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a joint venture of



NINE MILE POINT
NUCLEAR STATION

October 27, 2011

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station
Unit No. 1; Docket No. 50-220

Submittal of Revision 22 to the Final Safety Analysis Report (Updated),
10 CFR 50.59 Evaluation Summary Report, Technical Specifications Bases Changes,
and Report Consistent with 10 CFR 54.37(b)

Pursuant to the requirements of 10 CFR 50.71(e), 10 CFR 50.59(d)(2), and the Nine Mile Point Unit 1 (NMP1) Technical Specifications (TS) Bases Control Program (TS 6.5.6), Nine Mile Point Nuclear Station, LLC (NMPNS) hereby submits the following:

- The NMP1 Final Safety Analysis Report (Updated) (UFSAR), with Revision 22 incorporated,
- The NMP1 10 CFR 50.59 Evaluation Summary Report, and
- NMP1 Technical Specifications Bases Changes.

The entire UFSAR, with Revision 22 incorporated, is contained on the enclosed compact disc (CD). The UFSAR revision contains changes made since the submittal of Revision 21 in October 2009. The revision reflects all changes up to April 2011. Attachment 1, 10 CFR 50.59 Evaluation Summary Report, covering the same time interval as the UFSAR revision, contains a brief description of changes, tests, and experiments, and includes summaries of the associated 10 CFR 50.59 evaluations. None of the 10 CFR 50.59 evaluations involved obtaining a license amendment as defined in 10 CFR 50.59(c)(1).

Attachment 2 contains the revised Technical Specifications Bases pages, which incorporate changes made since April 2009.

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Consistent with 10 CFR 54.37(b), Attachment 3 contains a report describing how the effects of aging of newly-identified structures, systems, or components will be managed, such that the intended functions described in 10 CFR 54.4 will be effectively maintained during the license renewal period of extended operation.

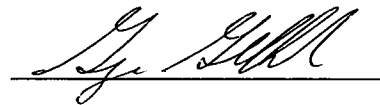
Should you have any questions regarding the information in this submittal, please contact John J. Dosa, Licensing Director, at (315) 349-5219.

Very truly yours,



CERTIFICATION:

I, George Gellrich, certify that I am Vice President-Nine Mile Point and that the information contained in this submittal accurately presents changes made since the previous submittal necessary to reflect information and analysis submitted to the Commission or prepared pursuant to Commission requirement.



GG/LWB

Enclosure: Final Safety Analysis Report (Updated) in CD format

Attachments: 1. 10 CFR 50.59 Evaluation Summary Report
2. Revised Technical Specifications Bases Pages
3. Report Consistent with 10 CFR 54.37(b) on How Effects of Aging of Newly-Identified Structures, Systems, or Components are Managed

cc: NRC Project Manager
NRC Resident Inspector
NRC Regional Administrator

ATTACHMENT 1

10 CFR 50.59 EVALUATION SUMMARY REPORT

ATTACHMENT 1
10CFR50.59 EVALUATION SUMMARY REPORT

Safety Evaluation No.: 2011-02

Implementation Document No.: ECP-10-000337

UFSAR Affected Pages: Sections IV, X, XV

System: Reactor Core, Spent Fuel Pool, Other Various Systems

Title of Change: NMP1 Reload Methods Changes

Description of Change:

Nine Mile Point Unit 1 (NMP1) stability restricted regions are predicted based on ODYSY methodology with a conservative 0.15 decay ratio adder. General Electric Topical Report NEDE-33213P-A, April 2009, "ODYSY Application for Stability Licensing Calculations Including Option 1-0 and II Long Term Solutions," as approved by the NRC, provided additional qualification calculations to support the elimination of the 0.15 adder.

The NRC-approved PRIME thermal/mechanical methods will replace GESTR-LOCA and GESTR Mechanical.

The TRACG04P computer code was used to perform plant- and cycle-specific DIVOM analysis. DIVOM is one component of the detect and suppress licensing methodology for stability Option II Long-Term Solutions. The original generic DIVOM analysis used TRACG02.

The method of evaluation used to generate core radionuclide inventories used in Chapter XV accident alternate source term dose analyses has changed for NMP1 Cycle 20. Previous analyses utilized the ORIGEN2 code, while the analyses performed for GNF2 implementation in Cycle 20 utilized the latest version of this code, ORIGEN-ARP. The fundamental difference between ORIGEN-ARP and ORIGEN2 is the burnup dependence of the cross-section libraries utilized in the point depletion calculation. ORIGEN2 does not update the cross-section library with increasing burnup during the depletion, but rather requires the user to initially select from a small set of pre-generated generic BWR cross-section libraries for the burnup of interest. ORIGEN-ARP allows assembly specific, burnup dependent, cross-section libraries to be utilized during the depletion.

