NEDO-31446
CLASS I
JUNE 1987

NINE MILE POINT UNIT ONE SAFER/CORECOOL/GESTR-LOCA LOSS-OF-COOLANT ACCIDENT ANALYSIS

P. WEI
H. R. FORBES

GENERAL ELECTRIC

NEDO-31446
Class I
June 1987

NINE MILE POINT
NUCLEAR GENERATING STATION
UNIT ONE

SAFER/CORECOOL/GESTR-LOCA
LOSS-OF-COOLANT ACCIDENT ANALYSIS

P. Wei
H. R. Forbes

Approved: L. D. Noble, Manager
Reload Nuclear Engineer-1

Approved: R. Artigas, Manager
Licensing & Consulting Services

NUCLEAR ENERGY BUSINESS OPERATIONS • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT

Please Read Carefully

The only undertakings of the General Electric Company respecting information in this document are contained in the contract between NMPC and General Electric Company for this report, and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than NMPC or for any purpose other than that for which it is intended, is not authorized; and with respect to any unauthorized use, General Electric Company makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

CONTENTS

LIST OF TABLFS

LIST OF FIGURES

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

1.0 INTRODUCTION

The purpose of this document is to provide the results of the loss-of-coolant accident (LOCA) analysis for the Nine Mile Point Nuclear Power Station Unit 1. The analysis was performed using the NRC approved General Electric (GE) SAFER LOCA code and application methodology for BWR/2 plants.

This analysis of postulated plant LOCAs is provided in accordance with NRC requirements and demonstrates conformance with the ECCS acceptance criteria of 10CFR50.46. The objective of the LOCA analysis contained herein is to provide assurance that the most limiting break size, break location, and single failure combination has been considered for the plant. The requirements for demonstrating that these objectives have been satisfied are given in Reference 1. The documentation contained in this report is intended to satisfy these requirements.

A description of the LOCA models and their application is contained in Reference 2. The Nine Mile Point Unit 1 values of the peak cladding temperature (PCT) and maximum oxidation fraction for use in licensing evaluations are calculated for the limiting break. The results conform to all the requirements of 10CFR50.46 and Appendix K. The methodology described in this report will serve as the evaluation basis for future Nine Mile Point-1 fuel designs.

2.0 DESCRIPTION OF MODEL

The General Electric evaluation model used for the Nine Mile Point Unit 1 loss-of-coolant accident (LOCA) analysis consists of three major computer codes. SAFER performs the long-term water level and inventory calculations and fuel rod heatup calculations with the gap conductance supplied by GESTR-LOCA. CORECOOL is used to analyze the transient after the core is uncovered and performs detailed evaluations of the core spray and radiation heat transfer and fuel rod heatup in the high power bundle. These models and their application are discussed in Reference 2. Figure 2-1 shows a flow diagram of the usage of these computer codes, indicating the major code functions and the transfer of major data variables.

2.1 LOCA ANALYSIS COMPUTER CODES

2.1.1 SAFER

The SAFER code is used to calculate the long-term system response of the reactor for reactor transients over a complete spectrum of hypothetical break sizes and locations. SAFER is compatible with the GESTR-LOCA fuel rod model for gap conductance and fission gas release. SAFER tracks, as a function of time, the core water level, system pressure response, ECCS performance, and other primary thermal-hydraulic phenomena occurring in the reactor. SAFER realistically models all regimes of heat transfer which occur inside the core during the event, and provides the outputs such as heat transfer coefficients and PCT as a function of time. SAFER also provides initial and boundary conditions, for the high power fuel bundle, to CORECOOL.

2.1.2 GESTR-LOCA

The GESTR-LOCA code is used to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA for input to SAFER. GESTR-LOCA also initializes the transient pellet-cladding gap conductance in SAFER.

2.1.3 CORECOOL

CORECOOL is a model for evaluation of core heatup transients for a fuel bundle during the period when the core is uncovered. It has detailed core spray heat transfer and thermal radiation models which can provide more realistic predictions of fuel rod heatup at high cladding temperatures (e.g., $\gtrsim$1700°F). The fuel rod model in CORECOOL includes the GESTR transient gap conductance model and the SAFER rod swelling/perforation model.

