PALISADES NUCLEAR PLANT
ENGINEERING ANALYSIS COVER SHEET

Title: Palisades Cycle 14 Disposition of Events Review

Initiation & Review

Calculation Status	Preliminary	Pending	Final	Superseded

Rev #	Description	Initiated		Init	Review Method			Technically Reviewed		Revr	SDR
0	Original Issue	By	Date	Appd By	Alt Calc	Detail Review	Qual Test	By	Date	Appd By	Appd
								J. Pleva			
1		[illegible]	6/15/98	[illegible]		X		[illegible]	6/17/98	[illegible]	
		[illegible] 6/16/98									

PURPOSE:

The purpose of this EA is to document the results of a Disposition of Licensing Basis Events Review as a result of changes made to the fuel and plant for Cycle 14 operation. All Palisades Licensing Basis Events are reviewed and dispositioned in this analysis.

Table 2.1 of reference 2.5 lists the PCS flow assumptions used in the safety analyses. Note 2 to this table lists lower minimum flows to be used in future analyses. After revision 0 of this EA was completed, a decision was made to reanalyze all events (see reference 2.33) which would be impacted by the use of lower minimum flows. Reference 2.33 of this EA provides dispositions or analyses for these lower minimum flows. Note that the control rod ejection (reference 2.31), the main steam line break (reference 2.30), the large break LOCA (reference 2.15), and the small break LOCA (reference 2.18) used these lower minimum flows.

SUMMARY OF RESULTS:

This EA reviewed all Palisades Licensing Basis Events (FSAR Chapter 14 [Ref. 2.2]) and determined whether the current analysis of record was bounding or required analysis as a result of changes being made to the fuel and plant for Cycle 14 operation. The effects on the Safety Analyses of the changes to the fuel and plant for Cycle 14 (see Background Section) meet all regulations and SRP requirements and are considered acceptable. Offsite doses were calculated for those events with radiological consequences that may not be bounded by past analyses and are also acceptable.

The effects on the Safety Analyses of the changes to the lower minimum flows for Cycle 14 meet all regulations and SRP requirements and are considered acceptable.

Total Sheets 15

1.0 OBJECTIVE:

The purpose of this EA is to document the results of a Disposition of Licensing Basis Events Review as a result of changes made to the fuel and plant for Cycle 14 operation. All Palisades Licensing Basis Events are to be reviewed and dispositioned in this analysis.

2.0 REFERENCES:

2.1 EMF-98-013, Revision 0, "Palisades Cycle 14: Disposition and Analysis of Standard Review Plan Chapter 15 Events," Siemens Power Corporation, May 1998.
2.2 Palisades Final Safety Analysis Report (FSAR).
2.3 Palisades Technical Specifications.
2.4 NUREG 0800, USNRC Standard Review Plan, Rev. 1 - July 1981 (Various Sections).
2.5 EMF-97-051, "Palisades Cycle 14 Principal Plant Parameters," January 1998.
2.6 ANF-90-078, "Palisades Cycle 9: Analysis of Standard Review Plan Chapter 15 Events," September 1990; Cart/Frame C667/1283.
2.7 ANF-90-181, "Review and Analysis of SRP Chapter 15 Events for Palisades with a 15% Variable High Power Trip Reset," November 1990.
2.8 EMF-95-022, "Palisades Cycle 12: Disposition & Analysis of Standard Review Plan Chapter 15 Events," Siemens Power Corporation, April 1995.
2.9 EA-TAM-96-04 Rev. 0, "Offsite Radiological Dose Consequences of a Cask Drop in the Spent Fuel Pool," October 1996; Cart/Frame G781/2304.
2.10 EMF-93-086(P), "Palisades Loss of Load Analysis," Siemens Power Corporation - Nuclear Division, April 1993.
2.11 EMF-87-150(NP), "Palisades Modified Reactor Protection System Report: Analysis of Chapter 15 Events," Volume 2, Advanced Nuclear Fuels Corporation, June 1988.
2.12 EA-TAM-95-03 Rev. 1, "Cycle 12 Off-Site Radiological Dose Consequences for the Main Steam Line Break," August 1995; Cart/Frame: G352/2379.
2.13 deleted
2.14 EA-TAM-95-01 Rev. 0, "Cycle 12 Off-Site Radiological Dose Consequences for the Control Rod Ejection Accident," July 1995; Cart/Frame: G331/2475.
2.15 EMF-98-026, Revision 1, "Palisades Large Break LOCA/ECCS Analysis, April 1998.
2.16 EA-TAM-95-05 Rev. 0, "Radiological Consequences for the Palisades Maximum Hypothetical Accident & Loss of Coolant Accident," November 1995; Cart/Frame G534/0032.
2.17 EA-TAM-96-02, Revision 1, "Palisades Control Room Habitability Following FSAR Chapter 14 Accidents with Radiological Consequences", May, 1998.
2.18 "SBLOCA Evaluation of Palisades Cycle 14 Fuel Design Change," A-PAL-FE--0001, Rev 00, February 1998.
2.19 EA-TAM-96-07, Owners Review - Palisades Updated Steam Generator Tube Rupture Analysis, ABB CENO 001-ST96-C-007 Rev. 00," December 1996; Cart/Frame H032/0975.
2.20 001-ST96-C-007 Rev. 00, " Palisades Updated Steam Generator Tube Rupture Analysis," November 1996, ABB CENO.

