, Volume 1 -Methodology Description, Volume 2 - Benchmarking Results," Siemens Power Corporation, January 1997.

3. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR2)," XN-NF-74-5(P)(A), Revision 2 and Supplements 3-6, Exxon Nuclear Company, October 1986, (8029/2447), (8029/2171), (8029/2390), (8029/2648), (8029/2515), (8029/2785) and (8029/2633).

4. "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-NF-75-21(P)(A), Revision 2, Exxon Nuclear Company, January 1986, (2741/1496).

5. EMF-2507, Revision 0, "Palisades Cycle 16 Safety Analysis Report," Framatome ANP Richland, Inc, March 2001.

6. EMF-2259, Revision 2, "Palisades Cycle 15 Safety Analysis Report," Siemens Power Corporation, September 2000.

7. ANF-84-73(P)(A), Revision 5, Appendix B(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Siemens Power Corporation, July 1990.

14.5 CORE BARREL FAILURE

14.5.1 EVENT DESCRIPTION

A circumferential rupture of the core support barrel has the same low probability of occurrence as a major rupture of the primary system piping. In the event of such an occurrence, the core stop supports serve to support the barrel and the reactor core by transmitting all loads directly to the vessel. The clearance between the core barrel and the supports is approximately one-half inch at operating temperatures. Thus, coolant flow through the core continues although at a somewhat reduced rate and the core sustains a small reactivity transient induced by the motion of the core relative to the inserted rod bank(s).

The worst possible axial location of the barrel rupture is at the midplane of the vessel nozzle penetrations. This forms a direct flow path between the inlet and exit nozzles in parallel with the path that goes through the core. Thus, the maximum possible amount of bypass flow will occur.

14.5.2 THERMAL-HYDRAULIC ANALYSIS

The core barrel failure event is bounded by the consequences of the control rod ejection event (FSAR Section 14.16). Both of these events are classified as Limiting Faults. Specifically, the reactivity insertion rate and radial power redistribution for the control rod ejection are worse than what occurs during a core barrel failure. In addition, the control rod ejection analysis assumed a coincident loss of offsite power which leads to a coastdown of the primary coolant pumps. The flow decrease due to the coastdown of the PCS pumps is more severe than the flow reduction due to increased core bypass from the core barrel failure.

14.5.3 RADIOLOGICAL CONSEQUENCES

The radiological consequences are bounded by the control rod ejection event.

14.5.4 CONCLUSIONS

Since the control rod ejection event meets the acceptance criteria for a limiting fault event, the results of a core barrel failure also meet the same criteria.

14.6 CONTROL ROD MISOPERATION

14.6.1 MALPOSITION OF THE PART-LENGTH CONTROL ROD GROUP

14.6.1.1 Event Description

The four part-length control rods contain neutron poison only over approximately 25% of their length. The original purpose of these rods was to control the axial power distribution as determined by the upper and lower sections of excore ion chambers by manual alignment within the core.

The part-length control rods are not connected to any reactor trip circuit and will not drop into the core on a reactor trip or loss of power, but a mechanical failure in a rod mechanism could cause an individual rod to drop into the lower region of the core. If the drop is caused by a mechanical failure in the brake mechanism, the rod position lower limit switch will be actuated. The limit switch will supply a signal that actuates the rod drop protection circuit.

14.6.1.2 Thermal-Hydraulic Analysis

A thermal-hydraulic analysis is not applicable for this event (Reference 3).

14.6.1.3 Radiological Consequences

A radiological consequences analysis is not applicable for this event.

14.6.1.4 Conclusions

Technical Specifications do not allow the use of the part-length control rods for the above purpose. During power operation, the part-length control rod group is maintained in the fully withdrawn position and is not used and mispositioning of these rods is not a credible event.

14.6.2 STATICALLY MISALIGNED CONTROL ROD/BANK

14.6.2.1 Event Description

The static misalignment events occur when a malfunction of the control rod drive mechanism causes a control rod to be out of alignment with its bank. Misalignment occurs when the rod is either higher or lower than any of the other control rods in the same bank or when a bank(s) is out of alignment with the Power Dependent Insertion Limit (PDIL). The reactor is at steady state, rated full-power or part-power conditions with enhanced power peaking. This event is classified as a moderate frequency occurrence.

In the evaluation of the static rod misalignment event, the control banks are inserted to the HFP PDIL but one of the control rods is assumed to remain in a fully withdrawn condition. This results in a local increase of the radial power peaking factor and a corresponding reduction in the DNB margin. The radial power redistribution consequences of a reverse misalignment, ie, one rod is fully inserted while the bank remains withdrawn, are essentially the same as the Dropped Control Rod or Bank event. Extreme misalignments can result in core radial power redistribution characterized by peaking factors in excess of design limits.

14.6.2.2 Thermal-Hydraulics Analysis

14.6.2.2.1 Analysis Method

In this event, the radial redistribution of power in the core can result in radial peaking factors in excess of Core Operating Limits Report (COLR) limits. The power peak associated with this event is characterized through an augmentation factor which relates the maximum power peak to the steady-state power peak. The steady-state power distributions and augmentation factors are calculated with the PRISM (Reference 1) reactor simulator. DNB calculations are performed using the XCOBRA-IIIC code (Reference 2).

This single rod misalignment event is analyzed at the rated power condition with conservative allowances applied in a direction to minimize DNBR.

In the analysis of the statically misaligned rod, primary system pressure, core inlet temperature and coolant flow rate at the rated full-power operating point are input into the XCOBRA-IIIC code to calculate MDNBR. The rated full-power core average clad surface heat flux is input to the MDNBR calculation after being adjusted for a radial peaking augmentation factor that bounds the radial power redistribution of a misaligned rod.

14.6.2.2.2 Bounding Event Input

A conservative radial-peaking augmentation factor corresponding to the most limiting static misalignment; ie, Bank 4 fully inserted with one rod fully withdrawn (Bank 4 is 99 inches out of alignment with rated power PDIL) is used in the DNB calculation. By determining the radial peaking augmentation factor in this manner, MDNBRs for this event are conservatively calculated.

14.6.2.2.3 Analysis of Results

The bounding MDNBR and peak LHR for this event for the current fuel cycle are given in Table 14.1-5.

14.6.2.3 Radiological Consequences

A radiological consequences analysis is not applicable for this event.

14.6.2.4 Conclusion

The results of the analysis demonstrates that:

The predicted MDNBR (Table 14.1-5) is greater than the HTP correlation limit presented in Table 14.1-1. This ensures that, with 95% probability and 95% confidence, DNB is not expected to occur and no fuel failures are expected.

The predicted peak linear heat rate (Table 14.1-5) is less than fuel centerline melt criterion presented in Table 14.1-1.

Therefore, all the applicable acceptance criteria for the event are met.

REFERENCES

1. EMF-96-029(P)(A), Volumes 1 and 2, “Reactor Analysis System for PWRs, Volume 1 - Methodology Descriptions, Volume 2 - Benchmarking Results,” Siemens Power Corporation, January 1997.

2. "XCOBRA-IIIC: A Computer Code To Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-NF-75-21(P)(A), Revision 2, Exxon Nuclear Company, January 1986, (2741/1496).

3. "Palisades Cycle 8: Disposition and Analysis of Standard Review Plan Chapter 15 Events," ANF-88-108, Revision 1, Advanced Nuclear Fuels, Inc, September 1988, (C254/1343).

4. Deleted

5. Deleted

6. Deleted

14.7 DECREASED REACTOR COOLANT FLOW

14.7.1 LOSS OF FORCED REACTOR COOLANT FLOW

14.7.1.1 Event Description

This event is characterized by a total loss of forced reactor coolant flow which is caused by the simultaneous loss of electric power to all of the reactor primary coolant pumps. Following the loss of electrical power, the reactor coolant pumps begin to coast down.

If the reactor is at power when the event occurs, the loss of forced coolant flow causes the reactor coolant temperatures to rise rapidly. This results in a rapid reduction in DNB margin, and could result in DNB if the reactor is not tripped promptly. Also, as the reactor coolant temperatures rise, the primary coolant expands, which causes an insurge into the pressurizer, a compression of the pressurizer steam space, and a rapid increase in reactor coolant system pressure. The primary system overpressurization will be mitigated by the action of the primary system safety valves and the reduction in core power following reactor trip. The reactor trip is initiated by the low primary coolant flow signal.

The minimum DNBR is controlled by the interaction of the primary coolant flow decay and the core power decrease following reactor trip. The power to flow ratio initially increases, peaks and then declines as the challenge to the Specific Acceptable Fuel Design Limits (SAFDL's) is mitigated by the decline in core power due to the reactor trip. If a reactor trip can be obtained promptly, the power to flow ratio will first peak and then decrease during the transient such that the SAFDL's will be no longer challenged.

The pump coastdown characteristics and the timing of the low primary coolant flow reactor trip, trip delays and scram rod insertion characteristics are key parameters in the initial stage of the event. Later, natural circulation flow is developed in the primary system and the steam generators are available to remove the decay power. Therefore, long-term cooling of the core can be achieved.

The primary concern with this event is the challenge to the SAFDL's. The event is analyzed to verify that the reactor protection system can respond fast enough to prevent penetration of the DNB SAFDL.

14.7.1.2 Thermal-Hydraulic Analysis

14.7.1.2.1 Analysis Method

The overall response of the primary and secondary systems in Reference 2 for this event is calculated by the PTSPWR2 computer code (Reference 3). The MDNBR for the event is calculated using the thermal hydraulic conditions from the PTSPWR2 calculation as input to XCOBRA-IIIC (Reference 4). The analysis for this event is described in Reference 1.

The event is initiated by simultaneously tripping of all of the reactor coolant pumps. The pump coastdown is governed by a conservative estimate of the pump flywheel inertia, the homologous pump curves and the loop hydraulics. Reactor trip is delayed until the low reactor coolant loop flow signal is obtained. This trip set point is conservatively reduced to account for uncertainties in flow measurement.

14.7.1.2.2 Bounding Event Input

This event is analyzed from full power initial conditions with the reactor control rod system in manual. The core thermal margins are at a minimum at full power conditions. This is the bounding mode of operation for this event. One case is analyzed for this event to assess the challenge to the DNB SAFDL. The event analysis is biased to minimize DNBR. The steam line bypass and the atmospheric dump valves are both assumed not to operate, which provide the greatest challenge to the DNB SAFDL. The fuel centerline melt SAFDL is not seriously challenged by the small power increase typical of this event.

14.7.1.2.3 Analysis of Results

The transient is initiated by tripping all four primary coolant pumps. As the pumps coast-down, the core flow is reduced, causing a reactor scram on low flow. As the flow coasts-down, primary temperatures increase before being turned around due to the power decrease following reactor scram. This increase in temperature causes a subsequent power rise due to moderator reactivity feedback. The temperature increase also causes an insurge into the pressurizer and resultant pressurization of the reactor coolant system. The primary challenge to DNB is from the decreasing flow rate and resulting increase in coolant temperatures.

The transient response from Reference 2 is shown in Figures 14.7.1-1 through 14.7.1-5. Table 14.7.1-1 lists the sequence of events for this

transient. The bounding MDNBR and the peak LHR for this event for the current fuel cycle are given in Table 14.1-5.

14.7.1.3 Radiological Consequences

Not required for this event.

14.7.1.4 Conclusions

The results of the analysis demonstrates that:

- The predicted MDNBR (Table 14.1-5) is greater than the HTP correlation limit presented in Table 14.1-1. This ensures that, with 95% probability and 95% confidence, DNB is not expected to occur and no fuel failures are expected.

- The predicted peak linear heat rate (Table 14.1-5) is less than fuel centerline melt criterion presented in Table 14.1-1.

Therefore, all the applicable acceptance criteria for the event are met.

14.7.2 REACTOR COOLANT PUMP ROTOR SEIZURE

14.7.2.1 Event Description

The locked rotor event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal.

Following the reactor trip, the heat stored in the fuel rods continues to be transferred to the reactor coolant. Because of the reduced core flow, the coolant temperatures will begin to rise, causing expansion of the primary coolant and consequent pressurizer insurge flow and PCS pressurization. As the pressure increases, pressurizer sprays and safety valves would act to mitigate the pressure transient.

The rapid reduction in core flow and the increase in coolant temperature may seriously challenge or penetrate the DNBR SAFDL. The event is thus evaluated to assess the DNBR challenge. The fuel center line melt SAFDL is not seriously challenged by the small power increase typical of this event. PCS pressurization criteria have not been approached in analyses of this event. No case addressing pressurization is therefore performed.

The event as simulated is structured to provide a bounding determination of MDNBR for both the locked rotor and broken shaft (15.3.4) events.

14.7.2.2 Thermal-Hydraulic Analysis

14.7.2.2.1 Analysis Method

The transient response of the plant is calculated using PTSPWR2. The MDNBR is calculated using the XCOBRA-IIIC code. The analysis is described in Reference 1.

14.7.2.3 Bounding Event Input

One case is analyzed for this event to maximize the challenge to the DNB limit. The bounding operating mode for this event is full power initial conditions. The analysis uses a low flow trip setpoint of 93% of four primary coolant pump flow.

14.7.2.4 Analysis of Results

This event is initiated by the seizure of a rotor in a primary coolant pump. This analysis assumes a locked pump loss coefficient consistent with the homologous pump curves at zero pump speed. As the core flow is reduced, a reactor scram on low flow occurs. The average core temperature increases before being turned around due to the power decrease following reactor scram. This increase in temperature causes a subsequent power rise due to moderator reactivity feedback. The temperature increase also causes an insurge into the pressurizer and resultant pressurization of the primary coolant system. The primary challenge to DNB is from the decreasing flow rate and resulting increase in coolant temperature. The transient response from Reference 2 is shown in Figures 14.7.2-1 through 14.7.2-5.

Table 14.7.2-1 lists the sequence of events for this transient. The bounding MDNBR and peak LHR for this event for the current fuel cycle are given in Table 14.1-5.

14.7.2.5 Radiological Consequences

Not required for this event.

14.7.2.6 Conclusions

The results of the analysis demonstrates that:

- The predicted MDNBR (Table 14.1-5) is greater than the HTP correlation limit presented in Table 14.1-1. This ensures that, with 95% probability and 95% confidence, DNB is not expected to occur and no fuel failures are expected.

- The predicted peak linear heat rate (Table 14.1-5) is less than fuel centerline melt criterion presented in Table 14.1-1.

Therefore, all the applicable acceptance criteria for the event are met.

REFERENCES

1. ANF-84-73(P)(A), Revision 5, Appendix B(P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Siemens Power Corporation, July 1990.

2. "Palisades Cycle 9: Analysis of Standard Review Plan Chapter 15 Events," ANF-90-078, Advanced Nuclear Fuels, Inc., September 1990, (C677/1283).

3. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," XN-NF-74-5(P)(A), Revision 2 and Supplement 1-6, Exxon Nuclear Company, October 1986, (8029/2447), (8029/2171), (8029/2390), (8029/2648), (8029/2515), (8029/2785), and (8029/2633).

4. "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-NF-75-21(P)(A), Revision 2, Exxon Nuclear Company, January 1986, (2741/1496).

5. Deleted

6. Deleted

7. Deleted

14.8 START-UP OF AN INACTIVE LOOP

14.8.1 EVENT DESCRIPTION

This event is initiated by the start-up of an inactive primary coolant pump. The startup of an inactive pump can lead to an introduction of colder primary coolant into the reactor core. The lower coolant temperature, together with a negative moderator temperature coefficient, can cause an increase in core power and a degradation of DNB margin. Sufficient protection is available to reduce the consequences of this event.

The manual flowtrip setpoint switch that allowed changing the low flow and variable high power trip setpoints automatically while at power was removed by FC-888 (Reference 1). The low flow trip setpoints can be manually changed to allow for 3 pump operation, however, plant operating procedures require 4 pumps to be operating when the reactor is operated above hot shutdown. For this reason, the "Startup of an Inactive Loop" event was dispositioned by the fact that this mode of operation is administratively prohibited.

REFERENCES

1. "Upgrade Reactor Protection System Bistable Trip Units and Power Supplies," FC-888, June 30, 1992.

14.9 EXCESSIVE FEEDWATER INCIDENT

Deleted - Bounded by FSAR Section 14.10, "Increase in Steam Flow."

14.10 INCREASE IN STEAM FLOW (EXCESS LOAD)

14.10.1 EVENT DESCRIPTION

The Increase in Steam Flow event is initiated by an increase in steam demand. The increased steam demand may be initiated by the operator or by regulating valve malfunction. The step increase in steam flow results from a rapid opening of the turbine control valves, atmospheric dump valves or the turbine bypass valve to condenser.

The event initiator is a step increase in steam flow at full power. The maximum step increase in steam flow is a 30% increase due to the full opening of the four atmospheric dump valves. The feedwater regulating valves open to increase the feedwater flow in an attempt to match the increased steam demand and maintain steam generator water level. In response to the increased steam flow, the secondary system pressure decreases, resulting in an increase in the primary-to-secondary heat transfer rate. The primary side steam generator outlet temperature decreases due to the enhanced heat removal. As a consequence, the primary system core average temperature decreases and the primary system fluid contracts, resulting in an outsurge of fluid from the pressurizer. The pressurizer level and pressure decrease as fluid is expelled from the pressurizer. If the moderator temperature coefficient is negative, the reactor core power increases as the moderator temperature decreases due to the mismatch between the power being removed by the steam generators and the power being generated in the core.

Depending on the magnitude of the increase in steam demand, a reactor trip may not be activated. Instead, the reactor system will reach a new steady-state condition at a power level greater than the initial power level which is consistent with the increased heat removal rate. The final steady-state conditions which are achieved will depend upon the magnitude of the moderator temperature coefficient. If the moderator temperature coefficient is positive, the reactor power would decrease as the core average coolant temperature decreases, and this event would not produce a challenge to the acceptance criteria.

The TM/LP and variable overpower trips are available to prevent violation of the acceptance criteria. The limiting DNB case will typically trip on the variable overpower trip. The variable overpower trip auctioneers between the highest of the ΔT-power and NI power signals, both of which are decalibrated. The ΔT-power signal is decalibrated due to the hot leg and cold leg RTD lag times. The NI signal is decalibrated due to excore downcomer decalibration (shadowing of excore detectors by the cold water) as the coolant in the downcomer decreases in temperature during the event. The limiting DNB and peak LHR case will occur when the auctioneered ΔT-power and NI power signals both reach the variable high power trip setpoint at essentially the same time, which effectively results in the most delayed reactor trip time and the maximum overshoot and highest core power level.

The key criterion for this event is demonstrating that fuel integrity is maintained by ensuring that the SAFDLs are not exceeded by assuring that the MDNBR is not less than the HTP correlation limit and that the peak LHR is less than the fuel centerline melt limit.

The Increase in Steam Flow event analysis provides transient shifts (over/undershoots) on power, hot leg temperature, cold leg temperature, and pressure which are used as inputs to the TM/LP statistical setpoint verification analysis.

14.10.2 THERMAL-HYDRAULIC ANALYSIS

14.10.2.1 Analysis Method

The transient response of the reactor system was calculated using the S-RELAP5 computer code and the S-RELAP5 non-LOCA methodology (Reference 1). The core thermal hydraulic boundary conditions from the S-RELAP5 calculation were used as input to the XCOBRA-IIIC code (Reference 2) to predict the minimum DNBR for the event.

14.10.2.2 Bounding Event Input

This event is characterized by an increase in power and a decrease in PCS pressure. The SAFDLs are most challenged by initiating this event at full power (2580.6 MWt including power measurement uncertainty). The analysis at full power bounds operation at lower power levels.

The limiting MDNBR case was found by analyzing the event at the maximum increase in steam flow of 30% (representing the full opening of the four atmospheric dump valves) over a range of negative MTCs from BOC to EOC conditions. Other cases were analyzed over a range of excess load increases, negative MTCs, and assumed axial ASIs which have a propensity to trip on the TM/LP trip rather than upon the variable high power trip. Some of these cases did not trip at all, and merely caused a restabilization of the reactor. For cases tripping on the TM/LP trip, the sets of transient shifts in power, hot leg temperature, cold leg temperature, and pressure were evaluated in the TM/LP statistical setpoint verification analysis.

14.10.2.3 <u>Analysis of Results</u>

The Excess Load analysis is described in Reference 3. Table 14.10-1 provides a sequence of events for the limiting MDNBR case. Figures 14.10-1 through 14.10-5 show results for the limiting MDNBR case.

The event was initiated at HFP since the margin to the SAFDLs is the smallest at HFP. The event was initiated by a step increase in steam flow of 30% of the initial steam flow with an MTC of – 12 pcm/°F. The steam generator pressure began to decrease immediately. As the steam generator pressure decreased, the main feedwater flow increased to maintain steam generator level. Also, as the steam generator pressure decreased, the heat transfer rate from the PCS to the secondary increased causing a decrease in PCS temperatures as shown in Figure 14.10-2. As the PCS temperatures decreased, fluid was expelled from the pressurizer and the pressurizer level and pressure began to decrease as shown in Figures 14.10-4 and 14.10-3. With a negative MTC and a decreasing PCS temperature, the core power began to increase as shown in Figure 14.10-1. Figure 14.10-1 also shows the decalibrated ΔT-power and NI power signals that are auctioneered, the highest of which was compared to the VHPT setpoint. Figure 14.10-5 shows the moderator, Doppler, scram, and total reactivity feedbacks. The VHPT provides the reactor trip signal at 22.99 sec. Following signal processing time and scram delay time, scram insertion began at 24.08 sec. The peak rod surface heat flux occurred shortly after scram insertion began, and the MDNBR occurred at 24.2 sec.

The bounding MDNBR and peak LHR for this event for the current fuel cycle are given in Table 14.1-5.

14.10.3 RADIOLOGICAL CONSEQUENCES

Not required for this event.

14.10.4 CONCLUSIONS

The results of the analysis demonstrates that:

- The predicted MDNBR (Table 14.1-5) is greater than the HTP correlation limit presented in Table 14.1-1. This ensures that, with 95% probability and 95% confidence, DNB is not expected to occur and no fuel failures are expected.

- The predicted peak linear heat rate (Table 14.1-5) is less than fuel centerline melt criterion presented in Table 14.1-1.

Therefore, all the applicable acceptance criteria for the event are met.

REFERENCES

1. EMF-2310(P)(A) Revision 0, SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors, Framatome ANP, May 2001.

2. "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-NF-75-21(P)(A), Revision 2, Exxon Nuclear Company, January 1986, (2741/1496).

3. EMF-3097 Revision 0, Palisades Cycle 18 Safety Analysis Report, June 2004.

14.11 POSTULATED CASK DROP ACCIDENTS

14.11.1 EVENT DESCRIPTION

In 2003, Facility Change FC-976 modified the main hoist of the Fuel Building Crane to increase the capacity to 110-tons, and to meet single failure criteria in accordance with NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” and NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants.” Postulated load drops from the spent fuel pool area cranes are analyzed, unless the crane and lifting devices are designed and specified to be single failure proof in accordance with NUREG 0612 and NUREG-0554. Since the main hoist has been upgraded to meet single-failure-proof criteria, analyses of postulated load drops from the main hoist are no longer required.

Although the main hoist of the spent fuel crane is designed and operated in accordance with single-failure-proof criteria for cask handling activities, there may be situations in which lifting devices used with the main hook do not meet these requirements or single-failure-proof features of the main hoist may be disabled. In these situations, the crane no longer meets single failure proof requirements, and load drops are postulated. Therefore, this section contains an outline of the methodology and evaluations used to document the consequences of postulated fuel transfer cask drop accidents in the fuel handling area of the Palisades plant.

The postulated drops of the loaded Sierra Nuclear Multi-Assembly Sealed Basket Transfer Cask (MTC) are the bounding evaluated cask drops, and are assumed to be the result of hypothetical failures in the crane system (main hook) or load handling devices. The load drops were evaluated for structural and radiological effects.

The following is a list of the structural and radiological calculations that were used to provide for the safe use of the loaded MTC in the spent fuel pool area:

EA-FC-864-09 "MSB Transfer Cask drop and Impact Limiter design" (Ref. 4)

EA-FC-864-11 "Evaluation of the MTC/MSB for drop on the VCC with the MTC/MSB C.G. located outside the VCC boundary" (Ref. 5)

EA-FC-864-41 "MTC Over Turn Calculation in the Spent Fuel Pool" (Ref. 7)

NAI-1149-026 “Palisades Design Basis Cask Drop Accident AST Radiological Analysis,” Revision 0 (Ref. 15)

Since the 15-ton auxiliary hoist of the spent fuel pool crane is not single failure proof, postulated load drops from the auxiliary hoist have been evaluated in accordance with NUREG-0612. Heavy Loads handled with the auxiliary hoist are limited to designated safe load paths. The Bechtel Load Drop analysis for a 25-Ton cask (Reference 1) bounds the structural consequences from any drop from the Auxiliary Hoist.

The evaluation of the radiological consequences for a postulated load drop of a loaded Multi-Assembly Sealed Basket Transfer Cask (MTC) from the main hoist, contained within this section, bounds the radiological consequences from postulated load drops from the auxiliary hoist.

14.11.2 STRUCTURAL ANALYSIS

Load drop analyses of a 25-ton cask were performed by Bechtel using the methodology described in Bechtel Topical Report BC-TOP-9 (Reference 1). The analyses considered 25-ton cask load drops in four specific locations in the spent fuel pool area (References 13). The Bechtel analyses show that energy absorbing material is not required to reduce the amount of energy transferred to the floors of the cask loading area in the spent fuel pool and the washdown pit for this load.

Prior to the upgrading of the spent fuel crane to single failure proof, FC-864 (Reference 2) analyzed drops of a loaded MTC due to an Operating Requirements Manual requirement that movement of loads greater than 25 tons over the main part of the spent fuel pool require an evaluation in accordance with Section 5.1 of NUREG-0612. Postulated drops of the loaded MTC/MSB were performed for the washdown pit, the spent fuel pool cask loading area, specific load transfer configurations in the Track Alley and MTC tip over on to the 11 X 7 fuel racks west of the cask loading area (See Fig 14.11-1). The evaluation concluded that impact limiting pads were required in the cask loading area of the spent fuel pool and the cask washdown pit to reduce the amount of energy transferred to the floors. It should be noted that, since the main hoist is now single-failure-proof, impact limiting pads are not required in these areas, provided the lifting devices and interfacing lift points meet the requirements of NUREG-0612, Section 5.1.6. If any of the single failure features of the main hoist are disabled, impact limiting pads are, therefore, required for lifts greater than 25-tons in the cask loading area of the spent fuel pool and the washdown pit.

A listing of the MTC/MSB cask drop results is in Table 14.11-1.

Heavy load lifts that have not been analyzed are acceptable due to compliance with the requirements of NUREG-0612 and Generic Letter 85-11 in that the probability for load drops is extremely small.

14.11.2.1 Analysis Method

The previously performed structural evaluation of postulated load drops described within this section only apply if the lifting devices and interfacing lift points do not meet the requirements of NUREG-0612, Section 5.1.6, or if any of the single-failure-proof features of the main hoist are disabled. As discussed in Section 14.11.2, impact limiting pads are not required to reduce the amount of energy transferred to the floors of the cask loading area and the washdown pit for a 25-ton load drop.