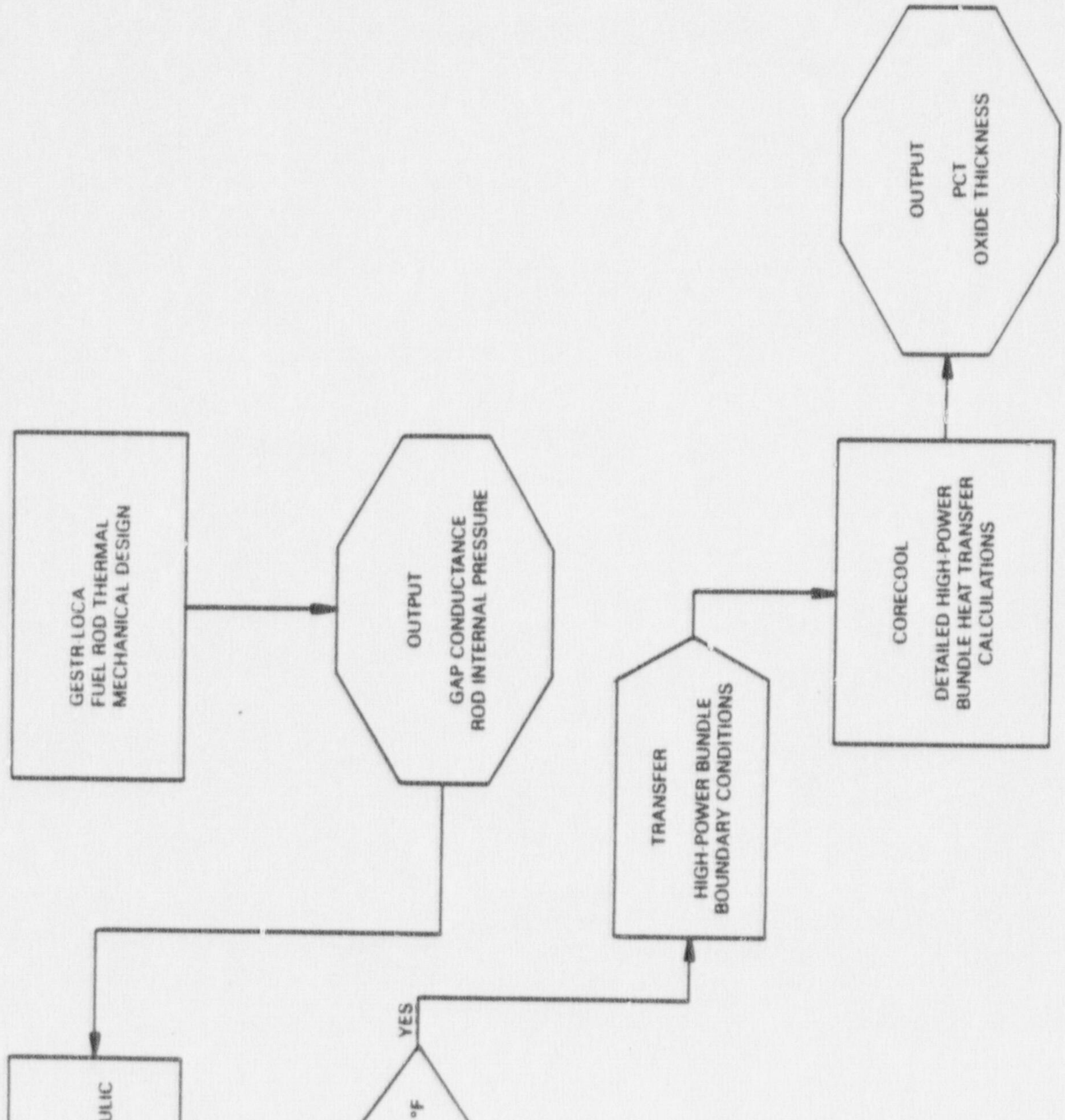

Figure 2-1. Flow Diagram of BWR/2 LOCA Analysis Using SAFER

3.0 ANALYSIS PROCEDURE

3.1 BWR/2 GENERIC ANALYSIS

For the BWR/2 product line, the limiting break was determined from the nominal break spectrum as that break size, location, and ECCS component failure combination that yielded the highest nominal PCT. An Appendix K calculation, utilizing the required features of 10CFR50 Appendix K, was performed for the limiting break.

____ *
______________________ was found to be the limiting break in the nominal break spectrum for the BWR/2 product line. As a result, this case was used to perform the Appendix K calculation. The results of the Appendix K calculation demonstrate that a discharge coefficient of ___ in the Moody Slip Flow Model yields the highest calculated PCT.

Comparison of the Appendix K licensing basis and the upper bound (95th percentile) results demonstrated the conservatism of the BWR/2 licensing application methodology.

3.2 NINE MILE POINT-1 SPECIFIC ANALYSIS

3.2.1 Break Spectrum Evaluation

The plant-specific analysis performed for Nine Mile Point-1 consisted of break sizes ranging from 0.05 ft^2 to the maximum of a DBA recirculation line break ($\sim$5.45 ft^2). This plant-specific analysis evaluated recirculation line and non-recirculation line breaks, as well as an assessment of limiting break location and ECCS component failure. The analysis assumptions (nominal and Appendix K) are presented in Table 3-1.