2.21 EA-A-NL-91-169-01 Rev. 2, "Offsite Dose Calculations of Fuel Handling Accident," March 1993; Cart/Frame F385/1446
2.22 deleted
2.23 deleted
2.24 deleted
2.25 FC-970, Palisades Nuclear Plant Facility Change, "Reload R," May 1998.
2.26 EA-LOCA-98-01, "Containment Response to an LOCA Using CONTEMPT-LT/28.", May, 1998.
2.27 EA-MSLB-98-01, "Containment Response to an MSLB Using CONTEMPT-LT/28.", May, 1998.
2.28 DA-001-NT90-014, Rev. 00, "Palisades Steam Generator Tube Rupture for the Replacement Steam Generator Project," MGolbabai, 11MAR91.
2.29 EMF-96-140, "Palisades Cycle 13: Disposition & Analysis of Standard Review Plan Chapter 15 Events," Siemens Power Corporation, November 1996
2.30 EMF-98-012, Revision 0, "Main Steam Line Break Analysis for Palisades", May 1998.
2.31 EMF-98-021, Revision 0, "Palisades Control Rod Ejection Analysis", April 1998.
2.32 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 GA Baustian.
2.33 EMF-98-042, Revision 0, "Palisades Cycle 14: Disposition and Analysis of Standard Review Plan Chapter 15 Events for Reduced Primary Coolnat Flow," Siemens Power Corporation, June 1998.
2.34 Memo RIW:98:190 Responses to Consumers' Comments on EMF-98-042 dated June 16, 1998

3.0 BACKGROUND:

The changes for Cycle 14 that impact the disposition of the Chapter 15 events and the verification of the setpoints, MDNBR and fuel centerline melt (FCM) analyses consist of the following (See Reference 2.1):

3.1 Plant Operating Conditions Changes Implemented

Maximum nominal T_{inlet} of 544°F was added.
The nominal steam generator flow decreased from 11.57 Mlbm/hr to 10.982 Mlbm/hr.
The nominal HZP steam generator pressure of 900 psia was added.
The HFP and HZP steam generator masses were changed:
The HFP liquid mass increased from 120,385 lbm to 133,595 lbm and the vapor mass decreased from 8,779 lbm to 8,545 lbm.
The HZP liquid mass of 203,783 lbm and the HZP vapor mass of 6,976 lbm were added.
The maximum steam generator tube plugging was assumed to be 15% for analysis purposes.
The shutdown margin with less than four reactor pumps operating was reduced from 3.75% to 3.5%.

The minimum flows listed in note 2 to table 2.1 of reference 2.5 were bounded for all events (see references 2.33 and 2.34). These minimum flows are:

PCS flow Rate Technical Specification Limit, gpm	352,000
PCS Flow Rate Analytical Limit, gpm	341,400
Core Flow Rate Analytical Limit, gpm	331,200

Per a telecon with Siemens (Bill Nutt) on June 16, 1998, for all MDNBR events renanalyzed at lower flowrates, the core average mass flux in the existing XCOBRA input files was reduced by 5% to bound the above minimum flowrates.

3.2 Plant Control Systems Changes Implemented

The delay time for the opening or closing of the normal feedwater regulating valves increased from 20.5 sec to 22 sec.
The letdown capacity increased to 120 gpm.
The minimum charging pump capacity of 33 gpm.
The maximum pressurizer spray flow rate increased from 280 gpm to 500 gpm.

