U.S. NUCLEAR REGULATORY COMMISSION
DOCKET 50-410
LICENSE NPF-69

NINE MILE POINT NUCLEAR STATION UNIT 2

UPDATED SAFETY ANALYSIS REPORT

OCTOBER 2016

REVISION 22

Appendix A

LIST OF EFFECTIVE FIGURES

Figure No.	Revision Number
F A.0-1	R19

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RELOAD ANALYSIS

A.0 INTRODUCTION

Reload Cycles

The Updated Safety Analysis Report (USAR) describes in detail the design and analysis that forms the licensing basis for Nine Mile Point Nuclear Station - Unit 2 (Unit 2). It is primarily based on information that bounded Cycle 1, the initial fuel load. Appendix A represents the cycle-specific information and analytical results for each reload, which includes the specific fuel loaded in the core and the respective safety analysis. Appropriate cross-references are provided within the USAR chapters and this appendix.

The reload safety analysis is based on the General Electric (GE) report, General Electric Standard Application for Reactor Fuel (GESTAR II), described in Reference 1. GESTAR II represents generic information relative to the GE fuel design and analysis and consists of a description of the fuel design and fuel thermal-mechanical, nuclear, and thermal-hydraulic analyses bases. It provides information and methods used to determine reactor limits that are independent of a plant-specific application. Plant-specific information and the transient and accident methods used are given in the United States (US) supplement. Proposed changes to GESTAR II are submitted to the appropriate regulatory body for review and approval. A listing of Nuclear Regulatory Commission (NRC)-approved amendments is provided in GESTAR II. All approved changes are incorporated as a revision into the text.

This appendix to the USAR reflects the reload cycle design and analysis. Since the reload reflects the fuel-related description and analytical results, the cycle-specific information affects Chapters 4, 5, 6, and 15. This Appendix reflects changes to only those chapters with an Appendix A designator.

Unit 2 was originally licensed to operate at reactor core power levels not in excess of 3,323 MWt. Beginning with Cycle 5, the licensed core thermal power limit was increased by 4.3 percent to 3,467 MWt. In Cycle 14, the licensed core thermal power limit was increased by another 15.0 percent to 3,988 MWt. The cycle-specific transient analyses and core operating limits described in this appendix include the effects of the power uprates.

The Unit 2 design is intended to be valid for the licensed life of the plant. The supplemental cycle-specific safety analysis assures that the plant can be operated safely and not pose any undue risk to the health and safety of the public. This is accomplished by demonstrating that radioactive releases from

plants for normal operation, anticipated operational occurrences, and postulated accidents meet applicable regulations.

Unit 2 plant operation must meet various safety requirements defined in the Code of Federal Regulations (CFR). In order to evaluate the safety impact of the current cycle, fuel lattice physics calculations and 3-D simulation, transient, and accident evaluations were performed. The NRC-approved methodologies described in GESTAR II were used.

The transient analysis (Section A.15) is based on results which fully bound the licensed operating states. This analysis evaluates the extension of the operating domain to include the maximum extended load line limit analysis plus region (MELLLA+) and increased core flow (ICF) to 105 percent of rated core flow.

The core-wide nuclear and thermal reactivity characteristics, when combined with the rest of the plant systems and equipment, determine the normal steady-state operation, transient and accident performance of the plant. The limiting transients performed are described in detail in Section A.15.

The evaluation utilizes the following methodologies and boundary conditions. The items that are different from the Cycle 1 analysis are marked "*". These items were initially evaluated for application in Cycle 2, Reload 1. Analyses for subsequent reload cycles are performed to confirm the applicability of these operational conditions and methodologies to Unit 2. The results of these analyses are summarized in this Appendix. In addition, the unit was evaluated for concurrent operational options as shown on Figure A.0-1.

1. GEMINI system of methods.*

2. GEXL-PLUS correlation.*

3. Maximum extended load line limit analysis plus (MELLLA+).*

4. Increased core flow (ICF).*

5. One main steam isolation valve (MSIV) out of service (OOS) (closed).* Detailed discussion is provided in Appendix 15D.

6. End of cycle-recirculation pump trip (EOC-RPT) OOS.

7. Turbine bypass OOS.

8. Up to two safety relief valves (SRV) OOS (including automatic depressurization system [ADS]).

9. Operation with a single recirculation loop. Detailed discussion is provided in Appendix 15B.

For this reload analysis, the four main safety areas evaluated were:

1. Shutdown margin demonstration.
2. Transient minimum critical power ratio (MCPR) responses.
3. Overpressure protection.
4. Emergency core cooling system (ECCS) performance during a postulated design basis loss-of-coolant accident (LOCA).

The shutdown margin calculation is performed to demonstrate that the core is capable of being made subcritical with sufficient margin in the most reactive condition throughout the cycle with the strongest control rod withdrawn. In addition to controlling core reactivity with control rods, the standby liquid control system (SLCS) is designed to inject liquid boron in the core when needed to bring the core to a subcritical condition. These two calculations demonstrated that the shutdown margin requirement is met, as documented in the cycle-specific Supplemental Reload Licensing Report (SRLR).

The performance of the anticipated operational occurrences (moderate frequency events) was evaluated with the methodologies described in GESTAR II. The limiting events analyzed are determined by a sensitivity study described in Reference 1 that examines the impact of MCPR due to the change in fuel design. Based on results of the study, several limiting events have been identified and analyzed using the appropriate input parameters and boundary conditions. The MCPR results of these limiting transients form the basis of the MCPR operating limits. Implementation of these MCPR operating limits in the Technical Specifications ensures that the MCPR safety limit will not be exceeded during the most severe anticipated operational occurrences.

The overpressure protection evaluation demonstrated that the ASME Code limits and requirements are met (Section A.5).

A hot bundle heatup calculation resulting from the design basis LOCA demonstrated that the peak cladding temperature (PCT) and maximum oxidation fraction are bounded by compliance with the 10CFR50.46 acceptance criteria. The results of this calculation combined with the fuel design criteria determine the maximum average planar linear heat generation rate (MAPLHGR) operating limits.

Starting with Cycle 4, the LOCA evaluations were performed using the SAFER/GESTR-LOCA methodology described in Reference 1. The results of this analysis for the current fuel cycle show that all 10CFR50.46 acceptance criteria are met without restricting the MAPLHGR operating limits.

A.0.1 References

1. General Electric Co., General Electric Standard Application for Reactor Fuel, GESTAR II, NEDE-24011-P-A, and United States Supplement GESTAR II, NEDE-24011-P-A-US (latest approved revision as specified in the Core Operating Limits Report).

A.4 REACTOR

Chapter 4 provides information on the design of the reactor including the fuel, materials and reactivity controls. The fuel design description of Chapter 4 has been revised to reflect GESTAR II[1]. The description of the reactor material and reactivity control system is considered applicable throughout the life of the plant until new reactor materials or new control rod designs are used. The following subsections describe the cycle-specific information resulting from the current reload analysis.

A.4.1 Summary Description

This section provides general information on the reactor assembly internals. No change has been made to this section as a result of the reload.

A.4.2 Fuel System Design

This section provides general information on the fuel system design. No change has been made to this section as a result of the reload.

A.4.3 Nuclear Design

A.4.3.1 Design Basis

No change has been made to this section as a result of the reload.

A.4.3.2 Description

No change has been made to this section as a result of the reload.

A.4.3.2.1 Nuclear Design Description

The nuclear and thermal hydraulic characteristics of the fuel bundles are simulated in a GE lattice computer model and 3-D simulator for the development of the reload core loading pattern. This loading pattern considered the integrated effect of mixing the new bundles with the irradiated bundles in the core. The objective of this loading pattern is to optimize the fuel burnup efficiency. The consideration includes meeting predetermined target radial and axial power distributions, thermal limits, and fuel cycle exposures. The reload core reference loading pattern and its target cycle exposure are provided in the SRLR.

The reload cycle core loading is specified in the SRLR (Reference 6, Section A.4.4.7).

A.4.4 Thermal-Hydraulic Design

A.4.4.1 Design Bases

No change has been made to this section as a result of the reload.

A.4.4.2 Description of the Thermal-Hydraulic Design of the Reactor Core

No change has been made to this section as a result of the reload.

A.4.4.3 Description of the Thermal-Hydraulic Design of the Reactor Coolant System

No change has been made to this section as a result of the reload.

A.4.4.4 Evaluation

The thermal-hydraulic design of the reactor core and reactor coolant system (RCS) is based upon an objective of no fuel damage during normal operation or during anticipated operational occurrences.

The uncertainty for the inputs used in the bounding statistical analysis is discussed in Chapter 4, Section 4.4.2.9. The results of the analysis show that at least 99.9 percent of the fuel rods in the core are expected to avoid boiling transition if the MCPR is equal to or greater than the applicable value in the Technical Specifications. The safety limit MCPR value for the current fuel cycle is specified in the Technical Specifications for two-recirculation-loop operation and for single-loop operation (SLO).