First, the various breaks were evaluated using the nominal assumptions. The case with the highest PCT was determined to be the _______________ ____ which became the limiting

*GE Proprietary Information has been deleted.

nominal case. The limiting scenario for a spectrum of break sizes was then analyzed again with specifications for the Appendix K calculation (see Table 3-1). The results of the Nine Mile Point-1 nominal and Appendix K cases were compared to assure that the PCT trends as a function of break size are consistent with each other and with those of the generic BWR/2 break spectrum curves (Section 3.1). These results are presented in Section 5.0.

3.2.2 Fuel Exposure Considerations

As discussed in Reference 2, the ECCS acceptance criteria of 10CFR50.46 which are most significant to the BWR/2 LOCA analysis require that the calculated PCT following a postulated LOCA shall not exceed 2200°F and that the calculated maximum cladding local oxidation fraction shall not exceed 17%. For a BWR/2 plant the ECCS performance is limited by different factors as the fuel exposure increases.

Table 3-1

ANALYSIS ASSUMPTIONS FOR NINE MILE POINT-1 CALCULATIONS

	Nominal	Appendix K
1. Decay Heat	1979 ANS	1971 ANS + 20% (see Figure 3-1)
2. Transition Boiling Temperature	Iloeje Correlation	300°F
3. Break Flow	1.25 HEM (SUB) HEM (SAT)	Moody Slip
4. Metal-Water Reaction	Cathcart	Baker-Just
5. Core Power	100%	102%
6. MAPLHGR[a] (kW/ft) Low Exposure High Exposure		
7. ECCS Water Temperature	120°F	120°F
8. ECCS Flow	See Table 4-1	See Table 4-1
9. ECCS Flow to Hot Bundle (2 Core Sprays)		
10. Fuel Stored Energy	Best-Estimate GESTR	Best-Estimate GESTR
11. Rod Internal Pressure	Best-Estimate GESTR	Best-Estimate GESTR
12. Cladding Rupture Stress	BWR Design Values	BWR Design Values

[a]Based on PLHGR of _________ (low exposure) or _________ (high exposure). A multiplier of 1.02 was applied to the Appendix K values.