3.3 Plant Safety Systems Changes Implemented

The pressurizer pressure for actuation of the SI decreased from an allowable value of 1,593 psig to an allowable value of 1,590 psig.
The LPSI pump performance has generally degraded over the range of operating pressures.
A containment high-pressure trip setpoint of ≤ 3.7 psig was added.
The low steam generator level reactor trip has been changed to an allowable value of 18.77% of the narrow range reading.
The low steam generator level trip for AFW actuation has been changed to an allowable value of 25.9% of the narrow range reading.
SI actuation on high containment pressure was given an allowable range of 3.7 psig to 4.3 psig.
The MSSV capacities were made consistent with design capacities. The MSSV opening pressure and flow data are now staged by bank.
The steam assisted MSIV closure time of 2 sec.
The single flow rate value (2,125 gpm) for the containment sprays was replaced with a table of values that are based on the failure assumptions.
The minimum, maximum and nominal flow rates and the startup delay times for the AFWs were added.
The SIT water volume was increased from 2,000 ft^3 to 2,011 ft^3.

3.4 Thermal-Hydraulic Characteristics Changes

The lower tie plate design changed for Reload R. These tie plates have a FUELGUARD™ debris-resistant insert.
New pressure drop measurements were performed with the control blades fully inserted.

3.5 Fuel Rod Mechanical Design Changes

A new fuel design will be introduced for Reload R, incorporating the following changes:

0.009-inch increase in the pellet OD.
0.152-inch increase in the pellet length.
0.4% reduction in the pellet dish volume.
0.009-inch increase in the cladding ID.
0.8-inch increase in the fuel rod active length.
0.111-inch increase in the effective plenum length.
Initial rod internal pressure of the gadolinia bearing rods has increased.

3.6 Neutronics Design Changes

The enrichment pattern for the Reload R assemblies was modified and it produces different local power distributions.
The UO_2 enrichment in the gadolinia bearing fuel rods increased.
The maximum gadolinia concentration in the gadolinia-bearing fuel rods increased in Cycle 14 from 6 wt% to 8 wt%.
The U^{238} capture-fission ratios for Cycle 14 increased slightly, resulting in an increase in the decay heat.
The 50% and 100% power differential rod worths at EOC conditions increased.
The moderator density and Doppler reactivity tables changed for Cycle 14.
The boron endpoint concentrations for Cycle 14 changed for all conditions.

3.7 Uncertainties and Response Times

~~Cycle 14 does not contain a full core of fresh detectors.~~
Delay times for the HPSI and LPSI pumps were increased from 27 sec to 40 sec.
RTD delay times were increased from 12 to 15 sec.
The trip delays for the low reactor coolant flow, high pressurizer pressure , low pressurizer pressure, low steam generator pressure and TM/LP and VHP trips were increased by 0.2 sec.

The Technical Specifications total and assembly radial peaking factor limits for Reload R fuel, F_r^T = 2.04 and F_r^A = 1.76, are unchanged from Cycle 13 Reload Q. Radial peaking factors for fuel prior to Reload O are unchanged from those contained in (Core Operating Limits Report) COLR Rev. 3. A copy of the COLR can be found attached to Reference 2.3.

The aforementioned changes necessitate a review of all Palisades Licensing Basis Events, which are contained in Ch. 14 of the Palisades Final Safety Analysis Report (Reference 2.2). The majority of these events are dispositioned in Reference 2.1. Events that are not dispositioned by Reference 2.1 are dispositioned in this EA. This EA will review each Ch. 14 Event and its associated Standard Review Plan (SRP) Ch.

15 Section [Ref. 2.4], and will either provide a disposition or reference the disposition in Reference 2.1. It will be noted whether the event is bounded by an analysis of record or was analyzed for Cycle 14 operation.

4.0 ASSUMPTIONS:

1.) Each event is dispositioned relative to the known changes made to the Palisades Plant and its fuel for Cycle 14 operation.

2.) The Palisades Cycle 14: Disposition and Analysis of Standard Review Plan Chapter 15 Events, [Ref 2.1] has been technically reviewed.

5.0 ANALYSIS INPUT:

The only inputs used in this analysis are the changes to be made to the fuel and plant for Cycle 14 operation, as described in References 2.1, 2.5 and Ref. 2.25.

6.0 ANALYSIS:

Each FSAR Chapter 14 event is reviewed and dispositioned into one of the following categories: Unchanged from the previous analysis, Bounded by another analysis, Not applicable (outside the licensing basis, deleted from FSAR or SRP, or Not Credible), or analyzed for Cycle 14. Some events are a combination of these (i.e. the thermal hydraulic system response remains unchanged from previous analyses, but the MDNBR or LHGR may be calculated for the current cycle).

The following events are dispositioned in reference 2.33.

14.2 Uncontrolled Rod Withdrawal
14.4 Control Rod Drop
14.7 Decreased Reactor Coolant Flow
14.10 Increase in Steam Flow (Excess Load)

Each event is reviewed and dispositioned below in the order in which they appear in Ch. 14 of the Palisades FSAR (Reference 2.2). Standard Review Plan events that are dispositioned in Reference 2.1, but bounded by another Ch. 14 analysis, are also referenced.

