Enclosure to
NMP1L 0512

# NINE MILE POINT - UNIT 1

# SAFETY EVALUATION SUMMARY REPORT

# 1990

Docket No. 50-220
License No. DRP-63

| | |
|---|---|
| **Safety Evaluation No.:** | 74-01 |
| **Implementation Document No.:** | Mod. N1-74-001 |
| **UFSAR Affected Pages:** | XV-22, Fig. V-1 |
| **System:** | Recirculation System |
| **Title of Change:** | Wiring Change to Recirculation Bypass Valves |

**Description of Change:**

The FSAR (Updated) is being revised to reflect a change in the normal position of recirculation pump discharge bypass valves from closed to open. These changes were inadvertently omitted during previous FSAR updates. To eliminate the temperature differential between the bypass and the main recirculation line and, in addition, eliminate the "dead leg" condition which is a potential cause of stress corrosion cracking, a modification has been completed on the recirculation bypass valves. The bypass valves will be operated as before, but with an additional mode of operation, these valves could be open when the recirculation valves are open. This is accomplished by simple wiring changes to the interlock circuitry for the bypass valve.

**Safety Evaluation Summary:**

Supplement 1 to SIL No. 104 (original SIL issued October 18, 1974) presents the GE-NED safety evaluation for operation with the recirculation system isolation valve bypass line open.

The General Electric analysis has determined that:

1. The modification has no effect on primary system coolant flow.

2. The modification has no effect on total core steady state and transient operation.

3. No analytical results of any accident analysis have been changed.

Based on the analysis and evaluation performed, this modification does not involve an unreviewed safety question.

NOTE: This safety evaluation is being reported at this time in support of the changes to FSAR (Updated) Sections V and XV.

| | |
|---|---|
| **Safety Evaluation No.:** | 85-060, Rev. 4 |
| **Implementation Document No.:** | Mod. 84-42 |
| **UFSAR Affected Pages:** | N/A |
| **System:** | Motor Generator |
| **Title of Change:** | Motor Generator Set Reliability |

**Description of Change:**

This modification affects the control circuits to motor generator sets 161, 162, 167, 171 and 172. These changes involve several items:

1. Replacement of the 83 and 83A, and 83B relays,
2. Addition of potentiometers with numerical settings,
3. Installation of alarm contacts for loss of DC for 162, 172 & 167,
4. Removal of certain 83 relay contacts not needed for 161 & 171,
5. Removal of an 83B relay contact not needed for 162, 172 and 167.

The control circuits of motor generators sets 161, 162, 171 and 172 are safety-related. Motor generator set 167 control circuits are nonsafety-related..

**Safety Evaluation Summary:**

By performing this modification, the motor generator reliability and maintainability will be improved.

This modification to the MG Sets does not change the facility or procedures as described in the FSAR. The modification will not adversely affect the safe operation or shutdown of Nine Mile Point Unit 1 and does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 87-015

**Implementation Document No.:** Mod. N1-86-11

**UFSAR Affected Pages:** Table XV-4

**System:** Main Steam

**Title of Change:** Remove Warmup Valves

**Description of Change:**

Each steam line has an AC motor-operated isolation valve inside containment and an air-operated isolation valve outside containment. Each air-operated MSIV is equipped with a two-inch bypass line, each with an air-operated bypass valve. This bypass line is used to warmup the downstream steam lines. The bypass valves (01-05 and 01-06) have solenoids powered from the 125 VDC system. The valves "fail closed" on loss of air or DC power. The valves isolation on the same signals as the MSIVs.

This modification consists of removing the bypass valves and capping the line. This modification would leave the manual local leak rate test valves intact and would not affect the emergency condenser vent line connection (downstream of 01-05).

**Safety Evaluation Summary:**

The warmup valves have a history of failing the local leak rate tests. Many man-rem of exposure are expended each refueling to test, repair and retest these valves. Removal of these valves would result in a reduction of exposure and would also eliminate a potential leak path from the containment.

In certain post-accident conditions, containment heat removal could be accomplished by reopening the MSIVs and using the condenser as a heat sink. The MSIVs can be reopened without using the warmup valves under these postulated conditions without causing a thermal shock to the steam lines. Consequently, the warmup valves are not required.

This modification does not constitute an unreviewed safety question. Removal of the main steam line warmup valves will not affect the normal operation of the plant and will not affect the plant's ability to respond to accident situations. The modification will be conducted in accordance with the rules of the ASME Code, Section XI, 1983 Edition, Summer 1983 addenda. A hydrostatic test of the modification will be performed.

**Safety Evaluation No.:** 87-015 (Continued)

**Safety Evaluation Summary:** (Continued)

This change was reviewed and approved by the NRC in License Amendment No. 96, issued March 25, 1988, which revised Technical Specification Table 3.2.7 to delete the main steam warmup valves. The corresponding revision to FSAR (Updated) Table VI-3a was incorporated in FSAR Revision 7.