The method of evaluation used to perform the Boral rack criticality analyses discussed in Chapter X.J.2.1 has changed for NMP1 Cycle 20. Previous analyses approved with License Amendment 167 utilized the CASMO-3 code, while the analyses performed for GNF2 implementation in Cycle 20 utilized a newer version of this code, CASMO-4.

ATTACHMENT 1
10CFR50.59 EVALUATION SUMMARY REPORT

Safety Evaluation No.: 2011-02 (cont'd.)

Safety Evaluation Summary:

The proposed activity uses methods of evaluation that have been approved by the NRC in a Safety Evaluation (SE) and NRC Regulatory Guide 1.183. The subject methods of evaluation are appropriate for the intended application and the terms and conditions for their use, as specified in the NRC SE, have been satisfied. By definition (a)(2) of 10CFR50.59, the new methods of evaluation are not considered a departure from methods described in the UFSAR. Therefore, from a criterion (viii) review, the use of NEDE-33213P-A for restricted region calculations, the use of PRIME for the above applications, the use of TRACG04P for DIVOM calculations, the use of ORIGEN-ARP for the above applications, and the use of CASMO-4 for the above applications for NMP1 do not require prior NRC review.

ATTACHMENT 2

NMP1 TECHNICAL SPECIFICATIONS BASES PAGES

**ATTACHMENT 2
TECHNICAL SPECIFICATIONS BASES PAGES**

**NINE MILE POINT UNIT 1 TECHNICAL SPECIFICATIONS BASES
INSERTION INSTRUCTIONS**

Remove the pages listed in the Remove column and replace them with the pages listed in the Insert column.

If there is an additional page being added to the Technical Specification Bases, dashes (-----) will be shown in the Remove column. Likewise, if a page is being removed with no replacement dashes (-----) will be shown in the Insert column.

<u>REMOVE</u>	<u>INSERT</u>
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LEP-2	LEP-2
LEP-3	LEP-3
LEP-4	LEP-4
LEP-5	LEP-5
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**NMP1 FACILITY OPERATING LICENSE (FOL) AND
TECHNICAL SPECIFICATIONS (TS)**

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(1) (B) denotes Bases page.

BASES FOR 3.0 LIMITING CONDITION FOR OPERATION AND 4.0 SURVEILLANCE REQUIREMENT APPLICABILITY

specified conditions are satisfied. In this case, this would mean that for one division the diesel generator power system must be operable (as must be the components supplied by the diesel generator power system) and the diesel generator must be running. In addition, all of the redundant systems, subsystems, trains, components, and devices in the other division must be operable, or likewise satisfy Specification 3.0.1 (i.e., be capable of performing their design functions and have the diesel generator power system operable, but with the diesel generator not running). In other words, both diesel generator power systems must be operable, with one diesel generator running, and all redundant systems, subsystems, trains, components, and devices in both divisions must also be operable. If these conditions are not satisfied, the plant is required to be placed in the condition stated in the applicable individual specification(s).

Additionally, Specification 3.0.1 delineates the action to be taken for circumstances not directly provided for in the specification condition statements, and whose occurrences would violate the intent of the specification. For example, certain specifications call for both subsystems in a two subsystem design to be operable and provide explicit action requirements if one (1) subsystem is inoperable. Under the terms of Specification 3.0.1, if both of the required subsystems are inoperable, the plant is required to take actions consistent with the specification. It is assumed that the plant is to be in at least the required operational condition within the required times by promptly initiating and carrying out the appropriate action statement.

- 3.0.8 LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered.

Each use of LCO 3.0.8 requires confirmation that at least one train (or subsystem) of systems supported by the inoperable snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. LCO 3.0.8 does not apply to non-seismic snubbers. In addition, a record of the design function of the inoperable snubber (i.e., seismic vs. non-seismic), implementation of any high risk configuration restrictions, and the associated plant configuration shall be available on a recoverable basis for inspection.

