February 3, 2003

Mr. John T. Conway
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P. O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: DIESEL GENERATOR ALLOWED OUTAGE TIME (TAC NO. MB4612)

Dear Mr. Conway:

The Commission has issued the enclosed Amendment No. 179 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated March 27, 2002, as supplemented by letter dated October 7, 2002.

The amendment revises TSs Section 3.6.3, "Emergency Power Sources," to extend the current allowable outage time for an inoperable diesel generator from 7 days to 14 days, and Section 3.4.4, "Emergency Ventilation System," and Section 3.4.5, "Control Room Air Treatment System," to reflect the change to Section 3.6.3.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/RA/

Peter S. Tam, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures: 1. Amendment No. 179 to DPR-63
2. Safety Evaluation

cc w/encls: See next page

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Accession Number: **ML030340460**

OFFICE	PDI-1\PM	PDI-1\LA	EEIB\SC*	SPSB\SC*	OGC	PDI-1/SC	
NAME	PTam	SLittle	CHolden	MRubin	RWeisman	RLaufer	
DATE	1/9/03	1/9/03	12/24/02*	6/20/02*	1/30/03	2/3/03	

*SE transmitted by memo on the date shown.

OFFICIAL RECORD COPY

DATED: February 3, 2003

AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-63 NINE MILE POINT UNIT NO. 1

PUBLIC
PDI R/F
RLaufer
SLittle
PTam
SRichards
OGC
GHill (2)
WBeckner
SSaba
MWohl
GThomas
ACRS
BPlatchek, RI

cc: Plant Service list

NINE MILE POINT NUCLEAR STATION, LLC (NMPNS)

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 179
License No. DPR-63

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nine Mile Point Nuclear Station, LLC (the licensee) dated March 27, 2002, as supplemented by letter dated October 7, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-63 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, which is attached hereto, as revised through Amendment No. 179, is hereby incorporated into this license. Nine Mile Point Nuclear Station, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief, Section I
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 3, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 179

TO FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Replace the following pages of Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
173	173
178	178
256	256

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-63

NINE MILE POINT NUCLEAR STATION, LLC

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

By letter dated March 27, 2002, as supplemented by letter dated October 7, 2002, Nine Mile Point Nuclear Station, LLC (the licensee), proposed changes to the Technical Specifications (TSs) for Nine Mile Point Nuclear Station, Unit No. 1 (NMP1). The proposed changes would revise TSs Section 3.6.3, “Emergency Power Sources,” to extend the current allowable outage time (AOT) for an inoperable diesel generator (DG) from 7 days to 14 days. In addition, the licensee proposed to change Section 3.4.4, “Emergency Ventilation System,” and Section 3.4.5, “Control Room Air Treatment System,” to reflect the change to Section 3.6.3. The October 7, 2002, supplemental letter, which is the licensee’s response to the Nuclear Regulatory Commission (NRC) staff’s request for additional information dated August 20, 2002, provided clarifying information that did not change the scope of the requested action or initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) provides that nuclear plant TSs will be derived from the analyses and evaluations included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34 (which addresses, among other things, contents of the Final Safety Analysis Report (FSAR)). The existing TS requirements, as well as the licensee’s proposed amendment, are based on such analyses and evaluations. The licensee requested the subject amendment in accordance with the provisions of 10 CFR 50.90.

Nine Mile Point Unit No. 1 was constructed before the General Design Criteria of 10 CFR Part 50 were promulgated. However, when the licensee petitioned the Atomic Energy Commission to convert its provisional operating license to full-term operating license, the licensee stated (reference Technical Supplement to Petition for Conversion from Provisional Operating License to Full-Term Operating License, dated July 1972) that the unit meets the intended safety function of General Design Criterion (GDC)-17, “Electric Power System” of Appendix A, “General Design Criterion of Nuclear Power Plants” to 10 CFR Part 50. GDC-17 requires, in part, that nuclear power plants have an onsite and offsite electrical power system to permit the functioning of structures, systems and components important to safety. The onsite system is required to have sufficient independence, redundancy and testability to perform its safety function, assuming a single failure. The offsite system is required to have two independent supply circuits to the onsite system. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as the result of loss of power from the unit, the offsite transmission network, or the onsite power supplies. GDC-18, "Inspection and Testing of Electric Power System," requires that electric power systems important to safety be designed to permit appropriate periodic inspection and testing.