The generation of the MCPR operating limit requires a statistical analysis of the core near the limiting MCPR condition. The statistical analysis is used to determine the MCPR corresponding to the transient design requirement given in GESTAR II. The TRACG methods were introduced in Cycle 15. All previous reloads used GEMINI methods except Cycle 1, which used GENESIS.

The TRACG methods replace the previous ODYN GEMINI methods. With both the TRACG and the GEMINI methods, the MCPR response for each event is determined using statistically determined scram times. Event-unique adders are applied to adjust for Technical Specification scram times and other uncertainties and conservatism to develop the operating limit MCPR values from the analytically determined MCPR responses.

Unit 2 has implemented the Boiling Water Reactor Owners' Group (BWROG) Long-Term Stability Solution Option III (Oscillation Power Range Monitor - OPRM) as described in Reference 7.

Plant-specific analysis incorporating the Option III hardware is described in Reference 8.

Reload validation has been performed in accordance with Reference 9. The two conditions evaluated are for a postulated oscillation at 45 percent core flow steady state operation and following a two recirculation pump trip from the limiting full power operation state point.

Unit 2 uses the backup stability protection (BSP) methodology (Reference 10) in the event that the OPRM system is declared inoperable. The BSP region boundaries are calculated for each cycle. The endpoints of the proposed BSP regions are connected using the modified shape function (MSF) as defined and approved in Reference 11. For the MSF, the calculated BSP region boundary was validated at the midpoint against the decay ratio acceptance criterion.

A.4.4.4.1 Critical Power

The GEXL critical power correlation was utilized for Cycle 1 fuel (P8x8R) in thermal-hydraulic evaluations. For current fuel cycles, the GEXL-PLUS correlation is used. This correlation is discussed in Reference 1.

A.4.4.5 Testing and Verification

No change has been made to this subsection as a result of the reload.

A.4.4.6 Instrumentation Requirements

No change has been made to this subsection as a result of the reload.

A.4.4.7 References

1. General Electric Co., General Electric Standard Application for Reactor Fuel, including United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision as specified in the Core Operating Limits Report).

2. General Electric Co., General Electric Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application, NEDO-10958A, January 1977.

3. Deleted.

4. Letter, C. O. Thomas (NRC) to H. C. Pfefferlen (GE), Acceptance for Referencing of Licensing Topical Report NEDE-24011, Revision 6, Amendment 8, Thermal Hydraulic Stability Amendment to GESTAR II, April 24, 1985.

5. General Electric Co., Service Information Letter No. 380, Revision 1, BWR Core Thermal Hydraulic Stability, February 14, 1984.

6. Global Nuclear Fuel Report, Supplemental Reload Licensing Report for Nine Mile Point Unit 2, as specified in the Core Operating Limits Report source documents.

7. NEDC-33075P-A, Revision 8, Licensing Topical Report, "General Electric Boiling Water Reactor Detect and Suppress Solution Confirmation Density," November 2013.

8. Deleted.

9. Deleted.

10. Deleted.

11. Deleted.

12. Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, NEDE-32906P, Supplement 3-A, Revision 1, April 2010.

13. TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses, NEDE-32906P-A, Revision 3, September 2006.

A.5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Chapter 5 provides information on the RCS and pressure-containing components that define the reactor coolant pressure boundary (RCPB). Current fuel cycle changes include the ASME upset category overpressure protection analysis and the associated SRV setpoint and valve capacity (Table A.5.2-1) assumed in the analysis. Only Chapter 5, Sections 5.2.2.2.2 and 5.2.2.2.3, are affected and are provided in the following subsections. (The following subsections have been numbered to correspond to the affected sections of Chapter 5 for clarity.)

A.5.2.2.2.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the SRVs based on the following assumptions.

Table A.5.2-2 lists the systems that could initiate (in addition to the SRVs) during the design basis overpressure event.

Operating Conditions

The Overpressure event is performed using rated power (3983 MWt) and dome pressure (1020 psig) under the TRACG-AOO methodology. A pressure adder is applied to the cycle-specific result as described in References 4, 5 and 6.

A transient analysis study has been performed for a typical BWR to investigate the effects of increasing the initial reactor pressure on the peak transient vessel pressure. Two models, one from the REDY and one from the ODYN codes, were used in the study. The model in the REDY code is more conservative than that in the ODYN code. The conclusion, even for the more conservative model, was that increasing the initial operating pressure up to the high-pressure scram setpoint results in an increase of peak system pressure of less than half the initial pressure increase for the overpressure design transient (i.e., all MSIV closure with indirect high neutron flux scram). The same general trend is expected to exist for Unit 2. There is a significant margin (greater than 50 psi with two SRVs OOS by comparing the peak vessel pressure with the ASME Code upset limit of 1,375 psig) for Unit 2.

Transients

The overpressure protection system must accommodate the most severe pressurization event described in Section S.3 of GESTAR II[1], which is the MSIV closure with flux scram, based on the installed SRV capacity (with the lowest two valves assumed to be OOS). The SRVs include two modes of operation; the safety mode, which is spring actuated and the relief mode, which is solenoid actuated. The overpressure protection analyses conservatively take credit for only the spring mode of operation.

Safety/Relief Valve Transient Analysis Specification

1. Valve groups: Spring-action safety mode - 5 groups

2. Spring pressure setpoint (maximum safety limit) and number of valves per group:

   Group 1: 1,200 psig2 SRVs
   Group 2: 1,210 psig4 SRVs
   Group 3: 1,221 psig4 SRVs
   Group 4: 1,231 psig4 SRVs
   Group 5: 1,241 psig4 SRVs

The assumed spring setpoints in the analysis given above are about 3 percent above the actual nominal setpoints to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Conservative SRV response characteristics are also assumed.

<u>Safety/Relief Valve Capacity</u>

Sizing of the SRV capacity was based on establishing an adequate margin from the peak vessel pressure to the vessel code upset limit (1,375 psig) in response to the reference transients.

Reference 2 provides sufficient information and documentation to show compliance with all requirements of Article NB-7000 of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 1, 1971 Edition, with Addenda to and including Winter 1972, in the area of overpressure protection design of the Unit 2 nuclear pressure vessel and other RCPB components. The effects of valve capacity on the pressure transients are shown also in Reference 2.

Additional analyses are performed to account for the reload core, the operating options and equipment OOS conditions given in Figure A.0-1.

The overpressure protection analysis also includes the simulation of anticipated transient without scram (ATWS) RPT on high reactor pressure.

A.5.2.2.2.3 Evaluation of Results

<u>Safety/Relief Valve Capacity</u>

The required SRV capacity is determined by analyzing the pressure rise from a MSIV closure with flux scram transient as documented in GESTAR II[1]. Adequacy of the SRV capacity has been reconfirmed for current fuel cycle operation (Reference 3). The Unit 2 Technical Specifications allow two SRVs to be OOS. This allowance was conservatively assumed to apply also to the spring action of the valves and no credit was taken for the two valves with the lowest setpoint. Results of this analysis demonstrate that the peak pressure obtained in the reactor vessel is below

the ASME upset criteria of 1375 psig even with two SRVs OOS and assuming a 3-percent drift in the SRVs setpoints.

The adequacy of the SRVs has also been confirmed for operation with one MSIV OOS (closed) concurrent with two SRVs OOS (see Section A.15D).

Pressure Drop in Inlet and Discharge

Pressure drop on the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures. Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent backpressure on each SRV from exceeding 40 percent of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each SRV has its own separate discharge line.

A.5.2.3 Reactor Coolant Pressure Boundary Materials

No change has been made to this subsection as a result of the reload.

A.5.2.4 In-service Inspection and Testing of Reactor Coolant Pressure Boundary

No change has been made to this subsection as a result of the reload.

A.5.2.5 Reactor Coolant Pressure Boundary and ECCS Leakage Detection System

No change has been made to this subsection as a result of the reload.

A.5.2.6 References

1. General Electric Co., General Electric Standard Application for Reactor Fuel, including United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision as specified in the Core Operating Limits Report).

2. General Electric Co. Design Report 22A7122, Overpressure Protection Report, Revision 2.

3. Global Nuclear Fuel Report, Supplemental Reload Licensing Report for Nine Mile Point Unit 2, as specified in the Core Operating Limits Report source documents.

4. Migration to TRACG04/PANAC11 from TRACG02/PANAC for TRACG AOO and ATWS Overpressure Transients, NEDE-32906P, Supplement 3-A, Revision 1, April 2010.

5. TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses, NEDE-32906P-A, Revision 3, September 2006.

6.

TABLE A.5.2-1

NUCLEAR SYSTEM SAFETY/RELIEF SETPOINTS

No. of Valves	Nameplate Nominal Spring Set Pressure (psig)	ASME Rated Capacity @ 103% of Nameplate Spring Set Pressure (lb/hr each)	Nominal Pressure Setpoint for Power-Actuated Mode (psig)
2	1,165	895,000	1,103
4	1,175	902,000	1,113
4	1,185	910,000	1,123
4	1,195	917,000	1,133
4	1,205	925,000	1,143

NOTE: Seven of the SRVs are used for the automatic depressurization function.