14.2 Uncontrolled Rod Withdrawal

14.2.1 Uncontrolled Control Rod Bank Withdrawal from a Sub-Critical or Low Power Start-Up Condition.

1. SRP 15.4.1
2. Dispositioned in Reference 2.1
3. System response was analyzed for Cycle 13, [Ref. 2.29]. MDNBR and maximum fuel temperature were analyzed for Cycle 14 [Ref. 2.33].

14.2.2 Uncontrolled Control Rod Bank Withdrawal at Power

1. SRP 15.4.2
2. Dispositioned in Reference 2.1
3. System response unchanged from analyses in References 2.6 & 2.7. MDNBR and maximum LHGR were analyzed for Cycle 14 [Reference 2.33].

14.2.3 Single Control Rod Withdrawal

1. SRP 15.4.3
2. Dispositioned in Reference 2.1
3. System response unchanged from analyses in References 2.6 & 2.7. MDNBR and maximum LHGR were analyzed for Cycle 14 [Reference 2.33].

14.3 Boron Dilution

Note: All Boron Dilution scenarios (except at Hot Standby and Reactor Critical) were addressed for Cycle 12 [Ref. 2.8] due to changes in the initial and critical boron concentration for each operating mode. Dilution during Hot Standby and Reactor Critical operating modes is taken into account in the analysis for SRP Sec 15.4.2 (Uncontrolled Control Rod Bank Withdrawal at Power). The analysis for Cycle 12. [Ref. 2.8] demonstrated significant margins in the times to criticality. The analysis for Cycle 12 bounds Cycle 14.

14.3.1 Dilution During Refueling

1. SRP 15.4.6
2. Dispositioned in Reference 2.1
3. Shutdown Margin Adequacy is addressed in Reference 2.1.

14.3.2 Dilution During Start-Up

1. SRP 15.4.6
2. Dispositioned in Reference 2.1
3. Shutdown Margin Adequacy is addressed in Reference 2.1.

14.3.3 Hot Standby or Reactor Critical

1. SRP 15.4.6
2. Dispositioned in Reference 2.1
3. Event is bounded by SRP 15.4.2 (Uncontrolled Control Rod Bank Withdrawal at Power) in Reference 2.1.

14.3.4 Dilution During Power Operation

1. SRP 15.4.6
2. Dispositioned in Reference 2.1
3. Event is bounded by the analysis for SRP 15.4.2 (Uncontrolled Control Rod Bank Withdrawal at Power) found in Reference 2.1.

14.3.5 Failure to Add Boron to Compensate for Reactivity Changes After Shutdown

1. SRP 15.4.6
2. Event is dispositioned in Reference 2.8.

14.4 Control Rod Drop

14.4.1 Dropped Rod Event

1. SRP 15.4.3
2. Dispositioned in Reference 2.33.
3. System response unchanged from analysis in Reference 2.6. MDNBR analyzed in Reference 2.33.

14.4.2 Rod Bank Drop Event

1. SRP 15.4.3
2. Dispositioned in Reference 2.33.
3. System response unchanged from analysis in Reference 2.6. MDNBR analyzed in Reference 2.33.

14.5 Core Barrel Failure

1. SRP 15.4.3
2. Dispositioned in Reference 2.1
3. Event is bounded by the Control Rod Ejection Event, (SRP Sec. 15.4.8). See Section 14.16 below.

14.6 Control Rod Misoperation

14.6.1 Malposition of the Part-Length Control Rod Group

1. SRP 15.4.3
2. Dispositioned as not applicable to Palisades in Reference 2.1

Note: Current T.S. (Reference 2.3) do not allow the use of part-length rods. These rods are maintained in a fully withdrawn position and misalignment of the rods is not considered a credible event.

14.6.2 Statically Misaligned Control Rod/Bank

1. SRP 15.4.3
2. Dispositioned in Reference ~~2.1~~ 2.33.
3. System response unchanged from analysis in Reference 2.6. MDNBR analyzed in Reference 2.33.

14.7 Decreased Reactor Coolant Flow

14.7.1 Loss of Forced Reactor Coolant Flow

1. SRP 15.3.1
2. Dispositioned in Reference 2.33.
3. System response unchanged from analysis in Reference 2.6. MDNBR analyzed in Reference 2.33

14.7.2 Reactor Coolant Pump Rotor Seizure

1. SRP 15.3.3
2. Dispositioned in Reference 2.33
3. System response unchanged from analysis in Reference 2.6. MDNBR analyzed in Reference 2.33

14.8 Start-up of an Inactive Loop

1. SRP 15.4.4
2. Dispositioned in Reference 2.1
3. Bounded by Rated Power MDNBR with four primary coolant pumps operating as stated in Reference 2.1.