3.0 TECHNICAL EVALUATION

The onsite emergency alternating current (AC) power system at NMP1 consists of two redundant and independent diesel generators (DG 102 and DG 103) which provide onsite emergency electrical power. Each DG is designed to independently start and carry the maximum anticipated emergency loads. For the design-basis accident (DBA) loss-of-coolant accident (LOCA) concurrent with a loss of offsite power, the DGs will automatically start and load onto their power boards, followed by the automatic sequential loading of the emergency core cooling system (ECCS) loads. DG 102 and 103 are alternate power supplies for power boards 102 and 103, respectively, which supply the 4160-V loads (i.e., for the ECCS and other engineered safeguards and safety-related loads). The 115-kV reserve bus feeds reserve station service transformers which are sized for plant startup, shutdown or operation. The 115-kV/4160-V stepdown reserve transformers are the normal power supplies for power boards 102 and 103.

3.1 Deterministic Evaluation

3.1.1 TS Section 3.6.3.c, "Emergency Power Sources"

The licensee proposed to revise Section 3.6.3.c by increasing the AOT from 7 days to 14 days. Section 3.6.3.c will read:

> One diesel-generator power system may be inoperable provided two 115 kV external lines are energized. If a diesel-generator power system becomes inoperable, it shall be returned to an operable condition within 14 days.

Defense-in-depth philosophy requires multiple means or barriers to be in place to accomplish safety functions and prevent the release of radioactive material. The safety-related equipment required to mitigate the consequences of postulated accidents consists of two independent divisional load groups. Each load group can be powered from three independent sources: either one of the two offsite sources, or the associated DG. The loss of an entire load group will not prevent the safe shutdown of the plant in the event of a DBA. In addition, with one DG out of service, two offsite power sources on the affected load group and the entire unaffected load group will remain available. This is ascertained by the balance of Section 3.6.3.c, which states that "[i]f a diesel-generator power system becomes inoperable coincident with a 115 kV line de-energized, that diesel generator power system shall be returned to operable condition within 24 hours." Thus, the unavailability of a single DG does not reduce the amount of available equipment to a level below that necessary to mitigate a DBA. The remaining power sources and safety-related equipment are designed with adequate independence, capacity, and capability to provide power to the necessary equipment during postulated accidents. Specifically, with one DG out of service, two offsite power sources on the affected load group and the entire unaffected load group will remain available.

The NRC staff evaluated the licensee's common cause failure modes analysis for the proposed 14-days AOT and determined that the proposed AOT does not introduce new potential common cause failure modes, and does not compromise protection against common cause failure modes previously considered.

The NRC staff re-visited the licensee's station blackout (SBO) analysis. SBO is defined as the complete loss of AC electric power to the switchgear buses in a nuclear power plant. An SBO would result from a loss of offsite power sources, concurrent with a turbine generator trip and unavailability of the onsite emergency AC power system. To address the potentially significant risk of core damage associated with an SBO event, the NRC previously issued the SBO Rule, promulgated as 10 CFR 50.63, "Loss of All Alternating Current Power," and Regulatory Guide (RG) 1.155. The SBO Rule requires that a licensed nuclear power plant be able to withstand an SBO for a specified time and recover from the SBO. The ability to cope with an SBO for a certain time period provides additional defense-in-depth should both offsite and onsite emergency AC power systems fail concurrently. For an SBO, NMP1 was classified as a 4-hour duration coping plant. The proposed 14-day AOT will not have an impact upon the previous SBO coping analysis because the DGs are not assumed to be available during the coping period, but one or both of the DGs would be restored during the coping period.

