

NMP1L 0426  
Enclosure 1

NIAGARA MOHAWK POWER CORPORATION  
NINE MILE POINT NUCLEAR STATION UNIT 1

DETERMINATION OF  
REGULATORY GUIDE 1.97  
TYPE A VARIABLES

*Report Date: July 28, 1989*

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DETERMINATION OF REGULATORY GUIDE 1.9A TYPE A VARIABLES  
FOR  
NINE MILE POINT UNIT 1

Page 1

BACKGROUND

Revision 2 of US NRC Regulatory Guide (RG) 1.9A, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environmental Conditions During, and Following an Accident," Section C, Paragraph 1, defines "Type A Variables" as follows:

"Type A variables to be monitored shall provide the primary information to permit the control room operator to take the specified actions. Type A variables shall be monitored continuously and their safety function for design basis accident events."

In the context of this regulation, primary information is limited to information that is essential for the direct accomplishment of the safety function. It does not include those variables that are associated with contingency actions that may also be identified in written procedures.

Section 1.9A of Regulatory Guide 1.9A states that the determination of Type A Variables is plant-specific and will depend on the operations that the designer chooses for planned manual action."

DETERMINATION OF REGULATORY GUIDE 1.97 TYPE A VARIABLES  
FOR  
NINE MILE POINT UNIT 1

BACKGROUND

Revision 2 of US NRC Regulatory Guide (RG) 1.97. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Section C, Paragraph 1.1, defines Type A Variables as follows:

"Those variables to be monitored that provide the primary information required to permit the control room operators to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety function for design basis accident events."

In the context of this definition, primary information is limited to information that is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures.

Revision 2 of Regulatory Guide 1.97 further states that the determination of Type A Variables is plant-specific "and will depend on the operations that the designer chooses for planned manual action."

COMPARISON OF EVALUATION RESULTS

The results of the evaluation are presented in Table IV of the report. The results show that the evaluation of the system is a complex task and requires a thorough understanding of the system and its components. The results also show that the evaluation of the system is a continuous process and requires ongoing monitoring and evaluation.

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## SUMMARY OF EVALUATION RESULTS

Based on the safety analysis results documented in Section XV of the Nine Mile Point Unit 1 (NMP-1) Final Safety Analysis Report (FSAR) and compliance with requirements for associated Limiting Safety System Settings specified in the NMP-1 Technical Specifications (Appendix A to Facility Operating License No. DPR-63), no manual operator action is required to assure that safety systems perform their safety functions for design basis accident events.

It is therefore concluded that there are no plant parameters which need be classified Type A Variables, as defined in RG 1.97 (Revision 2), for NMP-1. The detailed evaluation supporting this conclusion follows.

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## SAFETY FUNCTION DEFINITION

For the purpose of determining Type A Variables, consistent with the definition provided in Section C of RG 1.97 (Revision 2), the accomplishment of core and containment safety functions is defined as follows:

- Fuel cladding integrity is maintained (i.e., the core remains adequately cooled), and
- Reactor coolant system integrity is maintained, and
- Primary containment integrity is maintained.

The actions required to assure that these principal safety functions are accomplished for design basis accident events for NMP-1 are separately reviewed in the three sections which follow.

ACTIONS TO ASSURE FUEL LOADING INTEGRITY

... 99V water level above the top of the active fuel is sufficient  
... this assurance is substantiated by work  
... the Government of the general RMR Owners Group  
... (EFC) and is documented in the appendices to the

... technical specification 1.1.1 states that, with the  
... core level is assured for 99V water level as  
... core height.

... technical specification 1.1.1 documents the results of  
... (DNR Section VII) which  
... the operation of a core spray  
... core cooling for the range of  
... conditions resulting from design state accidents.

... also assured by WMP-1  
... (Safety Limit for Fuel Loading Integrity)  
... variables associated with  
... limits on the important  
... to assure that the integrity of the fuel  
... Assurances of core cooling is included within the  
... thermal behavior.

... system settings are established to  
... The associated  
... the adequacy of such  
... by design  
... resulting from design  
... limiting safety system settings  
... and associated automatic  
... (fuel loading integrity)

## ACTIONS TO ASSURE FUEL CLADDING INTEGRITY

Maintaining RPV water level above the top of the active fuel is sufficient to assure adequate core cooling. This assurance is substantiated by work performed during the development of the generic BWR Owners' Group Emergency Procedure Guidelines (EPGs) and is so documented in the appendices to the EPGs.