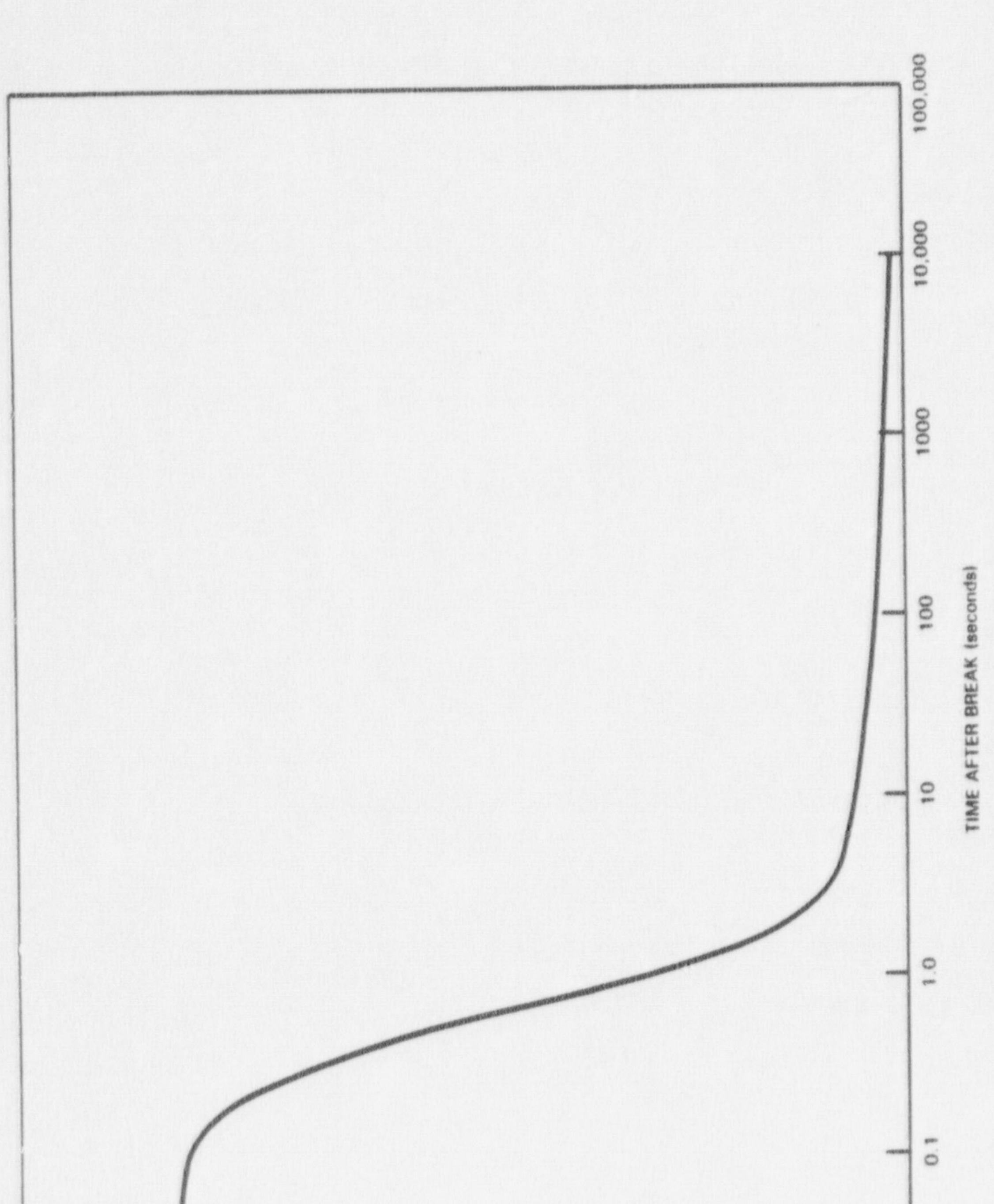

Figure 3-1. Normalized Power (Appendix K)

4.0 INPUT TO ANALYSIS

4.1 PLANT-SPECIFIC PARAMETERS

A list of the significant Nine Mile Point-1 plant-specific input parameters to the LOCA analysis is presented in Table 4-1. Table 4-2 identifies the break locations and corresponding single-failure/system available combinations specifically evaluated for Nine Mile Point-1.

4.2 TIMING FOR THE ONSET OF BOILING TRANSITION

The current Nine Mile Point-1 LOCA licensing report (Reference 3) concludes, from the results of the LAMB and SCAT evaluations (based on an initial MCPR of ____, that nucleate boiling is maintained prior to core uncovery for small recirculation line breaks ______________. For large breaks (DBA to____ DBA), where there is very rapid flow coastdown, the duration of nucleate boiling following the break is calculated using the GE dryout correlation (in the CHASTE code), which is based on instantaneous flow stagnation conditions. (The LAMB, SCAT and CHASTE models and applications are described in Reference 4.)

In this Nine Mile Point-1 SAFER Analysis, for break sizes from DBA to ______, the break spectrum evaluation (results summarized in Section 5.0) utilized a timing for the onset of boiling transition based on the GE dryout correlation. For break sizes smaller than _______, nucleate boiling was assumed to be maintained until core uncovery.

This approach established that the Nine Mile Point-1 SAFER-LOCA licensing evaluation is dependent upon the LAMB and SCAT analyses __________________ only for those break sizes less than _______. Results of the break spectrum evaluation (Section 5.0) indicate that the small break PCTs are significantly below those of the larger recirculation line breaks. Therefore, future potential changes in MCPR limit are not expected to affect the limiting LOCA scenario and the resultant MAPLHGR calculations.

Table 4-1

OPERATIONAL AND ECCS PARAMETERS

A. Plant Parameters

Core Thermal Power (MWth)

Nominal	1850	(100% of Rated)
Appendix K	1887	(102% of Rated)

Vessel Steam Output (lbm/hr)

Vessel Steam Dome Pressure (psia)

Maximum Recirculation Line Break Area (ft^2) 5.446

Initial MCPR

Initial Water Level Scram Trip Level

B. Emergency Core Cooling System Parameters

Core Spray System

Vessel Pressure at Which Pump Can Deliver Flow (psid)

System Flow at Vessel Pressure (psid) for One Loop (gpm)

Initiating Signals and Setpoints
Low Water Level
or
High Drywell Pressure (psig)

Runout Flow (at Zero psid) for Each Loop

Maximum Allowable Delay Time from Initiating Signal to Pump at Rated Speed (sec)

Injection Valve Stroke Time (sec)

Pressure Permissive at Which Injection Valve Opens (psid)

Core Spray Flow to Hot Bundle (2 headers) (gpm)

Table 4-1 (Continued)

OPERATIONAL AND ECCS PARAMETERS

ADS

Total Number of Valves in System

Number of Valves Assumed in Analysis

Minimum Flow Capacity of 3 Valves (lbm/hr)
at Vessel Pressure (psig)

Initiating Signals
Low Water Level
and
High Drywell Pressure (psig)

Time Delay After Initiating Signals (sec)

Emergency Condensers

Total Number of Emergency Condensers

Initiating Signals
Low Water Level
or
High Vessel Pressure (psia)

Maximum Isolation Valve Stroke Time (sec)

Maximum Operating Pressure (Vessel) (psia)

Initial Operating Temperature on
Shell Side of Condenser (°F)

Initial Water Mass on Shell
Side of Each Condenser (Gallons)

Surface Heat Transfer Area
of Each Condenser (ft^2)

Table 4-2

SINGLE-FAILURE EVALUATION FOR NINE MILE POINT-1

Break Location	Single Failure	Available Systems
Recirculation Line		
Feedwater and Main Steamlines		
Core Spray Line		

EC = Emergency Condenser

CS = Core Spray

ADS = Automatic Depressurization System

5.0 PLANT-SPECIFIC RESULTS

5.1 BREAK SPECTRUM CALCULATIONS

5.1.1 Recirculation Line Breaks

A sufficient number of break sizes and ECCS failure combinations were evaluated using nominal input conditions. The results (Table 5-1) identified the ______________________________ as limiting. Analyses with Appendix K input assumptions were performed for four break sizes from the limiting scenario determined by the nominal break spectrum. Table 5-2 lists the Appendix K PCT results. Figure 5-1 shows a comparison of these two break spectrums and, in both cases, the highest calculated PCT is associated with the largest break area.