With implementation of the proposed change, the licensee will establish appropriate restrictions and compensatory measures to assure that system redundancy, independence, and diversity are maintained during the extended AOT. These include TS and Maintenance Rule programmatic requirements, as well as administrative control in accordance with the licensee's Configuration Risk Management Program (CRMP, see Section 3.2 of this safety evaluation). Existing requirements (TS Section 3.6.3.g) require all emergency equipment aligned to an operable DG to have no inoperable components while the other DG is removed from service. Furthermore, the licensee stated that NMP1 procedures under its CRMP include provisions to implement the following compensating measures and configuration risk management controls when a DG is removed from service for any extended AOT duration (greater than 7 days and up to 14 days):

> The redundant DG and equipment will be verified operable and no elective testing or maintenance activities will be scheduled on the redundant (operable) DG.
>
> No elective testing or maintenance activities will be scheduled in the 115 kV switchyard or on the 115 kV power supply lines and transformers which could cause a line outage or challenge offsite power availability.
>
> The NMP1 diesel-driven firewater pump (DFP) will be verified operable as a feedwater makeup source to the NMP1 reactor pressure vessel (RPV).
>
> The Nine Mile Point Unit 2 (NMP2) DFP and cross-tie to NMP1 will be verified operable as a feedwater makeup source to the NMP1 RPV.
>
> Operators will take actions (credited in the SBO coping analysis) to activate the automatic depressurization system and use the DFP to assure that sufficient water inventory is maintained in the vessel for core cooling.

> Weather conditions will be considered so that no preplanned maintenance will be scheduled when severe weather conditions are expected.
>
> Testing and maintenance of equipment, load dispatching, equipment availability will be considered.

Inasmuch as the licensee uses the CRMP to satisfy the requirements of the Maintenance Rule (10 CFR 50.65(a)(4)), the above measures that are necessary to manage the increase in risk from the proposed DG maintenance activities will be appropriately controlled.

NMP1 is a boiling-water reactor (BWR)/2 design with an emergency cooling system (isolation condensers), a core spray system, automatic depressurization system (ADS) electromatic relief valves (EMV) and a high pressure core spray system (the feedwater system is not credited in the LOCA analysis). There are 2 isolation condensers, 4 core spray pumps, 16 EMVs and 2 motor-driven feedwater pumps. Isolation condenser and ADS EMV operations are independent of AC power.

Since the NMP1 design does not include jet pumps, core cooling following a design basis recirculation line break relies solely upon core spray. Adequate spray cooling is thus defined to exist in NMP1 when design spray flow requirements (TS Section 3.1.4) are satisfied. The licensee confirmed that, for the proposed extended outage time of 14 days, requirements in the TSs will ensure core spray pump operability during the outage (i.e., there is no change in the licensing basis and TS requirements for the core spray system).

For long-term operation of the isolation condensers, the minimum water level in the shell side is maintained with 10,680 gallons of water. This is sufficient to provide about 8 hours of continuous system operation. This time is sufficient to restore additional makeup water from the two 200,000 gallon condensate storage tanks.

Thus, the systems described above are capable of maintaining NMP1 in a safe condition under all operating conditions, and defense-in-depth is maintained during the proposed extended DG AOT. The NRC staff finds the proposed change to extend the DG AOT from the current 7 days to 14 days to be acceptable as set forth above.