TABLE A.5.2-2

SYSTEMS THAT MAY INITIATE* DURING OVERPRESSURE EVENT

System	Initiating/Trip Signal(s)
RPS	Reactor trips with high APRM neutron flux
RCIC	ON with reactor low water level
	OFF with reactor high water level
HPCS	ON with reactor low water level
	OFF with reactor high water level
Recirculation	Transfer to low speed with reactor high pressure
	OFF with reactor low water level
RWCU	OFF with reactor low water level

* In addition to the SRVs.

A.6 ENGINEERED SAFETY FEATURES

Chapter 6 provides information on the engineered safety feature (ESF) systems and components that are designed to ensure that the LOCAs are mitigated and the radioactivity releases from these accidents are limited. The ECCS performance analyses resulting from the LOCA have been reexamined for the current cycle to show conformance with 10CFR50.46 acceptance criteria. LOCA consideration is one of the requirements for fuel design. The following is the information resulting from the reload analysis on Chapter 6.

The safety and plant performance design bases for the nuclear design are described in GESTAR II[1]. Specifically, the new fuel design meets all the fuel thermal, mechanical and LOCA design and analysis criteria using GE design and analysis codes approved for these applications.

Starting with the Cycle 15 mid-cycle implementation of MELLLA+, the SAFER/PRIME-LOCA application methodology was used for LOCA evaluation. For information on current ECCS-LOCA analysis using SAFER/PRIME-LOCA see References 2 and 3. Significant fuel parameters utilized in the analyses are listed in Table A.6-2.

The results of this evaluation for the current fuel cycle show that the plant ECCS will perform its function, meeting the 10CFR50.46 2200°F PCT and 17-percent maximum oxidation fraction acceptance criteria for all normal operating conditions, and with allowable equipment OOS. The summary of Unit 2 LOCA evaluation results.

A.6.1 References

1. Global Nuclear Fuel Report, Supplemental Reload Licensing Report for Nine Mile Point Unit 2, as specified in the Core Operating Limits Report source documents.

2. Nine Mile Point Nuclear Station Unit 2 MELLLA+: ECCS-LOCA SAFER/PRIME, 0000-0162-4214-R0, August 2013.

3. Nine Mile Point Unit 2 GNF2 ECCS-LOCA Evaluation, 002N4205, Revision 0, December 2015..

TABLE A.6-1

THIS TABLE HAS BEEN DELETED

TABLE A.6-2

FUEL PARAMETERS

LOCA ANALYSIS(1)(2)(3)

Fuel Type	GE 11	GE 14	GNF2
Fuel Bundle Geometry	9x9	10x10	10 x 10
Lattice	C	C	C
Number of Fuel Rods per Bundle	66 (Full Length) 8 (Part Length)	78 (Full Length) 14 (Part Length)	78 14
PLHGR (kW/ft)			
Appendix K Analysis	14.4 x 1.02	13.4 x 1.02	14.40 x 1.02
Nominal Analysis	13.8	12.8	13.75
MAPLHGR (kW/ft)	12.9	12.82	13.78
Worst-Case Pellet* Exposure for ECCS Evaluation (MWd/MTU)	14,600	16,000	14,600
Initial MCPR			
Appendix K	1.25 ÷ 1.02	1.25 ÷ 1.02	1.20 ÷ 1.02
Nominal	1.25 + 0.02	1.25 + 0.02	1.20 + 0.02

* Represents the limiting operating condition resulting in the maximum calculated PCT at any time in the fuel lifetime.

A.15.0 GENERAL

As in Chapter 15, Section A.15 examines the effects of the anticipated process disturbances and postulated component failures to determine their consequences, and to evaluate the capability built into the plant to control or accommodate such failures and events.

Events discussed in Chapter 15, Sections 15.3, 15.5, 15.6, 15.7 and 15.8, are not limiting events or are addressed generically in GESTAR II[1] and are not reanalyzed for each reload. GE evaluates the entire spectrum of events in order to establish the most limiting events in a meaningful manner. It is these events that are quantified in this section.

The scope of the events analyzed includes anticipated (unplanned, but expected) operational occurrences (e.g., loss of electrical load); off-design abnormal (unexpected or infrequent) transients that induce system disturbances, postulated accidents of low probability (e.g., the sudden loss of integrity of a major component); and, finally, hypothetical events of extremely low probability (e.g., an anticipated transient without the operation of the entire control rod drive [CRD] system).

A.15.0.1 Analytical Objective

The spectrum of postulated initiating events is divided into categories based upon the type of disturbance and the expected frequency of the initiating occurrence. The limiting events in each combination of category and frequency are quantitatively analyzed. The plant safety analysis evaluates the ability of the plant to operate within regulatory guidelines without undue risk to public health and safety.

For reload cycles, only events which define the operating limits in the previous cycle or are close to the limiting cases are reanalyzed. The unanalyzed events still remain in the licensing basis of the plant.

A.15.0.2 Analytical Categories

Transient and accident events are discussed in individual categories in Chapter 15, Section 15.0.2. Each event evaluated is assigned to one of the analytical categories listed below:

1. Decrease in core coolant temperature
2. Increase in reactor pressure
3. Decrease in reactor core coolant flow rate
4. Reactivity and power distribution anomalies
5. Increase in reactor coolant inventory
6. Decrease in reactor coolant inventory
7. Radioactive release from a subsystem or component
8. Anticipated transients without scram

A.15.0.3 Event Evaluation

A.15.0.3.1 Identification of Causes and Frequency Classification

Situations and causes which lead to the initiating event analyzed are described within the categories designated above. The frequency of occurrence of each event is summarized on the bases of available operating plant history for the transient event. Events for which inconclusive data exist are discussed separately within each event section.

Each initiating event within the major groups is assigned to one of three frequency groups defined in Chapter 15, Section 15.0.3.1.

A.15.0.3.2 Sequence of Events and System Operations

Each transient or accident is discussed and evaluated in terms of:

1. Step-by-step sequence of events from initiation to final stabilized condition.

2. Extent to which normally operating plant instrumentation and controls are assumed to function.

3. Extent to which plant and reactor protection systems (RPS) are required to function.

4. Credit taken for the functioning of normally operating plant systems.

5. Operation of engineered safety systems that is required.

6. Effect of a single failure or an Operator error on the event.

A.15.0.3.3 Core and System Performance

The analyses documented in this Appendix are for the reload cycle core used for the nuclear evaluations given in Section A.4.4.

A.15.0.3.3.1 Introduction

The models used to analyze the core and system performance during abnormal operational transients are given in Reference 1. An acceptable criterion was determined to be that ≥99.9 percent of the fuel rods in the core would not be expected to experience boiling transition[2]. This criterion is met by demonstrating that incidents of moderate frequency do not result in a MCPR less than the safety limit MCPR specified in the Technical Specifications.

Determination of the steady-state operating limit is accomplished as described in References 4, 5 and 6. TRACG best-estimate methodology was implemented in Cycle 15 for AOO events. Under TRACG-AOO, the OLMCPR is determined for the given AOO events, as in previous methods, however, TRACG uses more a more realistic 3-D hydrodynamic and 3-D neutron kinetics model. The cycle-specific event ΔCPR/ICPR is combined with the bias and uncertainty determined within the implementation report (Reference 4) as input into GESAM to calculate the transient OLMCPR. The resultant OLMCPR assures that <0.1% of the rods will be susceptible to boiling transition in the event of the worst case AOO transient.

The resultant operating limit MCPR values are given in the SRLR[3] and the COLR for the limiting transients, including the equipment OOS limiting transients.

The operating limit MCPR is the maximum value of the event MCPRs calculated from the transient analysis. Maintaining the MCPR operating limit at or above this operating limit assures that the safety limit MCPR is never violated.

A.15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

The limiting events analyzed used values for input parameters and initial conditions as specified in Table A.15.0-4. Analyses which assume data inputs different from these values are designated accordingly in the appropriate event discussion.

The dynamic parameters assumed in Section A.15 are more conservative than the normal operating values (e.g., the analytical limit is used for setpoints).

A.15.0.3.3.3 Initial Power/Flow Operating Constraints

The analysis basis for the most limiting transient safety analyses uses an adjusted simulation of licensed (100 percent) thermal power at increased core flow (105 percent). At least 102 percent of rated power is considered either directly as the initial condition or as a statistical adjustment to the output. This is the highest power operating condition of a bounded operating power/flow map which, in response to most classified operational transients, yields the minimum pressure and thermal margins of any operating point within the bounded map.

Certain special events are evaluated at other than the above-mentioned conditions. These conditions are discussed as they pertain to the appropriate event.

The power/flow map is illustrated in Chapter 15, Figure 15.0-2a.

A.15.0.3.3.4 Results

The results of analyzed transient events for the reload core are presented in the SRLR[3]. The calculated MCPR values for these events are used to generate the operating limit MCPR values that are contained in the COLR. These limits include consideration of the effects of certain special operating conditions involving equipment OOS (see Section A.0).