14.11.2.1.1 Analysis of Cask Drop Scenarios

EA-FC-864-09 was written to address each of the following drop scenarios: Drop of the Multi-Assembly Sealed Basket Transfer Cask (MTC) into the spent fuel pool, into the washdown pit, and on top of the Ventilated Concrete Cask (VCC) which is on to the Load Distribution System (LDS). This analysis also evaluates whether modifications to existing auxiliary building structures are necessary to facilitate cask loading. The following areas required evaluation:

A - The fuel pool cask loading area in the northeast corner of the spent fuel pool was evaluated to determine whether modifications were needed to prevent damage to the spent fuel pool liner in the event that the loaded MTC falls from the crane.

B - The washdown pit was evaluated to determine whether modifications were needed to prevent damage due to a postulated drop of the loaded MTC in to the washdown pit.

C - The Track Alley was evaluated to determine whether modifications were needed for a drop of the loaded MTC/MSB on to the VCC in the Track Alley. Also, the MTC door rails were evaluated to determine the damage to them as a result of the postulated drop of the loaded MTC on to the VCC.

14.11.2.1.2 Cask Overturn Due to Seismic Event

For a MTC/MSB cask, EA-FC-864-41 was written to address the impact of an earthquake on the MTC/ MSB during the loading of the MSB and whether the cask could tip over and damage fuel in the adjacent fuel racks.

EA-MOD-2003-019-06 evaluated cask overturning due to a seismic event for the Transnuclear 32PT-S125 Dry Storage Canister (DSC) and OS-197 Transfer Cask (TC). The evaluations addressed the necessity of any modifications to prevent a cask from tipping over in the cask loading area and damaging fuel due to an earthquake.

14.11.2.2 <u>Bounding Event Input</u>

14.11.2.2.1 Analysis of Cask Drop Scenarios

The important input parameters used in drop analysis EA-FC-864-09 are:

A - Height of the MTC/MSB above the spent fuel pool water level at 646'= 48". The temperature of the spent fuel pool during fuel loading ≤100°F

B - Height of the MTC/MSB above the washdown pit floor at 634' = 192"

C - Height of the MTC/MSB above the VCC in Track Alley = 70"

The important input parameters called out in EA-FC-864-011 are as follows:

- Drop height = 5.83'
- MTC / MSB weight = 93.5 tons

14.11.2.2.2 Cask Overturn Due to Seismic Event

For a MTC/MSB, the important input parameters used in the drop analysis EA-FC-864-41 are:

The Horizontal acceleration = 0.30g
The Vertical acceleration = 0.14g

For a TC/DSC, the important parameters used in the cask overturn analysis EA-MOD-2003-019-06 are:

The Horizontal acceleration = 0.30g
The Vertical acceleration = 0.136g

14.11.2.3 Analysis Results

14.11.2.3.1 Analysis of Cask Drop Scenarios

The conclusion of EA-FC-864-09 is as follows:

A - The recessed area in the spent fuel pool northeast corner requires an Impact Limiting Pad. This prevents damage to the spent fuel pool structure and liner.

Note: As discussed in Section 14.11.2, since the main hoist is single failure proof, an impact limiting pad is not required provided the lifting devices meet the requirements of NUREG-0612, Section 5.1.6.

B - The washdown pit requires an Impact Limiting Pad. This prevents damage to the washdown pit floor.

Note: As discussed in Section 14.11.2, since the main hoist is single failure proof, an impact limiting pad is not required provided the lifting devices meet the requirements of NUREG-0612, Section 5.1.6.

C - The Load Distribution System will structurally withstand the drop of a loaded Transfer Cask on to the VCC. Local yielding of the MTC door rail as a result of the MTC drop is acceptable. The MTC can still be lifted and placed in the spent fuel pool and the fuel in the MSB unloaded.

The conclusion from EA-FC-864-11, Revision 3, is that the MTC will not drop below the top of the VCC and that the slab at 649'-0" is capable of resisting the impact as long as the VCC is centered $\pm$ 6" with the center of the Track Alley hatch.

14.11.2.3.2 Cask Overturn Due to Seismic Event

For MTC/MSB, the conclusion from EA-FC-864-41 is as follows:

The MTC/MSB fully loaded with fuel or empty will not tip over in the spent fuel pool due to a seismic event of horizontal accelerations of 0.30 g's or less and a vertical acceleration of 0.14 g's or less.

For TC/DSC, the conclusion from EA-MOD-2003-019-06 is as follows:

The TC/DSC fully loaded with fuel or empty will not tip over in the spent fuel pool due to a seismic event of horizontal accelerations of 0.30 g's or less and a vertical acceleration of 0.136 g's or less.

A listing of the cask drop and overturning results is in Table 14.11-1.

14.11.3 RADIOLOGICAL CONSEQUENCES

14.11.3.1 Analysis Method

14.11.3.1.1 Radiological Consequences of a Cask Drop in the Spent Fuel Pool

EA-TAM-96-04 updated the offsite radiological dose consequences resulting from a cask drop in the spent fuel pool as a result of C-PAL-96-0789 and C-PAL-96-0956 (References 8 and 9). These condition reports document the possibility of previously unanalyzed unfiltered leak paths from the fuel pool area ventilation. The offsite doses for EA-TAM-96-04 (Reference 3) were acceptable, however, when the control room doses were determined for the release rates of EA-TAM-96-04, it was determined that the time at which operators must switch the CR-HVAC to emergency mode was too short to be practically achievable. Hence, procedure changes were made to eliminate isolatable leak paths prior to heavy load moves over non-fueled areas of the main fuel pool zone. These procedure changes were accounted for in EA-CDA-98-01, which updated the offsite radiological release rates resulting from a cask drop in the spent fuel pool. In addition, EA-CDA-98-01 incorporated new offsite short-term X/Q's per Reference 11 in the determination of offsite dose consequences.

NAI-1149-026 (Reference 15) updated the offsite and onsite radiological consequences to incorporate the alternative source term methodology (Reference 16).

Three cask drop scenarios were analyzed in NAI-1149-026 to encompass all Cask Drop Scenarios:

1. A cask drop onto 30 day decayed fuel with the Fuel Handling Building (FHB) Charcoal Filter operating with a conservative amount of unfiltered leakage. All "Isolatable Unfiltered Leak Paths" are assumed to be isolated prior to event initiation.

2. A cask drop onto 30 day decayed fuel with the Fuel Handling Building (FHB) Charcoal Filter operating with an increased amount of unfiltered leakage. This scenario increases the amount of leakage that bypasses the FHB filters with respect to scenario #1 in order to provide a sensitivity on filter bypass fraction. This scenario also assumes isolation of all Isolatable Unfiltered Leak Paths prior to event initiation.

3. A cask drop onto 90 day decayed fuel **without** the FHB Charcoal Filter operating. This scenario needs no assumptions as to unfiltered leakage or post-accident unfiltered leak path isolation times, since all radiation is assumed to be released unfiltered from the FHB.

Note: The radiological consequences of a cask drop in the spent fuel pool do not take credit for an Impact Limiting Pad in the spent fuel pool.

14.11.3.1.2 Impact on MTC/MSB Due to Postulated Drop on the VCC in the Track Alley

EA-FC-864-011 analyzes the resultant impact on the MSB from dropping the loaded MTC on to the VCC. The scenario is that the MTC will first impact the VCC and then will topple and strike the adjacent floor at the 649' elevation.

14.11.3.2 Bounding Event Input

14.11.3.2.1 Radiological Consequences of a Cask Drop in the Spent Fuel Pool

The important input parameters used in NAI-1149-026 are given in Tables 14.11-2 and 14.11-3.

For Scenarios 1 & 2, charcoal filter bypasses of 10% and 17.5% for Scenarios 1 and 2 respectively, are assumed to exist for the entire duration of the release. For Scenario 3, 100% charcoal filter bypass is assumed for the entire duration of the release since the charcoal filters are not operating. Note that fuel in the MSB does not fail.

14.11.3.3 Analysis Results

14.11.3.3.1 Radiological Consequences of a Cask Drop in the Spent Fuel Pool

NAI-1149-026 shows that the MTC/MSB drop meets the acceptance criteria for all scenarios. The results of these calculations are listed in Table 14.1-6.

14.11.3.3.2 Drop of the Loaded MTC on to the VCC in the Track Alley

EA-FC-864-011 shows that the MTC/MSB will survive the postulated drop onto the VCC and will also withstand the tipping and coming to rest against the concrete slab at the 649' elevation.

The assessment of the MTC shows that the deflection of the shell, due to impact on the slab at 649'-0", is very small and insignificant. Therefore, the shell stresses are acceptable. No radiation will be released and the MTC/MSB can be placed back into the spent fuel pool and unloaded.

14.11.4 CONCLUSIONS

The structural and radiological evaluations of the impact of dropping the MTC / MSB in the auxiliary building have shown that the postulated drops are within design specifications and regulatory requirements.

REFERENCES

1. BC-TOP-9 Bechtel Topical Report, Design of Structures for Missile Impact, Revision 1, July 1973 (4320/2236), Revision 2, September 1974 (F567/0239).

2. Facility Change FC-864, "Dry Storage of Spent Nuclear Fuel," (G250/0266).

3. EA-TAM-96-04 Rev. 0, "Offsite Radiological Dose Consequences of a Cask Drop in the Spent Fuel Pool", October 1996, (G781/2304).

4. EA-FC-864-09, "MSB Transfer Cask Drop Analysis and Impact Limiter Design, Rev. 7", September 1995, (G707/2215).

5. EA-FC-864-11, "Evaluation of the MTC/MSB drop on the VCC with the MTC/MSB C.G. located outside the VCC boundary, Rev 4", September 1994, (G707/2458).

6. EA-FC-864-11, "Evaluation of the MSB for droploads for a hypothetical drop on the Load Distribution System (LDS) in the Track Alley. Rev. 2" April, 1993, (G253/0188).

7. EA-FC-864-41, "MTC Overturn Calculation in Spent Fuel Pool, Rev 0", April 1993, (G252/0492).

8. C-PAL-96-0789 "Potential Airborne Release in VRS and VRS Barrel Storage Areas," July 18, 1996, (4506/1841).

9. C-PAL-96-0956, "Failure to Follow Intent of Refueling & Heavy Load Movement Technical Specifications," August 21, 1996, (4166/0081).

10. EA-CDA-98-01, Revision 1, "Offsite Radiological Dose Consequences of a Cask Drop in the Spent Fuel Pool," February 7, 2001, (4931/1906).

11. EA-C-PAL-97-0808A, Revision 0, "Determination of Offsite Atmospheric Dispersion Factors (X/Q's) for Radiological Dose Consequence Analyses," June 1998, (4297/1547).

12. Facility Change FC-976, “Modify L-3 to Single Failure Proof and Increase Capacity from 100 Tons to 110 Tons.”

13. Bechtel Report for 25-Ton Cask, “Evaluation of Postulated Cask Drop Accidents,” August 1974, (9436/0118).

14. EA-MOD-2003-019-06, “Evaluation of Decontamination Pit and Spent Fuel Pool Areas for Loads from TN Transfer Cask,” Rev. 2, May 2004.

15. NAI-1149-026, Revision 0, “Palisades Design Basis Cask Drop Accident AST Radiological Analysis,” June 2005.

16. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, July 2000.

14.12 LOSS OF EXTERNAL LOAD

14.12.1 EVENT DESCRIPTION

A Loss of External Load event is initiated by either a loss of external electrical load or a turbine trip. Upon either of these two conditions, the turbine stop valve is assumed to rapidly close. Normally, a reactor trip would occur on a turbine trip; however, to calculate a conservative system response, the reactor trip on turbine trip is disabled. The steam dump system (atmospheric dump valves - ADV's) is assumed to be unavailable. These assumptions allow the Loss of External Load event to bound the consequences of SRP Event 15.2.2 (Turbine Trip) and SRP Event 15.2.4 (Closure of both MSIV's).

The Loss of External Load event primarily challenges the acceptance criteria for both primary and secondary system pressurization and DNBR. The event results in an increase in the primary system temperatures due to an increase in the secondary side temperature. As the primary system temperatures increase, the coolant expands into the pressurizer causing an increase in the pressurizer pressure. The primary system is protected against overpressurization by the pressurizer safety and relief valves. Pressure relief on the secondary side is afforded by the steam line safety/relief valves. Actuation of the primary and secondary system safety valves limits the magnitude of the primary system temperature and pressure increase.

This event is analyzed conservatively assuming a positive BOC moderator temperature coefficient, increasing primary system temperatures result in an increase in core power. The increasing primary side temperatures and power reduces the margin to thermal limits (i.e., DNBR limits) and challenges the DNBR acceptance criteria.

14.12.2 THERMAL-HYDRAULIC ANALYSIS

14.12.2.1 Analysis Method

This event is analyzed with the ANF-RELAP computer program (Reference 2) as described in References 1, 4, 5 and 7.

A loss of load event also challenges thermal margin limits. This subevent was dispositioned as being bounded by other more limiting AOO events. Thus, the DNBR for this event is not evaluated.

14.12.2.2 Bounding Event Input

The objectives in analyzing this event are to demonstrate that the primary pressure relief capacity is sufficient to limit the pressure to less than 110% (2750 psia) of the design pressure and that the secondary side pressure relief capacity is capable of limiting the pressure to less than 110% (1100 psia) of design pressure. A steam generator tube plugging level of 25% is assumed for the analysis. No credit is taken for direct reactor trip on turbine trip, the turbine bypass system or the steam dump system. Also, credit from the pressurizer PORV's is conservatively excluded from the Reference 1 analysis, but included in the Reference 5 analysis. The analysis used an initial indicated pressurizer level of 67.8%. This value was generated in Reference 4 and corresponds to the HFP high level alarm setpoint of 62.8% +5% for instrument uncertainty. In general, the parameters and equipment operational states are selected to maximize the system pressure.

The pressurizer safety valve setpoints used in the analysis are as follows: 2657.4 psia for RV-1039, 2616.2 psia for RV-1040 and 2575.0 psia for RV-1041. These setpoints include a 3% tolerance. The rated capacity used in the analysis for each pressurizer safety valve is 230,000 lb/hr at 2575.0 psia plus 3% accumulation.

The secondary safety valves in the analysis are split into groups of eight valves at each of three setpoints: 1030.0 psia, 1050.6 psia, and 1071.2 psia. A 3% tolerance is included in these setpoints. The rated capacity used in the Reference 1 analysis for each of these valves is 486,600 lb/hr at 1030.0 psia plus 3% accumulation. The Reference 5 secondary side overpressurization case used slightly higher capacities and slightly different setpoints for these valves.

The primary and secondary safety valves are modeled as achieving the fully opened position upon reaching the setpoint pressure. Full relieving capacity is reached within the rated accumulation above the setpoint pressure. Blowdown is conservatively ignored in the analysis.

The Loss of External Load is credible only for rated power and power operation events because there is no load on the turbine at other reactor conditions. At rated power conditions, the moderator temperature coefficient is negative. The rated power conditions bound the consequences for other reactor power operating conditions because of the increased stored energy. The higher the stored energy in the primary system, the more severe the consequences of this event.

14.12.2.3 Analysis of Results

The maximum primary pressurization case initiates with a rapid closure of the turbine stop valve in 0.1 seconds. The value of 0.1 seconds was chosen to simulate instantaneous closure of the valve. Even if the valve closure time were less than 0.1 seconds the plant response to this event would not change.

Steam line pressure increases until the relief valves open. The secondary side pressure relief valves contain sufficient capacity to limit the secondary pressure to less than 110% (1100 psia) of design pressure. The pressurization of the secondary side results in decreased primary to secondary heat transfer, and a substantial rise in primary system temperature. The primary coolant temperature increases which results in a large insurge into the pressurizer, compressing the steam space and pressurizing the primary system. The reactor trips on high pressure and the pressurizer safety valves open. The increase in coolant temperature also causes the core power to rise due to positive moderator feedback. The transient is terminated shortly after reactor scram due to decreasing primary coolant temperature and pressure.

The capacity of one valve is enough to contain the pressurizer pressure to a value less than the 2750 psia limit. The responses of key system variables from the Reference 7 analysis are given in Figures 14.12-1 to 14.12-5. Table 14.12-1 lists the sequence of events for this transient.

The References 1 and 7 analyses evaluated the maximum primary system over-pressure assuming all MSSVs were operational. Palisades Technical Specifications allow one MSSV to be inoperable. An evaluation of the impact of operation with an inoperable MSSV (Reference 6) confirmed that it would not have a significant impact on the primary system pressure for this event.

Secondary side pressurization was analyzed for the Loss of External Load event subsequent to the Reference 1 analysis. This analysis (Reference 5) was performed assuming one of the MSSVs to be inoperable. The event summary for the secondary side over-pressurization analysis is presented in Table 14.12-1.

14.12.3 RADIOLOGICAL CONSEQUENCES

Not required for this event.

14.12.4 CONCLUSIONS

The MDNBR for the Loss of External Load event is bounded by other more limiting events that have been shown to meet acceptance criteria. Thus, the DNB SAFDL is not penetrated for this event. This event does not pose a significant challenge to the FCM criterion (Reference 7). The maximum pressurizer and secondary side pressure remains below 110% of design pressure. Applicable acceptance criteria for the event are therefore met.

REFERENCES

1. "Palisades Loss of Load Analysis," EMF-93-086(P), Siemens Power Corporation, April 1993, (F853/0165).

2. "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," ANF-89-151(NP)(A), Advanced Nuclear Fuels Corporation, May 1992, (F859/2215).

3. ANF-87-150(NP), Volume 2, Palisades Modified Reactor Protection System Report: Analysis of Chapter 15 Events, Advanced Nuclear Fuels Corporation, June 1988, (C029/1767).

4. EA-DDC-93-001, "Development of Palisades Plant Pressurizer Liquid Volume as a Function of Indicated Level to Support Loss of External Load Analysis Initial Conditions Input," March 1993. (F524/2429)

5. EMF-98-013, Revision 2, "Palisades Cycle 14: Disposition and Analysis of Standard Review Plan Chapter 15 Events," Siemens Power Corporation, January 1999.

6. Letter, H. G. Shaw to R. J. Gerling, "Impact of an Inoperable Secondary Valve on the Palisades Loss of Load Analysis," HGS:268:93, July 26, 1993.

7. EMF-2507, Revision 0, “Palisades Cycle 16 Safety Analysis Report,” Framatome ANP Richland, Inc, March 2001.

14.13 LOSS OF NORMAL FEEDWATER

14.13.1 EVENT DESCRIPTION

A Loss of Normal Feedwater Flow (LNFF) event is initiated by the trip of the Main Feedwater (MFW) pumps or a malfunction in the feedwater control valves. The loss of MFW flow decreases the amount of subcooling in the secondary side downcomer leading to an increase in the primary coolant system (PCS) temperature. As the PCS temperature increases, the coolant expands into the pressurizer which increases the pressure by compressing the steam volume.

Steam generator liquid levels, which steadily drop after termination of MFW flow, soon reach the low steam generator level reactor trip setpoint and the low steam generator level Auxiliary Feedwater (AFW) actuation setpoint. This initiates the starting sequence for the AFW pumps and initiates a reactor scram, which ends the short-term-heatup phase of the event. When the delivery of AFW begins, the rate of level decrease in the steam generators slows.

The automatic turbine trip at reactor scram and the continuing primary-to-secondary transfer of the decaying core power and the reactor coolant pump heat cause steam generator pressures to rapidly increase. When steam generator pressures and coolant temperatures have increased to the appropriate values, the steam dump system and/or the Main Steam Safety Valves (MSSVs) serve to limit the increase in steam generator pressures.

Eventually, a long-term-heatup phase of the event may begin if primary-to-secondary heat transfer degrades as a result of steam generator tube uncovery.

As the decay heat level drops, liquid levels in the fed steam generators stabilize and then begin to rise. Also, reactor coolant temperatures stabilize and then begin to decrease. These conditions mark the end of the challenge to the event acceptance criteria.

The non-safety grade Main Steam Atmospheric Dump and Turbine Bypass systems are usually disabled for licensing basis analyses. However, the Palisades licensing basis allows credit for Atmospheric Dump Valve (ADV) operation for the LNFF event as discussed in FSAR Section 9.7.3. As a result, enabling the steam dump system can potentially produce the bounding minimum steam generator level early in the event. While AFW flow is controlled by flow control valves, excessive differential pressure between AFW pump discharge and the steam generators can limit flow. When ADVs are credited for the LNFF event, they maintain steam generator pressure low, providing for more AFW flow from AFW Pump P-8C. However, at reactor trip, the ADVs rapidly cool the primary coolant system. This rapid heat removal is accomplished by boil-off of steam generator liquid inventory, increasing the potential for an early steam generator dry-out early in the event, before the longer term effect of increased integrated auxiliary feedwater flow can preclude steam generator dryout.

The short-term impacts of the Loss of Normal Feedwater Flow event challenges the DNB and the primary system overpressurization acceptance criteria. The Loss of Forced Reactor Coolant Flow event (Event 15.3.1) (FSAR Section 14.7.1) bounds the short-term DNB consequences of a Loss of Normal Feedwater Flow transient. After the reactor trip system is activated, the core power is drastically reduced, alleviating the challenge to DNB. The Loss of External Load event (Event 15.2.1) (FSAR Section 14.12) bounds the short-term primary system overpressurization challenge of this event.

The long-term effects of this event primarily challenge the pressurization limits of the primary system due to the filling of the pressurizer and steam generator dryout. If the pressurizer were to fill completely solid with liquid, the primary system pressure control would be lost and primary liquid would be expelled through the pressurizer safety valves. The dryout of a steam generator causes the loss of a primary-to-secondary system heat sink exacerbating the primary-side heatup. The long-term consequences of a Loss of Normal Feedwater Flow event were analyzed in Reference 1. While the steam generators are designed to withstand the thermal loading imposed by a total loss of water inventory and subsequent refill transient (FSAR Section 4.3.4), the LNFF analysis does not credit this capability. Because of concern that steam generator dryout could lead to adverse effects on specified acceptable fuel design limits, the LNFF analysis conservatively uses an acceptance criterion of no steam generator dryout during the LNFF event.

14.13.2 THERMAL-HYDRAULIC ANALYSIS

14.13.2.1 Analysis Method

The analysis was performed using the S-RELAP5 Non-LOCA transient methodology described in Reference 2. The S-RELAP5 code was used to model the reactor system thermal and hydraulic responses to demonstrate that acceptance criteria are satisfied.

14.13.2.2 Bounding Event Input

Event Classification

The LNFF event is classified as an ANS Condition II event, Faults of Moderate Frequency. The LNFF event is further classified by the NRC Standard Review Plan (SRP) (Reference 3), as a Decrease in Heat Removal by the Secondary System event.

Acceptance Criteria

The following acceptance criteria are addressed in this analysis.

1. Pressure in the PCS and main steam system should be maintained below 110% of the design values. PCS overpressure criteria is satisfied by demonstrating that the pressurizer does not over-fill or reach a solid condition.

2. Fuel cladding integrity should be maintained by ensuring that the minimum departure from nucleate boiling ratio (MDNBR) remains above the 95/95 DNBR limit and that fuel centerline melt does not occur. These criteria are satisfied by demonstrating that the steam generators do not dry out.

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. This criteria is satisfied by demonstrating that the pressurizer does not over-fill.

If it can be demonstrated that the pressurizer does not fill solid during the LNFF event, the Loss of External Load (LOEL) analysis (FSAR Section 14.12) is also the bounding event for PCS pressurization (Acceptance Criterion 1). The biases applied in the LOEL analysis ensure that PCS pressure is maximized. The rapid pre-trip heatup associated with the LOEL event causes significantly higher primary system pressure than the slower developing LNFF scenario.

Description of Analysis Cases

At least two separate cases are required to ensure compliance with the above criteria. The principal difference is the effect of a Primary Coolant Pump (PCP) trip on the transient results. If the PCPs remain on, the pump heat imposes a significant heat load on the system. If the PCPs are tripped (due to a loss of offsite power), primary-to-secondary heat removal is dependent upon natural circulation. The two cases collectively demonstrate compliance with the acceptance criteria. Because the Palisades licensing basis allows credit for ADV operation for the LNFF event, a third case analyzes the actuation of the ADV to examine the trade-off between increased inventory loss and increased AFW flow from the lower generator pressure. The biases and initial conditions for the cases are selected to maximize the pressurizer level increase and to minimize the steam generator liquid inventory.

The Palisades AFW system consists of two independent motor-driven pumps (P-8A and P-8C) and a steam-driven AFW pump (P-8B). When the setpoint for the AFAS is received, P-8A is started. If it fails to start, P-8C is started. No credit is taken for the steam-driven AFW pump P-8B. The piping configuration allows each AFW pump to supply both steam generators simultaneously. The AFW system can provide a minimum of 135 gpm to each generator within 120 seconds after the Auxiliary Feedwater Actuation Signal (AFAS) occurs.

The LNFF event is analyzed from full power (plus uncertainty) conditions. This is the bounding case for initial power level. The full power initial condition maximizes the core decay heat that must be removed in the post-scram period. A primary concern in simulating this event is to demonstrate adequate long-term cooling capability.

Initial Conditions

Table 14.13-1 lists the initial conditions for the steady state S-RELAP5 model. These conditions are the plant conditions immediately preceding the LNFF event. This event is not sensitive to the level of steam generator tube plugging and no biasing is required by the Reference 2 methodology. Zero steam generator tube plugging was assumed for each case analyzed.

Discussion of Operator Action Assumptions

For those cases with offsite power available, an operator action is credited 1,200 seconds (20 minutes) into the event to adjust the AFW controller to obtain the maximum AFW flow. Furthermore, at 1,500 seconds (25 minutes) into the event the operators are assumed to trip all of the PCPs. No operator action is assumed for the case with a loss of offsite power.

In the LNFF event, the critical parameter for the operators is steam generator water level. The operators mitigate the event by minimizing inventory loss and maximizing water addition to the steam generators. Actions to minimize inventory loss include

- tripping primary coolant pumps (elimination of pump heat that must be removed from the primary coolant)
- isolating steam generator blowdown

Actions to maximize water addition to the steam generators include

- manually controlling the AFW flow controllers to full open
- reducing steam generator pressure to permit more flow from the centrifugal AFW pumps
- starting P-8B, the turbine driven AFW pump.

The operator actions assumed in the accident analysis and their timing were based on maintaining primary-to-secondary heat transfer such that the plant can be stabilized and brought to a safe shutdown in a controlled manner. If the operators perform any of the emergency operating procedure mitigating actions in such a manner as to prevent steam generator dry-out, the accident analysis is satisfied.

In a Loss of Feedwater event, the operators are directed by emergency operating procedures to initiate Once Through Cooling (OTC) when loss of steam generator heat sink is indicated. Once Through Cooling is a success path not credited in the LNFF analysis. Successful completion of OTC also satisfies the accident analysis.

14.13.2.3 Analysis of Results

Off-Site Power Available and Steam Dump System Disabled (Limiting Relative to Maximum Pressurizer Level)

Figures 14.13-1 through 14.13-10 show the transient response of the primary and secondary systems. Table 14.13-2 presents the sequence of events for the transient.