3.1.2 TS Sections 3.4.4, “Emergency Ventilation System” and 3.4.5, “Control Room Air Treatment System”

The licensee proposed to revise paragraph a. of both these sections by eliminating the phrase “and the diesel generators required for operation of such circuits.” According to the current wording, the unit would be required to be shut down for an inoperable emergency ventilation system or control room air treatment system due solely to the inoperability of an emergency power source (i.e., the associated DG). This is contrary to the power source provisions of TS Section 3.0.1, which allows an inoperable system, subsystem, train, or device to be considered operable for satisfying the requirements of the applicable LCO when it is determined that the inoperability is solely because its emergency or normal power source is inoperable. The proposed change removes the DG operability requirement from the subject sections. The proposed change is necessary to accommodate the proposed extension of the DG AOT from 7 to 14 days, which the NRC staff has already found acceptable above. Therefore, the NRC staff finds the proposed change to Sections 3.4.4 and 3.4.5 acceptable on the basis that it has no adverse effect on the availability of the subject systems, that it is consistent with TS Section 3.0.1, and consistent with the extension of the DG AOT from 7 to 14 days.

3.2 Probabilistic Risk Assessment (PRA) Evaluation

To assess the probabilistic impact on plant safety of the proposed 14-day AOT, the licensee performed a PRA consistent with the guidance discussed in RG 1.177, “An Approach for Plant-Specific Risk-Informed Decisionmaking: Technical Specifications.” The PRA generates a quantitative evaluation of the risk associated with the proposed 14-day AOT in terms of the change in (or delta) average Core Damage Frequency (CDF) and delta average Large Early Release Frequency (LERF). The licensee’s evaluation included consideration of its Maintenance Rule program established pursuant to the requirements of 10 CFR 50.65(a)(4) to assess and manage the increase in risk that may result from the performance of proposed maintenance activities. As applied to the requested amendment, the Maintenance Rule requires consideration of configurations associated with other potentially high-risk tasks during a DG outage, as well as consideration of specific compensatory measures to minimize risk. All these elements were included in the licensee’s risk evaluation using the three-tiered approach suggested in RG 1.177:

Tier 1 - PRA Capability and Insights
Tier 2 - Avoidance of Risk-Significant Plant Configurations
Tier 3 - Risk-Informed CRMP

Evaluations addressing each of these three tiers are provided below. The PRA model serves as the primary licensee tool for these evaluations.

3.2.1 PRA Model Development

The NMP1 PRA is based on a detailed model of the plant that was developed from the NMP1 Individual Plant Examination (IPE) and the NMP1 Individual Plant Examination for External Events (IPEEE) projects. The PRA model has undergone staff review and Boiling Water Reactor Owners Group (BWROG) certification. The model was recently updated to incorporate review comments, current plant design, current procedures, recent plant operating data, current PRA techniques, and general improvements identified by the licensee’s PRA team.

Key milestones for the development of the licensee’s PRA model were:

– IPE submitted to the staff in July 1993.
– IPE staff evaluation report (SER) received from the NRC staff in April 1996.
– IPEEE submitted to the NRC staff in July 1996.
– BWROG certification issued in June 1998.
– IPEEE SER received from the NRC staff in July 2000.
– Licensee PRA model update completed in June 2001-Model U1PRA01A.
– Licensee PRA model update for proposed 14-day DG AOT completed in January 2002-Model U1PRA01B.

The PRA model update used by the licensee to support the proposed 14-day DG AOT was a partial update that addressed only the inclusion of additional initiating events, DG test data, and DFP test data through 2001. The licensee has not yet re-evaluated the risk metric results presented in this application and, upon completion, should incorporate them into the next PRA update. The present updated PRA model incorporates the following:

– Plant-specific unreliability and unavailability data (Note: Failure data was evaluated through December 31, 2000, with a small gap in the 1991-1993 time frame between the IPE and the start of the institute of Nuclear Power Operations (INPO) Equipment Performance and Information Exchange System (EPIX). Unavailability is based on data between August 1, 1998, and July 31, 1999, which is considered representative). Specifically, DG and DFP unavailability data updated through December 31, 2001.

– Plant-specific initiating event data developed through December 31, 2001.

– Plant-specific configuration information (design and operation) as of December 31, 2001.

– Insights from several years of plant-specific applications utilizing the original IPE and IPEEE.

