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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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 operator 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.

Section C 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."

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

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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."

SUMMARY OF EVALUATION RESULTS

[illegible] 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 Setting [illegible] the NMP-1 Technical Specifications (Appendix A to Facility [illegible] License No. DPR-63), no manual operator action is required to [illegible] safety systems perform [illegible] safety functions for design basis [illegible] events.

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

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.

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.

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 documented in the appendices to the EPGs.

The basis for NMP-1 Technical Specification [illegible] states that, with the reactor shut down, adequate core cooling is assured for RPV water level as low as 2/3 of core height.

The basis for NMP-1 Technical Specification [illegible] 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 sufficiency 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.

The NMP-1 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 events. NMP-1 Fuel Cladding Integrity Limiting Safety System Settings ([illegible] of key plant parameters) and associated automatic actions [illegible] adequate core cooling (and thus fuel cladding integrity) [illegible].

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

Condition	Automatic Action	Remarks
High reactor [illegible]	Scram	Terminates power operation.
[illegible]	Scram	Terminates power operation
[illegible] water [illegible]	Scram, High Pressure Coolant 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. NM-1 IS 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 rejection	Scram	Anticipates rapid increase in RPV pressure and neutron flux resulting from fast closure of the turbine control valves, terminates power operation.

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 neutorn flux resulting from fast closure of the turbine control valves; terminates power operation.

Conclusion For Adequate Core Cooling

[illegible] events which could potentially threaten maintenance of [illegible] core [illegible] cladding integrity have been analyzed for [illegible], and the results are documented in the FSAR. Automatic actions [illegible] as specified by [illegible] Limiting Safety System Settings and [illegible] protection system actuation devices (e.g., reactor scram for [illegible] 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. In this case, it is therefore determined that there are no Type A variables as defined in Regulatory Guide 1.97 (Revision 2) for this safety function, Function [illegible].

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.

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

Condition	Automatic Action	Remarks
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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[illegible]

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[illegible]

[illegible]

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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.

[illegible]

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[illegible]
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[illegible]
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[illegible]
[illegible]
[illegible]
[illegible]

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.

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.

[illegible]

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.

REFERENCES

1. US NRC Regulatory Guide 1.97, "Instrumentation For Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During And Following An Accident," Revision 2, dated December 1980

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