In order to address all of the identified transient events in the eight analytical categories (refer to Section A.15.0.2), the limiting transients identified in GESTAR II[1] were established based on the analysis of all events included in the original FSAR. Each was assigned to one of these categories. Most of the events result in fairly mild plant disturbances. Thus, only a few events are severe enough to be potentially limiting. Furthermore, although the most limiting event may differ from plant to plant and reload to reload, it is GE's experience that the most limiting transients always can be expected to come from the same selected group of transient events. Therefore, most of the events analyzed need not be reanalyzed or reassessed for plant-specific reload core licensing application. The selected group of generically limiting events (Reference 1) consists of:

1. Turbine generator trip without bypass

2. Loss of feedwater heating

3. Feedwater controller failure

4. Control rod withdrawal error

Subsequent anticipated operational occurrence analyses verified the results of the above sensitivities. Descriptions of the typical analyses performed for the above limiting events are discussed in the following subsections.

A.15.0.3.3.4.1 Analysis Uncertainties

Model uncertainties are documented in Reference 1.

A.15.0.3.4 Barrier Performance

This section primarily evaluates the performance of the RCPB and the containment system during transients and accidents. During transients that occur with no release of coolant to the containment, only RCPB performance is considered. If release to the containment occurs, as in the case of limiting faults, then challenges to the containment are evaluated as well.

A.15.0.3.5 Radiological Consequences

In this section, the consequences of radioactivity release during the three types of events are considered:

1. Incidents of moderate frequency (anticipated operational occurrences)

2. Infrequent incidents (abnormal operational occurrences)

3. Limiting faults (design basis accidents [DBA])

For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For nonlimiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

A.15.0.4 Nuclear Safety Operational Analysis (NSOA) Relationship

No change has been made to this section as a result of the reload.

A.15.0.5 Loss of Instrument Air

No change has been made to this section as a result of the reload.

A.15.0.6 Effect of Nonsafety-Grade Equipment

The performance of some structures, systems, and components that are not identified as safety related is assumed in transient analyses.

Table 15.0-5 lists the nonsafety-grade equipment that has been assumed in the mitigation of the consequences. The assumed performance of these nonsafety-related equipment is based on extensive failure rate data for equipment of similar design and quality requirements.

Among these nonsafety-grade systems, the failure of the Level 8 trip and the failure of the turbine bypass are the events that would affect ΔCPR. For the load rejection with bypass failure transient, the unavailability of the turbine bypass valves is already assumed. For the feedwater controller failure maximum demand, the assumed failure of the Level 8 trip would have negligible effect on the final CPR such that the analytical conclusion, with respect to the fuel integrity, as presented in Section A.15.1.2, is still valid. Should failure of the turbine bypass occur during the feedwater controller failure maximum demand, the ΔCPR could increase such that it could momentarily drop below the MCPR safety limit. No fuel damage would result for such a momentary occurrence. If the bypass is known to be OOS, the Unit 2 Technical Specifications and COLR establish a revised operating limit MCPR to preclude the possibility. The Unit 2 Technical Specifications and COLR also include appropriate provisions regarding availability, setpoints, and surveillance testing of the remaining equipment.

The MCPR of the rod withdrawal error (RWE) could be affected by a failure of the rod block monitor (RBM). However, the RBM is designed to be single-failure proof so that complete loss of function is highly unlikely. In the event of loss of electrical power supply, the RBM becomes inoperative and will initiate rod block to prevent any rod motion. Without rod motion, LHGR and MCPR margins are preserved and the fuel thermal margin is adequately assured for the RWE event.

The peak vessel pressures for the analysis of transients not taking credit for nonsafety-grade systems and components are bounded by the peak pressure limit of the overpressure protection analysis described in Chapter 5.

A.15.0.7 References

1. General Electric Co., General Electric Standard Application for Reactor Fuel, including United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision as specified in the Core Operating Limits Report).

2. General Electric Co., General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application, NEDO-10959 and NEDE-10958, November 1973.

3. Global Nuclear Fuel Report, Supplemental Reload Licensing Report for Nine Mile Point Unit 2, as specified in the Core Operating Limits Report source documents.

4. Nine Mile Point Nuclear Power Plant, Unit 2 TRACG Implementation for Reload Licensing Transient Analysis (T1309), 0000-0157-9895-R1, October 2013.

5. Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, NEDE-32906P, Supplement 3-A, Revision 1, April 2010.

6. TRACG Application for Anticipated Operational Occurrences

7. (AOO) Transient Analyses, NEDE-32906P-A, Revision 3, September 2006.

TABLE A.15.0-1
TABLE A.15.0-2
TABLE A.15.0-3

THESE TABLES HAVE BEEN DELETED

TABLE A.15.0-4

RELOAD CYCLE INPUT PARAMETERS AND INITIAL CONDITIONS
FOR TRANSIENT ANALYSIS

1.	Thermal power level, MWt	
	Warranted value	3,988
	Analysis value	3,988 (6)
2.	Steam flow, lb/hr	
	Warranted value	$17.64x10^{6}$
	Analysis value	$17.64x10^{6}$ (6)
3.	Core Flow, lb/hr (1)	$113.9x10^{6}$
	% Rated	105
4.	Feedwater flow rate, lb/hr	
	Warranted value	$17.6x10^{6}$
	Analysis value	$17.6x10^{6}$
5.	Feedwater temperature, °F	440.5 (2)
6.	Vessel dome pressure, psig	1020
7.	Vessel core pressure, psig	1036.3
8.	Turbine bypass capacity, % NBR	18.5
9.	Core coolant inlet enthalpy, Btu/lb	528.9
10.	Turbine inlet pressure, psig	976.0
11.	Fuel lattice	SRLR (Ref. 3)
12.	Example Core average gap conductance, Btu/sec-ft^{2}-°F	0.407
13.	Core bypass flow, %	14.8
14.	Required MCPR operating limit	SRLR (Ref. 3)

TABLE A.15.0-4 (Cont'd.)

15.	MCPR safety limit	SRLR (Ref. 3)
16.	Doppler coefficient (-)¢/°F	
	Nominal EOC	(3)
	Analysis data	(3)
17.	Void coefficient (-)¢/% rated voids	
	Nominal EOC	(3)
	Analysis data for power increase events	(3)
	Analysis data for power decrease events	(3)
18.	Core average rated void fraction, %	(3)
19.	Scram reactivity, $ K	
	Analysis data	(3)
20.	Control rod drive speed,	
	Position versus time	Figure 15.0-3
21.	SRV capacity, % NBR	91% @ 1145 psig
	Manufacturer	Dikkers
	Quantity installed	18
22.	Relief function delay, sec	0.4
23.	Relief function response	
	Time constant, sec	0.1
24.	Safety function delay, sec	0.0
25.	Safety function response stroke	
	Time, sec	0.3
26.	Setpoints for SRVs	
	Safety function (lowest), psig	1200
	Safety function (all others), psig	1210, 1221, 1231, 1241
	Relief function, psig	1121, 1131, 1141, 1151, 1161

TABLE A.15.0-4 (Cont'd.)

27. Number of valve groupings simulated	
Safety function, No.	5[4]
Relief function, No.	5[4]
28. High flux trip, % NBR	
Analysis setpoint	123
29. High pressure scram, psig	
Analysis setpoint	1,086
30. Vessel level trips, ft above bottom of separator skirt bottom	
Level 8 - (L8)	6.133
Level 4 - (L4)	3.75
Level 3 - (L3)	1.225
Level 2 - (L2)	-4.708
31. APRM simulated thermal power trip, % NBR	
Analysis setpoint	118
32. Recirculation pump trip delay, sec	0.199
33. Recirculation pump trip inertia time constant for analysis, sec[5]	4.0 to 6.0
34. Total steam line volume, ft^3	4036

(1) Nominal core flow is 108.5×10^6.

(2) For feedwater controller failure, 420.5°F is conservatively assumed.

(3) Values are calculated within the TRACG-AOO code for EOC condition.

(4) Deleted.

(5) The inertia time constant is defined by the expression:

$$t = \frac{2 \ \pi \ J_0 \ n}{g \ T_0}$$

TABLE A.15.0-4 (Cont'd.)

Where

t = Inertia time constant, sec
J_o = Pump motor inertia, lb-ft
n = Rated pump speed, rps
g = Gravitational constant, ft/sec
T_o = Pump shaft torque, lb-ft

(6) MCPR analysis applies a bias and uncertainty to the Delta CPR/ICPR transient result. Overpressure analysis applies an adder to peak vessel pressure. The basis for these biases and uncertainties are found in the NMP2 TRACG-AOO implementation report (References 3, 4 and 5). Other analyses such as LOCA are performed at >/= 102% of 3,988 MWt.

A.15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

Five transients were evaluated for Cycle 1 (Chapter 15, Section 15.1) under the decrease in reactor coolant temperature analytical category:

1. Loss of feedwater heating
2. Feedwater controller failure
3. Pressure regulator failure
4. Inadvertent SRV opening
5. Inadvertent residual heat removal (RHR) shutdown cooling operation.