– Insights from NRC staff review comments.

Key safety-related goals of the licensee's PRA model development process were to:

– Understand the underlying plant risks and key sources of uncertainty.

– Develop a tool to quantify nuclear safety and support a comprehensive risk management program.

– Establish an in-house risk analysis capability to support plant decision-making.

An independent assessment of the licensee's PRA, using the self-assessment process as part of the BWROG peer review certification program, was completed to assure that the licensee's PRA was comparable to other PRA programs in use throughout the industry. The licensee's PRA was certified by the BWROG in June 1998 following an inspection and review by a PRA peer review certification team. The certification review results were documented and evaluated for inclusion in the licensee's last PRA model update. The review findings related primarily to improvements in the areas of guidance, documentation, models, and capturing of plant changes. Overall, the certification review provided high technical marks on the PRA, and there were no findings that significantly affected the PRA results. The certification team approved the suitability of the licensee's PRA for applications such as single TS actions if supported by deterministic evaluations.

3.2.2 PRA Model Maintenance

The PRA model is applied and controlled as defined in the licensee's administrative procedure NIP-REL-02, "Probabilistic Risk Assessment Program," and engineering department procedure NEP-REL-01, "Evaluations, Analysis, and Update of the Probabilistic Risk Assessment (PRA) Program." Ongoing assessments of the PRA model and reports are part of the normal duties of the PRA engineers. When a change to plant procedures, plant design, or operational data is identified that impacts the PRA model, the PRA engineer uses the guidance in the following table to prioritize the change and assist in the development of an implementation schedule.

GRADE	DEFINITION	ACTION
1	Extremely important and necessary to address to assure the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process.	Immediate update considered.
2	Important and necessary to address, but may be deferred to the next planned PRA update.	Consider in next planned update.
3	Considered desirable to maintain maximum flexibility in PRA applications and consistency with the industry, but is not likely to significantly affect results or conclusions.	Consider in next 2-3 planned updates.
4	Editorial or minor technical item, low priority.	Consider as update opportunity exists.

Planned updates to the PRA model are scheduled on a regular basis by the PRA team. Planned updates include an information gathering phase that is intended to capture plant changes that had not been previously identified by the PRA team. The normal scheduled (planned) update considers all aspects of the PRA. An unplanned update is undertaken when a Grade 1 item is identified for immediate update. An unplanned update may also be undertaken to address a need for a specific application of the PRA. An unplanned update is considered a limited scope update and does not necessarily include a detailed plant information review or consideration of all aspects of the PRA, according to the licensee. This type of update is intended to augment the PRA between normal planned updates, as needed.

The update of the initiating event frequency, DG reliability data, and DFP reliability data for the proposed extension of the DG AOT represents an unplanned update which was limited in scope to these changes.

3.2.3 PRA Model Application, Including External Events

The licensee's level 2 PRA model was used to determine the risk associated with removing a DG from service for planned maintenance in accordance with the proposed 14-day AOT. The risk metrics used are delta CDF, delta LERF, incremental conditional core damage probability (ICCDP), and incremental conditional large early release probability (ICLERP). The NMP1 baseline CDF is 2.6E-05/yr and the baseline LERF is 2.2E-06/yr. The PRA model is a consolidation of the NMP1 IPE and the NMP 1 IPEEE, which explicitly includes fires and seismic events, screening out other external events.

The PRA model is used by the licensee's work control and operations personnel throughout the online work planning and implementing processes. The PRA model is implemented through the use of a Safety Monitor as described in administrative procedure GAP-PSH-03, "Control of On-

Line Work Activities." The results obtained from the PRA model are used by the licensee along with other inputs, such as TS requirements and operator system knowledge, in a blended approach to determine the final work schedule. The PRA model is currently not applicable to shutdown conditions. Thus, risk assessments for work activities during plant outages are performed consistent with the defense-in-depth philosophy as described in licensee administrative procedure NIP-OUT-01, "Shutdown Safety."