____

___ is the limiting break for the nominal break spectrum with a calculated peak cladding temperature of ______ (low exposure). The corresponding PCT for this break with Appendix K specified models was calculated to be ______ and ______ for the ____________________ respectively. Plots showing system responses for all break spectrum cases are presented in the Appendix to this report.

5.1.2 Non-Recirculation Line Breaks

Evaluations were also performed for some of the non-recirculation line breaks. These breaks (including feedwater, core spray and main steam lines) were evaluated with the nominal input conditions and maximum line break sizes. PCT results (Table 5-3) show that these non-recirculation line guillotine breaks sustain essentially no heatup and are far from becoming candidates for the limiting event. The same conclusion applies for break sizes smaller than the guillotine break for these lines. The system responses of these breaks are also presented in the Appendix.

Nine Mile Point-1 plant-specific evaluations were not performed for other non-recirculation line breaks (e.g., EC lines, liquid instrument lines, cleanup system lines, etc.). These non-recirculation line breaks will not become candidates for the limiting event, since they are essentially the same as small recirculation or steam line breaks.

5.2 TECHNICAL SPECIFICATION MAPLHGR LIMITS

GE BWR MAPLHGR limits (as a function of fuel exposure) are based on the ____
____

For BWR/2 plants, in general, and the Nine Mile Point-1 plant, specifically, the MAPLHGR calculated from the ____________________ is limiting and determines the Technical Specification limits.

The MAPLHGR limits for the P8DRB299 fuel bundle were evaluated as a function of exposure with the limiting scenario ________________________________ ____ identified in the break spectrum analyses.

Table 5-4 lists the P8DRB299 MAPLHGR limits (five-loop operation) along with the calculated PCT and peak local oxidation fraction. The highest PCT calculated was ______ at an exposure of ___________, while the highest peak local oxidation fraction calculated was ____________________

Based on the results for the P8DRB299 bundle, it is expected that the SAFER evaluation methodology would show that the current MAPLHGR limits for fuel types 8DNB277 and P8DNB277 (calculated by Reference 3) are conservative at all exposures. Therefore, no change will be made in the MAPLHGRs for these two fuel types because they are not expected to limit core operation.

For fuel types in future Nine Mile Point-1 reloads, this SAFER/LOCA report can serve as an evaluation basis for the plant system responses, and supplemental calculations can be performed to determine the fuel type specific MAPLHGRs.

5.3 ALTERNATE OPERATING MODE CONSIDERATIONS

5.3.1 Three- and Four-Loop Operation

The Reference 3 analysis of three- and four-recirculation-loop operation identified two main differences in the LOCA analysis as compared to the normal five-loop case:

The effects of these differences on the SAFER calculations will depend on the break size.

Evaluation results for four-loop and three-loop operation, with the inoperative loops isolated, are summarized in Table 5-5a. The DBA break (low and high exposure) and the 0.1 ft^2 break (highest small-break PCT from the nominal analysis) cases were calculated with a ______________________________ compared with the five-loop base cases. The results show that a __________ __________ is adequate to compensate for the effect of loss of inventory and the faster coastdown for the large and small breaks. Table 5-5b summarizes the four-loop and three-loop MAPLHGR multipliers for the P8DRB299 fuel.

5.3.2 <u>Reduced Core Flow Operation (ELLLA)</u>

The impact, on MAPLHGR limits, of operating at rated reactor power and reduced core flow [i.e., in the Extended Load Line Limit Analysis (ELLLA)

Table 5-1

SUMMARY OF RECIRCULATION LINE BREAK RESULTS - NOMINAL EVALUATION(1)

Break Size (ft^2)	PCT (°F)	Peak Local Oxidation (%)	Core-Wide Metal-Water Reaction (%)
Discharge Break - Low Exposure (14 GWd/MTU)			
Discharge Break - High Exposure (22 GWd/MTU)			
Suction Break - Low Exposure (14 GWd/MTU)			

(2)Less than Appendix K low exposure DBA.

Table 5-2

SUMMARY OF RECIRCULATION LINE BREAK RESULTS - APPENDIX K EVALUATION(1)

Break Size (ft^2)	PCT (°F)	Peak Local Oxidation (%)	Core-Wide Metal-Water Reaction (%)
Discharge Break - Low Exposure (14 GWd/MTU)			
Discharge Break - High Exposure (22 GWd/MTU)			

(2)Less than Appendix K low exposure DBA.