Following the loss of feedwater flow, there is a mismatch between primary heat generation by the core and PCPs and heat removal by the secondary system. This causes a primary side heatup. The resultant coolant thermal expansion causes an insurge into the pressurizer which activates the spray flow for a short period of time. AFAS on low steam generator level is reached at 22 seconds. A reactor trip on low steam generator water level occurs after a 0.8 second delay at 22.8 seconds and control rods begin to drop at 23.3 seconds. The steam generator liquid mass inventory at the time of reactor trip is approximately 86,400 lbm. The turbine trips automatically following the reactor trip, closing the turbine stop valves. The MSSVs open briefly to relieve the secondary pressure increase caused by the turbine trip. Then the lowest setpoint bank of MSSVs cycle open and closed to maintain steam pressure at approximately 1,000 psia. This steam relief through the MSSVs controls PCS temperatures as the remaining steam generator inventory continues to boil off.

The reactor trip initiates a temporary reduction in PCS temperature and coolant thermal contraction causes a decrease in pressurizer level and pressure.

A single motor-driven AFW pump starts at 142.0 seconds (120 seconds after the AFAS signal at 22.0 seconds). The cooling capability of the AFW system is initially constrained by the physical requirement to purge stagnant water that is at or near the initial steam generator saturation temperature from a section of the AFW piping. The purge volume is cleared in approximately 3 seconds at 145 seconds.

A steady decrease in steam generator inventory continues after AFW actuation. At 1,500 seconds, the PCPs are turned off, resulting in a primary heatup that establishes a natural circulation flow in the primary system. Since the Pilot Operated Relief Valves (PORVs) are blocked, the maximum primary pressure is limited by the actuation of the pressurizer safety relief valves (SRVs). The maximum pressurizer level reached is 65.4% at 26 seconds. The pressurizer level rose again to 65.3% at 3,284 seconds due to an increase in PCS temperature that causes expansion of primary coolant. The minimum steam generator liquid mass inventory is reached at 4,318 seconds in steam generator B (9,885.8 lbm), after which steam generator inventory begins to increase. By 5,000 seconds, steam generator inventory is increasing gradually and the AFW flow is sufficient to ensure continued cooling of the PCS. PCS temperatures remain nearly constant after this time, verifying that the decay heat can be removed.

The results from this case verify that there is adequate capacity to remove PCS sensible heat, PCP heat, and core decay heat from the primary system and maintain subcooling margin in the PCS. Additionally, the AFW system and the credited operator response times are adequate to prevent steam generator dryout and to prevent pressurizer over-fill.

Off-Site Power Available with Steam Dump System Available (Limiting Relative to Minimum Steam Generator Level)

Palisades is equipped with four ADVs (two per steam generator) and one turbine bypass valve (TBV). The ADVs are located on the main steam headers upstream of the main steam isolation valves (MSIVs) and direct steam to the atmosphere. The TBV is connected to the Main Steam system between the MSIVs and the turbine control valves and dumps steam into the condenser.

Figures 14.13-11 through 14.13-20 show the transient response for the primary and secondary systems. Table 14.13-3 presents the sequence of events for the transient.

Following the loss of feedwater flow, there is a mismatch between primary heat generation by the core and PCPs and heat removal by the secondary system. This causes a primary side heatup. The resultant coolant thermal expansion causes an insurge into the pressurizer but does not activate the spray flow. AFAS on low steam generator level occurs at 21.95 seconds. A reactor trip on low steam generator water level occurs after a 0.8 second delay at 22.75 seconds and the control rods begin to drop at 23.28 seconds. The steam generator liquid mass inventory at the time of reactor trip is approximately 86,400 lbm. The turbine trips automatically following the reactor trip, closing the turbine stop valves. The ADVs and TBV open to maintain steam pressure at approximately 900 psia. This pressure drops slowly as the remaining steam generator inventory continues to boil off.

The reactor trip initiates a temporary reduction in PCS temperature and coolant thermal contraction causes a decrease in pressurizer level and pressure.

A single motor-driven AFW pump starts at 141.95 seconds (120 seconds after the AFAS signal at 21.95 seconds). The cooling capability of the AFW system is initially constrained by the physical requirement to purge stagnant water that is at or near the initial steam generator saturation temperature from a section of the AFW piping. The purge volume is cleared in approximately 3 seconds at 145 seconds.

A steady decrease in steam generator inventory continues after AFW actuation. At 1,500 seconds, the PCPs are turned off, resulting in a primary heatup that establishes a natural circulation flow in the primary system. The ADVs and TBV continue to maintain average coolant temperature within the control band causing a decrease in the generator pressure, thus lowering core inlet temperature. Since there is no primary heatup, the primary pressure does not rise and there is no pressurizer insurge. The maximum pressurizer level reached is 64.7% at 26 seconds, just after the reactor trip. The minimum steam generator liquid mass inventory is reached at 1,708 seconds in steam generator A (9,605.8 lbm), after which steam generator inventory begins to increase. By 2,000 seconds, steam generator inventory is increasing appreciably and it is evident that AFW flow is sufficient to ensure continued cooling of the PCS. PCS temperatures remain nearly constant after this time, verifying that stable progress towards safe shutdown is occurring.

The results from this case are more severe in terms of minimum generator liquid inventory, but less severe in terms of maximum pressurizer level than in the previous case which did not model the ADVs and TBV. The actuation of these components lowers the generator pressure sufficiently to significantly increase AFW flow after the controller is run up to the maximum setting.

Off-Site Power Unavailable and Steam Dump System Disabled (Limiting Relative to Maximum Pressurizer Level)

This case was performed to conservatively bound SRP (Reference 3) Event 15.2.6, Loss of Nonemergency AC Power. An instantaneous loss of all MFW flow and a Loss of Off-Site Power (LOOP) coincident with reactor scram on low steam generator level initiate the event, and the plant is conservatively assumed to maintain a 100% (plus uncertainty) power steam demand until reactor trip on low steam generator level. Figures 14.13-21 through 14.13-28 show the transient response for the primary and secondary system responses. Table 14.13-4 presents the sequence of events for the transient.

Following the loss of feedwater flow, there is a mismatch between primary heat generation by the core and heat removal by the secondary system. This causes a short-lived primary side heatup. The resultant coolant thermal expansion causes an insurge into the pressurizer which activates the spray flow for a short period of time. AFAS on low steam generator level is reached at 22.0 seconds. Reactor trip on low steam generator water level occurs after 0.8 seconds delay at 22.8 seconds and the control rods begin to drop at 23.3 seconds. The steam generator liquid mass inventory at the time of reactor trip is approximately 86,400 lbm. The PCPs are automatically tripped at 33.8 seconds coincident with reactor trip following an 11 second generator-assisted coastdown. The turbine trips automatically following the reactor trip, closing the turbine stop valves. The MSSVs open briefly to relieve the secondary pressure increase caused by the turbine trip, then the lowest setpoint bank of MSSVs cycle open and closed to maintain steam pressure at approximately 1,000 psia. This steam relief through the MSSVs controls PCS temperatures as the remaining steam generator inventory continues to boil off.

The reactor trip initiates a temporary reduction in PCS temperature and coolant thermal contraction causes a decrease in pressurizer level and pressure. The temperatures, pressure and level decrease to a quasi-steady state as the natural circulation flow is established. These all continue to decrease throughout the transient as decay heat falls. The maximum pressurizer level reached is 65.4% at 26 seconds, just after the reactor trip.

A single motor-driven AFW pump starts at 142.0 seconds (120 seconds after the AFAS signal at 22.0 seconds). The cooling capability of the AFW system is initially constrained by the physical requirement to purge stagnant water that is at or near the initial steam generator saturation temperature from a section of the AFW piping. The purge volume is cleared in approximately 3 seconds at 145 seconds.

A slow decrease in steam generator inventory continues after AFW actuation. The minimum steam generator liquid mass inventory is reached at 6,674 seconds in steam generator A (11,851 lbm), after which steam generator inventory begins to increase slowly. By 8,000 seconds, steam generator inventory is increasing steadily and the AFW flow is sufficient to ensure continued cooling of the PCS. PCS temperatures remain nearly constant or are decreasing after this time, verifying that stable progress towards safe shutdown is occurring.

The results from this case are less severe in terms of minimum generator inventory than in the offsite-power-available cases. It is, however, as severe in terms of maximum pressurizer level as the offsite-power-available case with steam dumps disabled. In addition, no operator action to increase the AFW flow controller setting is credited in this analysis. The lack of PCP heat as well as removal of less sensible heat from the PCS makes this transient less severe than the other cases.

Conclusion

The capacity of AFW system and associated reactor trip and AFW actuation setpoints were shown to maintain primary-to-secondary heat transfer such that the decay heat can be removed. For all cases analyzed, there was no steam generator dryout or significant heatup of the primary system and pressurizer over-fill did not occur. Natural circulation flow is established to remove heat form the primary-to-secondary by natural convection as long as the steam generators are being fed after the loss of forced primary coolant flow.

Analysis results are compiled in Table 14.13-5.

14.13.3 RADIOLOGICAL CONSEQUENCES

Not required for this event.

14.13.4 CONCLUSIONS

The results verify that with a minimum of 135 gpm of auxiliary feedwater delivery to each steam generator at 120 seconds after the auxiliary feedwater actuation signal on low steam generator level, and increasing the flow controller to a maximum setting at 20 minutes after the loss of feedwater (plus the reactor coolant pump trip at 25 minutes) there is adequate capacity to remove PCS sensible heat, reactor coolant pump heat, and core decay heat from the primary system and maintain the primary-to-secondary heat sink. Additionally, the AFW System and the credited operator response times are adequate to prevent pressurizer level from exceeding allowable limits. While the LNFF analysis credits certain operator actions, any operator action that prevents steam generator dryout satisfies the accident analysis.

In a Loss of Feedwater event, the operators are directed by the emergency operating procedures to initiate Once Through Cooling (OTC) when loss of steam generator heat sink is indicated. Once Through Cooling is a success path not credited in the safety analysis. Successful completion of OTC also satisfies the accident analysis.

A Loss of Normal Feedwater event does not result in the violation of SAFDLs, peak pressurizer pressure does not exceed 110% of the design rating, and primary liquid is not expelled through the pressurizer safety valves. Adequate cooling water is supplied by the auxiliary feedwater system to allow a safe and orderly plant shutdown and to prevent steam generator dryout, assuming minimum auxiliary feedwater capacity.

REFERENCES

1. EMF-2845, Revision 0, Palisades Loss of Normal Feedwater Flow Analysis, Framatome ANP, January 2003.

2. EMF-2310(P)(A), Revision 0, SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors, Framatome ANP, May 2001.

3. NUREG-0800, LWR Edition, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, July 1981.

14.14 STEAM LINE RUPTURE INCIDENT

14.14.1 EVENT DESCRIPTION

The Steam Line Break event is initiated by a postulated double-ended guillotine break in a main steam line[a]. This leads to an uncontrolled release of steam from the steam system.

The resultant pressurization of the containment system (if the break is located inside the reactor containment) or depressurization of the steam system causes the Main Steam Isolation Valves (MSIVs) (See Section 10.2) to close and, if the plant is operating at power when the event is initiated, causes the reactor to be scrammed.

If the break is located between the steam generator outlet nozzle and the MSIV, the affected steam generator will continue to blow down after the MSIVs close. However, closure of the MSIVs will terminate the blowdown of the unaffected steam generator[b].

The increase in energy removal through the steam system when the break occurs results in a severe overcooling of the PCS. In conjunction with a negative Moderator Temperature Coefficient (MTC), this cooldown causes positive reactivity to be inserted. If the plant is operating at End-of-Cycle (EOC) conditions (with a large negative MTC) when the event is initiated, the scram negative reactivity can be overcome, and the reactor can return to power. This return to power and the high power peaking factors associated with the post-scram control rod insertion configuration which must be assumed for this event (i.e., the most reactive control rod stuck in the fully withdrawn position and the other control rods fully inserted) can lead to fuel centerline melting and/or Departure from Nucleate Boiling (DNB).

The power excursion is countered by Doppler reactivity feedback and is eventually terminated by either (1) borated water reaching the core (if the PCS depressurizes sufficiently below the High Pressure Safety Injection (HPSI) pump shutoff head during the event to purge the Safety Injection [SI] lines) or (2) moderator reactivity feedback (when the affected steam generator dries out and the Primary Coolant System (PCS) temperatures begin to increase).

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[a] The fastest blowdown, and therefore the most rapid reactivity addition, occurs when the break is at a steam generator nozzle. Inadvertent opening of valves in the main steam system is discussed in Section 14.10.

[b] Before the MSIVs close, steam flows from the unaffected steam generator through the unbroken main steam line to the turbine header and from there through the broken main steam line (in the reverse-of-normal flow direction) to the break.

In the event of a rupture in the main steam system, the continued integrity of the Primary Coolant System barrier is assured since the steam generators are designed to withstand the pressure differential of Primary Coolant System operating pressure and atmospheric pressure. The minimum allowed steam generator tube wall thickness is determined using the Knight's criteria (Reference 1).

The containment building response to a main steam line break inside the building is discussed in Section 14.18.

14.14.2 THERMAL-HYDRAULIC ANALYSIS

14.14.2.1 <u>Analysis Method</u>

In accordance with the Framatome ANP Non-LOCA Transient / Steam Line Break methodology (Reference 2), the S-RELAP5 plant transient thermal-hydraulic code has been used to calculate the plant response to the event for several bounding cases, based on detailed hydraulic models of the PCS and steam system and a point kinetics model of the core. Asymmetric thermal-hydraulic and related reactivity feedback effects have been accounted for by dividing the S-RELAP5 model of the core into an affected sector and an unaffected sector. Also, the reactor vessel upper head region has been modeled in such a way so as to promote flashing in that region as the PCS depressurizes, thus retarding the depressurization and delaying and minimizing the injection of borated water.

Based on the overall core conditions (i.e., total power, inlet temperature and flow distributions, and outlet pressure distribution) calculated by S-RELAP5 for each case at the post-scram point in time most limiting for the fuel Linear Heat Rate (LHR) and DNB Ratio (DNBR), the PRISM core neutronics code (Reference 3) and the XCOBRA-IIIC core thermal-hydraulic code (Reference 4) have been run in a coupled mode to iteratively calculate consistent core power peaking and coolant temperature and flow distributions for that point in time. The peak LHR has been evaluated using the S-RELAP5-calculated core power and the PRISM-calculated maximum local peaking factor to determine the margin to fuel centerline melting. The Minimum DNB Ratio (MDNBR), based on the Modified Barnett correlation (Reference 6), has been evaluated using the S-RELAP5-calculated core power, PRISM-calculated radial and axial peaking factors for the hot assembly, and XCOBRA-IIIC-calculated temperature and flow distributions for the hot assembly to determine the margin to DNB. *(Also, the PRISM-calculated change in reactivity from transient initiation to the post-scram LHR-limiting and DNBR-limiting point in time has been compared with that calculated by S-RELAP5 to demonstrate that the S-RELAP5 reactivity modeling is conservative.)*

Reference 4 is not specifically listed as an approved methodology in Technical Specification Section 5.6.5. However, it is referenced by References 2 and 8, which are listed in Section 5.6.5 of the Palisades Technical Specifications. Therefore, Reference 4 is incorporated by reference into the Palisades Technical Specifications.

14.14.2.2 Bounding Event Input

The following cases have been evaluated (Reference 10):

- Hot Full Power (HFP), offsite power available, inside containment break
- HZP, offsite power lost when break occurs, inside containment break
- Hot Zero Power (HZP), offsite power available, inside containment break
- HZP, offsite power available, inside containment break

The results for these cases bound the results for cases with other possible initial conditions, PCS flow conditions and break locations.[c]

The most limiting Linear Heat Rate (LHR) case is a break inside containment at Hot Zero Power (HZP) with offsite power available throughout the transient. The most limiting Departure from Nucleate Boiling Ratio case is a break inside containment at HZP with a coincident loss of offsite power.

The single failure assumed in the engineered safeguards system is the failure of a High Pressure Safety Injection (HPSI) pump. Theoretically, in the loss-of-offsite-power cases, the most limiting single failure could be considered to be failure of an emergency diesel generator to start. However, the only engineered safeguards equipment credited for MSLB mitigation is a HPSI pump. Whether an emergency diesel is assumed to fail or operate, the effect on credited engineered safeguards equipment is unchanged.

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[c] Outside-containment-break cases, with non-harsh-condition Engineered Safety Feature (ESF) and Reactor Protection System (RPS) setpoints, have less limiting thermal-hydraulic results and have not been evaluated in this analysis.

The Palisades Plant employs a swing disc stop valve in each main steam line as the main steam isolation valve. Failure of the MSIV in the unbroken line to close could allow blowdown of both generators via reverse flow through the swing disc MSIV check valve in the broken line for a break upstream of the MSIV. This failure has not been considered in this analysis. Another single failure not considered in the analysis is a failure of a main feedwater isolation valve or feedwater bypass valve to close. In Reference 12, CP Co transmitted a Probabilistic Risk Assessment (PRA) and a cost benefit evaluation for modifications to prevent these single failures. In an SER dated February 28, 1986 (Reference 5), the NRC found that a double steam generator blowdown or a single steam generator blowdown with continued feedwater, although more severe than the licensing basis MSLB, is not expected to result in unacceptable consequences. The risk evaluation determined that the potential offsite consequences are low and that the proposed modifications would not provide a substantial improvement in plant safety. These potential single failures, therefore, are not required to be considered in this analysis.

Key input parameters and assumptions of the analysis are summarized in Table 14.14-1.

14.14.2.3 Analysis of Results

LHR-Limiting Case

The LHR-limiting case is initiated by an inside-containment break at HZP and has offsite power available throughout the transient. *(See Table 14.14-2 for a sequential list of key events and the times when they occur during the transient.)*

The steam flow through the break from the affected steam generator (see Figure 14.14-1) and the pressure and heat transfer rate in the affected steam generator (see Figure 14.14-2 and Figure 14.14-3) decay initially and then equilibrate (until the affected steam generator begins to dry out), as the affected steam generator blows down. The steam flow through the break from the unaffected steam generator (by backflow from the turbine header through the affected main steam line) is terminated when the MSIVs close (on a Low Steam Generator Pressure ESF signal[d]) at 19.8 seconds (6.0 seconds after reaching the Main Steam Isolation Signal [MSIS] setpoint).

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[d] A High Containment Pressure ESF signal would have initiated steam generator isolation sooner than the Low Steam Generator Pressure signal, but High Containment Pressure signals have not been credited.

The affected steam generator's secondary-side total fluid inventory (see Figure 14.14-4) is governed by the steam flowing from it to the break and by its feedwater delivery. The unaffected steam generator's secondary-side total fluid inventory prior to MSIV closure is governed by the steam flowing from it to the break.

The release of high-energy steam through the break causes the primary coolant to cool down (see Figure 14.14-5). The cooldown of the loop with the unaffected steam generator ends when the MSIVs close, but the cooldown of the loop with the affected steam generator continues until the affected steam generator begins to dry out. *(The PCS cooldown is also influenced by the PCS flow conditions. With offsite power available throughout the transient, forced flow conditions are maintained in the PCS [see Figure 14.14-6].)*

As the primary coolant contracts, particularly during the initial phase of the cooldown, the pressurizer pressure (see Figure 14.14-7) and liquid level (see Figure 14.14-8) drop. When the primary coolant pressure reaches the saturation pressure, coolant in the reactor vessel upper head begins to flash and thereby stabilizes the primary coolant pressure at the saturation pressure. In accordance with the worst-single-active-failure analysis requirement, it is postulated that one of the two HPSI pumps required to be in service fails. The other HPSI pump starts (on a Low Pressurizer Pressure ESF signal), reaches full speed at 50.4 seconds (30.0 seconds after reaching the Safety Injection actuation signal [SIAS] setpoint), and begins to fill the SI lines with borated water (see Figure 14.14-9). At 158.7 seconds, borated water has filled the SI lines and begins to enter the PCS cold legs. By 160.0 seconds, the borated water front has passed through the core.

Just after the transient is initiated, the reactor is at an all-rods-in-except-most-reactive-rod HZP condition (see Figure 14.14-10). However, as the PCS cooldown progresses, the shutdown worth is eroded by moderator and Doppler feedback. which is accentuated at the EOC conditions assumed for the analysis. At 32.2 seconds the shutdown worth has been fully overcome by moderator and Doppler feedback.

As the shutdown worth is eroded by moderator and Doppler feedback, the reactor power begins to increase above the decay power level (see Figure 14.14-11), but this post-scram power increase is countered by negative Doppler feedback (as the reactor power increases) and borated water injection (after the SI lines have been purged) and is eventually terminated by negative moderator feedback (after the affected steam generator begins to dry out). At 158.0 seconds the reactor power peaks at 26.92% of the rated power with most of the power produced in the stuck rod region. *(See Table 14.14-3 for a list of the overall core conditions at the time of the peak LHR.)*[e] No fuel failures are predicted to occur as a result of excessive LHR (Fuel Centerline Melt).

DNBR-Limiting Case

The DNBR-limiting case is initiated by an inside-containment break at HZP and has a coincident loss of offsite power. *(See Table 14.14-4 for a sequential list of key events and the times when they occur during the transient.)*

The PCS and steam system responses for this case (see Figure 14.14-12 through Figure 14.14-22) generally resemble those for the corresponding pumps-on case, but the timing and magnitudes of the responses are affected by the PCS natural-convection flow conditions.

The degraded steam generator primary-side heat transfer coefficients result in reduced primary-to-secondary heat transfer rates which, in turn, lead to reduced steam generator pressures, reduced break flow rates, and a delayed time of steam generator dryout. The reduced steam generator pressures result in reduced PCS temperatures, but this happens later in the transient because of the reduced PCS flow rates. Moderator reactivity and the reactor power increase more gradually (relative to the corresponding pumps-on case) early in the transient for the same reason. The peak post-scram power is only 13.23% of the rated power.

However, the MDNBR for this case is more limiting than that for the corresponding pumps-on case, because of the reduced PCS flow rates. *(See Table 14.14-5 for a list of the overall core conditions at the time of the MDNBR.)*[f] No fuel failures are predicted to occur as a result of exceeding Critical Heat Flux.

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[e] See Table 14.1-5 for the peak LHR value.

[f] See Table 14.1-5 for the MDNBR value.

14.14.3 RADIOLOGICAL CONSEQUENCES

14.14.3.1 Analysis Method

For the radiological consequences, the steam line break event was analyzed considering conditions that maximized the release of fission products to the environment from the primary and secondary coolant systems (Reference 3). The release of radioactivity to the environment is a function of initial primary and secondary coolant radioactivity concentrations, the mass released from blowdown of the secondary system, the primary to secondary coolant leak rate, and the fuel failures resultant from the event.

The fission product activity released during the event is calculated by the methods described in Regulatory Guide 1.183 (Reference 14).

14.14.3.2 Bounding Event Input

To bound the consequences of the steam line break event, the break is assumed to occur between the containment wall and the main steam isolation valve. A break in this location results in complete blowdown of the affected steam generator to the environment. With the exception of the break location, the event is assumed to be identical to that described in Section 14.14.2, resulting in 0.5% fuel failures due to penetration of DNB limits. Note that the 0.5% fuel failure assumption conservatively bounds the actual result of no fuel failures and bounds the pre-existing and event generated iodine spiking calculations.

The major assumptions used for evaluating the radiological consequences are:

1. The activity associated with fuel failure resulting from the event is instantaneously released to and thoroughly mixed in the primary coolant.

2. There is no dilution of the primary coolant activity concentration due to charging or high pressure safety injection.

3. The release of all of the auxiliary feedwater delivered to the affected steam generator prior to operator action to isolate auxiliary feedwater to the affected generator is bounded by the use of hot zero power inventory for the affected steam generator.

4. A primary to secondary leak is assumed to exist in the affected steam generator for the duration of the transient. The leak rate is 0.3 gpm. All of the primary fluid that leaks to the secondary side flashes to steam and is released through the break location with no decontamination factor.

5. Operators begin PCS cooldown when the auxiliary feedwater is terminated. The cooldown rate is assumed such that shutdown cooling entry conditions are reached in 8 hours. Using a low cooldown rate increases the radiological releases by extending the primary to secondary leak for a longer period of time.

Tables 14.14-6 and 14.14-7 list the parameters used in the radiological consequence analysis.

14.14.3.3 Analysis of Results

The radionuclide releases from the primary coolant and the secondary coolant are analyzed separately, since each has a different radioactivity concentration. The secondary coolant radioactivity concentration is assumed to be the Technical Specification value of 0.1 μCi/gm dose equivalent I-131, with a negligible amount of noble gas. The primary coolant radioactivity concentration is assumed to be that associated with 0.5% fuel failures is added.

Two releases of secondary coolant to the environment are considered: initial blowdown through the break location and use of the Atmospheric Dump Valves (ADVs) on the unaffected steam generator to cool down the plant. The initial blowdown through the break location is assumed to be 210,759 lbm, which corresponds to the maximum HZP steam generator water mass and encompasses the secondary side fluid of the affected steam generator, the initial fluid contents of the steam line, backflow from the unaffected steam generator, and auxiliary feedwater that flashes upon addition to the affected steam generator.

PCS Cooldown using the unaffected steam generator releases approximately 800,000 lbm of secondary coolant to the environment through the ADVs. Although the amount of secondary coolant released to the environment is significant, the impact on the offsite doses from the secondary coolant release is relatively small. This is due to the low iodine activity concentration of the secondary coolant.

The main contributor to the offsite doses is the primary coolant leakage in the affected steam generator since it contains the high radioactivity released from the failed fuel. The primary to secondary leak rate is 0.3 gpm.

The TEDE doses for 0 to 2 hours at the Site Boundary (SB) and the duration of the event at the Low Population Zone (LPZ) and the control room are shown in Table 14.1-6. The limit for the offsite doses at the SB and LPZ from a steam line break event in which fuel failures occur is 25 rem. The limit is derived from 10 CFR 50.67. Therefore, the calculated offsite doses from the steam line break event meet the acceptance criteria.

14.14.4 CONCLUSIONS

No fuel failures are predicted to occur due to penetration of DNB limits. No fuel failures are predicted to occur as a result of penetration of the centerline melt limit.

For the radiological consequences, the offsite doses from the steam line break event are well within the 10 CFR 50.67 limit. The doses to control room personnel are discussed in Section 14.24 and Reference 13.

REFERENCES

1. USNRC Regulatory Guide 1.121 dated August 1976, Bases for Plugging Degraded PWR Steam Generator Tubes

2. EMF-2310(P)(A), Revision 0, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” Framatome ANP, May 2001.

3. EMF-96-029(P)(A), Volumes 1 and 2, “Reactor Analysis System for PWRs, Volume 1 - Methodology Description, Volume 2 - Benchmarking Results,” Siemens Power Corporation, January 1997.

4. "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-NF-75-21(P)(A), Revision 2, Exxon Nuclear Company, January 1986, (2741/1496).