The Bases for NMP-1 Technical Specification 2.1.1 states that, with the reactor shut down, adequate core cooling is assured for RPV water level as low as two-thirds core height.

The Bases for NMP-1 Technical Specification 3.1.4 documents the results of additional plant-specific licensing analyses (FSAR Section VII) which confirm that, with the reactor shut down, the operation of one core spray system is sufficient to assure adequate core cooling for the range of conditions resulting from design basis accidents.

The maintenance of adequate core cooling is also addressed by NMP-1 Technical Specification 2.1.1, "Safety Limit for Fuel Cladding Integrity." This Safety Limit applies to the interrelated variables associated with fuel thermal behavior, and establishes limits on the important thermal-hydraulic variables to assure that the integrity of the fuel cladding is maintained. Assurance of core cooling is included within the scope of analyzed fuel thermal behavior.

Technical Specification Limiting Safety System Settings are established to prevent exceeding the Fuel Cladding Integrity Safety Limit. The associated Bases document the results of analyses confirming the adequacy of automatic actions to prevent rupture of fuel cladding integrity (by assuring adequate core cooling) for the range of conditions resulting from design basis accidents. NMP-1 Fuel Cladding Integrity Limiting Safety System Settings (status or condition of key plant parameters) and associated automatic actions to assure adequate core cooling (and thus fuel cladding integrity) are listed on the next page.

Automatic Actions for Core Cooling

REMARKS	AUTOMATIC ACTION	INITIAL ACTION
Terminates power operation	Setpoint	High pressure
Terminates power operation	Setpoint	Low pressure
RPV is designed to provide adequate flow to the RPV for small reactor cooling rate errors which exceed the design flow. The reactor flow drive circuit and which are not large enough to depressurize the RPV. The circuit for core flow to be effective (RPV flow is 1.13 gpm)	20 gpm high pressure (RPV) System activation	20 gpm high pressure (RPV) System activation
In conditions where flow is 100% capacity core system are available and backup power generation is available and power is available and automatically supplied if necessary to the core system. No core flow is maintained.	Core flow (CS) System activation	Core flow (CS) System activation
Direct response to high pressure product activity released from the core terminates power operation and releases source of radioactivity	High pressure (HP) System activation	High pressure (HP) System activation
Assured sufficient steam during closure to account for the water contained in the reactor and the primary system as a result of reactor steam.	Setpoint	High pressure (HP) System activation
Activates and reduces the reactivity from fast closure of the turbine control valves to maintain power operation.	Setpoint	High pressure (HP) System activation

### Automatic Actions For Core Cooling

Condition	Automatic Action	Remarks
High reactor power	Scram	Terminates power operation.
High RPV pressure	Scram	Terminates power operation.
Low RPV water level	Scram, High Pressure Injection (HPCI) System initiation	HPCI is designed to provide makeup flow to the RPV for small reactor coolant line breaks which exceed the capability of the control rod drive pumps and which are not large enough to depressurize the RPV fast enough for core spray to be effective. (Ref. NMP-1 TS 3.1.8 Bases)
Low-low RPV water level	Core Spray (CS) System initiation	Two completely redundant 100% capacity core spray systems are installed. Backup diesel generator power is available, and automatically supplied if necessary, to all core spray system motor-operated components.
Main steam line high radiation	Scram, main steam line isolation valve closure	Direct response to high fission product activity released from the core; terminates power operation and isolates source of radioactivity release.
High scram dump volume water level	Scram	Assures sufficient scram dump volume to accommodate the water discharged from the control rod drive hydraulic system as a result of reactor scram.
Generator load reject	Scram	Anticipates rapid increase in RPV pressure and neutron flux resulting from fast closure of the turbine control valves; terminates power operation.

Operation for Abnormal Core Cooling

to ... events which could ... maintain ...  
... of ... the results are ...  
... as soon as ...  
... reaction system ...  
... PV water level ...  
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### Conclusion For Adequate Core Cooling

Design Basis events which could potentially threaten maintenance of adequate core cooling and fuel cladding integrity have been analyzed for NMP-1, and the results are documented in the FSAR. Automatic actions, initiated as specified by associated Limiting Safety System Settings and other reactor protection system setpoint devices (e.g., reactor scram for high reactor power conditions, Core Spray injection for low RPV water level conditions), provide adequate assurance that the "maintain adequate core cooling (fuel cladding integrity)" safety function is accomplished for Design Basis events without any need for additional manual operator actions. On this basis it is therefore determined that there are no Type A Variables, as defined in Regulatory Guide 1.97 (Revision 2), for this safety function for NMP-1.