BASES FOR 3.0 LIMITING CONDITION FOR OPERATION AND 4.0 SURVEILLANCE REQUIREMENT APPLICABILITY

LCO 3.0.8 can only be used if one of the following two means of heat removal is available (high risk configuration restrictions):

- (1) At least one high pressure makeup path (e.g., High Pressure Coolant Injection) and heat removal capability (e.g., Electromagnetic Relief Valves with Containment Spray in Torus Cooling Mode, or Emergency Condensers), including a minimum set of supporting equipment required for success, not associated with the inoperable snubber(s),

OR

- (2) At least one low pressure makeup path (e.g., Core Spray) and heat removal capability (e.g., Electromagnetic Relief Valves with Containment Spray in Torus Cooling Mode, or Emergency Condensers, or shutdown cooling), including a minimum set of supporting equipment required for success, not associated with the inoperable snubber(s).

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

BASES FOR 3.0 LIMITING CONDITION FOR OPERATION AND 4.0 SURVEILLANCE REQUIREMENT APPLICABILITY

Specifications 4.0.1 through 4.0.3 establish general requirements applicable to all specifications in Sections 4.1 through 4.7 and apply at all times, unless otherwise stated.

4.0.1 Specification 4.0.1 establishes the requirement that SRs must be met during the applicable reactor operating or other specified conditions for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This specification is to ensure that surveillances are performed to verify the operability of systems and components, and that variables are within specified limits. Failure to meet a surveillance within the specified frequency, in accordance with Specification 4.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire surveillance is performed within the specified frequency.

Systems and components are assumed to be operable when the associated SRs have been met. Nothing in this specification, however, is to be construed as implying that systems or components are operable when either:

- a. The systems or components are known to be inoperable, although still meeting the SRs; or
- b. The requirements of the surveillance(s) are known to be not met between required surveillance performances.

Surveillances do not have to be performed when the unit is in a reactor operating or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a special test exception LCO are only applicable when the special test exception LCO is used as an allowable exception to the requirements of a specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given reactor operating or other specified condition.

Surveillances, including surveillances invoked by LCO actions, do not have to be performed on inoperable equipment because the applicable individual specifications define the remedial measures that apply. Surveillances have to be met and performed in accordance with Specification 4.0.2, prior to returning equipment to operable status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment operable. This includes ensuring applicable surveillances are not failed and their most recent performance is in accordance with Specification 4.0.2. Post maintenance testing may not be possible in the current reactor operating or other specified conditions in the LCO due to the necessary unit parameters not having been established. In these situations, the equipment may be considered operable provided

BASES FOR 3.0 LIMITING CONDITION FOR OPERATION AND 4.0 SURVEILLANCE REQUIREMENT APPLICABILITY

testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a reactor operating or other specified condition where other necessary post maintenance tests can be completed.

- 4.0.2 Specification 4.0.2 establishes the limit for which the specified time interval for SRs may be extended. It permits an allowable extension of the surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with a 24 month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the SRs. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.
- 4.0.3 Specification 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been completed within the specified frequency. A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time it is discovered that the surveillance has not been performed in accordance with Specification 4.0.2, and not at the time that the specified frequency was not met. This delay period permits the completion of a surveillance before complying with LCO actions or other remedial measures that might preclude completion of the surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.

When a surveillance with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to power operation, or in accordance with the 10 CFR 50 Appendix J Testing Program Plan, etc.) is discovered to not have been performed when specified, Specification 4.0.3 allows for the full delay period of up to the specified frequency to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity.

BASES FOR 3.0 LIMITING CONDITION FOR OPERATION AND 4.0 SURVEILLANCE REQUIREMENT APPLICABILITY

Specification 4.0.3 provides a time limit for, and allowances for the performance of, surveillances that become applicable as a consequence of operating condition changes imposed by LCO actions.

Failure to comply with specified frequencies for surveillance requirements is expected to be an infrequent occurrence. Use of the delay period established by Specification 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend surveillance intervals. While up to 24 hours or the limit of the specified frequency is provided to perform the missed surveillance, it is expected that the missed surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required or shutting the plant down to perform the surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance. The risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determines the risk management action thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed surveillances will be placed in the Corrective Action Program.

If a surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable then is considered outside the specified limits and entry into the applicable LCO actions begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and entry into the applicable LCO actions begin immediately upon failure of the surveillance.

Completion of the surveillance within the delay period allowed by this specification, or within the times allowed by LCO actions, restores compliance with Specification 4.0.1.