Only the loss of feedwater heating and feedwater controller failure transients in this analytical category have been analyzed for the reload (per Reference 2). The following subsections describe how each of the five transients is affected by the reload.

A.15.1.1 Loss of Feedwater Heating

The loss of feedwater heating transient due to the steam extraction line to the heater being closed or condensate being bypassed around the heater results in the reactor vessel receiving cooler feedwater. This event has been conservatively estimated to incur a loss of up to 100°F of the feedwater heating capability of the plant and causes an increase in core inlet subcooling.

Detailed description of this transient is provided in Chapter 15, Section 15.1.1. The unheated feedwater starts to raise core power level, but the automatic flow control system automatically reduces core flow to maintain initial steam flow.

The frequency classification of this event is categorized as an infrequent incident. However, because of the lack of a sufficient frequency data base, this transient disturbance is analyzed as an incident of moderate frequency. The operation of the required systems is described in Chapter 15, Section 15.1.1.2, and remains the same.

The significant features of the mathematical model used to predict dynamic behavior of this transient are described in Chapter 15, Section 15.1.1.3.1. For the reload analysis the 3D BWR simulator described in Section 4.3.1.2.2 of GESTAR II is used, which does not explicitly credit the flow-biased simulated thermal power scram.

This transient is evaluated for each reload core to determine if it could potentially alter the previous cycle MCPR operating limit. If it does, the results are presented in the SRLR.

A.15.1.2 Feedwater Controller Failure, Maximum Demand

A.15.1.2.1 Identification of Causes and Frequency Classification

A.15.1.2.1.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

A.15.1.2.1.2 Frequency Classification

This event is considered to be an incident of moderate frequency.

A.15.1.2.2 Sequence of Events and Systems Operation

A.15.1.2.2.1 Sequence of Events

The excess feedwater flow increases the water level to the high-level setpoint, at which time the feedwater pumps and the main turbine are tripped. Scram, bypass opening, and RPT (to low speed) are initiated from the turbine trip. Changes in key variables for this event, including cases both with and without turbine bypass system operation, are shown in the SRLR[3].

The Operator should verify automatic functions and monitor all parameters, but no Operator actions are necessary for reactor protection.

A.15.1.2.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, and RPS. Important system operational actions for this event are the tripping of the main turbine and feedwater pumps on high water level; RPT, scram, and opening of the bypass from turbine trip; opening of the SRVs as needed following the turbine trip; and low water level initiation of the reactor core isolation cooling (RCIC) system and the high-pressure core spray (HPCS) system to maintain long-term water level control following tripping of feedwater pumps.

A.15.1.2.2.3 Effect of Single Failures and Operator Errors

The first sensed event to initiate corrective action to the transient is the vessel high water level (L8) trip. Multiple level sensors are used to sense and detect when the water level reaches the L8 setpoint. At this point in the logic, a single failure will not initiate or prevent a turbine trip signal.

Turbine trip signal transmission, however, is not built to single-failure criterion.

The result of a failure at this point would have the effect of delaying the pressurization "signature". However, high moisture levels entering the turbine would be detected by high levels in the turbine moisture separators, resulting in a trip of the unit. In addition, excessive moisture entering the turbine may cause vibration to the point where it too will trip the unit.

Scram signals from the turbine stop valve and control valve closures are designed so that a single failure will neither initiate nor impede a reactor scram trip initiation (see Appendix 15A for detailed discussion of this subject). The same is true for the turbine trip RPT which is initiated in parallel with the scram. However, continued operation with the RPT OOS is permitted (with increased operating license MCPR [OLMCPR]) based on the analysis included below.

A similar option to operate Unit 2 (with increased OLMCPR) is provided based on the analysis given below for the case with the turbine bypass system OOS. The RCIC and HPCS systems provide redundancy for long-term water level protection. Either system is designed to be capable of maintaining coverage of the core without any Operator action.

A.15.1.2.3 Core and System Performance

A.15.1.2.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer-simulated, analytical model of a generic direct-cycle BWR. This model is described in detail in NEDO-24154[1]. This computer model has been improved and verified through extensive comparison of its predicted results with actual BWR test data.

The nonlinear computer-simulated analytical model is designed to predict associated transient behavior of this reactor. Some of the significant features of the model are:

1. An integrated one-dimensional core model which includes a detailed description of hydraulic feedback effects, axial power shape changes, and reactivity feedbacks, is assumed.

2. The fuel is represented by an average cylindrical fuel and cladding model for each axial location in the core.

3. The steam lines are modeled by eight pressure nodes incorporating mass and momentum balances that predict any wave phenomenon present in the steam line during pressurization transient.

4. The core average axial water density and pressure distribution is calculated using a single channel to represent the heated active flow and a single channel to represent bypass flow. A mode, representing liquid and vapor mass and energy conservation and mixture

momentum conservation, is used to describe the thermal-hydraulic behavior. Changes in the flow split between the bypasses and active channel flow are accounted for during transient events.

5. Principal controller functions such as feedwater flow, recirculation flow, reactor water level, pressure, and load demand are represented together with their dominant nonlinear characteristics.

6. The ability to simulate necessary RPS functions is provided.

7. The control systems and RPS models are, for the most part, identical to those employed in the point reactor model[(1)], and used in analysis for other transients.

See References 5 and 6 for detailed discussion on TRACG model.

A.15.1.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table A.15.0-4 for TRACG event analysis.

EOC nuclear characteristics are assumed. The SRV action is conservatively assumed to occur with higher than nominal setpoints. The transient is simulated by assuming an upper limit failure in the feedwater system, such that the maximum runout feedwater flow occurs at a pressure of 1,010 psig. The assumed feedwater runout capacity function has been validated for EPU (Reference 7).

The temperature of the feedwater flow is conservatively assumed to be 420.5°F (rather than nominal 440.5°F) to provide additional conservatism for change in CPR. Credit is only taken for 16 of 18 installed SRVs where the two valves with the lowest setpoints are assumed to be OOS.

A.15.1.2.3.3 Results

<u>Base Case</u>

The simulated feedwater controller transient is shown in the SRLR[(3)]. The high water level turbine trip and feedwater pump trip are initiated. Scram occurs simultaneously and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs.

The turbine bypass system opens to limit the peak vessel bottom pressure such that the nuclear system process barrier pressure limit is not compromised. If necessary, the SRVs would open to relieve vessel pressure; however, no credit for the relief mode of operation was taken in this analysis. The bypass valves

subsequently close to reestablish pressure control in the vessel during shutdown. Events caused by low water level trips, including initiation of HPCS and RCIC core cooling system actions, are not included in the simulation. Should these events occur, they will follow some time after the primary concerns of fuel thermal margin and overpressure effects have occurred. These events are less severe than those already experienced by the system.

<u>Results with Equipment Out of Service</u>

Analyses of the feedwater controller failure event were also performed considering the impact of selected equipment OOS, as permitted by the Unit 2 Technical Specifications. As noted in Section A.15.1.2.3.2, all feedwater controller failure events are analyzed with two SRVs assumed to be OOS (the two lowest-set valves are conservatively neglected). This OOS option is, therefore, demonstrated to be acceptable concurrently with the other OOS options discussed in the following sections.

The response of Unit 2 to this event with the turbine bypass system OOS is summarized in the SRLR[3]. Conservative model inputs (including EOC characteristics) were used which bound the licensing basis case. The initial part of the case is virtually identical to the base case. However, the response when the high level turbine trip occurs is more severe since bypass opening does not occur. Peak neutron flux is higher causing higher peak fuel surface heat flux. The unadjusted change in CPR is larger than for the base case, requiring modified MCPR operating limits (specified in the COLR) when the bypass is OOS. Peak vessel pressures are also higher, opening some of the SRVs, but the vessel bottom peak pressure is still well within the ASME limit of 1375 psig. The final portion of this event will be like an isolation with loss of feedwater. Water level will be maintained by RCIC and/or HPCS.

Another case was previously evaluated for operation with the turbine generator RPT OOS. The load rejection without bypass case shown in the SRLR[3] is more limiting for this OOS option. Peak pressures are controlled by actuation of the SRVs.

The feedwater controller failure event was also analyzed for operation of the unit with one steam line isolated (one MSIV OOS, closed) and for single recirculation loop operation. Appendices 15D and 15B, respectively, address these options for Cycle 2. The results of the analysis were found to be milder than the load rejection event (Section A.15.2.2).

A.15.1.2.3.4 Consideration of Uncertainties

Although the TRACG methodology used in the analysis uses nominal initial power, reactivity characteristics, and scram speed, the MCPR evaluation includes statistical allowances for uncertainties in simulation and plant parameters (including power and scram

speed). See References 5 and 6 for additional details on combination of uncertainties within the TRACG model.