The guidance contained in RGs 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and 1.177 was utilized by the licensee to ensure that the quantitative risk metric results of the PRA model would be acceptable to support the proposed 14-day DG AOT. The licensee expressed confidence that the results of its risk evaluations are technically sound and consistent with the expectations for quality set forth in RG 1.177. The licensee expressed confidence that the scope, level of detail and quality of its PRA are sufficient to support a technically defensible and realistic evaluation of the risk changes involved in the proposed change. Based on the foregoing discussion, the NRC staff agrees with the licensee.

3.2.4 Tier 1: PRA Capability and Insights

The risk-informed support for the proposed AOT for an inoperable DG is based on PRA calculations performed to quantify the change in average CDF and average LERF. To determine the effect of the proposed change with respect to plant risk, the licensee used the guidance provided in RGs 1.174 and 1.177.

An evaluation was performed based on the assumption that the full extended AOT (14 days) would be applied once per DG refueling cycle. The ICCDP and ICLERP were calculated as recommended in RG 1.177. The results of the risk evaluation were compared against the risk significance criteria in RG 1.174 for changes in the annual average CDF and LERF, and RG 1.177 for ICCDP and ICLERP. The ICCDP and ICLERP were calculated for both DG 102 and DG 103, which indicates that an outage of DG 103 is more limiting. The following table summarizes the results of the risk evaluation:

Risk Metric Guideline	Acceptance Guideline	Evaluation Results
delta CDF (Avg)	less than 1.0E-06/yr	2.2E-07/yr
delta LERF (Avg)	less than 1.0E-07/yr	7.7E-09/yr
ICCDP (102)	less than 5.0E-07	1.1E-07
ICCDP (103)	less than 5.0E-07	3.2E-07
ICLERP (102)	less than 5.0E-08	5.3E-09
ICLERP (103)	less than 5.0E-08	9.6E-09

These results consider the information recently included in the PRA model update for the case where the plant is operating with a DG out of service. The licensee completed this model update in January 2002 as part of the evaluation for the proposed extension of the DG AOT and includes select data and plant changes through 2001. The previous PRA update was completed in June 2001 with plant data current to the end of 2000.

The proposed change to extend the DG AOT to 14 days will reduce the probability of an unplanned manual shutdown initiated by online DG unavailability. The risk associated with an unplanned manual shutdown has been included in the NMP 1 PRA update and can be considered here. The licensee includes unplanned manual shutdowns in the scram initiators (SCRAM and BSCRAM). These initiators have a frequency of 4.8/yr in the PRA and account for a total CDF of 4.28E-07/yr. As a result, one manual shutdown would contribute approximately 8.23E-08/yr (4.28E-07/yr/4.8=8.23E-08/yr) to overall plant core damage risk.

3.2.5 Tier 2: Avoidance of Risk-Significant Configurations

A CRMP is in place at NMP1 for compliance with the Maintenance Rule (10 CFR 50.65), and in particular, for compliance with paragraph (a)(4) of the rule. The CRMP provides assurance that risk-significant plant equipment configurations are precluded or minimized when plant equipment is removed from service. Thus, any increase in risk posed by the removal of a DG from service and the potential combinations of other equipment out of service will be managed in accordance with the CRMP. Additional compensating measures and configuration risk management controls that will apply when entering the proposed extended DG AOT (greater than 7 days and up to 14 days) are listed in Section 3.1 (above).

While in the proposed extended DG AOT, additional elective equipment maintenance or testing that requires equipment to be removed from service will be evaluated and proposed activities that result in unacceptable risk results will be avoided.

The dominant sequences in the NMP1 PRA are evaluated to ensure that important equipment is identified and evaluated when a DG is out of service. Two types of evaluations are considered:

1. Important systems and equipment are assessed to determine whether their unreliability has increased since the last PRA update based on plant operational experience.