---

**Safety Evaluation No.:** 88-003, Rev. 1

**Implementation Document No.:** Mod. N1-87-066

**UFSAR Affected Pages:** IX-24

**System:** Fuel Oil Handling and Storage - System 82

**Title of Change:** Emergency Generators Diesel Fuel Storage Tank Replacement

**Description of Change:**

The Environmental Protection Agency (EPA) and New York State Department of Environmental Conservation (DEC) regulations require tightness testing of underground petroleum storage tanks. Tanks which are not tested must be replaced. NMPC considered it prudent to replace the existing tanks. The existing tanks will remain in service until the new double-walled tanks are installed. They will be installed north of the existing tanks. An oil spill collection tank and concrete paved spill pad will be added around the fill points of the new tanks.

**Safety Evaluation Summary:**

The replacement of the existing diesel generator fuel oil storage tanks with new double-walled tanks does not constitute an unreviewed safety question. This modification will upgrade the tanks so they conform with the DEC and EPA regulations.

---

**Safety Evaluation No.:** 88-011

**Implementation Document No.:** N/A

**UFSAR Affected Pages:** N/A

**System:** Process Computer

**Title of Change:** Software Enhancement

**Description of Change:**

The current version of process computer software (version GEXL-PLUS-15) cannot take advantage of new features built into the Reload 11 fuel. General Electric developed improved software (NFD/PC, the New Fuel Design Process Computer model) specifically to support Reload-11's GE8-model fuel as well as all previous designs.

Previous GE fuel designs used one enrichment (with natural uranium ends) and at most two gadolinia concentrations throughout the bundle, but the Reload-11 and later fuel designs will have multiple enrichment and gadolinia regions. The GEXL-PLUS-15 software could not have handled all of the datasets necessary nor modeled all the regions independently, and would have had to use overly-conservative correlations for the critical regions of the fuel. The new NFD/PC software will model each region with the correlations appropriate for that fuel, thus eliminating unnecessary conservatisms.

**Safety Evaluation Summary:**

The process computer does not initiate any safety systems; however, it is used to calculate fuel thermal limits, which are covered by Technical Specifications 3.1.7 and 4.1.7 and required daily. (The acceptability of the accuracy of these calculations is determined by General Electric monthly or bi-monthly, when they compare the results of independent computer analyses to results from the process computer.)

Based on the analysis, the planned extensive testing (both on-site and elsewhere), the historical core performance, and the use in other BWRs, this change does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 88-018

**Implementation Document No.:** Temporary Procedure N1-88-6-12

**UFSAR Affected Pages:** N/A

**System:** Core Spray

**Title of Change:** Core Spray Pump Recirculation Line Operability Test

**Description of Change:**

This operability test will determine flows at which acceptable piping vibration levels can be attained during the quarterly surveillance test and ensure core spray pump operability in accordance with Bulletin 88-04 requirements while system flow is only through the pump recirculation line. The Core Spray Operability Test is divided into two parts: part one, Piping Vibration Test and part two, Pump Recirculation Test.

**Safety Evaluation Summary:**

This test will be run during the 1988 Refueling Outage while the core is off-loaded. The applicable parts of the core spray system will be declared (per procedure) inoperable during the test. These requirements will allow the plant to run the test without having a Limiting Condition for Operation (LCO) per Technical Specifications 3.1.4a, 3.1.4b and 3.1.4d. Phase 4 and 5 of the test will confirm the system has not been damaged. If any damage has occurred it will be repaired prior to allowing the system to be declared operational.

Since portions of the core spray system will be declared inoperable and the core off-loaded during this test, the test will not adversely affect the safe operation or shutdown of Nine Mile Point Unit 1 and does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 89-004, Rev. 1

**Implementation Document No.:** LDCN U-N68

**UFSAR Affected Pages:** VII-62

**System:** High Pressure Coolant Injection (Feedwater)

**Title of Change:** FSAR Revision

**Description of Change:**

The FSAR text is updated to reflect calculations and other Niagara Mohawk letters responding to SSFI questions. This change to the FSAR adds an explanation of the HPCI off-site power requirements and distinguishes between normal off-site power and limited off-site power from Bennetts Bridge Hydro Station that limits HPCI to one train of feedwater system pumps.

**Safety Evaluation Summary:**

These changes in the FSAR text for the HPCI/Feedwater system are administrative in nature and do not affect the safe operation or shutdown capability of Nine Mile Point Unit 1.

| | |
|---|---|
| **Safety Evaluation No.:** | 89-006 |
| **Implementation Document No.:** | N/A |
| **UFSAR Affected Pages:** | N/A |
| **System:** | N/A |
| **Title of Change:** | Vice President - Quality Assurance Reporting to Executive Vice President - Nuclear Operations |

**Description of Change:**

The Vice President - Quality Assurance formerly reported to the President of Niagara Mohawk. This was changed as of March 1, 1989, when the President revised the organization. The Vice President - Quality Assurance will now report to the Executive Vice President - Nuclear Operations.