14.9 Excessive Feedwater Incident (Deleted from FSAR)

1. SRP 15.1.2
2. Dispositioned in Reference 2.1
3. Event is bounded by the Increase in Steam Flow Event (SRP Sec. 15.1.3). See Section 14.10 below.

14.10 Increase in Steam Flow (Excess Load)

1. SRP 15.1.3
2. Dispositioned in Reference 2.33
3. System response unchanged from analysis in Reference 2.6. MDNBR analyzed in Reference 2.33

14.11 Postulated Cask Drop Accident

1. SRP 15.7.5
2. Dispositioned in Reference 2.1

3. Radiological Consequences of this event were recalculated during Cycle 12 due to the discovery of unanalyzed leak paths from the fuel handling building. See EA-TAM-96-04 Rev. 0 [Ref. 2.9].

14.12 Loss of External Load

1. SRP 15.2.1
2. Dispositioned in Reference 2.1
3. System response is bounded by EMF-93-086(P), "Palisades Loss of Load Analysis," [Ref. 2.10]. MDNBR is bounded by analysis in Reference 2.11.

14.13 Loss of Normal Feedwater

1. SRP 15.2.7
2. Dispositioned in Reference 2.1
3. Event is bounded by analysis in ANF-87-150(NP) Vol. 2, "Palisades Modified Reactor Protection System Report: Analysis of Chapter 15 Events," Reference 2.11.

14.14 Steam Line Rupture Incident

1. SRP 15.1.5
2. Dispositioned in Reference 2.1
3. Analyzed in Reference 2.30. Radiological Consequences of a Main Steam Line Break Outside Containment are bounded by EA-TAM-95-03 Rev. 1, [Ref. 2.12] which assumed 2% fuel failures. The Cycle 14 MSLB analysis (reference 2.30) determined no fuel failures.

14.15 Steam Generator Tube Rupture With a Loss of Offsite Power

1. SRP 15.6.3
2. Dispositioned in Reference 2.1
3. Event is bounded by analysis in FSAR. See [Ref. 2.28].

In Reference 2.32, ABB provides a disposition for the lower assumed minimum flowrates. The current FSAR analysis [Ref. 2.28] remains the bounding SGTR analysis of record for Palisades. Radiological Consequences were calculated as part of the analysis in Reference 2.17.

Note: A Updated SGTR analysis [Ref. 2.19] was performed by ABB/CE for Cycle 13 to examine the feasibility of using the HPSI system to feed the auxiliary pressurizer sprays as opposed to the charging pumps. The results of this analysis were found to be acceptable and are detailed in Reference 2.20.

14.16 Control Rod Ejection

1. SRP 15.4.8

2. Dispositioned in Reference 2.1
3. Analyzed in Reference 2.31. Radiological consequences of this event are bounded by EA-TAM-95-01, [Ref. 2.14], which assumed 14.7% fuel failures. The Cycle 14 CRE analysis (reference 2.31) determined no fuel failures.

14.17 Loss of Coolant Accident

The results of the Cycle 14 reanalysis for both the SBLOCA and LBLOCA ECCS performance calculations show that the calculated values for peak clad temperature, cladding oxidation, hydrogen generation, and core geometry are within the allowed limits of 10 CFR 50.46 (see Reference 2.25).

14.17.1 Large Break LOCA

1. SRP 15.6.5
2. Dispositioned in Reference 2.1
3. A LBLOCA analysis was performed by Siemens for Cycle 14 (Reference 2.15). Radiological Consequences are bounded by the current MHA analysis EA-TAM-95-05 Rev. 0, Reference 2.16.

14.17.2 Small Break LOCA

1. SRP 15.6.5
2. Dispositioned in Reference 2.17
3. A SBLOCA analysis was performed by ABB/CE for Cycle 14. The results of this analysis were found to be acceptable and are detailed in Reference 2.18. Radiological Consequences are bounded by the current MHA analysis EA-TAM-95-05 Rev. 0, Reference 2.16.

14.18 Containment Pressure and Temperature Analysis

14.18.1 LOCA Analysis

1. SRP 6.2.1
2. Dispositioned in Reference 2.26
3. In Reference 2.32, ABB provides a disposition for the lower assumed minimum flowrates on the mass and energy release which is used as an input to reference 2.26. Event will be reanalyzed for Cycle 14 (see reference 2.26).