2. Important equipment and human actions are assessed to determine whether compensating measures can be credited to reduce risk while the DG is out of service.

The licensee has identified the following as contributors to risk when a DG is out of service:

(1) Loss of offsite power initiating event.
(2) Scram initiating event.
(3) Seismic initiating event.
(4) Fire initiating event.
(5) Loss of DC power.
(6) Loss of redundant DG.
(7) Loss of diesel driven fire pump (DFP).
(8) Loss of DC battery on demand.

(9) Electromatic relief valves sticking open.
(10) Loss of reactor recirculation pump seals.
(11) Loss of emergency condensers.
(12) Failure of operators to:
- Shed DC loads.
- Align diesel firewater pump.
- Control RPV level from East/West instrument room.

Several operator actions have been identified as potentially important and total dependency is assumed in the PRA model for key operator actions (e.g., if the operators fail to shed DC loads early (O15), no credit can be taken for the operators maintaining level in the East/West instrument room via SOP-14. Important operator actions are:

- Shed DC loads (O15).
- Align the DFP (OR1).
- Control RPV level from the East/West instrument room (HRA, SOP-14).
- Recover AC power.
- Control emergency condenser makeup on loss of instrument air.

No credit was taken for operator reliability compensatory measures in the PRA model, except for the operator action to align the NMP2 DFP through the cross-tie. This action is currently proceduralized and the licensee assumes that the existing human error rate in the PRA applies without any compensatory measure. However, the licensee had not previously credited this cross-tie capability in the PRA and it has now been added as an important compensatory measure.

3.2.6 Tier 3: Risk-Informed CRMP

Consistent with 10 CFR 50.65(a)(4), and as indicated previously, the licensee has developed a CRMP which ensures that the risk impact of out-of-service equipment is properly evaluated prior to performing a work activity. The procedures and instructions governing this process are GAP-PSH-03, NAI-PSH-02, “Use of the Safety Monitor,” NIP-OUT-01, and GAI-OPS-11, “Shutdown Safety Review.” The guidance provided in GAP-PSH-03 provides assurance that the risk associated with planned online work activities is evaluated and that the work activities are scheduled appropriately. The CRMP includes an integrated review (i.e., both probabilistic and deterministic) to identify risk-significant equipment outage configurations in a timely manner during the online work management process for both planned and emergent work. The licensee gives appropriate consideration to equipment unavailability, operational activities (e.g., testing, load dispatching), and weather conditions. The CRMP includes provisions for performing a configuration-dependent assessment of the overall impact on risk of proposed plant configurations prior to, and during, the performance of online work activities that remove equipment from service. Risk is re-assessed if an equipment failure or malfunction, or other emergent condition, produces a plant configuration that had not been addressed previously.

For online work activities, a quantitative risk assessment is performed to ensure that the activity does not pose an unacceptable risk. This evaluation is performed using the Safety Monitor.

Emergent work is reviewed by work management and operations to evaluate the impact on the risk assessment performed during the schedule development process. Prior to beginning any work, the work scope and schedule are reviewed to ensure that nuclear safety and plant operations remain consistent with regulatory requirements as well as licensee management expectations.

3.2.7 Maintenance Rule Program Controls

To ensure that maintenance activities, which would include the proposed 14-day DG AOT, will not degrade operational safety, should equipment not meet its performance criteria, evaluation is required as part of the Maintenance Rule (10 CFR 50.65). The reliability and availability of the NMP1 DGs are monitored under the Maintenance Rule program as described in licensee administrative procedures NIP-REL-01, “Maintenance Rule,” S-MRM-REL-0101, “Maintenance Rule,” and N1-MRM-REL-0105, “Maintenance Rule Performance Criteria.” If the pre-established reliability and availability performance criteria are exceeded for the DGs, consideration must be given to 10 CFR 50.65(a)(1) actions, including increased management attention and goal setting in order to restore DG performance (i.e., reliability and availability) to an acceptable level. The performance criteria are risk-informed, and therefore, are a means to manage the overall risk profile of the plant. An accumulation of large integrated core damage probabilities over time is precluded by adherence to the performance criteria.