5. Letter, H R Denton (NRC) to F W Buckman (CP Co) dated February 28, 1986, (3719/1117).

6. IN-1412, A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia, Idaho Nuclear Corporation, July 1970.

7. "Palisades Cycle 9: Analysis of Standard Review Plan Chapter 15 Events," ANF-90-078, Advanced Nuclear Fuels, Inc, September 1990, (C677/1283).

8. XN-NF-82-21(P)(A), Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983.

9. Deleted.

10. EMF-2853, Revision 0, Palisades Steam Line Break Analysis, Framatome ANP, January 2003.

11. Deleted

12. Letter, B D Johnson (CP Co) to Director (NRR) dated May 23, 1985, Transmitting Assessment of Palisades Main Steam Line Break (MSLB) Single Failure Back Fits, (3717/1259).

13. NAI-1149-018, Revision 2, “Palisades Design Basis Main Steam Line Break AST Radiological Analysis”, September 2007.

14. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, July 2000.

14.15 STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER

14.15.1 EVENT DESCRIPTION

The steam generator tube rupture (SGTR) accident is a penetration of the barrier between the primary coolant system (PCS) and the main steam system which results from the failure of a steam generator U-tube. Integrity of the barrier between the PCS and the 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. Following a reactor trip and turbine trip, the radioactive fluid is released through the steam generator safety or atmospheric dump valves as a result of the loss of normal AC power.

An SGTR event results in a depressurization of the PCS, causing a Thermal Margin/Low Pressure (TM/LP) reactor trip. Prior to the reactor trip, the radioactivity is transported through the turbine to the condenser where the noncondensable radioactive materials would be released via the condenser air ejectors. As a result of the reactor trip, the turbine/generator trips and normal AC power may be lost. The electrical power would then be unavailable for the station auxiliaries such as the primary coolant pumps and the condensate pumps. Under such circumstances the plant would experience a loss of load, a loss of normal feedwater and forced primary coolant flow, a loss of condenser vacuum and steam generator blowdown. 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. The plant is brought to shutdown cooling entry conditions by the operator per plant operating procedures. The time to reach shutdown cooling entry conditions will vary based on the availability of the following components; steam generator atmospheric dump valves (ADVs), available pressurizer heaters, auxiliary pressurizer spray, safety injection system and the auxiliary feedwater system.

Diagnosis of the SGTR accident is facilitated by secondary side radiation monitors which inform the operator of abnormal activity levels and that corrective operator action is required. These radiation monitors are located in the condenser air ejector discharge line, steam generator blowdown line, main steam lines, fan room, and in the stack. Additional diagnostic information is provided by PCS pressure and pressurizer level response indicating a leak as well as a decrease in the volume control tank level and the starting of the standby charging pumps.

The SGTR accident was evaluated by the NRC under Systematic Evaluation Program (SEP) Topic XV-17 (Reference 10). In this SEP evaluation, the NRC performed an independent analysis of a SGTR accident using assumptions and procedures indicated in the Standard Review Plan. The analysis assumes the plant is cooled down by releasing secondary steam to the environment through the main steam safety valves and the atmospheric

steam dump valves. On the basis of the NRC analysis, the NRC concluded that the Palisades plant design is acceptable with respect to the radiological consequences of a postulated steam generator tube rupture, and that the risk presented by this accident was similar to that of plants licensed under the more current criteria that existed at the time of the SEP. See Section 1.8.1 for additional discussion of the SEP.

The NRC issued Task Interface Agreement (TIA) 2009-003 (Reference 11) to address the reliance on the non-safety related atmospheric dump valves in the steam generator tube rupture analysis. The TIA concluded that the NRC approved the use of the non-safety related atmospheric dump valves as an acceptable method of mitigating a steam generator tube rupture in the original safety evaluation report for Palisades, and that this method is part of Palisades' licensing basis.

14.15.2 THERMAL-HYDRAULIC ANALYSIS

14.15.2.1 Analysis Method

The thermal-hydraulic response of the Nuclear Steam Supply System (NSSS) to the steam generator tube rupture with a loss of offsite power was simulated using the CESEC-III computer program for the first 30 minutes. At this time the operator is assumed to take control of the plant. Operator actions to mitigate the effects of the SGTR event and bring the plant to shutdown cooling entry conditions were simulated using a CESEC-III based cooldown algorithm. The CESEC-III computer program is described in Reference (1).

14.15.2.2 Bounding Event Input

The initial conditions and input parameters employed in the analyses of the system response to a steam generator tube rupture with a concurrent loss of offsite power are listed in Table 14.15-1. Additional discussion on the input parameters and the initial conditions are provided below. Conditions were chosen to maximize the radiological releases.

A parametric study using CESEC was performed to determine the effect of reactor trip time on doses. The trip setpoint was raised to yield an earlier trip and compared to a SGTR analysis in which the reactor trip (low pressurizer pressure setpoint) was biased low. The result of this comparison showed that while the integrated tube leakage was higher for the delayed trip case at 1800 seconds, the early reactor trip case had higher doses at that time due to larger leakage during the times when the MSSVs were open. However, the integrated Exclusion Area Boundary (EAB) doses at 2 hours for both accident Generated Iodine Spike (GIS) and Pre-existing Iodine Spike (PIS) were larger for the delayed reactor trip case. This is due to decay heat being lower during the cooldown portion of the early reactor trip case. With a smaller PCS heat load, the PCS pressure is lower, which in turn reduces the tube leakage rate. In addition, the PCS begins an earlier cooldown (via the MSSVs) so that primary and secondary temperatures and pressures are lower during the

transient than for the case of the delayed reactor trip. Therefore, the delayed trip assumption was selected for analysis based on this parametric study using the CESEC-based algorithm.

The initial conditions include the maximum allowed PCS pressure, nominal initial pressurizer liquid volume, maximum core power, minimum core coolant flow, and maximum core coolant inlet temperature. Table 14.15-2 contains assumptions regarding the system setpoints used in the analysis.

The operator actions, event recovery strategy and use of specific plant components assumed in this analysis were chosen to maximize the radiological releases while cooling the plant to shutdown cooling entry conditions following a SGTR event. The actual actions taken, recovery strategy and components used may be different from those assumed in the analysis, but will result in lower radiological releases than calculated in the bounding analysis. The major operator actions assumed in the analysis are summarized below. The timing of operator actions was based on Reference (3), which specifies time response criteria for safety related operator actions. The first intervention by the operator was assumed at 30 minutes after event initiation. Subsequently, a time delay of two minutes between each discrete operator action was assumed.

1) An automatic Auxiliary Feedwater Actuation Signal (AFAS) is generated if the level in the SG falls below 23.7% NR. However, Auxiliary Feedwater (AFW) flow to the SG will not commence for 120 seconds after the signal. AFW flow to both steam generators is established prior to the first operator action at 30 minutes. The AFW system is left in the automatic mode until the affected steam generator (SG) is isolated.

2) At 30 minutes after event initiation, the operator opens the ADVs of both SGs to cooldown the PCS at a rate of 75 °F/hr or less. The affected SG may be isolated only after the hot leg temperature (T_h) reaches 525 °F.

3) The operator maintains the SG level between 30% and 70% NR in the unaffected generator. However, the AFW flow to the affected SG will be maintained until it is isolated. This assumption is conservative, since it worsens the affected SG overfilling problem, and will result in higher doses.

4) The operator isolates the affected steam generator when the hot leg temperature is 525 °F or less. The initial cooldown of the PCS is aimed at preventing re-opening of the MSSVs on the affected steam generator.

5) The operator initiates auxiliary spray flow in order to depressurize the PCS to the SG pressure, about 1000 psia, after the isolation of the affected steam generator. The operator uses the HPSI system,

available pressurizer heaters, and auxiliary pressurizer spray to control PCS inventory and subcooling.

6) After isolating the affected generator, the operator cools the PCS at 75 °F/hr or less, using the unaffected steam generator. Note that a cooldown of less than 50 °F/hr is used in the analysis because it is a more realistic number for natural circulation conditions. The cooldown rate is reduced to a value which is designed to depressurize the PCS to shutdown cooling entry conditions in 8 hours. This reduction in cooldown rate increases the radiological release during the long term cooldown by delaying entry into shutdown cooling until approximately 8 hours after event initiation, thereby maximizing the decay heat and the amount of radioactivity to be removed through the ADVs within the 0-8 hour time period.

7) The operator attempts to maintain a subcooling margin of greater than 25 °F during the cooldown.

8) The operator uses the ADV of the affected SG in order to keep the SG from overfilling. For this analysis, the radiological doses are conservatively high by keeping the affected SG level less than 90%.

14.15.2.3 <u>Analysis of Results</u>

Table 14.15-3 presents a chronological sequence of events which occurs during the steam generator tube rupture event with a loss of offsite power from the time of event initiation (the double-ended rupture of a steam generator U-tube) to the attainment of shutdown cooling entry conditions (Reference 5). The sequence presented demonstrates that the operator can cool the plant down to shutdown cooling entry conditions during the event.

The dynamic behavior of important NSSS parameters following an SGTR is presented in Figure 14.15-1 to 14.15-21. Figures 14.15-1 through 14.15-10 depict event parameters from event initiation to 1800 seconds (30 minutes) when operator actions commence. Figures 14.15-11 through 14.15-21 are plant parameters through eight hours.

For a double-ended rupture, the primary to secondary leak rate exceeds the capacity of the charging pumps. As a result, the pressurizer pressure gradually decreases from an initial value of 2110 psia. At about 705 seconds, a reactor trip signal is generated when pressurizer pressure falls low enough to activate the TM/LP (low pressurizer pressure) trip. The reactor trip is followed by a turbine/generator trip. The loss of offsite power occurs 2 seconds after the turbine trip. Following the loss of offsite power, the primary coolant pumps coast down and natural circulation flow is established in the PCS.

Following the turbine trip, with turbine bypass unavailable, the main steam system pressure increases until the MSSVs open to release steam. A maximum main steam system pressure of approximately 1040 psia occurs around 725 seconds (Figure 14.15-4). Subsequent to this peak in the pressure, the main steam system pressure decreases, resulting in intermittent opening and closing of the MSSVs until 30 minutes into the event, at which point the operator intercedes by opening the ADVs to commence a cooldown. From then on, the steam is released through ADVs only (Figure 14.15-18).

Prior to reactor trip, the main feedwater control system is assumed to be in the automatic mode supplying feedwater to the steam generators such that steam generator water levels are maintained. Following the reactor trip, the main feedwater flow is ramped down at 5%/second commencing approximately 6.5 seconds after the loss of offsite power. As the level in the steam generators decrease below 23.7% NR, an AFAS signal is generated approximately 24.5 minutes into the event. The AFW flow begins reaching the steam generators 2 minutes later at a rate of 200 gpm per generator. If the low level setpoint is reached in one generator (unaffected SG only), the AFW flow is assumed to go to both generators until the time at which operator action isolates the affected SG (50 minutes).

The pressurizer empties at 725 seconds (Figure 14.15-7) resulting in rapid decrease in the PCS pressure. Soon afterwards, the reactor vessel upper head region begins to void due to flashing caused by the continued PCS depressurization and the boil off of coolant caused by heat transfer from the upper head metal. Essentially, the upper head begins to act like a pressurizer. Consequently, the PCS pressure decreases at a slower rate (Figure 14-15-3).

At 716 seconds a safety injection actuation signal is generated, and by 811.5 seconds the safety injection flow begins to enter the PCS when the PCS pressure has decreased to below the shutdown head of the HPSI pumps. At 1800 seconds, the operator takes control of the plant. The first action is to open the ADVs to commence a cooldown. Twenty minutes later, the operator isolates the affected steam generator, securing feed and shutting the associated MSIV and ADVs. The operator adjusts the ADVs of the intact steam generator to cooldown the plant at a rate of 50 °F/hr or less. The AFW flow to the intact steam generator is adjusted to maintain levels between 30% and 70% NR.

After isolation of the affected SG, the two steam generator pressures diverge (Figure 14.15-14). The isolated steam generator pressure increases due to flashing of the PCS fluid through the tube break. The intact steam generator pressure continues to decrease due to steaming via the ADV.

The operator initiates auxiliary pressurizer spray in order to depressurize the PCS (Figure 14.15-13) to within 50 psi of the affected SG pressure and thus reduce the leak flow (Figure 14.15-16).

The operator also controls the safety injection flow, auxiliary pressurizer spray flow and the available pressurizer heaters to maintain a minimum subcooling of 25 °F on the qualified Core Exit Thermocouples (CETs) and a pressurizer level of 20% to 90%.

At 13000 seconds, the affected steam generator level has increased to 90% WR, and the operator opens the ADV to reduce the level. After this time, the operator periodically steams the affected steam generator to prevent it from overfilling. Steaming the affected SG is conservative in determining the radiological consequences.

After reaching shutdown cooling entry conditions and engaging the shutdown cooling system, it is assumed that no further steam release occurs from steam generators.

The maximum PCS and secondary pressures do not exceed 110% of design pressure following a steam generator tube rupture event with a loss of offsite power, thus, assuring the integrity of the PCS and the main steam system.

Figure 14.15-19 gives the integrated ADV releases from the affected and intact steam generators. At 1800 seconds when the operator takes control of the plant, 44,654 lbs of steam have escaped from the affected steam generator via the MSSVs. During the same time, 61123 lbs of primary liquid leaked into the affected steam generator. The integrated ADV steam releases and leak flow results for the 0-2 hour and 0-8 hr periods are shown in Table 14.15-4.

14.15.3 RADIOLOGICAL ANALYSIS

14.15.3.1 <u>Analysis Method</u>

The analysis of the radiological consequences considers the most severe release of secondary as well as primary system activity leaked from the tube break. The analysis is consistent with the methodology described in Regulatory Guide 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," (Reference 2). The inventory of fission product activity available for release to the environment is a function of the primary to secondary coolant leakage rate, the assumed increase in fission product concentration for Iodine GIS dose, and the mass of steam discharged to the environment.

The CESEC computer code was used to determine the mass and energy releases during the first 30 minutes of the event. As documented in Reference 8, an error was detected in the decay heat portion of the CESEC computer code, with the result being an underprediction of releases for the first 30 minutes. Also, an error in the assumed HPSI flow rate was discovered in the SGTR thermal hydraulic analysis (Reference 9). The SGTR offsite and Control Room doses were revised to correct for these errors (Reference 9). In addition, the SGTR Previous Iodine Spike (PIS) scenario

iodine release rates account for the Technical Specification limit of 40 μCi/gm. Table 14.15-5 provides the significant input parameters for the dose calculations.

14.15.3.2 Bounding Event Input

The assumptions and parameters employed for the evaluation of radiological releases are (Reference 7):

1) Doses are calculated for two different assumptions: (a) an event generated iodine spike (GIS) coincident with the initiation of the event, and (b) a pre-accident iodine spike (PIS).

2) A portion of the primary fluid that leaks into the faulted SG flashes into steam. The amount that flashes depends on the enthalpy of the primary liquid and the saturation enthalpy of the SG. The flashing portion has a decontamination factor calculated according to the methods described in Regulatory Guide 1.183, Appendix F (Reference 7). The non-flashing portion of the primary leak flow is assumed to mix uniformly with the liquid in the SG.

3) Following the accident, no additional steam and radioactivity are released to the environment when the shutdown cooling system is placed in operation.

4) The SG is assumed to have a decontamination factor of 100 in accordance with Regulatory Guide 1.183 (Reference 7), so that the radioactivity concentration in the steam phase is 1/100 of the concentration in the liquid phase.

5) A primary-to-secondary leakage rate of 432 gallons per day is assumed in the unaffected steam generator for the duration of the transient to conservatively calculate the radiation released.

6) Accident doses are calculated for two different assumptions, which are the Pre-existing Iodine Spike (PIS), and event Generated Iodine Spike (GIS). For the PIS conditions, the primary system activity is 40 μCi/gm. For the GIS case, an initial activity of 1 μCi/gm and a spiking factor of 335 is assumed. See the discussion on calculation of PCS activity below.

7) An initial secondary iodine activity of 0.1 μCi/gm is assumed (Technical Specifications Limit).

Table 14.15-5 lists the assumptions used in the radiological analysis.

14.15.3.3 Analysis of Results

The initial PCS activity is assumed to be the equilibrium concentration prior to the accident.

Regulatory Guide 1.183 indicates that dose calculations for two types of iodine spiking cases be considered. This is due to iodine concentration increasing following a PCS pressure transient, such as a reactor trip. This phenomenon is known as iodine spiking. The two types of iodine spiking cases that must be considered are event generated iodine spike, GIS, and pre-accident iodine spike, PIS. The iodine spiking factor is defined as the ratio of the appearance rate of I-131 in the PCS following the event to the appearance rate required to produce a steady state equilibrium concentration.

The GIS iodine spike is a direct consequence of the PCS depressurization and shutdown caused by the SGTR event. A spiking factor of 335 is used in accordance with Regulatory Guide 1.183. The analysis conservatively assumes a step change in the iodine rate of appearance at the initiation of the SGTR which lasts eight hours to maximize the impact on the EAB doses. The PCS initial radioactivity concentration was assumed to be the Technical Specification value of 1 μCi/gm for this analysis.

The PIS iodine spike is assumed to occur during a period of high PCS activity which was initiated by an independent event prior to the SGTR. The PCS activity remains high during the event and does not decrease further because of the event. An initial coolant activity of 40 μCi/gm was assumed for this analysis. A spiking factor of 1 was used in this part of the analysis.

For the GIS case, the initial PCS activity is the Technical Specification value of 1 μCi/gm. However, the primary activity increases steadily due to the large spiking factor. Since the large spiking factor is assumed to exist for a long period of time, the eight hour GIS case doses are higher than for the PIS case. The actual doses depend upon the timing of the radioactivity release to the atmosphere during the event.

The two-hour Exclusion Area Boundary (EAB) and the eight-hour Low Population Zone (LPZ) boundary doses for both the GIS and the PIS are presented in Table 14.1-6. For a postulated SGTR accident with an assumed PIS, the dose limits are 25 rem TEDE. For an assumed accident GIS, the dose limits are 2.5 rem TEDE. These limits apply to both the EAB and the LPZ. The dose acceptance criteria are derived from Regulatory Guide 1.183 and 10 CFR 50.67. The calculated EAB and LPZ doses are well within the acceptance criteria.

14.15.4 CONCLUSIONS

The radiological releases calculated for the SGTR event with a loss of offsite power are well below the limits for offsite doses. The doses to control room personnel are discussed in Section 14.24. Finally, the PCS and secondary system pressures during the SGTR remain below 110% of the design pressure limits, thus, assuring the integrity of these systems.

1. LD-82-001 (dated 1/6/82), "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," Enclosure 1-P to letter from A.E. Scherer to D.G. Eisenhut, December, 1981.

2. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, July 2000.

3. "Time Response Design Criteria for Safety-Related Operator Actions," American National Standard, ANSI/ANS-58.8-1984.

4. NUREG-0800, U.S. NRC "Standard Review Plan", Revision 2, dated July, 1981.

5. ABB Combustion Engineering letter from W. G. Dove Jr. to R. J. Gerling, "Results of Palisades SGTR Analysis for the replacement SG", NT-90-0985, July 27, 1990, (C621/1416).

6. ST98-C-326, "Assessment of Reduced PCS Flowrate on SGTR and Containment Analyses for Palisades Plant," June 15, 1998, memo from Michael J. Gancarz (ABB) to Greg A. Baustian.

7. NAI-1149-019, Revision 1, Palisades Design Basis Steam Generator Tube Rupture AST Radiological Analysis, January 2006.

8. Letter, ABB/CENO ST-97-0273, MJ Gancarz to GA Baustian, Consumers Energy, "Closeout of CESEC Decay Heat Issue for the Palisades Plant," dated May 19, 1997.

9. Letter CPAL-04-10 (Ref. LTR-TAS-04-14), entitled "Steam Generator Tube Rupture Analysis Calculation Note", from SP Swigart (Westinghouse) to GE Jarka (NMC), February 17, 2004; attachment DA-001-NT90-014, Revision 1, "Palisades Steam Generator Tube Rupture Event for the Replacement Steam Generator Project," Westinghouse Minimum Documentation Calculation, February 17, 2004.

10. Letter, from DL Ziemann (NRC) to DP Hoffman (CPCo) dated January 29, 1980, "Completion of SEP TOPIC XV-17 Radiological Consequences of Steam Generator Tube Failure (PWR)," (2614/0439).

11. Letter, from TB Blount (NRR) to KG O'Brien (NRC Region III) dated June 29, 2010, "Final Response to Task Interface Agreement – Reliance of Non-Safety-Related Atmospheric Dump Valves in FSAR Chapter 14 Analysis for Steam Generator Tube Rupture Accident (TIA 2009-003)."

14.15 STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER

14.15.1 EVENT DESCRIPTION

The steam generator tube rupture (SGTR) accident is a penetration of the barrier between the primary coolant system (PCS) and the main steam system which results from the failure of a steam generator U-tube. Integrity of the barrier between the PCS and the 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. Following a reactor trip and turbine trip, the radioactive fluid is released through the steam generator safety or atmospheric dump valves as a result of the loss of normal AC power.

An SGTR event results in a depressurization of the PCS, causing a Thermal Margin/Low Pressure (TM/LP) reactor trip. Prior to the reactor trip, the radioactivity is transported through the turbine to the condenser where the noncondensable radioactive materials would be released via the condenser air ejectors. As a result of the reactor trip, the turbine/generator trips and normal AC power may be lost. The electrical power would then be unavailable for the station auxiliaries such as the primary coolant pumps and the condensate pumps. Under such circumstances the plant would experience a loss of load, a loss of normal feedwater and forced primary coolant flow, a loss of condenser vacuum and steam generator blowdown. 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. The plant is brought to shutdown cooling entry conditions by the operator per plant operating procedures. The time to reach shutdown cooling entry conditions will vary based on the availability of the following components; steam generator atmospheric dump valves (ADVs), available pressurizer heaters, auxiliary pressurizer spray, safety injection system and the auxiliary feedwater system.

Diagnosis of the SGTR accident is facilitated by secondary side radiation monitors which inform the operator of abnormal activity levels and that corrective operator action is required. These radiation monitors are located in the condenser air ejector discharge line, steam generator blowdown line, main steam lines, fan room, and in the stack. Additional diagnostic information is provided by PCS pressure and pressurizer level response indicating a leak as well as a decrease in the volume control tank level and the starting of the standby charging pumps.

The SGTR accident was evaluated by the NRC under Systematic Evaluation Program (SEP) Topic XV-17 (Reference 10). In this SEP evaluation, the NRC performed an independent analysis of a SGTR accident using assumptions and procedures indicated in the Standard Review Plan. The analysis assumes the plant is cooled down by releasing secondary steam to the environment through the main steam safety valves and the atmospheric

steam dump valves. On the basis of the NRC analysis, the NRC concluded that the Palisades plant design is acceptable with respect to the radiological consequences of a postulated steam generator tube rupture, and that the risk presented by this accident was similar to that of plants licensed under the more current criteria that existed at the time of the SEP. See Section 1.8.1 for additional discussion of the SEP.

The NRC issued Task Interface Agreement (TIA) 2009-003 (Reference 11) to address the reliance on the non-safety related atmospheric dump valves in the steam generator tube rupture analysis. The TIA concluded that the NRC approved the use of the non-safety related atmospheric dump valves as an acceptable method of mitigating a steam generator tube rupture in the original safety evaluation report for Palisades, and that this method is part of Palisades' licensing basis.

14.15.2 THERMAL-HYDRAULIC ANALYSIS

14.15.2.1 <u>Analysis Method</u>

The thermal-hydraulic response of the Nuclear Steam Supply System (NSSS) to the steam generator tube rupture with a loss of offsite power was simulated using the CESEC-III computer program for the first 30 minutes. At this time the operator is assumed to take control of the plant. Operator actions to mitigate the effects of the SGTR event and bring the plant to shutdown cooling entry conditions were simulated using a CESEC-III based cooldown algorithm. The CESEC-III computer program is described in Reference (1).

14.15.2.2 <u>Bounding Event Input</u>

The initial conditions and input parameters employed in the analyses of the system response to a steam generator tube rupture with a concurrent loss of offsite power are listed in Table 14.15-1. Additional discussion on the input parameters and the initial conditions are provided below. Conditions were chosen to maximize the radiological releases.

A parametric study using CESEC was performed to determine the effect of reactor trip time on doses. The trip setpoint was raised to yield an earlier trip and compared to a SGTR analysis in which the reactor trip (low pressurizer pressure setpoint) was biased low. The result of this comparison showed that while the integrated tube leakage was higher for the delayed trip case at 1800 seconds, the early reactor trip case had higher doses at that time due to larger leakage during the times when the MSSVs were open. However, the integrated Exclusion Area Boundary (EAB) doses at 2 hours for both accident Generated Iodine Spike (GIS) and Pre-existing Iodine Spike (PIS) were larger for the delayed reactor trip case. This is due to decay heat being lower during the cooldown portion of the early reactor trip case. With a smaller PCS heat load, the PCS pressure is lower, which in turn reduces the tube leakage rate. In addition, the PCS begins an earlier cooldown (via the MSSVs) so that primary and secondary temperatures and pressures are lower during the

transient than for the case of the delayed reactor trip. Therefore, the delayed trip assumption was selected for analysis based on this parametric study using the CESEC-based algorithm.

The initial conditions include the maximum allowed PCS pressure, nominal initial pressurizer liquid volume, maximum core power, minimum core coolant flow, and maximum core coolant inlet temperature. Table 14.15-2 contains assumptions regarding the system setpoints used in the analysis.

The operator actions, event recovery strategy and use of specific plant components assumed in this analysis were chosen to maximize the radiological releases while cooling the plant to shutdown cooling entry conditions following a SGTR event. The actual actions taken, recovery strategy and components used may be different from those assumed in the analysis, but will result in lower radiological releases than calculated in the bounding analysis. The major operator actions assumed in the analysis are summarized below. The timing of operator actions was based on Reference (3), which specifies time response criteria for safety related operator actions. The first intervention by the operator was assumed at 30 minutes after event initiation. Subsequently, a time delay of two minutes between each discrete operator action was assumed.

1) An automatic Auxiliary Feedwater Actuation Signal (AFAS) is generated if the level in the SG falls below 23.7% NR. However, Auxiliary Feedwater (AFW) flow to the SG will not commence for 120 seconds after the signal. AFW flow to both steam generators is established prior to the first operator action at 30 minutes. The AFW system is left in the automatic mode until the affected steam generator (SG) is isolated.

2) At 30 minutes after event initiation, the operator opens the ADVs of both SGs to cooldown the PCS at a rate of 75 °F/hr or less. The affected SG may be isolated only after the hot leg temperature (T_h) reaches 525 °F.

3) The operator maintains the SG level between 30% and 70% NR in the unaffected generator. However, the AFW flow to the affected SG will be maintained until it is isolated. This assumption is conservative, since it worsens the affected SG overfilling problem, and will result in higher doses.

4) The operator isolates the affected steam generator when the hot leg temperature is 525 °F or less. The initial cooldown of the PCS is aimed at preventing re-opening of the MSSVs on the affected steam generator.