14.8.2 MSLB Inside Containment

1. SRP 6.2.1
2. Dispositioned in Reference 2.27.
3. In Reference 2.32, ABB provides a disposition for the lower assumed minimum flowrates on the mass and energy release which is used as an

input to reference 2.27. Event will be reanalyzed for Cycle 14 (see reference 2.27).

14.19 Fuel Handling Incident

1. SRP 15.7.4
2. Dispositioned in Reference 2.1
3. Radiological Consequences of this event are bounded by EA-A-NL-91-169-01 Rev. 2, [Ref. 2.21].

14.20 Liquid Waste Incident

1. SRP 15.7.3
2. Dispositioned in Reference 2.1
3. Event is bounded by the current FSAR analysis [Ref 2.2] as discussed in Reference 2.1.

14.21 Waste Gas Incident

14.21.1 Gas Decay Tank Rupture

1. SRP 15.7.2 (Deleted)
2. Dispositioned in Attachment 1 to this analysis
3. Event is bounded by the current FSAR analysis [Ref 2.2] as discussed in Attachment 1 to this analysis.

14.21.2 Volume Control Tank Rupture

1. SRP 15.7.2 (Deleted)
2. Dispositioned in [Ref 2.17]
3. Event is bounded by the current FSAR analysis [Ref 2.2]. VCT rupture dose consequence analysis was expanded to include an event Generated Iodine Spike (GIS). The results of this analysis showed that the Previous Iodine Spike (PIS) case remains bounding. Results are documented in EA-TAM-96-02 Rev 1 [Ref. 2.17] for both control room and off site doses for the VCTR.

14.22 Maximum Hypothetical Accident

1. SRP 15.6.5 Appendix A
2. Dispositioned in Attachment 1 to this analysis.
3. Event is bounded by EA-TAM-95-05 Rev. 0 Reference 2.16, as discussed in Attachment 1 to this analysis.

14.23 Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment (SMLBOC)

1. SRP 15.6.2
2. Dispositioned in Reference 2.1
3. Event is bounded by the current FSAR analysis [Ref 2.2] as discussed in Reference 2.1. SMBLOC dose consequences were expanded to include an event Generated Iodine Spike (GIS). The results show that the SMLBOC for off site doses meet the requirements of 10 CFR 100 and are within the requirements of Standard Review Plan section 15.6.2. The control room doses are within the requirements of General Design Criteria 19 and Standard Review Plan 6.4. Results are documented in EA-TAM-96-02 Rev 1 [Ref. 2.17].

14.24 Control Room Radiological Habitability

1. SRP 6.4
2. Dispositioned in Attachment 1 to this analysis.
3. Event is bounded by EA-TAM-96-02 Rev. 1, Reference 2.17, as discussed in Attachment 1 to this analysis.

7.0 CONCLUSIONS:

This EA reviewed all Palisades Licensing Basis Events (FSAR Chapter 14 [Ref. 2.2]) and determined whether the current analysis of record was bounding or required analysis as a result of changes being made to the fuel and plant for Cycle 14 operation. The effects on the Safety Analyses of the changes to the fuel and plant for Cycle 14 (see Background Section) meet all regulations and SRP requirements and are considered acceptable. Offsite doses for those events with radiological consequences are also acceptable.

The effects on the Safety Analyses of the changes to the lower minimum flows for Cycle 14 meet all regulations and SRP requirements and are considered acceptable.

8.0 LIST OF ATTACHMENTS:

1.) Event Disposition and Justification for FSAR Sections: 14.21, 14.22, & 14.24
2.) Admin Procedure 9.11 Rev. 9, Attachment 4, "Engineering Analysis Checklist"
3.) Admin Procedure 9.11 Rev. 9, Attachment 5, "Technical Review Checklist"
4.) Admin Procedure 9.11 Rev. 9, Attachment 6, "Engineering Analysis Review Sheet"

Attachment 1

**Event Disposition and Justification for
FSAR Sections
14.21, 14.22, & 14.24**

14.21 Waste Gas Incident

Event Disposition and Justification

The primary inputs to the analysis are the amount of fission product gases contained in the tanks. A conservative source term was assumed for this analysis based on a conservative percentage of the core assumed to contain leaking rods. This event has been deleted from the SRP and no longer needs to be dispositioned. However, the Cycle 14 fuel changes will not impact the results of the current FSAR analysis.

14.22 Maximum Hypothetical Accident

Event Disposition and Justification

This hypothesized event is designed to bound the radiological consequences of all postulated and anticipated accidents. The entire core is assumed to fail due to fuel melting. Due to this assumption, fuel cycle changes do not affect the source terms used in the reference analysis EA-TAM-95-05 Rev. 0 [Ref. 2.16]. Modifications to the plant for Cycle 14 operation will not alter the results of this analysis. Therefore EA-TAM-95-05 Rev. 0 [Ref. 2.16] remains bounding for Cycle 14 operation.