In practice, the licensee’s actual out of service time for the DGs is minimized to ensure that the Maintenance Rule reliability and unavailability performance criteria for the DGs are met. Overall DG unavailability will be minimized consistent with the Maintenance Rule performance criteria (currently 1.5%) such that the proposed 14-day AOT is expected by the licensee to have minimal impact on DG unavailability. Any change to the Maintenance Rule performance criteria will be evaluated by the licensee using the PRA model, consistent with the Maintenance Rule programmatic requirements.

Both NMP1 DGs are currently in the 10 CFR 50.65(a)(2) Maintenance Rule category (i.e., the DGs are meeting established performance criteria). Performance of DG overhaul maintenance online is not expected by the licensee to result in exceeding the current Maintenance Rule criteria for the DGs.

Pursuant to 10 CFR 50.65(a)(3), DG reliability and unavailability are monitored and periodically evaluated with respect to Maintenance Rule performance criteria. The Maintenance Rule unavailability performance criterion for the NMP1 DGs is currently 1.5%. For the rolling 24-month Maintenance Rule monitoring period ending January 31, 2002, DG 102 unavailability was 0.59% (99.41% availability) and DG 103 unavailability was 0.57% (99.43% availability). Since the Maintenance Rule program also includes monitoring for DG reliability, fault exposure is not included in the unavailability performance values. In the past 10 years, DG 103 has not experienced any fault exposure unavailability, while DG 102 has incurred 356.81 hours of fault exposure unavailability, primarily due to a recent failure that occurred in January 2002. Prior to that event, the last time DG 102 experienced fault exposure unavailability was in December 1998. Although fault exposure is not included in the Maintenance Rule program, with the January 2002 fault exposure added to the 3-year performance indicator for safety system unavailability (Nuclear Energy Institute (NEI)99-02), the licensee states that the overall DG unavailability is still acceptable at 1.3%. The licensee states that the NMP1 Maintenance Rule program establishes reliability criteria at the Functional Failure (FF) level rather than at the

Maintenance Preventable Functional Failure level. The licensee states that this provides assurance that all DG FFs are assessed for possible 10 CFR 50.65(a)(1) goal setting and monitoring under the Maintenance Rule program, regardless of maintenance preventability. The licensee's Maintenance Rule performance criteria for DG reliability is no more than 3 FFs in 20 demands, no more than 4 FFs in 50 demands, and no more than 5 FFs in 100 demands. DG 103 has had 104 consecutive satisfactory starts since its last FF, which occurred in March 1995. DG 102 has experienced 4 FFs in its last 100 starts, but only 1 in its last 50 and 20 starts. Prior to the failure in January 2002, DG 102 had 69 consecutive satisfactory starts. Only two of the previous three FFs were maintenance preventable functional failures, and all three FFs occurred more than 4 years ago (in 1997).

The licensee stated that its Maintenance Rule program provides a process to identify and correct adverse trends to ensure that the proposed 14-day DG AOT would not degrade operational safety over time. Compliance with the Maintenance Rule would not only optimize the reliability and availability of important equipment, but would also establish controls for the management of the risk associated with removing equipment from service for testing or maintenance in accordance with 10 CFR 50.65(a)(4). Based on the above, the NRC staff agrees with the licensee.

3.3 Summary of Technical Evaluation

Based on the above, the NRC staff concludes that the licensee's proposed elimination of the requirements of DG operability from TS Section 3.4.4.a and 3.4.5.a, and extension of the DG AOT from 7 days to 14 days is acceptable. With these changes, NMP1 will continue to meet current licensing requirements, specifically, GDC-17. Also, the impact on plant risk of allowing NMP1 to have a 14-day DG AOT is very small for both internal and external events.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (67 FR 21290). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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