5) The operator initiates auxiliary spray flow in order to depressurize the PCS to the SG pressure, about 1000 psia, after the isolation of the affected steam generator. The operator uses the HPSI system,

available pressurizer heaters, and auxiliary pressurizer spray to control PCS inventory and subcooling.

6) After isolating the affected generator, the operator cools the PCS at 75 °F/hr or less, using the unaffected steam generator. Note that a cooldown of less than 50 °F/hr is used in the analysis because it is a more realistic number for natural circulation conditions. The cooldown rate is reduced to a value which is designed to depressurize the PCS to shutdown cooling entry conditions in 8 hours. This reduction in cooldown rate increases the radiological release during the long term cooldown by delaying entry into shutdown cooling until approximately 8 hours after event initiation, thereby maximizing the decay heat and the amount of radioactivity to be removed through the ADVs within the 0-8 hour time period.

7) The operator attempts to maintain a subcooling margin of greater than 25 °F during the cooldown.

8) The operator uses the ADV of the affected SG in order to keep the SG from overfilling. For this analysis, the radiological doses are conservatively high by keeping the affected SG level less than 90%.

14.15.2.3 Analysis of Results

Table 14.15-3 presents a chronological sequence of events which occurs during the steam generator tube rupture event with a loss of offsite power from the time of event initiation (the double-ended rupture of a steam generator U-tube) to the attainment of shutdown cooling entry conditions (Reference 5). The sequence presented demonstrates that the operator can cool the plant down to shutdown cooling entry conditions during the event.

The dynamic behavior of important NSSS parameters following an SGTR is presented in Figure 14.15-1 to 14.15-21. Figures 14.15-1 through 14.15-10 depict event parameters from event initiation to 1800 seconds (30 minutes) when operator actions commence. Figures 14.15-11 through 14.15-21 are plant parameters through eight hours.

For a double-ended rupture, the primary to secondary leak rate exceeds the capacity of the charging pumps. As a result, the pressurizer pressure gradually decreases from an initial value of 2110 psia. At about 705 seconds, a reactor trip signal is generated when pressurizer pressure falls low enough to activate the TM/LP (low pressurizer pressure) trip. The reactor trip is followed by a turbine/generator trip. The loss of offsite power occurs 2 seconds after the turbine trip. Following the loss of offsite power, the primary coolant pumps coast down and natural circulation flow is established in the PCS.

Following the turbine trip, with turbine bypass unavailable, the main steam system pressure increases until the MSSVs open to release steam. A maximum main steam system pressure of approximately 1040 psia occurs around 725 seconds (Figure 14.15-4). Subsequent to this peak in the pressure, the main steam system pressure decreases, resulting in intermittent opening and closing of the MSSVs until 30 minutes into the event, at which point the operator intercedes by opening the ADVs to commence a cooldown. From then on, the steam is released through ADVs only (Figure 14.15-18).

Prior to reactor trip, the main feedwater control system is assumed to be in the automatic mode supplying feedwater to the steam generators such that steam generator water levels are maintained. Following the reactor trip, the main feedwater flow is ramped down at 5%/second commencing approximately 6.5 seconds after the loss of offsite power. As the level in the steam generators decrease below 23.7% NR, an AFAS signal is generated approximately 24.5 minutes into the event. The AFW flow begins reaching the steam generators 2 minutes later at a rate of 200 gpm per generator. If the low level setpoint is reached in one generator (unaffected SG only), the AFW flow is assumed to go to both generators until the time at which operator action isolates the affected SG (50 minutes).

The pressurizer empties at 725 seconds (Figure 14.15-7) resulting in rapid decrease in the PCS pressure. Soon afterwards, the reactor vessel upper head region begins to void due to flashing caused by the continued PCS depressurization and the boil off of coolant caused by heat transfer from the upper head metal. Essentially, the upper head begins to act like a pressurizer. Consequently, the PCS pressure decreases at a slower rate (Figure 14-15-3).

At 716 seconds a safety injection actuation signal is generated, and by 811.5 seconds the safety injection flow begins to enter the PCS when the PCS pressure has decreased to below the shutdown head of the HPSI pumps. At 1800 seconds, the operator takes control of the plant. The first action is to open the ADVs to commence a cooldown. Twenty minutes later, the operator isolates the affected steam generator, securing feed and shutting the associated MSIV and ADVs. The operator adjusts the ADVs of the intact steam generator to cooldown the plant at a rate of 50 °F/hr or less. The AFW flow to the intact steam generator is adjusted to maintain levels between 30% and 70% NR.

After isolation of the affected SG, the two steam generator pressures diverge (Figure 14.15-14). The isolated steam generator pressure increases due to flashing of the PCS fluid through the tube break. The intact steam generator pressure continues to decrease due to steaming via the ADV.

The operator initiates auxiliary pressurizer spray in order to depressurize the PCS (Figure 14.15-13) to within 50 psi of the affected SG pressure and thus reduce the leak flow (Figure 14.15-16).

The operator also controls the safety injection flow, auxiliary pressurizer spray flow and the available pressurizer heaters to maintain a minimum subcooling of 25 °F on the qualified Core Exit Thermocouples (CETs) and a pressurizer level of 20% to 90%.

At 13000 seconds, the affected steam generator level has increased to 90% WR, and the operator opens the ADV to reduce the level. After this time, the operator periodically steams the affected steam generator to prevent it from overfilling. Steaming the affected SG is conservative in determining the radiological consequences.

After reaching shutdown cooling entry conditions and engaging the shutdown cooling system, it is assumed that no further steam release occurs from steam generators.

The maximum PCS and secondary pressures do not exceed 110% of design pressure following a steam generator tube rupture event with a loss of offsite power, thus, assuring the integrity of the PCS and the main steam system.

Figure 14.15-19 gives the integrated ADV releases from the affected and intact steam generators. At 1800 seconds when the operator takes control of the plant, 44,654 lbs of steam have escaped from the affected steam generator via the MSSVs. During the same time, 61123 lbs of primary liquid leaked into the affected steam generator. The integrated ADV steam releases and leak flow results for the 0-2 hour and 0-8 hr periods are shown in Table 14.15-4.

14.15.3 RADIOLOGICAL ANALYSIS

14.15.3.1 Analysis Method

The analysis of the radiological consequences considers the most severe release of secondary as well as primary system activity leaked from the tube break. The analysis is consistent with the methodology described in Regulatory Guide 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," (Reference 2). The inventory of fission product activity available for release to the environment is a function of the primary to secondary coolant leakage rate, the assumed increase in fission product concentration for Iodine GIS dose, and the mass of steam discharged to the environment.

The CESEC computer code was used to determine the mass and energy releases during the first 30 minutes of the event. As documented in Reference 8, an error was detected in the decay heat portion of the CESEC computer code, with the result being an underprediction of releases for the first 30 minutes. Also, an error in the assumed HPSI flow rate was discovered in the SGTR thermal hydraulic analysis (Reference 9). The SGTR offsite and Control Room doses were revised to correct for these errors (Reference 9). In addition, the SGTR Previous Iodine Spike (PIS) scenario

iodine release rates account for the Technical Specification limit of 40 μCi/gm. Table 14.15-5 provides the significant input parameters for the dose calculations.

14.15.3.2 Bounding Event Input

The assumptions and parameters employed for the evaluation of radiological releases are (Reference 7):

1) Doses are calculated for two different assumptions: (a) an event generated iodine spike (GIS) coincident with the initiation of the event, and (b) a pre-accident iodine spike (PIS).

2) A portion of the primary fluid that leaks into the faulted SG flashes into steam. The amount that flashes depends on the enthalpy of the primary liquid and the saturation enthalpy of the SG. The flashing portion has a decontamination factor calculated according to the methods described in Regulatory Guide 1.183, Appendix F (Reference 7). The non-flashing portion of the primary leak flow is assumed to mix uniformly with the liquid in the SG.

3) Following the accident, no additional steam and radioactivity are released to the environment when the shutdown cooling system is placed in operation.

4) The SG is assumed to have a decontamination factor of 100 in accordance with Regulatory Guide 1.183 (Reference 7), so that the radioactivity concentration in the steam phase is 1/100 of the concentration in the liquid phase.

5) A primary-to-secondary leakage rate of 432 gallons per day is assumed in the unaffected steam generator for the duration of the transient to conservatively calculate the radiation released.

6) Accident doses are calculated for two different assumptions, which are the Pre-existing Iodine Spike (PIS), and event Generated Iodine Spike (GIS). For the PIS conditions, the primary system activity is 40 μCi/gm. For the GIS case, an initial activity of 1 μCi/gm and a spiking factor of 335 is assumed. See the discussion on calculation of PCS activity below.

7) An initial secondary iodine activity of 0.1 μCi/gm is assumed (Technical Specifications Limit).

Table 14.15-5 lists the assumptions used in the radiological analysis.

14.15.3.3 Analysis of Results

The initial PCS activity is assumed to be the equilibrium concentration prior to the accident.

Regulatory Guide 1.183 indicates that dose calculations for two types of iodine spiking cases be considered. This is due to iodine concentration increasing following a PCS pressure transient, such as a reactor trip. This phenomenon is known as iodine spiking. The two types of iodine spiking cases that must be considered are event generated iodine spike, GIS, and pre-accident iodine spike, PIS. The iodine spiking factor is defined as the ratio of the appearance rate of I-131 in the PCS following the event to the appearance rate required to produce a steady state equilibrium concentration.

The GIS iodine spike is a direct consequence of the PCS depressurization and shutdown caused by the SGTR event. A spiking factor of 335 is used in accordance with Regulatory Guide 1.183. The analysis conservatively assumes a step change in the iodine rate of appearance at the initiation of the SGTR which lasts eight hours to maximize the impact on the EAB doses. The PCS initial radioactivity concentration was assumed to be the Technical Specification value of 1 μCi/gm for this analysis.

The PIS iodine spike is assumed to occur during a period of high PCS activity which was initiated by an independent event prior to the SGTR. The PCS activity remains high during the event and does not decrease further because of the event. An initial coolant activity of 40 μCi/gm was assumed for this analysis. A spiking factor of 1 was used in this part of the analysis.

For the GIS case, the initial PCS activity is the Technical Specification value of 1 μCi/gm. However, the primary activity increases steadily due to the large spiking factor. Since the large spiking factor is assumed to exist for a long period of time, the eight hour GIS case doses are higher than for the PIS case. The actual doses depend upon the timing of the radioactivity release to the atmosphere during the event.

The two-hour Exclusion Area Boundary (EAB) and the eight-hour Low Population Zone (LPZ) boundary doses for both the GIS and the PIS are presented in Table 14.1-6. For a postulated SGTR accident with an assumed PIS, the dose limits are 25 rem TEDE. For an assumed accident GIS, the dose limits are 2.5 rem TEDE. These limits apply to both the EAB and the LPZ. The dose acceptance criteria are derived from Regulatory Guide 1.183 and 10 CFR 50.67. The calculated EAB and LPZ doses are well within the acceptance criteria.

14.15.4 CONCLUSIONS

The radiological releases calculated for the SGTR event with a loss of offsite power are well below the limits for offsite doses. The doses to control room personnel are discussed in Section 14.24. Finally, the PCS and secondary system pressures during the SGTR remain below 110% of the design pressure limits, thus, assuring the integrity of these systems.

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2. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, July 2000.

3. "Time Response Design Criteria for Safety-Related Operator Actions," American National Standard, ANSI/ANS-58.8-1984.

4. NUREG-0800, U.S. NRC "Standard Review Plan", Revision 2, dated July, 1981.

5. ABB Combustion Engineering letter from W. G. Dove Jr. to R. J. Gerling, "Results of Palisades SGTR Analysis for the replacement SG", NT-90-0985, July 27, 1990, (C621/1416).

6. ST98-C-326, "Assessment of Reduced PCS Flowrate on SGTR and Containment Analyses for Palisades Plant," June 15, 1998, memo from Michael J. Gancarz (ABB) to Greg A. Baustian.

7. NAI-1149-019, Revision 1, Palisades Design Basis Steam Generator Tube Rupture AST Radiological Analysis, January 2006.

8. Letter, ABB/CENO ST-97-0273, MJ Gancarz to GA Baustian, Consumers Energy, "Closeout of CESEC Decay Heat Issue for the Palisades Plant," dated May 19, 1997.

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14.17 LOSS OF COOLANT ACCIDENT

14.17.1 LARGE BREAK LOCA (LBLOCA)

14.17.1.1 Event Description

A Large Break Loss of Coolant Accident (LBLOCA) is initiated by a postulated large rupture of the Primary Coolant System (PCS) piping. Based on deterministic studies, the worst break location is in the cold leg piping between the Primary Coolant Pump (PCP) and the reactor vessel for the PCS loop containing the pressurizer. The break initiates a rapid depressurization of the PCS. A reactor trip signal is initiated when the low pressurizer pressure trip setpoint is reached; however, the reactor trip is conservatively neglected in the analysis. The reactor is shut down by coolant voiding in the core.

The plant is assumed to be operating normally at full power prior to the accident. The large cold leg break is assumed to open instantaneously. For this break, a rapid primary system depressurization occurs, along with a core flow stagnation and reversal. This causes the fuel rods to experience Departure from Nuclear Boiling (DNB). Subsequently, the limiting fuel rods are cooled by film convection to steam. The coolant voiding creates a strong negative reactivity effect and core fission ends. As heat transfer from the fuel rods is reduced, the cladding temperature rises.

Coolant in all regions of the PCS begins to flash. At the break plane, the loss of subcooling in the coolant results in substantially reduced break flow. This reduces the depressurization rate and may also lead to a period of positive core flow or reduced downflow as the PCPs in the intact loops continue to supply water to the vessel. Cladding temperatures may be reduced and some portions of the core may rewet during this period.

This positive core flow or reduced downflow period ends as two-phase conditions occur in the reactor coolant pumps, reducing their effectiveness. Once again, the core flow reverses as most of the vessel mass flows out through the broken cold leg.

Mitigation of the LBLOCA begins when the Safety Injection Actuation Signal (SIAS) is tripped. This signal is initiated by either high containment pressure or low pressurizer pressure. Regulations require that a worst active single-failure be considered for Emergency Core Cooling System (ECCS) safety analysis. This worst active single failure was determined generically in the Realistic Large Break LOCA (RLBLOCA) evaluation model (Reference 5) to be the loss of one ECCS train. The AREVA NP RLBLOCA methodology conservatively assumes a minimal time delay and a normal (no failure irrespective of the assumed worst single active failure) lineup of the containment sprays and fan coolers to reduce containment pressure and

increase break flow. The analysis assumes that one High Pressure Safety Injection (HPSI) pump, one Low Pressure Safety Injection (LPSI) pump, all containment spray pumps and all containment fan coolers are operational.

When the PCS pressure falls below the Safety Injection Tank (SIT) pressure, borated water from the SITs is injected into the cold legs. In the early delivery of SIT water, high pressure and high break flow will cause some of this fluid to bypass the core. During this bypass period, core heat transfer remains poor and fuel rod cladding temperatures increase. As PCS and containment pressures equilibrate, ECCS water begins to fill the lower plenum and eventually the lower portions of the core. This improves core heat transfer and cladding temperatures begin to decrease.

Eventually, the relatively large volume of SIT water is exhausted and core recovery relies solely on ECCS pumped injection. As the SITs empty, the nitrogen gas used to pressurize the SITs exits through the break. This gas release may result in a short period of improved core heat transfer as the nitrogen gas displaces water in the downcomer. After the nitrogen gas is expelled, the ECCS may not be able to sustain full core cooling temporarily because of the core decay heat and the higher steam temperatures created by quenching in the lower portions of the core. Peak fuel rod cladding temperatures may increase for a short period until additional energy is removed from the core by the LPSI and the decay heat continues to fall. Steam generated from fuel rod rewet will entrain liquid and pass through the core, vessel upper plenum, the hot legs, the steam generator and the PCP before it is vented out the break. The resistance of this flow path to the steam flow (including steam binding effects) is balanced by the driving force of water filling the downcomer. This resistance (steam binding) may act to retard the progression of core reflooding and postpone core-wide cooling. Eventually (within a few minutes of the accident), core reflooding will progress sufficiently to ensure core-wide cooling. Full core quench occurs within a few minutes after core-wide cooling. Long-term cooling is then sustained with the LPSI.

The purpose of the RLBLOCA analysis is to demonstrate that the following criteria of 10 CFR 50.46(b) (Reference 11) are met:

1. The calculated maximum fuel element cladding temperature shall not exceed 2,200 °F.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the

metal in the cladding cylinders surrounding the fuel excluding the cladding surrounding the plenum volume were to react.

14.17.1.2 Thermal Hydraulics Analysis

14.17.1.2.1 Analysis Method

The RLBLOCA methodology is documented in topical report EMF-2103, Realistic Large Break LOCA Methodology (Reference 5). The methodology follows the Code Scaling Applicability and Uncertainty (CSAU) evaluation methodology (Reference 6). This method outlines an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifies the uncertainties in a LBLOCA analysis.

The RLBLOCA methodology uses the following computer codes:

- RODEX3A for computation of the initial fuel stored energy, fission gas release, and fuel-cladding gap conductance.
- S-RELAP5 for the system calculation, including the containment pressure response.

The governing two-fluid (plus non-condensibles) model with conservation equations for mass, energy and momentum transfer is used. The reactor core is modeled in S-RELAP5 with heat generation rates determined from reactor kinetics equations (point kinetics) with reactivity feedback, and with actinide and decay heating.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction, and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ.

The modeling of plant components is performed by following guidelines developed to ensure accurate accounting for physical dimensions and that the dominant phenomenon expected during an LBLOCA event are captured. The basic building block for modeling is the hydraulic volume for fluid paths and the heat structure for a heat transfer surface. In addition, special purpose components exist to represent specific components such as the pumps or the steam generator separators. All geometries are modeled at a level of detail necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations.

A typical calculation using S-RELAP5 begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and

initial conditions for this steady-state calculation are chosen to reflect plant technical specifications or to match measured data. Additionally, the RODEX3A code provides initial conditions for the S-RELAP5 fuel models.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops (specifically, the loop with the pressurizer). The evolution of the transient through blowdown, refill, and reflood is computed continuously using S-RELAP5. Transient containment pressure is also calculated by S-RELAP5 using containment models derived from the CONTEMPT-LT code (Reference 7).

The methods used in the application of S-RELAP5 to the LBLOCA are described in Reference 5. A detailed assessment of this computer code was made through comparisons to experimental data, with many benchmarks with cladding temperatures ranging from 1,700 °F (or less) to above 2,200 °F. These assessments were used to develop quantitative estimates of the ability of the code to predict key physical phenomena in a PWR LBLOCA. Various models, for example, the core heat transfer, the decay heat model and the fuel cladding oxidation correlation, are defined based on code-to-data comparisons and are, hence, plant independent.

The RV internals are modeled in detail based on specific inputs supplied by Entergy. Nodes and connectivity, flow areas, resistances and heat structures are all accurately modeled. The location of the hot assembly/hot pin(s) is unrestricted; however, the channel is always modeled to restrict appreciable upper plenum liquid fallback.

The final step of the best-estimate methodology is to combine all the uncertainties related to the code and plant parameters and estimate the Peak Clad Temperature (PCT) at a high probability level. The steps taken to derive the PCT uncertainty estimate are summarized below:

1. Base Plant Input File Development

   First, RODEX3A and S-RELAP5 base input files for the plant (including a containment input file) are developed. Code input development guidelines are followed to ensure that the model nodalization is consistent with that used in the code validation.

2. Sampled Case Development

   The non-parametric statistical approach requires that many “sampled” cases be created and processed. For every set of input created, each “key LOCA parameter” is randomly sampled over a range established through code uncertainty assessment or expected operating limits (provided by plant technical specifications or data). Those parameters

considered "key LOCA parameters" are listed in Table 14.17.1-1. This list includes both parameters related to LOCA phenomena (based on the phenomena identification and ranking table provided in Reference 5) and to plant operating parameters.

3. Determination of Adequacy of ECCS

The RLBLOCA methodology uses a non-parametric statistical approach to determine values of PCT at the 95 percent probability level with 95 percent confidence (95/95). Total oxidation and total hydrogen generation are based on the 95/95 PCT case. The adequacy of the ECCS is demonstrated when these results satisfy the regulatory criteria set forth in Section 14.17.1.1.

14.17.1.2.2 Bounding Event Input

The plant analysis presented herein is for a CE-designed PWR, which has a 2x4-loop arrangement. There are two hot legs each with a U-tube steam generator and four cold legs each with a PCP. The PCS also includes one pressurizer connected to a hot leg. The core contains 204 15x15 AREVA NP fuel assemblies. The ECCS includes four SIT lines, each connecting to a cold leg pipe downstream of the pump discharge. The HPSI and LPSI lines tee into the SIT lines prior to their connection to the cold legs. The ECCS HPSI pumps are cross-connected. The single failure assumption renders one LPSI pump, two LPSI injection motor operated valves, and a HPSI pump inoperable. This results in one LPSI pump injecting through two valves into cold legs 1A (leg containing the break) and 1B, and one HPSI pump injecting through four valves in all four of the cold legs. This models the break in the same loop as the pressurizer, as directed by the RLBLOCA methodology. The RLBLOCA transients are of sufficiently short duration that the switchover to sump cooling water for ECCS pumped injection need not be considered.

The S-RELAP5 model explicitly describes the PCS, reactor vessel, pressurizer, and the ECCS. The model also describes the steam generator secondary side that is instantaneously isolated (closed main steam isolation valve and feedwater trip) at the time of the break. A steam generator tube plugging level of up to 15 percent per steam generator is assumed.

Plant input modeling parameters were provided by Entergy specifically for Palisades. As described in the AREVA NP RLBLOCA methodology, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. A list of the sampled parameters is given in Table 14.17.1-1. The LBLOCA phenomenological uncertainties are provided in Reference 5. Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in the analysis are given in Table 14.17.1-2. Plant data are analyzed to develop uncertainties for the process parameters sampled in the analyses. Table 14.17.1-3 presents a summary of the uncertainties used in the analyses. Two parameters, Safety Injection and Refueling Water Tank (SIRWT) temperature for ECCS pumped injection flows and diesel start time, are set at conservative bounding values for all calculations. Where applicable, the sampled parameter ranges are based on technical specification limits. Plant and design data are used to define range boundaries for some parameters, such as loop flow and containment temperature.

For the AREVA NP RLBLOCA evaluation model, significant containment parameters, as well as Nuclear Steam Supply System (NSSS) parameters, were established via a Phenomena Identification and Ranking Table (PIRT) process. Other model inputs are generally taken as nominal or conservatively biased. The PIRT outcome yielded two important (relative to PCT) containment parameters—containment pressure and temperature. As noted in Table 14.17.1-3, containment temperature is a sampled parameter. Containment pressure is indirectly ranged by sampling the containment volume (Table 14.17.1-3). The material, area, and thickness of the containment passive structural heat sinks, including new strainer sump, are given in Table 14.17.1-7. The containment initial and boundary conditions are given in Table 14.17.1-8. The containment-related technical specification minimum SIRWT temperature is used for the building sprays.

14.17.1.2.3 Analysis of Results

A case set of transient calculations was performed, 59 in total, which sampled the parameters listed in Table 14.17.1-1. For each transient calculation, PCT was calculated for a UO_2 rod and for gadolinia-bearing rods with concentrations of 2, 4, 6 and 8 w/o Gd_2O_3. The limiting PCT of 1,740 °F occurred in Case 22 for a 6 w/o Gd_2O_3 rod. The major parameters for the limiting transient are presented in Table 14.17.1-4. Table 14.17.1-5 lists the limiting PCT results for the hot fuel rod. The fraction of total hydrogen generated is conservatively bounded by the calculated total percent oxidation, which is well below the 1 percent limit. A nominal 50/50 PCT case, based on the 2 w/o Gd_2O_3 rod, was identified as Case 55. The nominal PCT is 1,403 °F. This result can be used to quantify the relative conservatism in the 95/95 result; in this analysis, it is 337 °F.

The hot fuel rod results are given in Table 14.17.1-5 and event times for the limiting PCT case are shown in Table 14.17.1-6, respectively.
Figure 14.17.1-1 shows linear scatter plots of the key parameters sampled for the calculations. Parameter labels appear to the left of each individual plot. These figures show the parameter sample ranges used in the analysis.
Figures 14.17.1-2 and 14.17.1-3 are PCT scatter plots versus the time of PCT and versus break size from the calculations, respectively.
Figure 14.17.1-4 and Figure 14.17.1-5 show the maximum oxidation and total oxidation versus PCT scatter plots for the 59 calculations, respectively.
Figures 14.17.1-6 through 14.17.1-16 present transient results for key parameters from the S-RELAP5 limiting case. Figure 14.17.1-6 is a PCT elevation-independent plot; this figure clearly indicates that the transient exhibits a sustained and stable quench.

14.17.1.3 Radiological Consequences

The radiological consequences from a loss of coolant accident are bounded by that for the Maximum Hypothetical Accident (MHA), described in Section 14.22.

14.17.1.4 Conclusions

A RLBLOCA analysis was performed for Palisades using NRC-approved AREVA NP RLBLOCA methods (Reference 5). Analysis results show that the limiting AREVA NP fuel case has a PCT of 1,740 °F, and a maximum oxidation thickness and hydrogen generation that fall well within regulatory requirements. Mixed-core effects are a non-issue since the core is completely fueled with 15x15 AREVA NP fuel assemblies.

The analysis supports operation at a nominal power level of 2,565.4 MWt (plus uncertainty), a steam generator tube plugging level of up to 15 percent in both steam generators, a Linear Heat Rate (LHR) of 15.28 kW/ft, an Total Radial Peaking Factor (F_r^T) of 2.04 with no axially-dependent power peaking limit and peak rod average exposures of up to 62,000 MWd/MTU. For a LBLOCA, all 10 CFR 50.46(b) criteria are met and operation of Palisades with AREVA NP-supplied 15x15 M5 clad fuel is justified.

14.17.2 SMALL BREAK LOCA

14.17.2.1 Event Description

The postulated SBLOCA is defined as a break in the PWR pressure boundary which has an area of up to approximately 10% of a cold leg pipe area. The most limiting break location is in the cold leg pipe on the discharge side of a Primary Coolant Pump (PCP). This break location results in the largest amount of inventory loss and the largest fraction of Emergency Core Cooling System (ECCS) fluid being ejected out through the break. This produces the greatest degree of core uncovery, the longest fuel rod heatup time, and consequently, the greatest challenge to the 10 CFR 50.46(b) criteria.

The SBLOCA event is characterized by a slow depressurization of the primary system with a reactor trip occurring on a Thermal Margin/Low Pressure (TM/LP) trip signal (low pressure floor). The Safety Injection Actuation Signal (SIAS) occurs when the system has further depressurized. The capacity and shutoff head of the High Pressure Safety Injection (HPSI) pumps are important parameters in the SBLOCA analysis. For the limiting break size, the rate of inventory loss from the primary system is large enough that the HPSI pumps cannot preclude significant core uncovery. The primary system depressurization rate is slow, extending the time required to reach the Safety Injection Tank (SIT) pressure or to recover core liquid level on HPSI flow. This tends to maximize the heatup time of the hot rod which produces the maximum Peak Cladding Temperature (PCT) and local cladding oxidation. Core recovery for the limiting break begins when the Safety Injection (SI) flow that is retained exceeds the mass flow rate out the break. For very small break sizes, the primary system pressure does not reach the SIT pressure.