14.24 Control Room Radiological Habitability

Event Disposition and Justification

General Design Criterion 19 of 10CFR50, Appendix A requires that the Control room HVAC be designed to provide protection of the Control Room personnel from radiation doses. The whole body radiation dose limit of 5 rem or its equivalent to any other part of the body (30 rem thyroid) for control room personnel is also specified in GDC-19. The results of this analysis do not change for modifications being performed for Cycle 14 operation. Therefore EA-TAM-96-02 Rev. 1 [Ref. 2.17] remains bounding for Cycle 14 operation.

PALISADES NUCLEAR PLANT
10CFR50.59 SAFETY REVIEW

PS&L Log No 98-13-19

Item Identification: No **EA-GEJ-98-01** Rev **1** Title **Palisades Cycle 14 Disposition of Events Review**		SE REV
Describe Issue/Change: **Reduced core and vessel flow assumptions in the safety analyses.**		
Reason for Issue/Change: **Analysis needed for proposed tech spec change.**		

	Yes	No
1. Does the item involve a change to procedures as described in the FSAR? FSAR Sections affected **None** FSAR Sections reviewed **See #2**		X
2. Does the item involve a change to the facility as described or implied in the FSAR? FSAR Sections affected **Tables 14.1-2, 14.15-1, 14.18.2-2 14.17.2-1, 14.17.1-1 References 14.1,14.15** FSAR Sections reviewed **Chapters 1, 4, and 14** DBD Sections reviewed **2.01 - 2.06**	X	
3. Does the item involve a test or experiment not described in the FSAR? FSAR Sections affected **None** FSAR Sections reviewed **See #2** DBD Sections reviewed **2.01 - 2.06**		X
4. Should the Technical Specifications or any of their Bases be changed in conjunction with this item? TS Sections affected none TS Sections reviewed **1, 2, 3, COLR, and Standing Orders**		X

Justify "NO" answers below if logic is not obvious:

Searched ZyIndex for: flow

If any Safety Review question listed above is answered "YES," perform a written USQ Evaluation according to Section 5.3.

If all Safety Review questions listed above are answered "NO," written USQ Evaluation is not required. However, this attachment shall accompany other review materials for the item to document that a Safety Evaluation was not required.

SE [signature] 6/17/98 Prepared By Date	[signature] 6/17/98 Reviewed By Date

PALISADES NUCLEAR PLANT
SAFETY ANALYSIS

Item Identification: No **EA-GEJ-98-01** Rev **1** Title **Palisades Cycle 14 Disposition of Events Review**

SECTION I	Yes	No
1. Will the probability of an accident previously evaluated in the FSAR be increased?		X
2. Will the consequences of an accident previously evaluated in the FSAR be increased?		X
3. Will the probability of malfunctions of equipment important to safety previously evaluated in the FSAR be increased?		X
4. Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?		X
5. Will the possibility of an accident of a different type than any previously evaluated in the FSAR be created?		X
6. Will the possibility of a malfunction of a different type than any previously evaluated in the FSAR be created?		X
7. Will the margin of safety as defined by Plant Licensing Bases be reduced?		X

A written Safety Analysis which documents the bases for these answers shall accompany this form. If any of the above questions are answered "YES," an unreviewed safety question is involved and the item shall not be implemented without prior NRC concurrence.

SECTION II	Yes	No
1. Should this be included in an FSAR update?	X	
2. Is an application for amendment to the Palisades Operating License required?		X
3. Is NRC approval required prior to implementation of this item?		X

SECTION III

G E Jackson , 6/16/98	J G Marks , 6/17/98
Prepared By Date	Reviewed By Date

PALISADES NUCLEAR PLANT
SAFETY ANALYSIS

Item Identification: No EA-GEJ-98-01 Rev 1 Palisades Cycle 14 Disposition of Events Review

DESCRIPTION OF ITEM Analyses and dispositions were performed to document acceptability of lower PCS flowrates in the safety analyses.

JUSTIFICATION

1. No. This is an analysis or disposition of a previously analyzed accidents and does not change any plant equipment, procedure, or process as described in the FSAR. Therefore, the probability of an accident previously evaluated in the FSAR will not be increased.

2. No. The results of the analyses or dispositions have demonstrated that all acceptance criteria are met. Therefore, the consequences of an accident previously evaluated in the FSAR will not be increased.

3. No. The analyses or dispositions do not change any plant equipment, process, or procedure. Therefore, the probability of malfunctions of equipment important to safety will not be increased.