A.15.1.2.4 Barrier Performance

As noted previously, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity and function as designed. If required, special MCPR operating limits are specified to maintain the thermal margin if the turbine bypass, turbine generator RPT or one MSIV are OOS (as shown in the SRLR[3] and the COLR).

A.15.1.2.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant Operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established Technical Specifications; therefore, this event, at worst, would only result in a small increase in the yearly integrated exposure level.

A.15.1.3 Pressure Regulator Failure, Open

Detailed description of this transient is provided in Chapter 15, Section 15.1.3.

This event was not identified as one of the transients that is significantly affected by the reload as stated in GESTAR II[2]. It has been shown in the Cycle 1 analysis that a high water level trip occurs quickly, and low pressure isolation valve closure stops the vessel depressurization. Reactor power decreases, responds to the turbine trip (less than that base event), and then decreases due to the scram and continued depressurization. The SRV operates intermittently to relieve the pressure rise from decay heat. No reduction in thermal limits occurs. For the reload, this event will behave similarly to the Cycle 1 analysis. It is bounded by other transients in this analytical category. No new analysis has been performed.

A.15.1.4 Inadvertent Safety/Relief Valve Opening

Detailed description of this transient is provided in Chapter 15, Section 15.1.4.

This event was not identified as one of the transients that is affected by reload in GESTAR II[2]. It has been shown in the

Cycle 1 analysis that the response of this event is very mild from the viewpoint of fuel protection. Thermal margins are unchanged and MCPR response remains essentially unchanged. For the reload, this event will behave similarly to the Cycle 1 analysis. It is bounded by other transients under this analytical category. No new analysis has been performed.

A.15.1.5 Spectrum of Steam System Piping Failures Inside and Outside of Containment in a PWR

No change has been made to this section as a result of the reload.

A.15.1.6 Inadvertent RHR Shutdown Cooling Operation

Detailed description of this transient is provided in Chapter 15, Section 15.1.6.

This event was not identified as one of the transients that is affected by the reload in GESTAR II[2]. No Cycle 1 analysis was performed for this transient since it was shown that if the Operator did not act to control the power level, a high neutron flux reactor scram would terminate the transient without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

A.15.1.7 References

1. Qualification of the One-Dimensional Core Transient Model for BWR, October 1978 (NEDO-24154).

2. General Electric Co., General Electric Standard Application for Reactor Fuel, including United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision as specified in the Core Operating Limits Report).

3. Global Nuclear Fuel Report, Supplemental Reload Licensing Report for Nine Mile Point Unit 2, as specified in the Core Operating Limits Report source documents.

4. Nine Mile Point Nuclear Plant, Unit 2 TRACG Implementation for Reload Licensing Transient Analysis (T1309), 0000-0157-9895-R1, October 2013.

5. Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, NEDE-32906P, Supplement 3-A, Revision 1, April 2010.

6. TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses, NEDE-32906P-A, Revision 3, September 2006.

7. Complication of Fuel Analyses Parameters, S0OPL001 (Latest Revision.

A.15.2 INCREASE IN REACTOR PRESSURE

Eight transients were evaluated for Cycle 1 (Chapter 15, Section 15.2) under the increase in reactor pressure analytical category:

1. Pressure regulator failure - closed
2. Generator load rejection
3. Turbine trip
4. MSIV closures
5. Loss of condenser vacuum
6. Loss of ac power
7. Loss of feedwater flow
8. Failure of RHR shutdown cooling

Only the generator load rejection transient in this analytical category has been analyzed for the reload (per Reference 1). The following subsections describe how each of the eight transients is affected by the reload.

A.15.2.1 Pressure Regulator Failure - Closed

Detailed description of this transient is provided in Chapter 15, Section 15.2.1.

This event was not identified as one of the transients that is affected by the reload for BWR/5 units in GESTAR II[1]. It has been shown in the Cycle 1 analysis that this event is bounded by the limiting transient in this analytical category. For the reload, this event will behave similarly to the Cycle 1 analysis; therefore, no new analysis has been performed.

A.15.2.2 Generator Load Rejection

A.15.2.2.1 Identification of Causes and Frequency Classification

A.15.2.2.1.1 Identification of Causes

Fast closure of the turbine control valves (TCV) is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The TCVs are required to close as rapidly as possible to prevent excessive overspeed of the turbine generator. Fast closure of the main TCVs causes a reactor scram and sudden reduction in steam flow which results in an increase in system pressure.

A.15.2.2.1.2 Frequency Classification

<u>Generator Load Rejection - Base Event</u>

This event is categorized as an incident of moderate frequency.

<u>Generator Load Rejection with Bypass Failure</u>

This event is categorized as an infrequent incident with the following characteristics:

Frequency	0.0036/plant year
Mean time between events (MTBE)	278 yr

Frequency Basis Thorough searches of domestic plant operating records have revealed three instances of bypass failure during 628 bypass system operations. This gives a probability of bypass failure of 0.0048. Combining the actual frequency of a generator load rejection with the failure rate of the bypass yields a frequency of a generator load rejection with bypass failure of 0.0036 event/plant year, or a MTBE of 278 yr. Although this event is classified as infrequent, it has been considered in the evaluation of operating CPR limits.

A.15.2.2.2 Sequence of Events and System Operation

A.15.2.2.2.1 Sequence of Events

Generator Load Rejection - With Bypass

A loss of generator electrical load from high power conditions produces the typical sequence of events listed in Table 15.2-1.

Generator Load Rejection with Bypass Failure

A loss of generator electrical load at high power with bypass failure produces a sequence of events that is similar to the case with bypass. The transient analysis results are shown in the SRLR[2] for both the base analysis and for the case with one MSIV OOS.

Identification of Operator Actions

The Operator should verify automatic functions and monitor all parameters.

No Operator actions are required for fuel protection. For an event with bypass failure, manual suppression pool cooling may be required.

A.15.2.2.2.2 System Operation

Generator Load Rejection - With Bypass

To simulate properly the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, and RPS unless stated otherwise.

TCV fast closure initiates a scram trip signal for power levels greater than 26 percent NBR. In addition, RPT is initiated. Both these trip signals satisfy single-failure criterion and

credit is taken for these protection features. The turbine bypass valves are also normally opened quickly when the turbine trip occurs.

The pressure relief system, which operates the relief valves independently when system pressure exceeds relief valve instrumentation setpoints, is assumed to function normally during the time period analyzed (two valves are assumed to be OOS).

Generator Load Rejection with Bypass Failure

This case is the same as generator load rejection with bypass except that failure of the main turbine bypass valves is assumed for the entire transient. In addition, the reload analysis did not credit the operation of the SRVs in the relief mode. This limiting case is also analyzed with only 16 of the 18 SRVs in service assuming that the two lowest setpoint valves are OOS.

A.15.2.2.2.3 The Effect of Single Failures and Operator Errors

Mitigation of the pressure-induced power increase, the basic nature of this transient, is accomplished by the RPS functions. TCV trip scram and RPT mitigate the effects of the power increase; they are designed to satisfy the single-failure criterion. An evaluation of the most limiting single failure (i.e., failure of the bypass system) was considered in this event. Details of single-failure analysis can be found in Appendix 15A.

A.15.2.2.3 Core and System Performance

A.15.2.2.3.1 Mathematical Model

The computer model described in Section A.15.1.2.3.1 was used to simulate this event.

A.15.2.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table A.15.0-4. EOC nuclear characteristics are used for the limiting cases (Reference 2). The cases are simulated at 105 percent core flow, the most limiting operating point due to the less effective scram reactivity characteristics at this point.

The turbine electrohydraulic control (EHC) system detects load rejection before a measurable speed change takes place and begins fast control valve closure.

The closure characteristics of the TCVs are modeled as they actually operate, in partial arc mode (beginning in cycle 14).

This operating mode maintains three of the valves at full open position while the fourth valve is throttled at 47% ± 7%. The

valves have a full sroke closure time, from fully open to fully closed, of 0.15 sec.

Auxiliary power is normally independent of any turbine generator overspeed effects and is continuously supplied at rated frequency, assuming automatic fast transfer (within 0.1 sec) to offsite power supplies. For the purpose of base case analysis, the automatic transfer is neglected and the recirculation pumps are assumed to remain powered from the main generator. Thus, they experience a slight increase in speed with the turbine generator overspeed until tripped by the RPT system (within 0.2 sec). Continued operation of Unit 2 is permitted with the turbine generator trip RPT OOS as analyzed below. In this case, the overspeed effect is more significant.

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation. Operation with the bypass OOS is covered by the cases analyzed with bypass failure. For all load rejection cases, credit is only taken for 16 of the 18 SRVs (the two lowest setpoint valves are assumed to be OOS).

Events caused by low water level trips, including initiation of HPCS and RCIC core cooling system functions (if they are needed), are not included in the simulation. If these events occur, they will follow sometime after the primary concerns of fuel margin and overpressure effects have passed and are expected to result in effects less severe than those already experienced by the reactor system.

The generator load rejection with bypass event is also analyzed for single recirculation loop operation (with and without RPT in service) and for operation with one MSIV OOS, as discussed in Appendices 15B and 15D, respectively. Results of these analyses for reload cores are summarized in Sections A.15B and A.15D.