Table 5-3

SUMMARY OF NON-RECIRCULATION LINE BREAK RESULTS - NOMINAL EVALUATION(1)

	PCT (°F)	Peak Local Oxidation (%)	Core Wide Metal-Water Reaction (%)
Core Spray Line break			
Steamline Break Inside Containment			
Steamline Break Outside Containment			
Feedwater Line Break			

(2)Less than Appendix K low exposure DBA.

Table 5-4

MAPLHGR vs. AVERAGE PLANAR EXPOSURE
FIVE-LOOP OPERATION

Plant: NMP-1 | Fuel Type: P8DRB299

Average Planar Exposure (GWd/MTU)	MAPLHGR (kW/ft)	PCT (°F)	Local Oxidation Fraction
0.22	10.9		
1.1	10.9		
5.5	10.9		
11.0	10.9		
14.0	10.8		
15.0	10.7		
16.5	10.4		
22.0	9.7		
27.5	9.6		
33.0	9.5		
38.5	9.3		
44.0	9.2		
50.0	9.0		

Table 5-5a

SUMMARY OF FOUR-LOOP AND THREE-LOOP MAPLHGR MULTIPLIERS EVALUATION
P8DRB299 FUEL, ISOLATED CONDITION

Parameter	Loops		
	Five*	Four*	Three**
MAPLHGR Multiplier			
DBA Break PCT (°F)			
Low Exposure			
High Exposure/Oxide Thickness (%)			
Small (0.1 ft^2) Break PCT (°F)			

Notes

*Four-loop and five-loop are evaluated with identical Appendix K conditions.

Table 5-5b

MAPLHGR MULTIPLIERS FOR P8DRB299 FUEL

*Results are applicable to all exposure ranges.

(GE PROPRIETARY FIGURE)

Figure 5-1. Nominal and Appendix K LOCA Recirculation Line Break Spectrum Comparison

6.0 CONCLUSIONS

The recirculation line break results presented in Section 5.1 demonstrate that a sufficient number of Nine Mile Point-1 plant-specific PCT points have been evaluated to verify that the trend of the PCT curves, for both the nominal and Appendix K calculations, is similar to the (Reference 2) generic PCT versus break size curves.

It has been demonstrated generically that the PCT calculated in accordance with the application methodology described in Reference 2 maintains margin for licensing evaluations (i.e., the licensing basis PCT is at least the upper 95th percentile PCT). This was verified by separate calculations to determine the upper 95th probability values of PCT at the most limiting conditions. These calculations were performed to qualify the "Appendix K Procedure" as being sufficiently conservative. The generic upper bound PCTs, which includes a 50°F conservatism (Δ5) assigned by the NRC (Reference 1), were ______ and ______ for low and high exposures, respectively. By comparison, the generic licensing basis Appendix K evaluation with SAFER/CORECOOL for the limiting conditions provided PCTs of ______ and ______ for low and high exposures, providing ____ (low exposure) and ____ (high exposure) margins to the upper bound requirements.

The Nine Mile Point-1 plant-specific Appendix K analysis will have similar margin to the 95th percentile PCT because of the following considerations:

(1) Nine Mile Point-1 has an ECCS configuration identical to that used in the generic BWR/2 analysis. Therefore, the limiting case LOCA for both Nine Mile Point-1 and the generic BWR/2 is the ____________ ____.

(2) The key operating parameters for the plant-specific Nine Mile Point Unit 1 analysis are similar to the inputs used in the calculations of the generic analysis PCT.

(3) The similarity between the generic and the plant-specific evaluations (in plant configuration and the operating parameters) is responsible for the similar PCTs calculated with SAFER/CORECOOL.

The generic nominal analysis reported ________ and ______ (for the limiting scenario), while the plant-specific analysis yielded ______ and ______ for the low and high exposures, respectively. The Appendix K licensing basis results were also very similar.

Therefore, it is confirmed that the generic assessment (Reference 2) is applicable to Nine Mile Point-1 and the Nine Mile Point-1 Appendix K licensing basis analysis exceeds the upper bound 95th percentile PCT. Also, the plant-specific results of the Appendix K licensing analysis of Section 5.0 meet the criteria of 10CFR50.46.

In conclusion, it is verified that the Nine Mile Point-1 plant-specific SAFER/CORECOOL/GESTR-LOCA analysis meets the explicit requirements of the Reference 1 NRC Safety Evaluation Report.

7.0 REFERENCES

1. Letter, A. C. Thadani (NRC) to H. C. Pfefferlen (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-30996-P, Volume II, 'SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet and Non-Jet Pump Plants'", May 1987.

2. "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants", NEDE-30996-P, June 1986.

3. "LOCA Analysis Report for Nine Mile Point Unit 1 NGS", NEDO-24348, August 1981.