BASES FOR 3.1.2 AND 4.1.2 LIQUID POISON SYSTEM

The liquid poison system also has a post-LOCA safety function to buffer the suppression pool pH in order to maintain the bulk pH above 7.0. This function is necessary to prevent iodine re-evolution consistent with the Alternative Source Term analysis methodology. Manual system initiation is used, and the minimum amount of sodium pentaborate solution required to be injected for suppression pool pH buffering is 1114 gallons at a minimum concentration of 9.423 weight percent. This volume consists of the minimum required volume of 1325 gallons minus the 197 gallons that are contained below the point where the pump takes suction from the storage tank and minus 14 gallons that are assumed to remain in the pump suction and discharge piping after injection stops. Operation of a single liquid poison pump can satisfy this post-LOCA function. This function applies to the power operating condition, and also whenever the reactor coolant system temperature is greater than 212°F except for reactor vessel hydrostatic or leakage testing with the reactor not critical.

Specification 3.1.2.e requires initiation of a normal orderly plant shutdown within one hour if Specifications 3.1.2.a through 3.1.2.d are not met. Specifically, the plant must be brought to a reactor operating condition in which the LCO does not apply. To achieve this status, the reactor coolant system temperature must be reduced to $\leq 212^\circ\text{F}$ by initiating a normal orderly shutdown using the normal plant shutdown procedure. Based on operating experience, the use of the normal plant shutdown procedure to achieve the plant shutdown results in a reasonable time to reach the required plant conditions from full power operating conditions in an orderly manner and without challenging plant systems.

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than 7 days. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

The system test specified demonstrates component response such as pump starting upon manual system initiation and is similar to the operating requirement under accident conditions. The only difference is that demineralized water rather than the boron solution will be pumped to the reactor vessel. The test interval between operating cycles results in a system failure probability of 1.1×10^{-6} (Fifth Supplement, p. 115)* and is consistent with practical considerations.

Pump operability will be demonstrated on a more frequent basis. A continuity check of the firing circuit on the explosive valves is provided by pilot lights in the control room. Tank level and temperature alarms are provided to alert the operator of off-normal conditions.

The functional test and other surveillance on components, along with the monitoring instrumentation, gives a high reliability for liquid poison system operability.

*FSAR

BASES FOR 3.2.1 REACTOR VESSEL HEATUP AND COOLDOWN

Design calculations reported in Volume I, Section V-A, 4.0 (page V-6)* have demonstrated that the heatup and cooldown rate of 100°F/hr considered in the fatigue analysis will result in stresses well within code limits. A series of calculations have demonstrated that various extreme heatup and cooldown transients result in thermal strains well within the ASME Code limits stated in Volume I, Section V-C, 3.0 (p. V-19)*. Cooldown incidents include: failure of the pressure regulator leading to a cooldown of 215°F in 5.5 minutes (Appendix E-I, 3.15 (p. E-45))* , inadvertent opening of a single solenoid-actuated pressure relief valve leading to a cooldown of 1050°F/hr sustained for 10 minutes (Vol. I, Section V-B, 1.3 (p. V-11))* , and finally, opening all six of the solenoid-actuated relief valves leading to a cooldown of 250°F in 7.5 minutes (Volume IV, Section I-B)*. Reactor vessel heatup of 300°F/hr (Volume IV, Section I-B)* also demonstrates stresses well within the code requirements.

The maximum allowable reactor vessel heatup and cooldown rates during normal startup and shutdown operations are specified in the Pressure and Temperature Limits Report (PTLR). These limits affect the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leakage and hydrostatic testing pressure-temperature (P-T) limit curves that are contained in the PTLR. Thus, operation within the specified limits for rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P-T limit curves.

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BASES FOR 3.2.2 AND 4.2.2 MINIMUM REACTOR VESSEL TEMPERATURE FOR PRESSURIZATION

The Pressure and Temperature Limits Report (PTLR) establishes the methodology for determining pressure-temperature (P-T) limits and contains the P-T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing. The heatup curve provides limits for both heatup and criticality. Each P-T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. The pressure values on the curves have been adjusted to account for instrument uncertainties and to reflect the calculated elevation head difference between the pressure sensing instrument locations and the pressure-sensitive area of the core beltline region. The temperature values on the curves have been adjusted to account for instrument uncertainties.

10 CFR 50, Appendix G, requires the establishment of P-T limits for material fracture toughness requirements of the reactor coolant pressure boundary materials. 10 CFR 50, Appendix G requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic and leak tests, and mandates that the P-T limits be at least as conservative as limits obtained by following the methods of Appendix G to Section XI of the ASME Code. The operating limits specified in the PTLR provide a margin to brittle failure of the reactor vessel and ensure that the requirements of 10 CFR 50 Appendix G are satisfied.