A.15.2.2.3.3 Results

<u>Generator Load Rejection - With Bypass</u>

Figure 15.2-1 shows the original Cycle 1 results of the generator trip from 104.3 percent of original rated power (3,323 MWt). This case is not reanalyzed for each reload as it is not a limiting transient (per Reference 1).

<u>Generator Load Rejection with Bypass Failure</u>

The SRLR[2] shows the reload analysis for the base event with bypass failure using TRACG methods. This event is one of the transients most likely to limit operation because of MCPR considerations. The MCPR for this event is reported in the SRLR[2] and may provide the basis for the OLMCPR values presented in the COLR.

Generator Load Rejection with Bypass Failure and Turbine Generator Trip RPT Out of Service

Analysis of the generator load rejection event with bypass failure was also performed for Unit 2 operation with the turbine generator trip RPT OOS. As in all the load rejection cases, the analysis also assumed that only 16 of the 18 installed SRVs are in service, so it demonstrates that operation with these features concurrently OOS (turbine generator RPT and 2 SRVs) is acceptable.

Nuclear characteristics and scram inputs assumed in the analysis conservatively bound EOC conditions. The recirculation pumps are eventually tripped (transferred to low speed) on high pressure but that occurs after the maximum change in CPR has already occurred. For the purpose of conservative analysis, no credit was taken for the automatic fast transfer of the pump power supply to the outside grid; the pumps are assumed to remain powered from the main generator and thus increase in speed with the turbine generator overspeed. Peak vessel pressure is essentially unchanged from the base event as it is limited by the SRVs (16 assumed to be in service). The resulting MCPR operating limits for the reload cycle operation with the RPT OOS are given in the SRLR.

A.15.2.2.3.4 Consideration of Uncertainties

The TRACG methodology used in the analysis of the generator load rejection cases with bypass failure uses nominal initial power EOC reactivity characteristics and scram speed. However, the MCPR evaluation includes statistical allowances for uncertainties in simulation and plant parameters (including power and scram speed).

Other systems utilized for protection in this event were assumed to have the most conservative allowable response. Anticipated plant behavior is therefore expected to reduce the actual severity of the transient.

The analysis assumed SRV setpoints that are 3 percent above the actual nominal setpoints. In addition, the peak pressure will be bounded by limiting overpressure transient analyzed in Section A.5.

A.15.2.2.4 Barrier Performance

Generator Load Rejection - With Bypass

Peak pressure remains within normal operating range and no threat to the barrier exists.

Generator Load Rejection with Bypass Failure

The peak nuclear system pressure at the bottom of the reactor vessel is well below the ASME upset transient pressure limit of

1375 psig for both the base event and the case with the one MSIV also OOS, as documented in the SRLR[2]. Fuel barrier performance is assured by the OLMCPRs that are established.

A.15.2.2.5 Radiological Consequences

While the consequence of this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation as described in Section 15.1.2.5.

A.15.2.3 Turbine Trip

The detailed description of this transient is provided in Chapter 15, Section 15.2.3.

These events (turbine trip with and without bypass) were not identified as transients that are affected by the fuel reload in GESTAR II[1] (for the standard analysis). It has been shown in the Cycle 1 analysis that transient responses of both these events are similar to but bounded by the generator load rejection event. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report.

The turbine trip event could potentially be limiting for the equipment OOS options identified on Figure A.0-1. For the reload cycle, analysis of the turbine trip without bypass event was performed for Unit 2 operation with the EOC-RPT OOS. The resulting OLMCPR for the reload cycle operation with the RPT-OOS are less limiting than those calculated for the load rejection with bypass failure event (Section A.15.2.2).

A.15.2.4 Main Steam Isolation Valve Closures

The detailed description of this transient is provided in Chapter 15, Section 15.2.4.

These events (closure of one or more MSIVs) were not identified as transients that are affected by the fuel reload in GESTAR II[1]. It has been shown in the Cycle 1 analysis that transient responses of both of these events are bounded by the generator load rejection event. For the reload, these events will behave similarly to the Cycle 1 analysis; therefore, no new analysis has been performed.

A.15.2.5 Loss of Condenser Vacuum

Detailed description of this transient is provided in Chapter 15, Section 15.2.5.

This event was not identified as one of the transients that is affected by the reload in GESTAR II[1]. It has been shown in the Cycle 1 analysis that the transient MCPR response of this event

is bounded by the generator load rejection and turbine trip events. For the reload, this event will behave similarly to the Cycle 1 analysis; therefore, no new analysis has been performed.

A.15.2.6 Loss of Ac Power

Detailed description of the transient is provided in Chapter 15, Section 15.2.6.

Loss of ac power events were not identified as transients that are affected by the reload in GESTAR II[1]. It has been shown in the Cycle 1 analyses that the initial transient for the loss of normal and preferred Station service transformers is similar to the RPT transient. The transient response for the loss of all grid connections takes on the characteristic response of the standard full load rejection event. For the reload, these events will behave similarly to Cycle 1. They are bounded by other transients under this same analytical category. No new analysis has been performed.

A.15.2.7 Loss of Feedwater Flow

Detailed description of this transient is provided in Chapter 15, Section 15.2.7.

This event was not identified as one of the transients that is affected by the reload in GESTAR II[1]. It has been shown in the Cycle 1 analysis that there is no increase in heat flux; therefore, thermal margins are not threatened. Reactor inventory is maintained by RCIC and/or HPCS. For the reload, this event will behave similarly to Cycle 1 analysis. It is bounded by other transients under the same analytical category. No new analysis has been performed.

A.15.2.8 Feedwater Line Break

No change has been made to this subsection as a result of the reload.

A.15.2.9 Failure of RHR Shutdown Cooling

Detailed description of this transient is provided in Chapter 15, Section 15.2.9.

This event was not identified as one of the transients that is affected by the reload in GESTAR II[1]. The Cycle 1 analysis demonstrated the capability to safely transfer fission product decay heat and other residual heat from the reactor core at such a rate that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded. For the reload, this event will behave similarly to the Cycle 1 analysis. It is bounded by other transients under the same analytical category. No new analysis has been performed.

A.15.2.10 References

1. General Electric Co., General Electric Standard Application for Reactor Fuel, including United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision as specified in the Core Operating Limits Report).

2. Global Nuclear Fuel Report, Supplemental Reload Licensing Report for Nine Mile Point Unit 2, as specified in the Core Operating Limits Report source documents.

3. Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, NEDE-32906P, Supplement 3-A, Revision 1, April 2010.

4. TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses, NEDE-32906P-A, Revision 3, September 2006.

A.15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Seven transients were evaluated for Cycle 1 (Chapter 15, Section 15.4) under this analytical category:

1. RWE - low power
2. RWE - at power
3. Control rod maloperation (system malfunction or Operator error)
4. Abnormal startup of idle recirculation pump
5. Recirculation flow control failure with increasing flow
6. Misplaced bundle accident
7. Control rod drop accident (CRDA)

Only the most limiting transient of moderate frequency, the control RWE at power, has been analyzed under this analytical category for the reload (per Reference 1). The following subsections describe how each of the seven transients is affected by the reload.

A.15.4.1 Rod Withdrawal Error - Low Power

The detailed descriptions of these postulated transients are provided in Chapter 15, Section 15.4.1.

These events were not identified as transients that are affected by the reload in GESTAR II[1]. It has been shown in the Cycle 1 analysis that the events will not result in an unacceptable positive reactivity insertion. For the reload, the events will behave similarly to the Cycle 1 analysis. They are bounded by

other analyzed transients under the same category. No new analysis has been performed.

A.15.4.2 Rod Withdrawal Error at Power

The control RWE at power condition has been identified in GESTAR II[1] as one of the more likely events to limit operation from MCPR consideration; therefore, it was analyzed for the reload (Reference 2).

A.15.4.2.1 Identification of Causes and Frequency Classification

A.15.4.2.1.1 Identification of Causes

While operating in the power range in a normal mode of operation, the Reactor Operator (RO) makes a procedural error and withdraws the maximum-worth control rod until the RBM system inhibits further withdrawal.

A.15.4.2.1.2 Frequency Classification

The probability of this event is considered low enough to warrant its categorization as an infrequent incident. However, because of the lack of sufficient frequency data base, this transient disturbance is analyzed as an incident of moderate frequency until its frequency can be further evaluated and justified.

A.15.4.2.2 Sequence of Events and Systems Operation

A.15.4.2.2.1 Sequence of Events

For this transient, no Operator actions are required. This event results in a local power increase due to a reactivity rise from the decrease in control rod poison material. The RBM system blocks the further withdrawal of the error control rod and terminates the event. A typical sequence of events is presented in Table 15.4-1.

A.15.4.2.2.2 Systems Operation

This event is localized to a small portion of the core. Although reactor control and instrumentation is assumed to function normally, credit is taken only for the RBM system. A discussion of the event follows.