4. "General Electric Standard Application for Reactor Fuel (U.S. Supplement)", NEDE-24011-P-A-8-US, May 1986.

APPENDIX

NINE MILE POINT 1
SYSTEM RESPONSE CURVES

APPENDIX

NINE MILE POINT 1 SYSTEM RESPONSE CURVES

This appendix contains the system response curves for Nine Mile Point 1. Table A-1 contains the figure numbering sequence for the recirculation line breaks, and Table A-2 contains the figure numbering sequence for the non-recirculation line breaks.

Table A-1

NINE MILE POINT-1 RECIRCULATION LINE BREAK FIGURE SUMMARY

	DBA (NOM)	DBA (App K)	DBA (NOM)	80% DBA (NOM)	80% DBA (App K)	60% DBA (NOM)	60% DBA (App K)	40% DBA (NOM)	40% DBA (App K)
Break	DSCG	DSCG	Suction	DSCG	DSCG	DSCG	DSCG	DSCG	DSCG
Failure									
Hot and Average Channel Water Level	1a	2a	3a	4a	5a	6a	7a	8a	9a
Reactor Vessel Pressure	1b	2b	3b	4b	5b	6b	7b	8b	9b
Peak Cladding Temperature	1c	2c	3c	4c	5c	6c	7c	8c	9c
Hot Channel Heat Transfer Coefficient	1d	2d	3d	4d	5d	6d	7d	8d	9d

Table A-1 (Continued)

	1.0 ft^2 (Nom)	0.5 ft^2 (Nom)	0.1 ft^2 (Nom)	0.05 ft^2 (Nom)	High Exposure DBA (Nom)	High Exposure DBA (App K)
Break	DSCG	DSCG	DSCG	DSCG	DSCG	DSCG
Failure						
Hot and Average Channel Water Level	10a	11a	12a	13a	14a	15a
Reactor Vessel Pressure	10b	11b	12b	13b	14b	15b
Peak Cladding Temperature	10c	11c	12c	13c	14c	15c
Hot Channel Heat Transfer Coefficient	10d	11d	12d	13d	14d	15d
Oxide Thickness	-	-	-	-	-	15e

Table A-2

NINE MILE POINT-1 NON-RECIRCULATION LINE BREAK FIGURE SUMMARY

	Core Spray Line	Steamline (Inside Containment)	Steamline (Outside Containment)	Feedwater Line
Failure				
Hot and Average Channel Water Level	16a	17a	18a	19a
Reactor Vessel Pressure	16b	17b	18b	19b
Peak Cladding Temperature	16c	17c	18c	19c
Hot Channel Heat Transfer Coefficient	16d	17d	18d	19d

Figure A-1. DBA DSCG (Nominal) -
a. Water level in Hot and Average Channel

Figure A-1. DBA DSCG (Nominal) -
b. Reactor Vessel Pressure

Figure A-1. DBA DSCG (Nominal) -
c. Peak Cladding Temperature

Figure A-1. DBA DSCG (Nominal) -
d. Heat Transfer Coefficient

Figure A-2. DBA DSCG (Appendix K) -
a. Water Level in Hot and Average Channel

Figure A-2. DBA DSCG (Appendix K) -
b. Reactor Vessel Pressure

Figure A-2. DBA DSCG (Appendix K) -
c. Peak Cladding Temperature

Figure A-2. DBA DSCG (Appendix K) -
d. Heat Transfer Coefficient

Figure A-3. DBA Suction (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-3. DBA Suction (Nominal) -
b. Reactor Vessel Pressure

Figure A-3. DBA Suction (Nominal) -
c. Peak Cladding Temperature

Figure A-3. DBA Suction (Nominal) -
d. Heat Transfer Coefficient

Figure A-4. 80% DBA DSCG (Nominal) -
a. Water level in Hot and Average Channel

Figure A-4. 80% DBA DSCG (Nominal) -
b. Reactor Vessel Pressure

Figure A-4. 80% DBA DSCG (Nominal) -
c. Peak Cladding Temperature

Figure A-4. 80% DBA DSCG (Nominal) -
d. Heat Transfer Coefficient

Figure A-5. 80% DBA DSCG (Appendix K) -
a. Water Level in Hot and Average Channel

Figure A-5. 80% DBA DSCG (Appendix K) -
b. Reactor Vessel Pressure

Figure A-5. 80% DBA DSCG (Appendix K) -
c. Peak Cladding Temperature

Figure A-5. 80% DBA DSCG (Appendix K) -
d. Heat Transfer Coefficient

Figure A-6. 60% DBA DSCG (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-6. 60% DBA DSCG (Nominal) -
b. Reactor Vessel Pressure

Figure A-6. 60% DBA DSCG (Nominal) -
c. Peak Cladding Temperature

Figure A-6. 60% DBA DSCG (Nominal) -
d. Heat Transfer Coefficient

Figure A-7. 60% DBA DSCG (Appendix K) -
a. Water Level in Hot and Average Channel