1-1000 TO BE THE REACTION COOLANT SYSTEM INTEGRITY

The design of the reactor coolant system (RCS) is based on the principle of passive safety. The system is designed to maintain the reactor core at a safe temperature and pressure without the need for active components. The design is based on the following assumptions:

- The reactor core is designed to operate at a maximum temperature of 300°C.
- The primary loop is designed to circulate the coolant at a flow rate of 10,000 gpm.
- The secondary loop is designed to circulate the coolant at a flow rate of 10,000 gpm.
- The system is designed to maintain a pressure of 15.5 MPa in the primary loop and 15.5 MPa in the secondary loop.
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- The primary loop is designed to circulate the coolant at a flow rate of 10,000 gpm.
- The secondary loop is designed to circulate the coolant at a flow rate of 10,000 gpm.
- The system is designed to maintain a pressure of 15.5 MPa in the primary loop and 15.5 MPa in the secondary loop.
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1-1000 TO BE THE REACTION COOLANT SYSTEM INTEGRITY

Component	Flow Rate (gpm)	Pressure (MPa)
Primary Loop	10,000	15.5
Secondary Loop	10,000	15.5
Reactor Core	10,000	15.5



## ACTIONS TO ASSURE REACTOR COOLANT SYSTEM INTEGRITY

Maintaining Reactor Coolant System (RCS) pressure below the NMP-1 Technical Specification (TS) Reactor Coolant System Safety Limit (TS 2.2.1) is sufficient to assure that RCS integrity is maintained. Conversely, maintenance of RCS integrity is no longer assured if pressure in the RPV exceeds the Safety Limit. The Reactor Coolant System Safety Limit is derived from the design pressures and applicable codes for the reactor pressure vessel and reactor coolant system piping. (Refer to the Bases for NMP-1 Technical Specification 2.2.1 - Reactor Coolant System Safety Limit).

Technical Specification Limiting Safety System Settings are established to prevent exceeding the Reactor Coolant System Integrity Safety Limit (NMP-1 Technical Specification 2.2.1). The Bases for these Limiting Safety System Settings documents the results of analyses confirming the adequacy of automatic actions to prevent rupture of RCS integrity for the range of design basis accident conditions (RCS break accidents obviously excepted). NMP-1 Reactor Coolant System Limiting Safety System Settings and associated automatic actions are listed below.

### Automatic Actions For Coolant System Integrity

<u>Condition</u>	<u>Automatic Action</u>	<u>Remarks</u>
High reactor power (RUN mode)	Scram	Terminates power operation, thereby limiting any further increase in RPV pressure.
High RPV pressure	Scram	Terminates power operation, thereby limiting any further increase in RPV pressure.
High-high RPV pressure	Safety valve opening	Relieves RPV overpressure condition, discharges to the drywell

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In addition to the Reactor Coolant System Limiting Safety System Settings listed above, other automatic reactor protection and safety system devices serve as secondary backup to the chosen Limiting Safety System Settings. These include the following:

Condition	Automatic Action	Remarks
High RPV pressure	Electromatic relief valve opening	Pressure relief, with discharge to the water in the torus; averts safety valve opening.
High primary containment pressure	Scram	Backup to high RPV pressure scram in the event of safety valve opening.
Low condenser vacuum	Scram	Anticipates high RPV pressure scram caused by loss of main reactor heat sink.
Main steam line isolation	Scram	Anticipates high RPV pressure scram caused by main steam line isolation valve closure.
High scram dump volume water level	Scram	Assures sufficient scram dump volume to accommodate the water discharged from the control rod drive hydraulic system as a result of reactor scram.

GENERAL INSTRUCTIONS FOR THE USE OF THE

The following instructions are intended to assist you in the proper use of the instrument. It is recommended that you read these instructions carefully before using the instrument. The instrument is designed to measure the concentration of a specific substance in a sample. The concentration is measured in parts per million (ppm). The instrument is calibrated to measure concentrations from 0 to 100 ppm. The instrument is used by drawing a sample of the substance into the instrument and measuring the concentration. The concentration is displayed on the instrument's display. The instrument is used by drawing a sample of the substance into the instrument and measuring the concentration. The concentration is displayed on the instrument's display. The instrument is used by drawing a sample of the substance into the instrument and measuring the concentration. The concentration is displayed on the instrument's display.