14.17.2.2 Thermal-Hydraulic Analysis

14.17.2.2.1 Analysis Models

The AREVA NP SBLOCA evaluation model for event response of the plant and hot fuel rod used in this analysis (Reference 1) consists of two computer codes. The appropriate conservatisms, as prescribed by Appendix K of 10 CFR 50, are incorporated. This methodology has been reviewed and approved by the NRC to perform SBLOCA analyses. The two AREVA NP computer codes used in this analysis are:

1. The RODEX2-2A code was used to determine the burnup-dependent initial fuel conditions for the system calculations.

2. The S-RELAP5 code was used to predict the thermal-hydraulic response of the primary and secondary sides of the reactor system and the hot rod response.

The gap conditions used to initialize S-RELAP5 are taken at end-of-cycle (EOC), although a sensitivity case for middle of cycle (MOC) axial power was also analyzed. The use of EOC fuel rod conditions along with an EOC power shape bounds beginning-of-cycle (BOC) because the gap conductance is higher at EOC, the power shape is more top-skewed at EOC, and because the initial stored energy, although higher at BOC, has a negligible impact on SBLOCA results because the stored energy is dissipated long before core uncovery.

Properties for M5 cladding are incorporated into AREVA NP methods by the Reference 4 approved topical report.

14.17.2.2.2 Plant Description and Summary of Analysis Parameters

Palisades is a Combustion Engineering (CE) designed two-by-four loop PWR with two hot legs, four cold legs, and two vertical U-tube Steam Generators (SGs). The reactor has a rated core power of 2580.6 MWt (including uncertainty). The reactor vessel contains a downcomer, upper and lower plenums, and a reactor core containing 204 fuel assemblies. The hot legs connect the reactor vessel with the vertical U-tube SGs. Feedwater is injected into the downcomer of each SG. There are three Auxiliary Feedwater (AFW) pumps, two motor-driven and one turbine-driven. The ECCS contains two HPSI pumps, four SITs, and two Low Pressure Safety Injection (LPSI) pumps.

The primary coolant system (PCS) of the plant was nodalized in the S-RELAP5 model into control volumes interconnected by flow paths or "junctions." The model includes four SITs, a pressurizer, and two SGs with both primary and secondary sides modeled. All of the loops were modeled explicitly to provide an accurate representation of the plant. It was assumed that 15% of the tubes in each steam generator were plugged. The HPSI system was modeled to deliver to the four cold legs in the S-RELAP5 model. LPSI flow was not modeled since system pressure was not expected to fall below the shutoff head of the LPSI pumps.

The heat generation rate in the S-RELAP5 reactor core model was determined from reactor kinetics equations with actinide and decay heating as prescribed by Appendix K.

The analysis (Reference 3) assumed loss of offsite power concurrent with reactor scram on low pressurizer pressure. This assumption bounds the energy input to the PCS relative to an assumption of Loss of Offsite Power (LOOP) at event initiation with an associated earlier scram time. The single failure criterion required by Appendix K was satisfied by assuming the loss of one Emergency Diesel Generator (EDG), which resulted in the disabling of one HPSI pump and one motor-driven AFW pump. Thus, a single HPSI pump was assumed to be operable. No charging pump was credited. Initiation of the HPSI system was delayed by 40 seconds beyond the time of SIAS, representing the maximum Technical Specification delay required for EDG startup, sequencer delays for pump and motor-operated valve operation, and pump startup. The disabling of a motor-driven AFW pump would leave one motor-driven pump and the turbine-driven pump available. The initiation of the motor-driven pump was delayed 120 seconds beyond the time of the Auxiliary Feedwater Actuation Signal (AFAS) indicating low SG level (23.7% narrow range). The operator startup of the turbine-driven AFW pump was not credited in the analysis.

The input model included details of both main steam lines from the SGs to the turbine control valve, including the short Main Steam Safety Valve (MSSV) inlet piping lines connected to the main steam lines. The MSSVs were set to open at their nominal setpoints plus 3% tolerance. The valves were modeled to account for a blowdown of 4% of the nominal opening pressure.

SG blowdown flow was not modeled since this has an insignificant effect on the SBLOCA event. The initial secondary pressure was adjusted to be consistent with a 15% SGTP level.

Important system parameters and initial conditions used in the analysis are given in Table 14.17.2-1

14.17.2.2.3 Analytical Results

The single failure considered in the analysis was a failure of an EDG coincident with the loss of offsite power (Reference 3). This results in the loss of one HPSI pump (leaving only a single HPSI pump in operation) and a motor-driven AFW pump. Because the operator startup of the turbine-driven AFW pump is not credited, this failure mode is limiting since it causes the minimum HPSI and AFW flow rates.

SBLOCA break spectrum calculations were performed for break sizes of 0.04 ft^2, 0.05 ft^2, 0.06 ft^2, 0.08 ft^2, 0.10 ft^2, and 0.15 ft^2. The PCT results are presented in Table 14.17.2-2. The limiting break size was determined to be 0.08 ft^2. The break spectrum sizes were chosen to be fine enough to identify the limiting break size and to capture different recovery phenomena. The 0.15 ft^2 break was the largest size considered since the PCT monotonically

decreased from the limiting 0.08 ft^2 break. Analysis of even larger sizes would give lower PCTs as all four loop seals clear and SITs initiate earlier. Predicted event times are summarized in Table 14.17.2-3. Hot rod results are presented in Table 14.17.2-4.

The results for the limiting case, 0.08 ft^2 break, S-RELAP5 calculation are shown in Figure 14.17.2-1 through Figure 14.17.2-13. The following discussion pertains to the limiting case. System behavior for the other cases was similar, although event timing was different.

The break flow rate is shown in Figure 14.17.2-1. Single phase liquid flow began at the initiation of the break and continued until about 150 seconds when primary pressure reached saturation pressure and the break flow became a two-phase mixture. The decrease in flow rate at about 300 seconds was due to the transition from two-phase flow to single-phase vapor flow, which occurred following loop seal clearing.

The primary and secondary pressure responses are shown in Figure 14.17.2-2. The primary pressure decreased immediately after break initiation. When the primary pressure reached the TM/LP trip low pressure floor setpoint of 1585 psia, reactor scram occurred (Figure 14.17.2-3), followed by a turbine trip[1]. The turbine trip caused the secondary pressure to increase rapidly until the MSSVs opened, causing the secondary pressure to stabilize. Credit was not taken for non-safety grade plant systems, such as atmospheric dump valves (ADVs) and turbine bypass valves.

The primary coolant pumps tripped at scram on the assumed loss of offsite power and began to coastdown in speed, resulting in decreasing loop flow.

The total HPSI flow rate is shown in Figure 14.17.2-4. At approximately 96 seconds, HPSI flow began and increased as primary system pressure decreased. SIT flow (Figure 14.17.2-5) did not begin until the primary pressure reached the SIT pressure of 215 psia at 1690 seconds.

At approximately 282 seconds, liquid was expelled from the cold leg 1B loop seal piping (Figure 14.17.2-6), allowing steam to flow directly to the break, which allowed the primary pressure to decrease more rapidly. The loop seal in 1B cleared at 282 seconds; the loop seals in 1A, 2A and 2B (broken loop) did not clear. Figure 14.17.2-7 shows that the break flow transitioned to single-phase steam following loop seal clearing.

The primary system and reactor vessel fluid masses, shown in Figure 14.17.2-8, declined rapidly after event initiation. After loop seal clearing at approximately 282 seconds, the system mass inventory continued

[1] In the SBLOCA model, the only steam demand is the turbine. Other turbine building steam system loads like air ejectors and gland seal are ignored. Therefore, when the turbine is tripped, there is no steam demand.

to decline, but at a reduced rate, and the reactor vessel mass reached a minimum at 1698 seconds. At approximately that time, SIT flow began to reach the reactor vessel, which significantly increased system inventories.

The hot-channel-core-collapsed liquid level is shown in Figure 14.17.2-9. The core collapsed liquid level fell below the top of the core (11.05 feet) immediately after the break opened. The rate at which the level fell decreased after the break flow became two-phase (approximately 150 seconds). The hot node uncovery began at approximately 990 seconds, as shown by the increasing hot rod temperature shown in Figure 14.17.2-10. The liquid level continued to fall until PCS pressure fell low enough to initiate SIT flow at 1690 seconds.

In the 0.04 ft^2, 0.05 ft^2, and 0.06 ft^2 cases, the primary system pressure did not fall low enough to initiate SIT flow. For breaks of 0.08 ft^2 and larger, SIT discharge, minimum PCS inventory, and PCT were nearly coincident.

The PCT was calculated to be 1734°F for the 0.08 ft^2 limiting case, as seen on Figure 14.17.2-10. The calculations for each case were continued until well past the time of PCT, when a significant decrease in hot rod temperature and increase in hot channel liquid level were observed.

Secondary side results are shown in Figures 14.17.2-11 through 14.17.2-13. The secondary liquid levels reached the AFAS setpoint (23.7% narrow range) at approximately 25 seconds. Flow from the one available motor-driven AFW pump started approximately 120 seconds later. A constant flow rate was then supplied to each steam generator. MSSV flow, shown in Figure 14.17.2-13, ended soon after AFW flow began.

A calculation was also performed using the limiting middle-of-cycle (MOC) axial power shape for the limiting 0.08 ft^2 break case. In that case, the PCT was 1663ºF, versus 1734ºF for the limiting EOC case.

These calculations indicate that the case with an EOC axial power shape provides a bounding SBLOCA analysis.

14.17.2.3 Radiological Consequences

A radiological consequence analysis is not applicable to this event.

14.17.2.4 Conclusion

Calculations were performed for a spectrum of break sizes and axial power shapes. The limiting scenario for those calculations was a break size of 0.08 ft^2, EOC axial power shape and gap conductance, and the loss of one diesel generator. While stored energy at BOC is higher than EOC, EOC conditions were chosen because stored energy is dissipated long before core

uncovery and has a negligible impact on PCT. The PCT for this limiting case was 1734°F.

The analysis supports full-power operation at 2580.6 MW_t including uncertainty, a steam generator tube plugging level of up to 15% in each steam generator, a total radial peaking factor of 2.04, and a maximum LHR of 15.28 kW/ft.

14.17.3 REACTOR INTERNALS STRUCTURAL BEHAVIOR FOLLOWING A LOCA

14.17.3.1 Event Description

In the original Combustion Engineering (CE) analysis of this event, the consequences and effects of a postulated loss of coolant accident on the reactor internal structures was analyzed for PCS pipe breaks up to a double-ended rupture of a 42-inch pipe. Following a pipe rupture, two types of loading occur sequentially. The first is an impulse load of 15 to 30 milliseconds duration caused by rapid system depressurization from initial subcooled conditions to saturated conditions. This initial blowdown phase is followed by a two-phase fluid blowdown which persists for time periods varying up to several seconds, depending on the size of the postulated rupture. In the early portion of the blowdown, acoustic waves propagate through the PCS. For the saturated portion of the blowdown, the loadings on the reactor internals are associated with the fluid drag forces imposed by the high-velocity, two-phase fluid in its flow to the break location. The short-term impulse forces are generally greater than the long-term drag forces except for the loads on some of the control rod shrouds in case of a pipe rupture near the pressure vessel outlet nozzle.

Later, the structural adequacy of reactor internals was further evaluated as part of the CE Owner's Group Asymmetric Loads Program (References 8, 9, and 10). The initial phase of the evaluation consisted of a comparison of the design verification LOCA loads used in the original analysis and the asymmetric LOCA loads considering vessel motion. The three components of the load, including the vertical and horizontal shear forces and the horizontal moment, were compared to the original loads to determine if any portion of the load increased. Any area of the reactor internals with a load component higher than the original LOCA loads analysis was evaluated by performing a new analysis using asymmetric loads. The results of the load comparison indicated that the only areas of the reactor internals which did not show an increase in loads with the asymmetric load analysis were the core support barrel upper and lower flanges. No further analysis was performed for these components. All other areas of the reactor internals were reanalyzed using the asymmetric loads to compute stress intensities.

14.17.3.2 Thermal-Hydraulic Analysis

14.17.3.2.1 Analysis Method

In the original analysis, the structural behavior of the core due to propagation of the blowdown acoustic waves following a pipe rupture was evaluated using the Waterhammer Code (WHAM). WHAM calculates the impulse-type pressure loadings which the system is subjected to during passage of the pressure waves through the system.

The asymmetric loads analysis used CEFLASH-4B for the reactor internal hydraulic loads calculations, and used various codes for the reactor internal structural loads calculations, as described in References 8, 9, and 10.

14.17.3.2.2 Bounding Event Input

The original analysis for this event was performed by CE for the original FSAR. All inputs and assumptions are contained in the original calculation package, an internal CE document.

The asymmetric loads analysis was also performed by CE. Inputs and assumptions are contained in the Asymmetric Loads Program Final Report (Reference 8).

14.17.3.2.3 Analysis of Results

The blowdown-induced stresses and deflections induced during the blowdown are well below failure conditions. The results from the original FSAR analysis are given in Table 14.17.3-1. The results from the asymmetric loads analysis are given in Table 14.17.3-2.

14.17.3.3 Radiological Consequences

A radiological consequences analysis is not applicable for this event

14.17.3.4 Conclusions

It is concluded that the reactor vessel internal structures can withstand the forces caused by a large loss-of-coolant accident.

REFERENCES

1. EMF-2328(P)(A), PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, Framatome ANP, March 2001.

2. Delete

3. ANP-2712, Revision 0, Palisades Small Break LOCA Analysis with M5 Cladding, AREVA NP, November 2008.

4. BAW-10240(P)(A), Revision 0, Incorporation of M5 Properties in Framatome ANP Approved Methods, Framatome ANP, May 2004.

5. AREVA NP Document, EMF-2103(P)(A), Revision 0, Realistic Large Break LOCA Methodology, Framatome ANP, Inc., April 2003.

6. Technical Program Group, Quantifying Reactor Safety Margins, NUREG/CR-5249, EGG-2552, October 1989.

7. Wheat, Larry L., "CONTEMPT-LT A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-Of-Coolant-Accident," Aerojet Nuclear Company, TID-4500, ANCR-1219, June 1975

8. CE Owner's Group Asymmetric Loads Program Report, "Reactor Coolant System Asymmetric Loads Evaluation Program Final Report," Volumes 1, 2 and 3, dated June 30, 1980.

9. Combustion Engineering Report, "Response to Questions on the Reactor Coolant System Asymmetric Loads Evaluation Program Final Report," submitted to the NRC on July 31, 1981.

10. Letter from AW De Agazio (NRC) to KW Berry (CP Co), "Safety Evaluation on Asymmetric LOCA Loads - MPA D-010 - Palisades (TAC No. MO8621)," October 27, 1989.

11. Code of Federal Regulation, Title 10, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and Appendix K to 10CFR50, "ECCS Evaluation Models."

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14.18 CONTAINMENT PRESSURE AND TEMPERATURE ANALYSIS

14.18.1 LOCA ANALYSIS

14.18.1.1 Event Description

The response of the containment building to the failure of primary coolant system piping is analyzed to determine the pressure and temperature response versus time. The pressure response curve is used to determine whether or not the design pressure limit for the containment building would be exceeded following a LOCA. The temperature response curve is used to determine the acceptability of the EEQ components inside containment.

A LOCA is initiated by the rupture of the primary coolant system piping. The primary coolant will flash to steam and escape through the break. As the steam is released to the containment building, the pressure and temperature of the containment atmosphere quickly increase. The structures in containment will absorb energy and condense steam, counteracting the initial pressure and temperature increase. The containment air coolers and containment spray system, which are activated by the pressure rise, then act to reduce the pressure and temperature and remove energy released from decay heat.

The Palisades pressure and temperature transient analyses for the LOCA are performed using the GOTHIC 7.2a computer program (Reference 30).

14.18.1.2 Description of GOTHIC

GOTHIC 7.2a (Generation of Thermal-Hydraulic Information for Containment) is a program that solves the conservation equations for mass, momentum and energy for multi-component, multi-phase flow. The phase balance equations are coupled by mechanistic models for interface mass, energy and momentum transfer that cover the entire flow regime from bubbly to flow to film/drop flow, as well as single phase flows. The interface models allow for the possibility of thermal non-equilibrium between phases and unequal phase velocities. GOTHIC includes full treatment of the momentum transport terms in multi-dimensional models, with optional models for turbulent shear and turbulent mass and energy diffusion. Conservation equations are solved for up to three primary fields and up to two secondary fields. The primary fields include steam/gas mixture, continuous liquid and liquid droplet. The secondary fields include mist and ice.

For the primary fields, GOTHIC calculates the relative velocities between the separate but interacting fluid fields, including the effects of two-phase slip on pressure drop. GOTHIC also calculates heat transfer between phases, and between surfaces and the fluid. Reduced equations sets are solved for the secondary fields by the application of appropriate assumptions. The three primary fluid fields may be in thermal non-equilibrium in the same computational cell.

The steam/gas mixture is referred to as the vapor phase and is comprised of steam and, optionally, up to eight different non condensing gases. Mass balances are solved for each component of the steam/gas mixture, thereby providing the volume fraction of each type of gas in the mixture.

The mist field is included to track very small water droplets that form when the atmosphere becomes super saturated with steam.

The principal element of a model is a computational volume. As a minimum, a GOTHIC model consists of at least one lumped parameter volume. Separate volumes communicate through what are referred to as junctions or flow paths. A separate set of momentum equations are solved for junctions. Mass, momentum and energy can be added or removed at boundary conditions which are connected to volumes by flow paths.

Solid structures are referred to in GOTHIC as thermal conductors. Thermal conductors are modeled as one-dimensional slabs for which heat transfer occurs between the fluid and the conductor surfaces and, within a conductor, perpendicular to the surfaces. GOTHIC_S includes a general model for heat transfer between thermal conductors and the steam/gas mixture or the liquid. There is no direct heat transfer between thermal conductors and liquid droplets. Thermal conductors can exchange heat by thermal radiation. Any number of conductors can be assigned to a volume. Nodalization of a conductor allows variation of material properties in the direction of heat transfer. Heat generation within thermal conductors may be specified on a node-by-node basis.

Conduction models in GOTHIC provide that thermal conductors can be modeled by one of three geometrical shapes. These shapes are flat plate (wall), cylindrical tube and solid rod.

For walls, heat transfer is normal to the surface. For tubes and rods, heat transfer is radial. Several surface heat transfer models are available for modeling condensation and convection. Boundary conditions that may be specified for a conductor surface are convection, heat flux, and temperature.

GOTHIC includes an extensive set of models for operating equipment. These items include pumps and fans, valves and doors, heat exchangers and fan coolers, vacuum breakers, spray nozzles, cooler and heaters, volumetric fans (annular fans, deck fans, etc), ignitors (spark device used to ignite hydrogen burns) and pressure relief valves (PRVs).

Trips are used to control the on/off status of these components. Sense variables used to activate trips include simulation time, relative time, or the value of certain computed variables such as pressure, pressure differential and temperature. Trips can also be activated by control variables.

Control variables are variables that are defined by the user and can be applied as the independent variable in a forcing function and as a variable available for plotting, in addition to their use to activate trips. Through trip conditions and forcing functions, control variables can be used to control the system under study, either directly or by evaluation of correlations that are not already programmed.

14.18.1.3 <u>Thermal-Hydraulic Analysis</u>

14.18.1.3.1 Analysis Method

The base GOTHIC containment model consists of four control volumes: the containment, the primary coolant system, the SIRW tank and a shutdown cooling heat exchanger outlet volume. All control volumes except the shutdown cooling heat exchanger outlet volume, which represents the containment spray and HPSI piping, consists of a vapor and a liquid region.

The mass and energy additions following a LOCA are comprised of blowdown, decay heat generation, and sensible heat energy from the primary coolant system metal structure. The blowdown is modeled using a boundary condition connected to the containment control volume, and the decay heat and sensible heat energy are modeled using liquid phase heaters located in the primary coolant system control volume.

The Containment Spray System supplies cooling water from the SIRW tank to the containment atmosphere. The spray is dispersed through a series of nozzles attached to two ring headers located in the dome region. The containment sprays are modeled in the GOTHIC code by volumetric pump controlled by an on trip that represents containment high pressure, a trip that represents RAS and a table of time vs. spray flow.

The reactor safety injection system supplies water to the reactor core, modeled by the primary coolant system control volume, for decay heat removal. The LPSI and HPSI pumps take suction from the SIRW tank and are modeled in the GOTHIC code by volumetric pump controlled by an “on” trip that represents containment high pressure, a trip that represents RAS and a table of time versus safety injection flow.

When the water in the SIRW tank reaches a minimum level setpoint (RAS), the containment spray pump suction is automatically switched to the containment sump (recirculation mode), and the containment spray pump discharges to the shutdown cooling heat exchangers, which cools the water before it is sprayed into the containment atmosphere. Following RAS, the LPSI pump is turned off and the HPSI pump suction is switched to the discharge of the containment spray pump downstream of the shutdown cooling heat exchangers.

The shutdown cooling heat exchanger is modeled as a shell and U-tube heat exchanger with a single shell pass. The heat exchanger is located on the flow path connecting the containment control volume and the shutdown cooling heat exchanger outlet control volume. The hot sump water is passed through the tube side of the heat exchanger for cooling. The shell side uses Component Cooling Water (CCW) as the cooling medium. Table 14.18.1-5 shows the shutdown cooling heat exchanger primary and secondary flow rates.

After RAS, both the containment spray pump model and the HPSI pump model take suction from the shutdown cooling heat exchanger outlet control volume, with the containment spray pump discharging into the containment control volume and the HPSI pump discharging into the primary coolant system control volume. Both the containment spray pump model and the HPSI pump model are volumetric pumps controlled by an “on” trip that represents RAS and tables of time versus flow rate.

The forced-air recirculation cooling system is comprised of 4 containment air coolers which remove energy from the atmosphere by condensing steam. Following a DBA, only 3 of the 4 containment air coolers are available to provide cooling since service water to VHX-4 is isolated upon activation of the safety injection signal (SIS). These coolers are described in the GOTHIC model by heat exchanger components, inlet and outlet control volumes, and an associated volumetric fan controlled by “on” and “off” trips. Moisture condensed by the air coolers is assumed to fall immediately into the liquid region (sump).

The containment building and its internal structures will act as heat sinks throughout the event absorbing energy and condensing steam. Their thermal behavior is described by a one-dimension, multi-region heat-conduction equation. These heat conduction sections can act as heat sources or sinks. The Palisades containment building and internal structures have been divided into 27 separate heat conductors and are presented in Table 14.18.1-1. Most of these heat conductors are combinations of several sinks grouped together for modeling purposes. Where a decontaminable coating is applied to a surface, the coating has been included as another layer for the heat to penetrate. Air gaps are included in heat conductors with steel liner plates to account for the contact resistance between the steel and concrete.

Three primary system walls are also modeled, but they act as a heat source as opposed to a heat sink. These walls are lumped combinations of various metal structures that will be at a higher temperature than the PCS at the end of the blowdown phase. The energy from these walls is added to the PCS liquid volume until the wall temperature is in equilibrium with the PCS fluid. Once the primary system metal walls come into equilibrium with the PCS, they simply transfer heat to follow the primary coolant temperature. These heat sources are listed in Table 14.18.1-1.

Heat transfer to the heat conductors exposed to the containment atmosphere is determined by the Diffusion Layer Model (DLM) heat transfer correlation. The DLM calculates the thermal convection and the condensation rate at the wall considering the liquid film resistance and steam diffusion through the boundary layer rich in noncondensing gases.

Heat transfer to the heat conductors exposed to the water on the containment floor and in the sump is determined by natural convection correlations.

Two cases for single failures and corresponding engineered safeguard equipment availability were evaluated in Reference 33 to show the containment response to a LOCA. One case evaluated the failure of diesel generator 1-2 to start and the other case evaluated the failure of diesel generator 1-1 to start. Both cases assume a loss of offsite power. The safeguards equipment used for each case is tabulated on Table 14.18.1-2. The failure of diesel generator 1-2 to start is the more limiting case.

14.18.1.3.2 Bounding Event Input

A LOCA can occur from the rupture of the primary coolant system piping in a hot leg, a cold leg on the suction side of the primary coolant pumps or a cold leg on the discharge side of the primary coolant pumps. Each of these break locations have different characteristics for blowdown and containment response. The design basis break for Palisades is a double ended guillotine break in a hot leg. The original analysis that established this design basis was performed using a very simplistic model which was the available technology at the time. It was later thought that a cold leg break may be more limiting due to the effects of reflood and refill. In 1992, ABB Combustion Engineering was contracted to evaluate each of the possible break locations in conjunction with the spectrum of available safety injection equipment (References 18, 19 & 20). The spectrum of break locations was evaluated to ensure that the double ended hot leg break was still the limiting break location using up to date methodology and computer codes. The spectrum of available safety injection equipment was evaluated to ensure minimum available equipment is worst case since there is a trade-off between adding more mass and energy and providing more cooling through safety injection. The analyses that ABB CE performed did demonstrate that the design basis double ended hot leg break with minimum available safety injection equipment is limiting for containment response analyses. In 2008, this analysis was updated by Westinghouse to account for the additional mass and energy release in the 0.0 to 0.5 second time range (Reference 36).

The same initial conditions are assumed for all LOCA cases and are summarized in Tables 14.18.1-3 and 14.18.1-5. These initial conditions are chosen to approximate conservative conditions both within the containment and for heat removal from the containment.

The following conservatisms have been incorporated into the GOTHIC containment response model:

1. Bounding high initial containment pressures and temperatures are assumed.

2. A bounding low initial SIRW tank level is assumed.

3. A bounding high SIRW tank level is assumed for the RAS setpoint.

4. A bounding high SIRW tank temperature is assumed.

5. A bounding high service water temperature is assumed.

6. Bounding low flow rates are assumed for containment spray, HPSI, LPSI, CCW and service water.

7. The containment air coolers are turned off before the end of the event.

8. No heat transfer is credited between the containment sump and atmosphere.

9. No heat transfer is credited between the containment building and the outside atmosphere.

10. An air gap between steel liner plates and the adjacent concrete is included which reduces the effectiveness of the associated heat conductors.

11. The decay power energy input assumes that the fission product decay activity is based on core operation at 2,580.6 MWt with a 100% capacity factor. The fission product decay energy model is taken from Auxiliary Systems Branch Technical Position 9-2 in NUREG-0800 (Standard Review Plan) Subsection 9.2.5.

12. Leakage from containment is neglected for purposes of calculating pressure reduction or heat loss.

13. The discharge of safety injection tank water into the PCS is neglected. This is conservative because the safety injection tank water is significantly cooler than the PCS and would cool it such that boil-off into the containment atmosphere would be reduced.

14.18.1.3.3 Analysis of Results

The results of the analysis for the cases of a failure of diesel generator 1-1 and a failure of diesel generator 1-2 during a double-ended hot leg break LOCA are presented in Table 14.18.1-4. Plots of pressure and temperature versus time are shown in Figures 14.18.1-1 and 14.18.1-2 (References 33 and 37). The predicted peak pressure for both cases remains below the containment building design limit of 55 psig.