Figure A-7. 60% DBA DSCG (Appendix K) -
b. Reactor Vessel Pressure

Figure A-7. 60% DBA DSCG (Appendix K) -
c. Peak Cladding Temperature

Figure A-7. 60% DBA DSCC (Appendix K) -
d. Heat Transfer Coefficient

Figure A-8. 40% DBA DSCG (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-8. 40% DBA DSCG (Nominal) -
b. Reactor Vessel Pressure

Figure A-8. 40% DBA DSCG (Nominal) -
c. Peak Cladding Temperature

Figure A-8. 40% DBA DSCG (Nominal) -
d. Heat Transfer Coefficient

Figure A-9. 40% DBA DSCG (Appendix K) -
a. Water Level In Hot and Average Channel

Figure A-9. 40% DBA DSCG (Appendix K) -
b. Reactor Vessel Pressure

Figure A-9. 40% DBA DSCG (Appendix K) -
c. Peak Cladding Temperature

Figure A-9. 40% DBA DSCG (Appendix K) -
d. Heat Transfer Coefficient

Figure A-10. 1.0 Ft^2 DSCG (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-10. 1.0 Ft^2 DSCG (Nominal) -
b. Reactor Vessel Pressure

Figure A-10. 1.0 Ft^2 DSCG (Nominal) -
c. Peak Cladding Temperature

Figure A-10. 1.0 Ft^2 DSCG (Nominal) -
d. Heat Transfer Coefficient

Figure A-11. 0.5 Ft^2 DSCG (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-11. 0.5 Ft^2 DSCG (Nominal) -
b. Reactor Vessel Pressure

Figure A-11. 0.5 Ft^2 DSCG (Nominal) -
c. Peak Cladding Temperature

Figure A-11. 0.5 Ft^2 DSCG (Nominal) -
d. Heat Transfer Coefficient

Figure A-12. 0.1 Ft^2 DSCG (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-12. 0.1 Ft^2 DSCG (Nominal) -
b. Reactor Vessel Pressure

Figure A-12. 0.1 Ft^2 DSCG (Nominal) -
c. Peak Cladding Temperature

Figure A-12. 0.1 Ft^2 DSCG (Nominal) -
d. Heat Transfer Coefficient

Figure A-13. 0.05 Ft^2 DSCG (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-13. 0.05 Ft^2 DSCG (Nominal) -
b. Reactor Vessel Pressure

Figure A-13. 0.05 Ft^2 DSCG (Nominal) -
c. Peak Cladding Temperature

Figure A-13. 0.05 Ft^2 DSCG (Nominal) -
d. Heat Transfer Coefficient

Figure A-14. DBA DSCG - High Exposure (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-14. DBA DSCG - High Exposure (Nominal) -
b. Reactor Vessel Pressure

Figure A-14. DBA DSCG - High Exposure (Nominal) -
c. Peak Cladding Temperature

Figure A-14. DBA DSCG - High Exposure (Nominal) -
d. Heat Transfer Coefficient

Figure A-15. DBA DSCG - High Exposure (Appendix K) -
a. Water Level in Hot and Average Channel

Figure A-15. DBA DSCG - High Exposure (Appendix K) -
b. Reactor Vessel Pressure

Figure A-15. DBA DSCG - High Exposure (Appendix K) -
c. Peak Cladding Temperature

Figure A-15. DBA ESCG - High Exposure (Appendix K) -
d. Heat Transfer Coefficient

Figure A-15. DBA DSCG - High Exposure (Appendix K) -
e. Oxide Thickness

Figure A-16. Core Spray Line (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-16. Core Spray Line (Nominal) -
b. Reactor Vessel Pressure

Figure A-16. Core Spray Line (Nominal) -
c. Peak Cladding Temperature

Figure A-16. Core Spray Line (Nominal) -
d. Heat Transfer Coefficient

Figure A-17. Steam Line Inside Containment (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-17. Steam Line Inside Containment (Nominal) -
b. Reactor Vessel Pressure

Figure A-17. Steam Line Inside Containment (Nominal) -
c. Peak Cladding Temperature

Figure A-17. Steam Line Inside Containment (Nominal) -
d. Heat Transfer Coefficient

Figure A-18. Steam Line Outside Containment (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-18. Steam Line Outside Containment (Nominal) -
b. Reactor Vessel Pressure

Figure A-18. Steam Line Outside Containment (Nominal) -
c. Peak Cladding Temperature

Figure A-18. Steam Line Outside Containment (Nominal) -
d. Heat Transfer Coefficient

Figure A-19. Feedwater Line (Nominal) -
a. Water Level in Hot and Average Channel

Figure A-19. Feedwater Line (Nominal) -
b. Reactor Vessel Pressure

Figure A-19. Feedwater Line (Nominal) -
c. Peak Cladding Temperature

Figure A-19. Feedwater Line (Nominal) -
d. Heat Transfer Coefficient