The P-T limit curves are established based on limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P-T limit curves, different locations are more restrictive. In addition, heatup operations represent a different set of restrictions than cooldown operations because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls. The P-T limit curves reflect the most restrictive results from consideration of heatup and cooldown operations. The criticality limits include the 10 CFR 50, Appendix G requirement that they be at least 40°F above the heatup curve or the cooldown curve and not lower than the minimum permissible temperature for inservice leakage and hydrostatic testing.

The reactor vessel head flange and vessel flange in combination with the double "O" ring type seal are designed to provide a leak-tight seal when bolted together. When the vessel head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the vessel head is flexed slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flanges. The minimum vessel flange and head flange temperature for bolting is established in the PTLR.

BASES FOR 3.2.7.1 AND 4.2.7.1 REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE (PIV) LEAKAGE

The function of reactor coolant system (RCS) PIVs is to separate the high pressure RCS from an attached low pressure system. This protects RCS pressure boundary described in 10 CFR 50.2 and 10 CFR 50.55a(c) (Refs. 1 and 2). The PIVs, which are listed in the NMP1 UFSAR (Reference 3), are designed to meet the requirements of Reference 4. During their service lives, these valves can exhibit varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. Leakage through these valves is not included in any allowable leakage specified in Specification 3.2.5, "Reactor Coolant System Leakage."

The RCS PIV Specification allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety. Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed event that could degrade the ability for low pressure injection. In Reference 5, it was concluded that periodic leakage testing of the PIVs can substantially reduce intersystem LOCA probability.

This Specification applies in the power operating and hot shutdown reactor operating conditions because the PIV leakage potential is greatest when the RCS is pressurized. In the cold shutdown, refueling, and major maintenance reactor operating conditions, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment. Accordingly, the potential for the consequences of reactor coolant leakage is far lower during these conditions.

Note 1 to Specification 3.2.7.1 has been provided that allows separate Condition entry for each affected RCS PIV flow path because the actions in Specification 3.2.7.1.b provide appropriate compensatory measures for separate, affected RCS PIV flow paths. Note 2 to Specification 3.2.7.1 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function. As a result, the applicable actions for systems made inoperable by PIVs must be entered. This ensures appropriate remedial actions are taken, if necessary, for the affected systems.

If leakage from one or more RCS PIVs is not within limit, the flow path must be isolated by at least one closed manual, deactivated automatic, or check valve within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the flow path if leakage cannot be reduced while corrective actions to reseal the leaking PIVs are taken. The 4 hours allows time for these actions and restricts the time of operation with leaking valves.

Specification 3.2.7.1.b.2 specifies that the double isolation barrier of two valves be restored by closing another valve qualified for isolation or restoring one leaking PIV. The 72 hour time limit considers the time required to complete the action, the low probability of a second valve failing during this time period, and the low probability of a pressure boundary rupture of the connected low pressure piping when overpressurized to reactor pressure (Ref. 6).

BASES FOR 3.2.7.1 AND 4.2.7.1 REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE (PIV) LEAKAGE

Valves used for isolation must meet the same leakage requirements as the PIVs and must be on the RCPB or the high pressure portion of the system.

If leakage cannot be reduced or the system isolated, the plant must be brought to an operating condition in which the Specification does not apply. To achieve this status, an orderly shutdown must be initiated within one hour and the plant be brought to the cold shutdown condition within 10 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside the containment.

Performance of leakage testing on each primary coolant system PIV is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve size up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition. For two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Reference 4 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential). The observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one-half power.

The 24-month Frequency required by the Inservice Testing Program is based on the ASME OM Code Frequency. Specification 4.2.7.1 is modified by a Note that states the leakage Surveillance is not required to be performed in the hot shutdown condition. Entry into this condition is permitted for leakage testing at high differential pressures with stable conditions which is not possible in the cold shutdown or refueling conditions.

References:

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. UFSAR, Section V-D.2.3.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. Letter from T. A. Ippolito (NRC) to D. P. Dise (NMPC) dated April 20, 1981, "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," included attached Technical Evaluation Report TER-C5257-237, Rev. 1, dated March 20, 1981.
6. NEDC-31339, "BWR Owners Group assessment of Emergency Core Cooling System Pressurization in Boiling Water Reactors," November 1986.