While operating in the power range in a normal mode of operation (except as noted in Section A.15.4.2.3.2), the RO makes a procedural error and withdraws the maximum worth control rod until the RBM system inhibits further withdrawal.

Under most normal operating conditions, no Operator action is required since the transient that would occur is very mild. Should the peak linear power design limits be exceeded, the nearest local power range monitor (LPRM) would detect this

phenomenon and sound an alarm. The Operator must acknowledge this alarm and take appropriate action to rectify the situation.

If the RWE were severe enough, the RBM system would sound alarms, at which time the Operator would acknowledge the alarms and take corrective action. Even for extremely severe conditions (i.e., those involving highly abnormal control rod patterns or operating conditions in which it is assumed that the Operator ignores all alarms and warnings and continues to withdraw the control rod), the RBM system would block further withdrawal of the control rod before the fuel reached the point of boiling transition or the 1 percent plastic strain limit imposed on the clad.

A.15.4.2.2.3 Effect of Single Failures and Operator Errors

No change has been made to this subsection as a result of the reload.

A.15.4.2.3 Core and System Performance

A.15.4.2.3.1 Mathematical Model

No change has been made to this subsection as a result of the reload.

A.15.4.2.3.2 Input Parameters and Initial Conditions

The number of possible RWE transients is extremely large due to the number of control rods and the wide range of exposures and power levels. In order to encompass all possible RWEs that could conceivably occur, a limiting analysis is defined that provides a conservative assessment of the consequences.

The conservative assumptions are:

1. The error is a continuous withdrawal of the maximum worth rod at its maximum drive speed.

2. The core is operating at rated conditions.

3. The reactor is in its most reactive state and devoid of all xenon. This ensures that the amount of excess reactivity that must be controlled by the movable control rods is maximum.

4. The Operator has fully inserted the maximum worth rod prior to its removal and selected the remaining control rod pattern in such a way as to approach thermal limits in the fuel bundles in the vicinity of the rod to be withdrawn. It should be emphasized that this control rod configuration would be highly abnormal and could only be achieved by deliberate Operator action or by numerous Operator errors.

5. The Operator has ignored all warnings during the transient.

6. Of the four LPRM strings nearest to the control rod being withdrawn, the two highest-reading LPRM strings have failed during the transient.

7. One of the two instrument channels is bypassed and OOS. The A and C LPRM chambers input to one channel while the B and D chambers input to the other. The channel with the greatest response is bypassed.

These conservative assumptions provide a high degree of assurance that the transient, as analyzed, bounds all RWEs that could possibly occur. The analyses have been performed, unless otherwise noted, with the plant conditions in Table A.15.0-4. Other parameters used in the analysis of this event are presented in the SRLR[2].

Rod Block Monitor System Operation

The RBM system minimizes the consequences of a RWE by blocking the motion of the control rod before the safety limits are exceeded.

The RBM has three adjustable trip levels (rod withdrawal permissive removed). The highest trip level is set so that the safety limit is not exceeded and is specified in the COLR. The lower two trip levels are intended to provide a warning to the Operator. The Operator may encounter up to three trip points depending on the starting power of a given control rod withdrawal. The lower two points may be passed up (reset) by manual operation of a push button. The reset permissive is actuated (and indicated by a light) when the RBM reaches 2 percent power less than the trip point. The Operator should then assess the local power and either reset or select a new rod. The highest (power) trip point may not be reset.

A.15.4.2.3.3 Results

The consequences of this transient are relatively mild and neither localized nor gross occurrence of boiling transition occur. Evaluation of the 1-percent plastic strain limit on the cladding and of the MLHGR is performed on a generic basis for each fuel type (Reference 1).

The results of the RWE analysis, including the assumed RBM setpoint, are tabulated in the SRLR[2]. This event is one of the transients most likely to limit operation because of MCPR considerations. The ΔCPR for this event is reported in the SRLR[2] and may establish the basis for the OLMCPR values specified in the COLR.

A.15.4.2.3.4 Considerations of Uncertainties

The conservative assumptions assuring that this event has been conservatively analyzed are discussed in Section A.15.4.2.3.2.

A.15.4.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since this is a localized event with very little change in the gross core characteristics. Typically, an increase in total core power is less than 5 percent and the changes in pressure are negligible.

A.15.4.2.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel.

A.15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)

The control rod malfunction results in a positive reactivity insertion into the core. It is stated in Chapter 15, Section 15.4.3, that this event is included in the evaluation cited in the two control RWE transients (low power and at power condition). This conclusion remains valid for the reload.

A.15.4.4 Abnormal Startup of Idle Recirculation Pump

Detailed description of this transient is provided in Chapter 15, Section 15.4.4.

This event was not identified as one of the transients that is affected by the reload in GESTAR II[1]. It has been shown in the Cycle 1 analysis that the transient response of this event is not limiting and it is bounded by the RWE. For the reload, this event will behave similarly to the Cycle 1 analysis; therefore, no new analysis has been performed.

A.15.4.5 Recirculation Flow Control Failure With Increasing Flow

Detailed description of this transient is provided in Chapter 15, Section 15.4.5.

This event was not identified as one of the transients that is affected by the reload in GESTAR II[1]. It was shown in the Cycle 1 analysis that the transient response of this event is bounded by the RWE. MCPR remains above the safety limit. For the reload, this event will behave similarly to the Cycle 1 analysis; therefore, no new analysis has been performed.

A.15.4.6 Chemical and Volume Control System Malfunctions

Not applicable to BWRs.

A.15.4.7 Misplaced Bundle Accident

Detailed description of this transient is provided in Chapter 15, Section 15.4.7.

This event was not identified as one of the transients that is affected by the reload in GESTAR II[1]. As approved for GESTAR II, the analysis of the mislocated bundle accident is performed only for initial cores and not performed for reload cores. Analysis of the mislocated bundle accident is performed for reload cores where the resultant CPR response may establish the operating limit MCPR, and the misoriented bundle accident is evaluated on a cycle-specific basis.

A.15.4.8 Spectrum of Rod Ejection Accidents

Not applicable to BWRs.

A.15.4.9 Control Rod Drop Accident

Detailed description of this transient is provided in Chapter 15, Section 15.4.9.

This event was not identified as one of the events to be analyzed by the reload in GESTAR II[1]. It was shown in the Cycle 1 analysis that the peak fuel enthalpy is within the design criterion of 280 cal/gm and that the radiological consequences are acceptable. GESTAR II[1] documents the generic acceptability of the banked position withdrawal sequence (BPWS) cores without cycle-specific reanalysis.

BPWS plants normally operate with the rod worth minimizer (RWM) alone, and for the first 50 percent of the rods, the effective withdrawal is in the form of (stepped) defined bank patterns. Beyond 50 percent, it is also in the form of stepped bank withdrawal.

The Technical Specifications require administration of this withdrawal sequence through the RWM and plant operating procedures.

Consequently, adherence to BPWS reduces control rod worth such that the postulated CRDA is well under the design criterion of 280 cal/gm.

A.15.4.10 References

1. General Electric Co., General Electric Standard Application for Reactor Fuel, GESTAR II, NEDE-24011-P-A, and United States Supplement GESTAR II, NEDE-24011-P-A-US, (latest approved revision as specified in the Core Operating Limits Report).

2. Global Nuclear Fuel Report, Supplemental Reload Licensing Report for Nine Mile Point Unit 2, as specified in the Core Operating Limits Report source documents.

A.15B RECIRCULATION SYSTEM SINGLE-LOOP OPERATION

Recirculation system single-loop operation (SLO) is discussed in detail in Appendix 15B. A summary of the affected transient analyses results is provided in Tables 15B.3-4 and 15B.3-5. The results for Cycle 2 demonstrate large margins between the two-loop limits and SLO limits. Due to these large margins, it is concluded that SLO analyses will remain bounded by two-loop operation for future reload cycles. A reanalysis of equipment OOS for SLO in the current fuel cycle confirms that the conclusions presented in Appendix 15B for Cycle 2 remain applicable. The limiting MAPLHGR reduction factor and the cycle-specific safety limit MCPR for SLO is documented in the SRLR[1] and the COLR.

<u>References</u>

1. Global Nuclear Fuel Report, Supplemental Reload Licensing Report for Nine Mile Point Unit 2, as specified in the Core Operating Limits Report source documents.

SECTION A.15C
(TWO SAFETY/RELIEF VALVES OUT OF SERVICE)
HAS BEEN DELETED

(Section A.0 already indicates that two SRVs out of service is one of the operational options that is included in the standard reload analyses.)

A.15D ONE MAIN STEAM LINE ISOLATION VALVE OUT OF SERVICE

Operation with one MSIV OOS is discussed in detail in Appendix 15D. With the power limitation specified in Section 15D.1, the limiting MCPR and overpressure protection events, with all equipment in service, bound the MSIV OOS case.

References

1. Global Nuclear Fuel Report, Supplemental Reload Licensing Report for Nine Mile Point Unit 2, as specified in the Core Operating Limits Report source documents.