14.18.1.4 Radiological Consequences

A radiological consequences analysis is not applicable for this event.

14.18.1.5 Conclusion

It is concluded that the design pressure limit of the containment building will not be exceeded following a LOCA inside the building.

14.18.2 MSLB INSIDE CONTAINMENT

14.18.2.1 Event Description

The response of the containment building to the failure of an internal main steam line is analyzed to determine the pressure and temperature response versus time. The pressure response curve is used to determine whether or not the design pressure limit for the containment would be exceeded during the main steam line break (MSLB). The temperature response curve is used to set the upper limits of the EEQ analysis performed on various components inside the containment building.

A main steam line break event is initiated by the rupture of the main steam line leading from one of the two steam generators (i.e. the ruptured steam generator). The contents of the ruptured steam generator, along with some fraction of the contents of the intact steam generator, escape into the containment building, raising the pressure and temperature of the atmosphere inside the containment. During this process, the containment air coolers and containment spray system, activated by the pressure rise, begin removing energy from the containment atmosphere, countering the pressure and temperature rise. After a rapid initial rise, the pressure and temperature reach their peak values and then fall back to their original values.

Prior to the installation of the replacement steam generators, the MSLB containment response was reanalyzed by ABB-CE (References 10, 11, and 12). The new steam generators were determined to have greater secondary side inventories than the old steam generators. This greater inventory, combined with decreased levels of tube plugging in the new steam generators, necessitated the performance of the revised analysis.

Subsequent to the installation of the replacement steam generators, the containment response portions of the MSLB analyses were redone at Consumers Energy with the CONTEMPT EI/28 computer program. The current analysis was performed using GOTHIC 7.2a (Reference 32).

14.18.2.2 Thermal-Hydraulic Analysis

14.18.2.2.1 Analysis Method

The replacement steam generator (RSG) MSLB analysis was performed in accordance with the Standard Review Plan (SRP) Sections 6.2.1.1.A and 6.2.1.4, (Reference 8), and the applicable sections of the Palisades Technical Specifications. The primary objective of the analysis was to determine peak containment pressure and compare this result with the containment design pressure criterion of 55 psig. For the RSG, a spectrum of power levels (102%, 75%, 50%, 25%, and 0%) were analyzed. In addition, the most severe case (75% power) was reanalyzed to verify the most limiting single active failure. The coupled blowdown and containment analysis was performed using the NRC-approved SGNIII coupled primary, secondary and CONTRANS containment analysis computer code. This code, designed for secondary system pipe break analysis, is described in References 5 and 6. All significant equations, including those for the calculation of primary-to-secondary, core-to-coolant, and metal-to-coolant heat transfer, and for the calculation of steam separation and moisture carryover, are also described in References 5 and 6.

Following a postulated MSLB inside the containment, the contents of the ruptured steam generator will be released to the containment. Most of the contents of the other steam generator (ie, the intact one) will be isolated by the main steam isolation valve (MSIV) when the MSIV bypass valves are shut. Containment pressurization following a secondary side rupture strongly depends on how much of the break flow enters the containment atmosphere as steam.

The break type modeled in this analysis is a double-ended guillotine break. Per SRP Section 6.2.1.4.II.2, a double-ended guillotine break should be used as an initial case analysis, with subsequent cases decreasing the break size until there is no liquid entrainment in the break flow. MSLB break flows can be pure steam or two-phase. With a pure steam blowdown, all of the break flow enters the containment atmosphere and is uniformly mixed. With a two-phase blowdown, part of the liquid in the break flow boils off in the containment and is also added to the atmosphere, while the rest falls to the sump and contributes nothing to containment pressurization. For MSLB cases with large break areas, steam cannot escape fast enough from the two-phase region of the ruptured steam generator, and the two-phase level rises rapidly to the steam line nozzle. A two-phase blowdown results. The duration of this blowdown is short; therefore, little primary-to-secondary heat transfer takes place and the break flow is largely liquid.

For MSLB cases with small break areas, steam can escape fast enough from the two-phase region of the ruptured steam generator so that the level swell does not reach the steam line nozzle. A pure steam blowdown results. Because the pressure-reducing effects of active and passive containment heat sinks are more significant for smaller break sizes (due to the longer blowdown), the highest peak containment pressure resulting from a MSLB typically occurs for that case where the break area is the maximum at which a pure steam blowdown can occur. The potential for steam generator two-phase level swell following a MSLB increases as power level decreases; therefore, a spectrum of power levels and break sizes is normally analyzed to determine which one results in the peak MSLB containment pressure.

A spectrum of break sizes was not analyzed for the replacement steam generators because the replacement steam generators are equipped with flow restrictors at the outlet nozzles. These flow restrictors have a flow area of 30% of the main steam line area or 1.89 ft^2. They limit the steam flow for all cases presented in this analysis such that the break flow is entirely steam regardless of the break size chosen.

To permit a determination of the effect of a MSLB upon containment pressure, the replacement steam generator analyses were performed at 102, 75, 50, 25 and 0% power. An analysis of a spectrum of power levels was necessary because of the competing effects of initial steam generator inventory and feed water flow for the various power levels. Initial steam generator inventories are highest at low power levels, but feedwater flow additions to the ruptured steam generator are the lowest. For a constant initial steam generator level, as initial power levels increase, the corresponding reduced steam generator pressure results in lower initial secondary mass inventory. However, the feedwater flow rate and associated integrated inventory addition to the ruptured steam generator increases. These competing factors are resolved by analyzing a complete spectrum of initial power levels.

The contribution to containment pressure from feedwater flow is determined by feedwater flow addition to the ruptured steam generator and the boiling off of the feedwater by primary to secondary heat transfer. The feedwater flow is the sum of the pumped main feedwater flow prior to isolation, the auxiliary feedwater flow of 200 gpm (165 gpm for the 0% power case), which is conservatively modeled to go entirely to the ruptured steam generator, plus the isentropic expansion of the fluid between the ruptured steam generator and its MFW regulating valve. The modeling of the feedwater flow prior to isolation of the steam generators is accomplished by a plant specific MFW flow algorithm developed by CPCo (Reference 7) and incorporated into SGNIII. This algorithm provides a ramping function for shutting the MFW regulating valves over a 22 second time period; it also addresses feedwater spiking by accounting for the diversion of main feedwater to the affected steam generator caused by the pressure differential between the steam generators. The algorithm conservatively ignored MFW pump ramp down following turbine trip.

Steam line capacity to the MSIVs is modeled by adding the line volume and steam mass to the steam generator steam space volumes and masses. Due to the SGNIII code input limitations relative to the operation of the MSIVs, all piping volume downstream of the MSIVs to the Turbine Admission Valves is modeled as a separate break flow, conservatively emptying its steam contents to the containment over a ten second interval. The section between the intact steam generator and its MSIV will be isolated from the containment when the MSIV shuts. Due to the operation of the MSIVs, the MSIV nearest to the ruptured steam generator will remain open.

Following closure of the MFW regulating valves, there is an inventory of feedwater between the regulating valve and the ruptured steam generator. As the ruptured steam generator depressurizes, this inventory starts to boil. As the steam in the line expands, the feedwater inventory is pushed into the steam generator and is boiled off by primary to secondary heat transfer. This isentropic expansion of the feedwater inventory into the ruptured steam generator has been modeled in the SGNIII code.

The original containment response calculation for the replacement steam generators determined that the 75% power case was limiting for pressure and the 102% power case was limiting for temperature. Subsequent to this analysis, it was recognized that the 0% power case did not analyze the condition where the MSIV bypass valves were open. Earlier 0% power MSLB cases assumed that the MSIV bypass valves were closed immediately after the MSIVs were opened. Thus, for the 0% power case, closure of the MSIV on Containment High Pressure (CHP) was sufficient to isolate the intact steam generator. EA-A-PAL-91-196 (Reference 15) pointed out that the MSIV bypass valves were kept open for a while at 0% power. These valves do not have an automatic closure on CHP. Thus, in the case of a MSLB at 0% power, the intact steam generator could continue to release steam via the MSIV bypass valve. This steam would travel down the steam line, through the cross-connect, up the steam line to the failed steam generator, and escape out the break into the containment. When accounting for this line-up, the 0% power case becomes limiting relative to peak pressure.

When the minimum technical specification required PCS flow rate was reduced, Reference 25 demonstrated that References 10, 11 and 12 remain valid.

The Consumers Energy analyses of the containment response are performed using GOTHIC 7.2a, which is discussed in more detail in Section 14.18.1.2. These GOTHIC calculations use the blowdown calculated by the CONTRANS code for the replacement Steam Generators.

14.18.2.2.2 Bounding Event Input

For the most limiting failure, the NRC found that a double steam generator blowdown or a single steam generator blowdown with continued feedwater, although more severe than the licensing basis MSLB, is not expected to result in unacceptable consequences (Reference 29). Therefore, the current single failure for the MSLB is the failure of the most limiting Safety Injection Signal or Containment

High Pressure electrical relay which disables the associated safety equipment. The current analysis is documented in Reference 32.

Nominal values for some important input parameters are presented in Tables 14.18.2-1 and 14.18.2-2. Table 14.18.2-1 gives the values for parameters which remain constant for all the cases analyzed. Table 14.18.2-2 provides values for those parameters which are power dependent. The heat conductor data used in this analysis, consisting of containment walls and structures and some equipment, is presented in Table 14.18.2-4.

Cases were conservatively run at 102% power, and at 0% power with the MSIV bypass valves open, with and without offsite power available and the loss of various electrical relays or failure of an emergency diesel generator. The most-limiting relay for pressure was determined in Reference 32 to be the failure of containment high pressure relay 5P-7. All failures (5P-7, 5P-8, SIS-5, failure of an emergency diesel generator and SIS-6) are evaluated relative to EEQ.

14.18.2.2.3 Analysis of Results

The results for the limiting MSLB Containment cases are presented in Table 14.18.2-3. The results are for those cases redone by Palisades using the GOTHIC code. The peak containment pressure case is the 0% power MSLB with open MSIV bypass valves for a failure of relay 5P-7 (Figure 14.18.2-1). The peak pressure is below the design maximum containment pressure of 55 psig. The limiting temperature profiles relative to EEQ are:

- Failure of containment high pressure relay 5P-7 at an initial power of 0%.
- Failure of containment high pressure relay 5P-7 at an initial power of 102%.
- Failure of safety injection signal relay SIS-6 at an initial power of 0%.
- Failure of safety injection signal relay SIS-6 at an initial power of 102%.
- Failure of the 1-2 emergency diesel generator at an initial power of 0%.

At different times, different failure conditions yield the highest temperature, as is shown in Figure 14.18.2-2 (References 32 and 37).

The temperatures profiles have been evaluated as acceptable, as discussed in Reference 32.

With a LOOP, the PCS flow drops and natural circulation develops in the primary system. Thus, the heat transfer from primary to secondary occurs more slowly, the ruptured steam generator blowdown takes more time, and the containment heat removal systems become more effective in maintaining lower containment pressure. This has the affect of making the cases with off site power available generally more bounding. However, in the 1-2 emergency diesel generator failure case, no containment heat removal is credited once the containment spray pumps are stopped on low SIRW level. Thus, this case becomes bounding long term with respect to the EEQ profile.

14.18.2.3 Radiological Consequences

A radiological consequence analysis is not performed as part of the MSLB containment analysis: the release of radionuclides is not a part of this event.

14.18.2.4 Conclusion

It is concluded that the containment response to an internal main steam line break does not exceed the design limits for the building.

14.18.3 CONTAINMENT INTERNAL STRUCTURE EVALUATION

14.18.3.1 Event Description

The GOTHIC computer program, as described in Section 14.18.1.2, was used to calculate the Pressure -Time transients within the reactor cavity and the two steam generator compartments following a loss of coolant accident up to a 42-inch double-ended LOCA.

14.18.3.2 Thermal Hydraulic Analysis

14.18.3.2.1 Analysis Method

GOTHIC was used to perform pressure vs. time transient calculations for the steam generator and reactor compartments.

The calculations are based on the conservation of mass, momentum, and energy. The blowdown mass is allowed to flash to the total cavity or compartment pressure. No heat transfer to the structures is allowed; however, a thermal equilibrium is established in the cavity or compartment atmosphere by heat transfer with no associated mass transfer.

The ensuing flow from the cavity or compartment follows orifice flow relations with the entrance and friction losses included in the flow coefficient for each case period. The vent area from the cavity or compartment controls the differential pressure transient. The peak differential pressures are summarized at the end of this section for the cases discussed below.

14.18.3.2.2 Bounding Event Input

The calculations (References 34 and 35) were performed for 4 separate structures, the Reactor Cavity, the Reactor Cavity floor, and the two Steam Generator compartments. Each structure is discussed on the following pages.

Reactor Cavity

There are three types of openings for flow out of the reactor cavity into the main containment volume:

1. Total clearance area around the main coolant piping 24.2 ft^2
2. 30-inch relief tube (after rupture disc bursts) 4.75 ft^2
3. Clearance around the refueling seal:
   a. Before seal breaks 4.77 ft^2
   b. After the seal breaks 82.23 ft^2

Immediately after a pipe rupture, the volume available to the expanding steam within the reactor cavity is limited by the refueling seal.

The refueling seal will break away when the differential pressure reaches 5.8 psi. The seal is assumed to lift vertically. The calculation includes the effect of the variation in flow area with time as the seal is displaced.

The additional flow area through the 30-inch access pipe becomes available as the standpipe rupture disc bursts. Sensitivity studies on the break away pressure for the insulation support structure demonstrate that the insulation support modeling details do not impact reactor cavity pressurization; therefore the insulation support is not explicitly modeled.

Both guillotine- and slot-type primary pipe failures have been considered, however, the slot type break has been found to be more limiting with respect to energy release for compartment pressurization analysis (Reference 36):

1. For the guillotine-type break the coolant pipes are partially restrained by the cavity walls and the shielding brick. Restraining the primary pipes in this manner gives flow areas of 0.62 and 0.26 ft^2 for guillotine failures of the 30-inch and 42-inch lines which result in cavity pressurization to 25.0 and 10.0 psi, respectively (Reference 31).

2. The slot-type failure considers a maximum failure length of two times the inside pipe diameter and a maximum failure area which is equal to the pipe cross-sectional area. Calculation of the actual flow area due to a failure considers two effects: first, as a slot opens, the flow area increases; and, second, as the slot opens, the flow area between the edges of the slot and the shielding brick decreases. These two effects result in a maximum area for flow from the failed pipe to the cavity. Slot failures are considered to have uniform width over their entire length. In all cases the failure is taken to originate at the reactor nozzle weld, which allows for one foot of slot within the cavity and the remainder in the tunnel. Total area discharging into the cavity is taken as one-half of the area inside the tunnel plus all of the area inside the cavity. Rupture of the 30-inch and 42-inch pipes yields flow areas of 1.55 and 2.0 ft^2 which results in cavity pressurization to 67.7 and 52.4 psig, respectively. These pressures are higher than those which result from the guillotine-type failure and they are therefore the design bases for the reactor cavity (Reference 34).

Reactor Cavity Floor

The containment sump is located directly below the reactor cavity. If a primary coolant pipe rupture occurs outside the cavity, the sump and the volume below the lower insulation support will pressurize due to flow inward from the main containment. Flow into the sump is through five 16-inch pipes and one 24-inch pipe (total area 10.1 ft^2). The six 4-inch floor drains which connect to the sump, and the two 4-inch sump vent lines were evaluated and found to have a negligible impact on the peak cavity floor differential pressure. The flow into the reactor cavity is through a 30-inch (0.25 inch wall) rupture disc (4.75 ft^2). This difference in available flow area will result in a higher pressure in the sump and a net upward force on the cavity floor.

For a 42-inch double-ended rupture in the main containment, the differential (uplift) pressure on the cavity floor does not exceed 0.4 psi, based on a rupture disc bursting pressure of 3.8 psi. This is less than the design pressure and, hence, the floor will not lift. The volumes, flow areas and orifice coefficients used for this calculation are shown in Table 14.18.3-1.

Steam Generator Compartment

For the steam generator compartments only the double-ended guillotine slot break in the hot leg is considered since during the time of interest in this analysis it provides the largest energy and mass release. The effective flow areas for the north and south compartments, their volumes and peak pressures are shown on Table 14.18.3-2. Orifice coefficents were developed independantly for each of the vent paths from the steam generator compartments based on their individual geometries. These values can be found in Reference 35.

14.18.3.2.3 Analysis of Results

In all the cases, the calculated maximum differential pressure is less than the design pressure. These pressures and the design value for the appropriate case are summarized in Table 14.18.3-3

14.18.3.3 Radiological Consequences

A radiological consequences analysis is not applicable for this event.

14.18.3.4 Conclusion

It is concluded that the containment internal structures will withstand the forces generated by a LOCA without failure.

REFERENCES

1. Deleted

2. Deleted

3. Deleted

4. Deleted

5. CESSAR, "Combustion Engineering Standard Safety Analysis Report", Combustion Engineering, Inc., docketed December 19, 1973, (2338/1082).

6. R. C. Mitchell topical report, CENPD-140-A, dated June 1976, "Containment Thermodynamic Analysis".

7. Engineering Analysis EA-P-SDW-90-002-01, "Derivation of a MSLB Main Feedwater Flow Algorithm," Consumers Power Company, June 1990, (D240/0095).

8. NUREG-0800, U.S. NRC "Standard Review Plan", Revision 2, dated July, 1981.

9. Deleted

10. M. J. Gancarz calculation 001-NT90-C-0002, dated 1/29/91, "Palisades MSLB Containment Analysis for Replacement Steam Generator Project," ABB-Combustion Engineering, (C621/0989).

11. W. G. Dove, Jr. (ABB-CE) letter NT-90-0975 to RJGerling (CPCo), dated July 27, 1990, "Results of Palisades Main Steam Line Break Analysis for the Replacement Steam Generator Project", (C457/1711).

12. R. C. Whipple letter OPS-91-0467 to RJGerling, (CPCo), dated April 29, 1991, "Transmittal of Calculation Revision for Main Steam Line Break PCS Flow Rate Sensitivity Study, 001-NT90-C-0002," dated April 12, 1991.

13. Deleted

14. Deleted

15. EA-A-PAL-91-196, "Palisades Containment Building Response to a Main Steam Line Break at Hot Zero Power with Open MSIV Bypass Valves," Consumers Power Company, February 1992, (F239/0004), Superseded.

16. Deleted

17. Deleted

18. Calculation Number 001-AS93-C-002, "CEFLASH-4A LOCA Containment Blowdown Analysis for Palisades at 102% Power (2530.6 MWt)," ABB Combustion Engineering, July 1993, (F555/0766).

19. Calculation Number 001-AS93-C-003, "Palisades Post-Blowdown LOCA Mass and Energy Release Analysis Using the FLOOD-MOD2 Code," ABB Combustion Engineering, July 1993, (F555/1660).

20. Calculation Number 001-AS93-C-004, "Palisades LOCA Containment Peak Pressure/Temperature Analysis," ABB Combustion Engineering, July 1993, (F555/1502).

21. Deleted

22. Deleted

23. Deleted

24. Deleted

25. ST98-C-326, "Assessment of Reduced PCS Flowrate on SGTR and Containment Analyses for Palisades Plant," June 15, 1996, memo from Michael J. Gancarz (ABB) to Greg A. Baustian.

26. Superseded by Reference 32.

27. Deleted

28. Superseded by Reference 33.

29. Letter from HR Denton (NRC) to FW Buckman (Consumers) dated February 28, 1986 (ie, Single Failure MSLB SER), (3719/1117).

30. NAI 8907-02, Revision 17, “GOTHIC Containment Analysis Package User Manual,” Version 7.2a, January 2006.

31. Letter from DJ Oliver (Bechtel) to WC Cooper (Consumers) dated May 29, 1968, "Compartment Pressurization Calculations," (0038/0636).

32. EA-GOTHIC-04-09, Revision 3, "Containment Response to an MSLB Using GOTHIC 7.2a.”

33. EA-GOTHIC-04-08, Revision 3, "Containment Response to a LOCA Using GOTHIC 7.2a."

34. NAI-1431-001, Revision 0, “Palisades Nuclear Plant Reactor Cavity Pressurization Evaluation”, December 2008.

35. NAI-1384-002 Rev. 0, “Palisades Nuclear Plant Containment Compartment Analysis”, March 2009.

36. CN-OA-08-54 Rev. 0, “CEFLASH-4A LOCA Containment Blowdown DEHLS Mass and Energy Release Information for Palisades”, September 2008.

37. Engineering Change EC13670, “EEQ New Temperature Design Basis Calculations for Normal and Shutdown Time Periods Outside Containment and Inside Containment Temperature Profiles.”

14.19 FUEL HANDLING INCIDENT

14.19.1 EVENT DESCRIPTION

The possibility of an accident with significant consequences during refueling is remote due to the many physical limitations imposed upon refueling operations. Administrative restrictions on refueling procedures provide additional margin.

Before refueling operations start, the boron concentration of the primary coolant is increased to refueling boron concentration and is verified by chemical analysis. With this boron concentration, a shutdown margin of $\geq 5\%\Delta\rho$ is maintained even with all control rods removed from the reactor. Prior to the removal of the reactor vessel head, verification of complete insertion of all control rods is obtained from a visual check of each control drive position indicator. Each control rod is then individually uncoupled from the control drive mechanism drive shaft. Positive indication of uncoupling is obtained by rotating the inner tool at least one full turn at completion of the uncoupling procedure. Rotation cannot occur if the control rod is connected. The control drive mechanism is then energized to withdraw. Once this procedure has been completed for each control rod and each drive mechanism is at its upper limit, the vessel head is removed. During control drive mechanism withdrawal and reactor vessel head removal, the count rate is monitored as additional assurance that control rods are not being inadvertently removed.

Fuel handling hoists and manipulators are designed so that it is not possible to raise fuel bundles above a position which provides the minimum water shield requirements. This constraint applies in fuel handling areas inside containment and in the spent fuel pool area. In addition to these safeguards, direct radiation monitors at fuel handling areas give operating personnel audible and visual warning of high radiation levels. Interlocks are also provided to prevent tilt machine rotation during insertion and removal of spent fuel bundles. Fuel storage pool integrity is assured by designing the pool and storage racks as Class 1 structures.

In the spent fuel storage area, the design of storage racks and inspection elevator is such that fuel is always in a subcritical geometrical array based on zero boron concentration in the fuel storage pool water. In addition, during refueling operations, fuel pool water will contain a minimum of 1,720 ppm of boron. This is well above the 850 ppm required to maintain k-effective ≤0.95 for two 4.95 wt% Uranium enriched fuel assemblies placed in the tilt machine (Reference 7). Adequate cooling of fuel during handling and storage is supplied by natural convection of the surrounding water. An adequate supply of cooled water is assured by the spent fuel pool cooling system. At no time during transfer from the reactor core to the storage location is the fuel removed from the water. The fuel handling equipment is described in detail in Section 9.11.

The criticality analysis performed to support crediting soluble boron in the Spent Fuel Pool (SFP) determined that a minimum boron concentration of 850 ppm would provide a k-effective of less than or equal to 0.95. In order to ensure that the design-basis k-effective of 0.95 is not exceeded due to potential dilution events, a boron dilution analysis to support this criticality analysis was also performed (Reference 10). As a result, it was established that a boron concentration of greater than or equal to 1720 ppm provides adequate margin for fuel assembly storage and movement within the SFP.

Based on the creditable dilution events evaluated, the 1 ½ inch fire hose station is the only system with practically an infinite water storage source (Lake Michigan) that could provide the necessary volume of water needed to dilute the SFP to 850 ppm. However, with this fire hose, it would take over 9 hours to dilute the SFP soluble boron concentration from 1720 ppm to 850 ppm. Thus, if an SFP dilution were to occur from this system, reasonable assurance exists that it would be identified and suppressed by an operator before the 0.95 k-effective limit is reached. As an additional measure, a fuel pool high level alarm was added to give an earlier warning of fuel pool increases which could lead to dilution of the soluble boron concentration.

The dilution analysis concluded that an unplanned or inadvertent event that would dilute the SFP is not credible for Palisades. Sufficient time is available to detect and suppress the worst dilution event that can occur from the minimum TS boron concentration to the boron concentration required to maintain the 0.95 k-effective design-basis limit.

Fuel failure during refueling as a result of inadvertent criticality or overheating during transfer is highly improbable. Similarly, damage to a fuel bundle as a consequence of external forces is also improbable. Operating procedures prohibit the handling of heavy objects such as shipping casks above the fuel storage rack. Inadvertent disengagement of the fuel bundle from the fuel handling machine is prevented by interlocks; consequently, the probability of dropping and damaging a fuel bundle is low.

14.19.2 THERMAL-HYDRAULIC ANALYSIS

A thermal-hydraulic analysis is not applicable for this event.

14.19.3 RADIOLOGICAL CONSEQUENCES

14.19.3.1 Analysis Method

For the purpose of defining the upper limit of the consequences of a fuel handling accident, it is assumed that the fuel bundle is dropped during handling. Because of interlocks and procedural and administrative controls, such an event is unlikely. However, if the bundle is damaged to the extent that a number of fuel rods fail, the accumulated fission gases and iodines in the fuel rod gap could be released into the surrounding water. Release of fission products which are not in the gap; ie, in the fuel matrix, is negligible because the low fuel temperature during refueling reduces diffusion through the fuel to an insignificant amount (Reference 1).

The fuel bundles are stored within the spent fuel rack which is an eggcrate structure at the bottom of the spent fuel pool. When the fuel bundles are resting in their normal position within the spent fuel rack, the top of the rack extends above the tops of the stored fuel bundles. Because of the configuration and construction of the spent fuel storage racks, a dropped fuel bundle can strike no more than one fuel bundle in the storage racks. Impact can occur only between the ends of the involved fuel bundles (the bottom end fitting of the dropped fuel bundle striking the top end fitting of a stored fuel bundle). In the reactor core, however, it would be possible for a dropped fuel bundle to strike other fuel bundles. The results of analyses of the energy absorption capability of the fuel bundles indicate that a fuel bundle dropped in this manner is capable of absorbing the kinetic energy of the drop without causing any fuel rod failures.

As part of References 4 and 5, the NRC performed an independent calculation and assessment for a fuel handling accident. Because of a concern for radiation embrittlement of fuel cladding material, the NRC analysis assumed all the fuel rods in the equivalent of an entire assembly failed (216 rods), as opposed to CPCo's original assumption that a single outer row (13 rods) failed.

In Reference 6, the fuel handling accident was reanalyzed to account for increased radial power peaking factors and higher fuel burnup levels and to be consistent with the methodology described in Regulatory Guide 1.183, Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," (Reference 2). Increased radial peaking factors affects the fission product inventory of the peak assembly and increased burnup levels can affect the amount of some fission products that migrate to the fuel-clad gap. This analysis remained consistent with the NRC's assumption that all of the fuel rods in the dropped assembly would fail.