### Conclusion For Reactor Coolant System Integrity

Design Basis events which could potentially threaten the integrity of the reactor coolant system have been analyzed for NMP-1, and the results are documented in the FSAR. Automatic actions, initiated as specified by Limiting Safety System Settings and other reactor protection system devices (e.g., reactor scram for main steam line isolation valve closure, electromatic relief valve operation for high RPV pressure conditions, etc.), provide adequate assurance that the "maintain reactor coolant system integrity" safety function is accomplished for Design Basis events without any need for additional manual operator actions. On this basis it is therefore determined that there are no Type A Variables, as defined in Regulatory Guide 1.97 (Revision 2), for this safety function for NMP-1.

2020-2021 BUDGETARY CONTROL SYSTEM

The budgetary control system is designed to provide a framework for the management of the organization's financial resources. It involves the preparation of budgets for each department, which are then compared against actual performance. This process helps in identifying areas of over-spending or under-spending, allowing for timely corrective action. The system also facilitates the allocation of resources to different departments based on their budgeted needs. The primary objective of the budgetary control system is to ensure that the organization's financial goals are achieved through efficient resource utilization.

The budgetary control system is a key component of the organization's financial management. It provides a clear and concise overview of the organization's financial performance over a specific period. By comparing actual results against budgeted figures, management can gain valuable insights into the organization's financial health. This information is used to make informed decisions regarding resource allocation, cost control, and overall financial strategy. The system also serves as a tool for accountability, as it holds department heads responsible for staying within their budgeted limits. In summary, the budgetary control system is essential for the organization's financial success and long-term sustainability.

## ACTIONS TO ASSURE PRIMARY CONTAINMENT INTEGRITY

The Primary Containment (drywell, vent system, and torus) is specifically designed and constructed to remain intact and fully functional for the most severe consequences (e.g., peak temperatures and pressures) of design basis accident events without any need for manual operator action. Compliance with normal plant operating procedures and Technical Specification Limiting Conditions for Operation assures that status of key primary containment parameters (suppression pool water temperature, suppression pool water level, and pressure suppression system pressure) does not exceed the worst-case initial conditions assumed in the accident analyses.

Although manual operator actions are specified in the EOPs to control the plant when out-of-specification primary containment conditions occur, thus mitigating the consequences of postulated accidents, such manual actions are not required in order to assure that the "maintain primary containment integrity" safety function is accomplished for the defined scope of design basis accidents. Safety Analysis results documented in Section XV of the NMP-1 FSAR fully substantiate this feature of plant design.

The Board of Directors of the Company has pleasure in presenting to you the following report on the operations of the Company during the year 1967. The Company has achieved a record year in terms of sales and earnings. This is due to the successful implementation of our expansion program and the continued loyalty of our customers. The Board is confident that the Company's strong financial position and diversified product line will enable us to continue our growth in the years ahead.



### Conclusion For Primary Containment Integrity

Design Basis accident events which could potentially threaten the integrity of the primary containment have been analyzed for NMP-1, and the results are documented in the FSAR. No requirement for manual operator action to assure primary containment integrity for design basis events has been identified. On this basis it is therefore determined that there are no Type A Variables, as defined in Regulatory Guide 1.97 (Revision 2), for this safety function for NMP-1.

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2. Nine Mile Point Nuclear Station Unit 1 Final Safety Analysis Report Section XV, "Safety Analysis"
3. Appendix A [Technical Specifications] To Facility Operating License No. DPR-63 For the Niagara Mohawk Power Corporation, Nine Mile Point Nuclear Station Unit 1, Docket No. 50-220
4. Nine Mile Point Nuclear Station Unit No. 1 Emergency Procedure Guidelines, (OEI Document 8309-2, Revision 5, dated April 18, 1989)
5. Nine Mile Point Nuclear Station Unit 1 Emergency Operating Procedures:
  - N1-EOP-1, "General Instructions,"
  - N1-EOP-2, "RPV Control"
  - N1-EOP-3, "Primary Containment Control"
6. NMP1L 0190 from Niagara Mohawk Power Corporation (T.E. Lempges) to U.S. Nuclear Regulatory Commission, dated October 5, 1987