BASES FOR 3.3.7 AND 4.3.7 CONTAINMENT SPRAY SYSTEM

In conjunction with containment spray pump operation during each operating cycle, the raw water pumps and associated cooling system performance will be observed. The containment spray system shall be capable of automatic initiation from simultaneous low-low reactor water level and high containment pressure. The associated raw water cooling system shall be capable of manual actuation. Operation of the containment spray system involves spraying water into the atmosphere of the containment. Therefore, periodic system tests are not practical. Instead separate testing of automatic containment spray pump startup will be performed during each operating cycle. During pump operation, water will be recycled to the suppression chamber. Also, tests to verify that the drywell and torus spray nozzles are free from obstructions will be performed following maintenance that could result in nozzle blockage. As an alternative, a visual inspection (e.g., boroscope) of the nozzles or piping could be utilized in lieu of an air test if a visual inspection is determined to provide an equivalent or more effective post-maintenance test. A visual inspection may be more effective if the potential for material intrusion is localized and the affected area is accessible. Maintenance that could result in nozzle blockage would be those maintenance activities on any loop of the containment spray system where the Foreign Material Exclusion program controls were deemed ineffective. For activities such as valve repair/replacement, a visual inspection would be the preferred post-maintenance test since small debris in a localized area is the most likely concern. An air test may be appropriate following an event where a large amount of debris potentially entered the system or water was actually discharged through the spray nozzles. Design features are discussed in Volume I, Section VII-B.2.0 (page VII-19)*. The valves in the containment spray system are normally open and are not required to operate when the system is called upon to operate.

The test interval between operating cycle results in a system failure probability of 1.1×10^{-6} (Fifth Supplement, page 115)* and is consistent with practical considerations. Pump operability will be demonstrated on a more frequent basis and will provide a more reliable system.

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ATTACHMENT 3

**REPORT CONSISTENT WITH 10 CFR 54.37(B) ON HOW EFFECTS OF AGING
OF NEWLY-IDENTIFIED STRUCTURES, SYSTEMS, OR COMPONENTS ARE
MANAGED**

ATTACHMENT 3
REPORT CONSISTENT WITH 10 CFR 54.37(b) ON HOW EFFECTS OF AGING OF
NEWLY-IDENTIFIED STRUCTURES, SYSTEMS, OR COMPONENTS
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This report is in lieu of adding a level of detail to the Nine Mile Point Unit 1 Updated Final Safety Analysis Report (UFSAR) that is greater than in the remainder of the UFSAR, including the License Renewal Supplement. An entry on the NRC website, "Frequently Asked Questions (FAQs) About License Renewal Inspection Procedure (IP) 71003, 'Post-Approval Site Inspection for License Renewal,'" relates to the amount of detail required per 10 CFR 54.37(b). It states, "The NRC staff will consider it acceptable if the summary information included in the FSAR update is consistent with the requirements of 10 CFR 54.21(d), and the guidance provided in Revision 1 of NUREG-1800, 'Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants' (SRP-LR), provided that the licensee has supplied the technical details (as described in RIS 2007-16) in another documented submittal to the NRC." The information in this report is consistent with the technical information previously submitted to the NRC with the Amended License Renewal Application (ALRA).

On July 14, 2005, Nine Mile Point Nuclear Station, LLC (NMPNS) submitted an ALRA to the NRC to renew the operating licenses for Nine Mile Point Nuclear Station Unit 1 (NMP1) and Unit 2 (NMP2) for an additional 20 years beyond the original expiration dates of August 22, 2009 (NMP1) and October 31, 2026 (NMP2). Within the ALRA, system tables were provided to define the component types, functions and the Aging Management Programs that applied. Lists of individual components within scope of license renewal were not required to be provided.

Subsequent to the completion of the necessary reviews, audits, responses to Requests for Additional Information (RAIs), and resolutions of other questions, the NRC published NUREG-1900, Safety Evaluation Report Related to the License Renewal of Nine Mile Point Nuclear Station, Units 1 and 2, in September of 2006, which documented the NRC staff's review of the information submitted to them through April 21, 2006. The renewed operating licenses for NMP1 and NMP2 were issued on October 31, 2006, extending the license for NMP1 to August 22, 2029, and NMP2 to October 31, 2046.

For holders of a renewed operating license, 10 CFR 54.37(b) requires that newly-identified Structures, Systems, or Components (SSCs) be included in the Final Safety Analysis Report (FSAR) update required by 10 CFR 50.71(e) describing how the effects of aging will be managed. Newly-identified SSCs are those SSCs that were installed in the plant at the time of the License Renewal of NMP1 and NMP2, but were not evaluated as part of the ALRA (as discussed in RIS 2007-16).

There were no newly-identified SSCs during the period of May 2009 to April 2011.