14.19.3.2 <u>Bounding Event Input</u>

To bound the consequences of the event, the fuel handling accident was assumed to occur in containment two days after shutdown, which is the earliest time that the Operating Requirements Manual allows fuel movement. No containment isolation was assumed to occur due to containment area radiation monitor alarms. Therefore, the released fission products escape to the environment with no credit for filtration. If the fuel handling accident were to occur in the spent fuel pool, the release would be filtered since the Technical Specifications require the charcoal filters to be operating for fuel moves with less than 30 days decay. Sensitivity cases were evaluated to determine the impact of various levels of release filtration.

The source term used is based on plant operation at 102% of 2,650 MWt. A conservative fuel gap activity based on the peak fuel assembly is assumed to be released to the refueling water. Credit for partial retention of iodine in the refueling water was taken. Significant parameters used in the fuel handling accident analysis are given in Table 14.19-1.

14.19.3.3 <u>Analysis of Results</u>

The offsite doses from the worst case fuel handling accident are shown in Table 14.1-6 to be within the Regulatory Guide 1.183 limits. These results are conservative considering that a fuel handling accident inside containment with a significant fission product release would alarm the containment area radiation monitors and trigger a containment isolation signal. This would significantly reduce the release to the environment and hence reduce the offsite doses.

If the fuel handling accident were to occur in the spent fuel pool, the radioactive release would pass through the charcoal filters of the fuel handling area exhaust. This would also significantly reduce the release to the environment and the offsite doses.

The results of the NRC analysis from References 4 and 5 concluded that the consequences of a fuel handling accident in the spent fuel area are acceptable with or without the charcoal filters operating. In Reference 5, the NRC stated, "The dose with the filter system operating was calculated to be 9 rem to the thyroid. If the filtration was not operating, the dose would have been 91 rem which is still 'appropriately within the guidelines' of 10 CFR 100 (ie, < 100 rem thyroid)." If the fuel has decayed for 30 days or greater, the dose consequences from a fuel handling accident would be of the same magnitude without the filters operating as the dose would be with the filters operating and the fuel decayed for only two days.

14.19.4 CONCLUSIONS

The potential offsite doses resulting from a credible fuel handling accident in the spent fuel pool area or containment building are less than the guidelines of Regulatory Guide 1.183 (Reference 6). The doses to control room personnel are discussed in Section 14.24.

REFERENCES

1. 1980 Palisades FSAR, Chapter 14, Section 19.2.

2. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, July 2000.

3. Deleted

4. Letter, Dennis L Zieman (NRC) to David Bixel (CP Co), dated June 21, 1979, Subject: Safety Evaluation of a Fuel Handling Accident Inside Containment, (2600/0540).

5. Letter, Walter Paulson (NRC) to David VandeWalle (CP Co), dated May 22, 1984, Subject: Technical Specification Amendment 81, (3143/1750).

6. NAI-1149-016, Revision 1, Palisades Design Basis Fuel Handling Accident AST Radiological Analysis, June 2005.

7. EA-SFP-99-03, Rev 0, “New Fuel Storage Fuel Pool and Fuel Handling Criticality Safety Analysis,” EC0013489.

8. Deleted.

9. Deleted.

10. EA-WJB-00-01, Revision 2, “Spent Fuel Pool Dilution Analysis,” EC0044749

14.20 LIQUID WASTE INCIDENT

14.20.1 EVENT DESCRIPTION

Accidents which might result in release of activity to the environs involve rupture or leakage from the liquid waste system components, or the accidental release to the circulating water discharge canal of the contents of one of the radioactive liquid storage tanks.

14.20.2 THERMAL-HYDRAULIC ANALYSIS

A thermal-hydraulic analysis is not applicable for this event.

14.20.3 RADIOLOGICAL CONSEQUENCES

14.20.3.1 Analysis Method

Activity release from the liquid waste system to the environs can occur only by accidental discharge to the circulating water discharge canal, by a rupture of the volume control tank with the ensuing gaseous release as discussed in Section 14.21, or by a failure of liquid storage tanks T-90 or T-91.

All components (except tanks T-90 and T-91 discussed below) of the liquid radwaste system are located within the containment and auxiliary buildings. Piping between the containment and the auxiliary building is run within the containment base slab and within the auxiliary building. Any leakage or spillage from any component will be collected by the floor drains and sumps and will drain to the waste receiver tanks in the liquid waste system. The liquid storage tanks and the volume control tank are protected against overpressurization and equipped with level instrumentation and alarms. The floor drains and sumps will contain the leakage or spillage within the containment or auxiliary buildings. Thus, the leakage or spillage will contaminate only the local area to which it spilled.

Liquid storage tanks T-90 and T-91 are not located within the containment or auxiliary buildings. However, both tanks are equipped with level indication and level alarms, and internal overflows back to the auxiliary building. Both tanks have administrative controls that maintain tank activity concentration within the limits described in Section 11.2.3.1, thereby ensuring that 10 CFR 20 dose limits would not be exceeded in the event of a tank failure.

Evaluation of the credibility of an accidental release of radioactive liquid to the circulating water canal is based on the following analysis of the waste discharge operating procedure, monitoring functions and failure consequence.

14.20.3.2 Bounding Event Input

In general, the procedure for discharging liquid wastes to the circulating water canal is as follows:

1. A batch of waste is collected in one of the waste tanks and the tank is isolated.

2. This batch is thoroughly mixed by recirculation and a sample is taken for radiochemical analysis.

3. If the sample analysis indicates that the quantity of activity to be released is within permissible limits as defined by 10 CFR 20, the quantity of activity is recorded corresponding to the tank volume and its activity concentration. If the release is not within permissible limits, the contents of the waste tank are retained in the tank or recycled through the treatment system.

4. If the activity can be released within permissible limits, three valves downstream of the waste radiation monitor must be opened by operator action. The first valve is a manually operated valve that is normally kept locked closed. The next two valves are air to open, fail closed. The first air operated valve in the flow direction is downstream from and controlled by a radiation monitor in the discharge line. This radiation monitor will automatically trip the valve closed on receipt of a high radiation signal. This monitor is described in Section 11.5. The second air operated valve is interlocked with the dilution water pumps and will close if these pumps are not running for dilution of the discharged wastes.

14.20.3.3 Analysis of Results

The discharge of liquid waste is governed by automatic interlock and administrative control. In addition, there is a monitor in the circulating water discharge canal to provide a backup to the discharge process monitor by alarming any discharge to Lake Michigan in excess of 10 CFR 20 limits.

These discharge monitors are provided with a check source to permit testing of a monitor and its circuitry from the control room prior to any liquid waste discharge. Failure of a monitor or its circuitry at any time is annunciated in the control room. Any failure in either the monitor or its circuitry will trip closed the valve in the liquid waste discharge line.

14.20.4 CONCLUSIONS

It is concluded from the discussion above that the Plant design and administrative controls ensure that radioactive liquid leakage or spillage will be retained within the facility or within 10 CFR 20 dose limits. Also, administrative controls and automatic interlocks, together with the fail safe design of the instrumentation and control devices, provide assurance against any discharge of liquid wastes to the environs in excess of 10 CFR 20 limits.

14.21 WASTE GAS INCIDENT

14.21.1 GAS DECAY TANK RUPTURE

14.21.1.1 Event Description

Six gas decay tanks contain compressed radioactive gases from the waste gas surge tank which collects gases vented from the Primary Coolant System, the volume control tank and the liquid waste system. These tanks contain the radioactive gases for decay and subsequent controlled release within the limits of 10 CFR 20. Three of the six decay gas tanks were added during the outage of 1971-1973.

The radioactive gases are compressed to a maximum of 115 psia for storage while allowing all the radioisotopes except Kr-85 and Xe-133 to decay to negligible activity levels.

The addition of the three tanks leaves unchanged the existing analysis for a gas decay tank rupture incident since the source term in the new tanks is the same as it is in the original tanks and administrative procedures require that the tanks be normally isolated from each other. This analysis shows that even under the worst expected meteorological conditions, the offsite doses in the event of a gas decay tank rupture are very low.

The waste gas decay tanks are designed for service at 135 psia and 550°F for expected operation at 115 psia and 90°F. Because components of the Waste Gas System are not subjected to high temperatures or high stresses, a failure or rupture of a gas decay tank is unlikely. However, a rupture of this tank is analyzed to define the radiation dose that would result from a malfunction of the Waste Gas System.

14.21.1.2 Thermal-Hydraulic Analysis

A thermal-hydraulic analysis is not applicable for this event.

14.21.1.3 Radiological Consequences

14.21.1.3.1 Analysis Method

The waste gas accident is defined as an uncontrolled release of the radioactive contents of the Waste Gas System to the atmosphere. Failure of a gas decay tank or the associated piping will result in the release of this radioactive gas.

14.21.1.3.2 Bounding Event Input

The activity released from a gas decay tank is assumed to be the maximum amount that would accumulate from operation with 1% failed fuel rods in the Primary Coolant System. This activity is obtained from the noble gases krypton and xenon by assuming no gas release from the gas decay tank between refuelings with an equilibrium core.

The principal active gas source is the bleeding and degassing of effluents from the Primary Coolant System. No carry-over of soluble and particulate fission products is assumed. The contents of the gas decay tank will be only noble gases.

The inventory of one gas decay tank containing all radioactive noble gases resulting from reactor operation with 1% failed fuel (gap inventory) is released to the auxiliary building. This activity is obtained from the noble gases krypton and xenon by assuming no gas release from the gas decay tank between refuelings with an equilibrium core radionuclide inventory. Partial decay is assumed corresponding to the time required, approximately 22 hours, to fill one waste gas decay tank. The release then travels unfiltered to the ventilation stack and then to the atmosphere over a period of 2 hours.

This value is conservative as the Plant operational control allows the radioactive gases to be released within limits on a continuous basis. Some of this activity would normally remain in the primary coolant and there would be more holdup time for decay, and the activity removed from the coolant would be distributed among several tanks.

14.21.1.3.3 Analysis of Results

The resultant offsite doses from the Gas Decay Tank Rupture are within the requirements of 10 CFR 100 and are presented in Table 14.1-6.

14.21.2 VOLUME CONTROL TANK RUPTURE

14.21.2.1 <u>Event Description</u>

The volume control tank in the Chemical and Volume Control System contains primary letdown coolant with its associated fission product concentrations. The volatile fission products collect in the volume control tank vapor space and are vented by operator action to the Waste Gas System. These volatile fission products primarily consist of noble gases. The halogens have a low volatility at the temperature and pH of the liquid in the volume control tank and therefore will remain in solution.

The volume control tank is designed for a differential pressure of 60 psi and a temperature of 250°F for normal operation at 10 psig and 120°F.

The volume control tank and the associated piping are not subjected to high temperatures or high stresses. Any failure in this system is very unlikely; however, a rupture of the volume control tanks is analyzed to define the radiation dose that might result in a failure in this system.

14.21.2.2 Thermal-Hydraulic Analysis

A thermal-hydraulic analysis is not applicable for this event.

14.21.2.3 Radiological Consequences

14.21.2.3.1 Analysis Method

The Volume Control Tank Rupture (VCTR) is very similar to the Small Line Break Outside of Containment (SMLBOC) except that the activity initially in the VCT is released and continued letdown flow after the rupture would be limited to 120 gpm (as opposed to 160 gpm for the SMLBOC) due to valving downstream of the containment penetration. Three scenarios are analyzed for offsite doses:

1. VCTR with equilibrium iodine in the primary coolant,

2. VCTR following a Previous Iodine Spike (PIS) in the primary coolant.

3. VCTR with a concurrent Event Generated Iodine Spike (GIS) in the primary coolant.

For all scenarios noble gas activities will be determined using the primary coolant gross specific activity limit of $100/\bar{E}\gamma$ (μCi/gm).

NOTE: The event generated iodine spike scenario was analyzed in the event the charging system becomes inoperable with a resultant PCS depressurization.

14.21.2.3.2 Bounding Event Input

The Volume Control Tank Rupture (VCTR) is defined as an uncontrolled release of the volatile contents of the VCT to the atmosphere. No decay is assumed prior to the rupture. The VCT volume of 4165 gallons is assumed to contain a radionuclide inventory corresponding to 1% defective fuel and is released to the Auxiliary Building over a 15 minute interval. A letdown flow rate, equal to the maximum value of 120 gpm, is assumed for 1 hour following the rupture. The primary coolant equilibrium iodine activity concentration is 1.0 μCi/gm dose equivalent (DE) I-131. The primary coolant Previous Iodine Spike activity concentration is 40 μCi/gm dose equivalent (DE) I-131. The fluid in the VCT has gross specific activity of $100/\bar{E}\gamma$ μCi/gm.

14.21.2.3.3 Analysis of Results

The resultant offsite doses from the Volume Control Tank Rupture are within the requirements of 10 CFR 100 and are presented in Table 14.1-6.

14.21.3 CONCLUSIONS

It is concluded from the foregoing that a rupture in the Waste Gas System or a rupture of the volume control tank would result in offsite doses well below the 10 CFR 100 limits, and would not present any undue risk to the health and safety of the public (References 1 and 2). The doses to control room personnel are discussed in Section 14.24.

REFERENCES

1. EA-TAM-96-02, Rev 3, "Palisades Control Room Habitability Following FSAR Chapter 14 Accidents with Radiological Consequences," (4953/1942).

2. EA-C-PAL-97-0808A, Revision 0, "Determination of Offsite Atmospheric Dispersion Factors (X/Qs) for Radiological Dose Consequence Analyses," (4297/1547).

14.22 MAXIMUM HYPOTHETICAL ACCIDENT

14.22.1 EVENT DESCRIPTION

Guidelines have been established which serve to identify some of the factors that must be considered in the evaluation of the suitability of sites for power reactors. The guidelines are set forth in Title 10, Chapter I, Code of Federal Regulations, Part 100 (10 CFR 100).

One of the requirements of 10 CFR 100 is to evaluate the total body and thyroid doses to the public following postulated incidents that result in the release of radioactive material from the Plant. The consequences of these hypothesized incidents are to be calculated as a function of the distance from the Plant and the time following the initiating event.

The Maximum Hypothetical Accident (MHA) is postulated to release substantially more fission products and result in more severe consequences than any incident considered credible. The evaluation is only meant to determine a reasonable upper bound of the consequences of an incident involving the release of radioactive material from the Plant site. The radiological consequences of the MHA are determined in a manner independent of any specific Plant transient sequence that might be postulated. To this end, the evaluation is performed in accordance with the guidelines and recommendations put forth by the NRC staff. For alternative source term analysis of the MHA, 10 CFR 50.67 defines the dose acceptance criteria. These guidelines define methods, assumptions and parameters which are nonmechanistic in nature and are meant to maximize the consequences of the MHA. (An important deviation from this overall philosophy is that the NRC guidance does not require the assumption of the loss of containment integrity.)

14.22.2 RADIOLOGICAL CONSEQUENCES

14.22.2.1 Analysis Method

The radiological consequences of the MHA have been calculated (Reference 9) using Regulatory Guide 1.183. Several plant modifications and tests were required prior to completion of the analysis in Reference 9. The most important of which include: replacement of high pressure safety injection system minimum flow recirculation valves, replacement of control room HVAC dampers, procedural changes to align the low pressure safety injection system cross-tie valves under certain scenarios, tests to quantify leakage past discharge and check valves into the SIRWT, and control room unfiltered inleakage tests. The TSP baskets have been since replaced with Sodium Tetraborate (NaTB), which functions similarly to TSP.

14.22.2.2 Bounding Event Input

Initially, the operation of the Containment Spray System in the injection mode will act to greatly reduce the concentration of particulate and elemental iodine species in the containment atmosphere. Upon reaching an appropriate decontamination factor, elemental iodine removal is assumed to end. However, removal of particulate iodine species continues as long as the containment sprays operate. After the receipt of a Recirculation Actuation Signal (RAS), the containment sump pH will be greater than 7.0 due to the passive addition of Sodium Tetraborate (NaTB) via baskets on the containment floor. This will prevent iodine re-evolution from the sump throughout the accident. After RAS, leakage begins outside of containment via ESF equipment leakage and inleakage to the SIRWT.

The incremental increase in the offsite radiological consequences due to the continuous containment vent path through a clean waste receiver tank has been analyzed and found to be insignificant, as the vent path is automatically isolated prior to the earliest predicted onset of cladding damage.

The sequence of events for the MHA radiological consequence analysis is listed in Table 14.22-1. Other parameters used in the analysis are listed in Table 14.22-2.

14.22.2.3 Analysis of Results

The total effective dose equivalent (TEDE) doses that have been calculated are listed in Table 14.1-6. These doses are within the guidelines of Regulatory Guide 1.183.

14.22.3 CONCLUSION

The potential offsite doses resulting from the maximum hypothetical accident are less than the 10 CFR 50.67 limits, and would not present any undue risk to the health and safety of the public. The doses to control room personnel are discussed in Section 14.24.

REFERENCES

1. Deleted

2. Deleted

3. Deleted

4. Deleted

5. Deleted

6. Deleted

7. Deleted

8. Deleted

9. NAI-1149-014, Revision 4, Palisades Design Basis AST MHA/LOCA Radiological Analysis, October 2007.

10. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research,” July 2000.

11. Deleted.

12. Deleted.

13. Deleted.

14.23 RADIOLOGICAL CONSEQUENCES OF FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT

14.23.1 EVENT DESCRIPTION

This section deals with the radiological consequences caused by the failure of small lines capable of carrying primary coolant outside the containment building. In this situation, radionuclides are released directly into the environment.

14.23.2 THERMAL-HYDRAULIC ANALYSIS

A thermal-hydraulic analysis is not applicable for this event.

14.23.3 RADIOLOGICAL CONSEQUENCES

14.23.3.1 Analysis Method

This event was analyzed by the Nuclear Regulatory Commission as part of the Palisades Systematic Evaluation Program (SEP). Consumers Power Company provided plant-specific information to the NRC (Reference 1). The NRC analyzed the event in accordance with Standard Review Plan Section 15.6.2 and transmitted the results to CPCo (References 2 and 3). The NRC offsite dose consequence analysis has been replaced by a more conservative analysis in Reference 5.

Consistent with Standard Review Plan Section 15.6.2, an event generated iodine spike scenario was analyzed in the event the charging system becomes inoperable with a resultant PCS depressurization. This event was analyzed with the methodology described in Regulatory Guide 1.183 (Reference 7), supplemented with specific guidance from SRP 15.6.2, as appropriate.

14.23.3.2 Bounding Event Input

Per NRC request (Reference 4), CPCo transmitted information about the various process lines carrying primary coolant outside the containment wall and the ventilation in the regions of the auxiliary building containing those lines. The process lines described included the following:

1. Letdown line, 2 inch and 3 inch, Containment Penetration 36
   120 gpm maximum flow rate.

2. Charging line, 2 inch, Containment Penetration 45 133 gpm maximum flow rate.

3. Reactor Coolant Sample line, 1/2 inch, Containment Penetration 40 50 gpm maximum flow rate.

4. Primary Coolant Pump Bleedoff, 3/4 inch, Containment Penetration 44 60 gpm maximum flow rate.

Per Reference 3, the NRC made the following assumptions during their evaluation:

1. All iodine contained in the leaked coolant was released to the Auxiliary Building atmosphere.

2. Primary Coolant System activities were at their upper limits of 1 μCi/gm I-131 dose equivalent and 100/E-bar μCi/gm.

3. A leak flow of 133 gpm was used.

4. The leak isolation time was assumed to be twenty minutes.

5. Because the leak flow matched the maximum charging flow and the leak isolation time was short, no reactor depressurization or trip was assumed to occur. An iodine spike was not assumed.

Following the NRC analysis a technical review of process lines carrying primary coolant outside of containment (Reference 6) revealed that a leak flow could be as great as 160 gpm and isolation time should not be credited until 1 hour following a small line break. The current analysis (Reference 5) assumes this higher flow rate and longer isolation time.

The assumptions for the updated analysis are listed in Tables 14.23-1 and 14.23-2.

14.23.3.3 <u>Analysis of Results</u>

The results of the analysis, listed in Table 14.1-6, demonstrates that the calculated doses are below the exposure guidelines of 10% of 10 CFR 50.67 limits consistent with the dose requirements specified in Standard Review Plan Section 15.6.2.

14.23.4 CONCLUSIONS

It is concluded that the radiological consequences of a failure of small lines carrying primary coolant outside containment fall below the applicable limits.

REFERENCES

1. Hoffman, David P, Nuclear Licensing Administrator, CP Co, to Crutchfield, Dennis M, Chief, Operating Reactors Branch 5, USNRC, "Response to SEP Top XV-16 - Radiological Consequences of Small Lines Carrying Primary Coolant Outside Containment," May 6, 1980, (2615/0130).

2. Crutchfield, Dennis M, Chief, Operating Reactors Branch 5, USNRC, to Hoffman, David P, "SEP Topics, Design Basis Events - Accidents and Transients," July 17, 1981, (2485/0729).

3. Crutchfield, Dennis M, Chief, Operating Reactors Branch No. 5, USNRC to Hoffman, David P, "Palisades - SEP Topics - Design Basis Events," November 3, 1981, (2525/1636).

4. Ziemann, DL, USNRC to Consumers Power Company, letter dated February 19, 1980, (4466/0690).

5. NAI-1149-020, Revision 0, Palisades Design Basis Small Line Break Outside Containment AST Radiological Analysis, June 2005.

6. AA 81-03, CPCo Internal Correspondence from AAzima to RWSinderman, AIR #A-NA-80-66F Palisades SEP Topic XV-16 Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment, 24APR81, (2570/0793).

7. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, July 2000.

14.24 CONTROL ROOM RADIOLOGICAL HABITABILITY

14.24.1 EVENT DESCRIPTION

As stated in Subsections 6.10.1 and 6.10.3, the design and operation of the Control Room Heating, Ventilation and Air Conditioning (CRHVAC) System is such that the dose limits of General Design Criterion 19 of Appendix A to 10 CFR 50 are not exceeded following any design basis accidents (DBAs).

The CRHVAC components and modes of operation are described in Subsections 9.8.1.4, 9.8.2.1, 9.8.2.2, 9.8.2.3.2 and 9.8.2.4.12.

14.24.2 THERMAL-HYDRAULIC ANALYSIS

The thermal-hydraulic analysis of each DBA, if applicable, is described in the appropriate sections of Chapter 14.

14.24.3 RADIOLOGICAL CONSEQUENCES

14.24.3.1 Analysis Method

To analyze the control room habitability, radionuclide releases are taken from, or are conservatively derived from, the analyses that form the basis for the current DBA descriptions. The DBA descriptions are listed in the appropriate sections of Chapter 14. The events are then evaluated to determine the time at which the CRHVAC System is placed in the emergency mode. For evaluating the time at which the CRHVAC System is placed in the emergency mode, the following factors must be determined:

1. The diesel generator sequencing time and control room pressurization time if offsite power is lost for the worst case event.

2. The time of signal generation after the start of the event if a containment high pressure (CHP) or containment high radiation (CHR) signal is generated. (Following a circuit delay time, the CRHVAC System is automatically switched to emergency mode for either CHP or CHR signal generation.)

3. The operator response time to manually switch the CR-HVAC into the Emergency Mode if a CHP or CHR signal is not generated. Twenty minutes is assumed in the analyses for an operator to diagnose the emergency and place the CR-HVAC in emergency mode. CAM radiation alarms are diagnostic tools to help an operator determine the need for emergency mode, but automatic switchover upon CAM alarm is not installed or credited.

Events utilizing the alternative source term (AST) were analyzed with the methodology described in Regulatory Guide 1.183 (Reference 2), as described in the corresponding sections of Chapter 14. Events retaining the technical information document (TID) source term (Reference 3) were analyzed with the methodologies described in the corresponding sections of Chapter 14 (see 14.20 – Liquid Waste Incident and 14.21 – Waste Gas Incident).

For the TID source term analysis, the CONDOSE computer code is then used to calculate the doses to control room personnel given the time dependent radionuclide release rates for the event and the control room occupancy and CRHVAC System parameters.

The CONDOSE code is a FORTRAN program developed to determine the habitability of the control room following design basis accidents that release radiation to the atmosphere. CONDOSE determines control room habitability only with respect to occupant radiation dose. The program calculates the time dependent concentration of radionuclides in the control room accounting for the following radiation source and removal mechanisms: outside air makeup, unfiltered air in-leakage, radioactive decay, iodine removal by filters (for outside air makeup and recirculation), and control room air exfiltration (out-leakage). From the time dependent radionuclide concentrations, time integrated occupant radiation doses are calculated.

CONDOSE has incorporated the full implementation of ICRP30 dose calculation methodology, which equivalences internal exposures to whole body exposures using organ weighting factors, the quantity of intake of each radionuclide, and the corresponding ALI (annual limit on intake) for the radionuclide. This committed effective dose equivalent is then added to the external whole body dose for a total effective dose equivalent. The ICRP30 methodology is the same as that used in the revision of 10CFR20 published in the Federal Register on May 21, 1991. However, for the calculation of design basis accident doses Standard Review Plan 6.4 states acceptance criteria for the dose to the thyroid and the whole body due to submersion.

While the total ICRP30 dose calculation methodology (equivalencing internal doses to external doses) is accepted by the NRC for occupational dose calculations (10CFR20), the NRC still requires that the SRP dose limits, in terms of dose to the thyroid and whole body from submersion, be applied for design basis accidents. CONDOSE also calculates these latter doses. Hence, the Thyroid-Inhalation dose from iodines and Whole Body Submersion from Noble Gases for the 5 iodine and 13 noble gas isotopes of main concern after all design basis accidents are presented in 14.24-2 for comparison to SRP Section 6.4 Dose limit requirement. Note that if the SRP 6.4 dose limits are met, then the GDC-19 dose limits are also, since the SRP method is more conservative.

For the AST analysis, the RADTRAD computer code was used to calculated control room (and offsite) doses, as recommended in Regulatory Guide 1.183.

14.24.3.2 Bounding Event Input

The bounding events with respect to control room habitability are the Main Steam Line Break and Maximum Hypothetical Accident. These events are described in Sections 14.16 and 14.22. The sequence of events and parameters used in evaluating these radionuclide releases are listed in Tables of Sections 14.14 and 14.22, respectively. The parameters and assumptions concerning the CR-HVAC system operation and dose calculation parameters used for control room habitability analyses are listed in Tables 14.24-1, 14.24-2, and 14.24-3.

14.24.3.3 Analysis of Results

The results of the control room habitability analyses for all DBAs are summarized in Table 14.1-6 (Reference 1 and the specific event references). The resultant doses to control room personnel were less than the limits of 10CFR50 Appendix A, General Design Criterion 19, and 10 CFR 50.67, as applicable.

14.24.4 CONCLUSION

The control room dose limits specified in 10 CFR 50.67 and 10CFR50 Appendix A, General Design Criterion 19, have not been exceeded for any design basis accident that releases radiation to the environment. The doses are summarized in Table 14.1-6.

REFERENCES

1. EA-TAM-96-02, Rev 3, "Palisades Control Room Habitability Following FSAR Chapter 14 Accidents with Radiological Consequences," (4953/1942).

2. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, July 2000.

3. TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, U.S. Atomic Energy Commission, U.S. Department of Commerce, March 1962.