4. No. The analyses or dispositions account for any required equipment failures. Therefore, the consequences of a malfunction of equipment important to safety will not be increased.

5. No. This is an accident analysis. It does not change any plant equipment, process, or procedure. Therefore, the possibility of an accident of a different type than previously evaluated in the FSAR will not be created.

6. No. This does not change any plant equipment, process, or procedure. Therefore, the possibility of a malfunction of a different type than previously evaluated in the FSAR will not be created.

7. No. Tne results of these analyses or dispostions demonstrate that the acceptance criteria are met for all events. The margin of safety as defined by the plant licensing basis will not be reduced.

PALISADES NUCLEAR PLANT
ENGINEERING ANALYSIS CHECKLIST

EA -GEJ-98-01 REV 1

SECTION I Items Affected By This EA	Affected Yes	Affected No	Revision Required	Identify*	Closeout
1.0 Other EAs	☐	☑			
2.0 Design Documents Electrical E-38 through E-49	☐	☑			
3.0 Design Documents Mechanical M239-M246, M249, M257-M261, M660, M664-M666	☐	☑			
4.0 LICENSING DOCUMENTS					
4.1 Final Safety Analysis Report (FSAR)	☑	☐		later - TS change	
4.2 Technical Specifications	☑	☒	TS 3.1		
4.3 Operating Requirements Manual	☐	☑			
5.0 PROCEDURES					
5.1 Administrative Procedures	☐	☑			
5.2 Operating Procedures (SOP, EOP, ONP, etc)	☐	☑			
5.3 Working Procedures	☐	☑			
5.4 Tech Spec Surveillance Test Procedures	☑	☒	RT 118		
6.0 OTHER DOCUMENTS					
6.1 Q-List	☐	☑			
6.2 Plant Drawings	☐	☑			
6.3 Equipment Data Base	☐	☑			
6.4 Spare Parts (Stock/MMS)	☐	☑			
Fire Protection Program Report (FPPR)	☐	☑			
6.6 Design Basis Documents	☐	☑			
6.7 Operating Checklists	☐	☑			
6.8 SPCC/PIPP Oil and Hazardous Material Spill Prevention Plan	☐	☑			
6.9 EQ Documents	☐	☑			
6.10 MOV/AOV Program Documents (Voltage, thrust, weak link, etc)	☐	☑			
6.11 Work Instructions	☐	☑			
6.12 Other ____	☐	☑			

SECTION II

Do any of the following documents need to be generated as a result of the conclusions reached in this EA:

	Yes	No	Reference
1. Corrective Action Document?	☐	☑	Reference
2. EQ Evaluation Sheet?	☐	☑	Reference
3. Safety Evaluation?	☑	☐	Reference Included
4. Design Basis Document Change Request?	☐	☑	Reference
5. FSAR Change Request?	☑	☐	Reference later
6. Verification Test Procedure (for changes to the Design Basis)?	☐	☑	Reference

Completed By DE Jarlan Date 6/16/98

hnical Reviewed By J G Miller Date 6/17/98

*Identify Section, No, Drawing, Document, etc.

TECHNICAL REVIEW CHECKLIST

EA - GEJ-98-01 REV 1

This checklist provides guidance for the review of engineering analyses. Answer questions Yes or No, or N/A if they do not apply. Document all comments on a EA Review Sheet. Satisfactory resolution of comments and completion of this checklist is noted by the Technical Review signature at the bottom of this sheet.

	(Y, N, N/A)
1. Have the proper input codes, standards and design principles been specified?	Y
2. Have the input codes, standards and design principles been properly applied?	Y
3. Are all inputs and assumptions valid and the basis for their use documented?	Y
4. Is Vendor information used as input addressed correctly in the analysis?	Y
5. If the analysis argument departs from Vendor Information/Recommendations, is the departure justification documented?	N/A
6. Are assumptions accurately described and reasonable?	Y
7. Are the design basis changes permitted by this EA bounded by the applicable Safety Review/Evaluation?	Y
8. Are all constants, variables and formulas correct and properly applied?	Y
9. Have all comments been documented on an EA Review Sheet and resolved, or have any minor (insignificant) errors been identified and their insignificance justified? **(Indicate "No Comments," if none were made.)**	Y
10. If the analysis involves welding, is the following information accurately represented on the analysis drawing (Output document)?	N/A

- Type of Weld
  - Size of Weld
  - Material Being Joined
  - Thickness of Material Being Joined
  - Location of Weld(s)
  - Appropriate Weld Symbology

11. Has the objective of the analysis been met?	Y
12. Have administrative requirements such as numbering, format, and indexing been satisfied?	Y

Technical Reviewer JG Metz Date 6/17/90