**Safety Evaluation Summary:**

The change in the reporting structure for the Vice President - Quality Assurance will not have any effect on the safe operation of any system or safe shutdown of the plants and does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 89-007

**Implementation Document No.:** Mod. 89-016

**UFSAR Affected Pages:** Fig. VI-24

**System:** Reactor Building Ventilation

**Title of Change:** Airlocks - Removal of Exhaust Air System

**Description of Change:**

Minor modification 89-016 was initiated as a result of Problem Report 918. The air exhaust duct in both Turbine Building/Reactor Building airlocks were "blocked" closed. This minor modification will remove the affected ductwork back to the exhaust system connections and seal the airlock exhaust vent openings in a permanent manner.

Engineering has determined that the air exhaust ducts are not required for ventilation since the airlocks are considered passageways utilized for momentary personnel transit only.

**Safety Evaluation Summary:**

This modification is necessary to maintain the integrity of the existing fire barriers. The removal of the air ducts will also enhance the integrity of the secondary containment.

The removal of the exhaust air ducts in the airlocks will not adversely affect the safe operation or shutdown of Nine Mile Point Unit 1 and does not constitute an unreviewed safety question.

| | |
|---|---|
| **Safety Evaluation No.:** | 89-011 |
| **Implementation Document No.:** | Mod. 85-092 |
| **UFSAR Affected Pages:** | N/A |
| **System:** | Reactor Water Clean-up System - System 33 |
| **Title of Change:** | 125 VDC System Cable Replacement for MOV 33-04 |

**Description of Change:**

This modification includes the upgrading of power supply cable to MOV Motor #33-04 and breaker interchange on the Valve Boards #11 and #12. The 100 AMP and 225 AMP breakers will feed Valve Boards #11 and #12, respectively. The power cable to MOV 33-04 will be replaced with a larger size cable to reduce the voltage drop, to ensure valve operability during degraded voltage condition. The MOV 33-04 and breakers in this modification are classified as safety-related.

**Safety Evaluation Summary:**

The changes proposed in this modification will not increase the probability of occurrence of an accident previously analyzed or reduce the margin of safety used in the basis for the Technical Specification as:

1. The cable replacement for MOV 33-04 will improve the operability by resolving the problem of minimum voltage requirement for starting torque.

2. The breaker interchange will improve the breaker coordination. Accordingly, system reliability will be enhanced.

The changes described in this modification neither create the potential for a new accident or malfunction, nor increase the consequences of a currently analyzed one. This modification does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 89-012

**Implementation Document No.:** LDCN U-N73, Rev. 1

**UFSAR Affected Pages:** Sections IV, V, VII, XV

**System:** Nuclear Fuel

**Title of Change:** Operation of Reload 11/Cycle 10

**Description of Change:**

The present core design consists of 532 bundles. These bundles are General Electric P8x8R bundles. Safety Evaluation 85-059 found these fuel designs and their present configuration acceptable. Reload 11 will consist of 176 fresh GE8x8EB (GE8 BD321B) fuel bundles, which is a different design from those used at Nine Mile Point Unit 1 in previous cycles. This design was approved by the NRC in GESTAR and reviewed by NMPC in Safety Evaluation 87-027.

This Safety Evaluation was originally based on refueling after shutdown within the established Cycle 9 core exposure range of 9750 to 10340 MWD/ST. Since Cycle 9 was terminated outside this range, a new core loading plan was developed and the limiting transients were re-analyzed to confirm the conclusions of the original safety evaluation. As a result of the transient and LOCA Basedeck Reviews, a need to place additional conservatisms in the transient and LOCA basedecks was discovered, and as a result the LOCA and feedwater controller failure analyses were re-run and new MAPLHGR and MCPR limits calculated.

**Safety Evaluation Summary:**

The majority of the changes caused by Reload 11 were approved in Safety Evaluation 87-027. However, General Electric has since provided new LOCA and transient analyses results. The new inputs to and results from the GE analyses were evaluated under this safety evaluation. These changes do not constitute an unreviewed safety question.

**Safety Evaluation No.:** 89-013, Rev. 1

**Implementation Document No.:** Mod. N1-89-131

**UFSAR Affected Pages:** Sections IV, VI, VII, XV

**System:** Containment Spray System

**Title of Change:** Appendix J - Water Seal
a) Cross-over Tie Valve Modifications
b) Establishment and Maintenance of Water Seal
c) Development of Water Seal Procedure

**Description of Change:**

Niagara Mohawk initially indicated that it would develop an accident procedure (modify the existing Containment Spray System Operating Procedure N1-OP-14) to accomplish the goal of establishing and maintaining the water seal on the valves during the worst case conditions. After evaluation of the various alternatives, it has been decided to provide a short term (Operating Cycle No. 11) water seal utilizing the containment spray pump to keep both the primary and secondary spray headers filled with water. In order to accomplish this, two of the torus test line tie valves between the primary and secondary spray loops will be open throughout all modes of operation. Operating Procedure N1-OP-14 Containment Spray System will be revised to incorporate this water seal methodology.

**Safety Evaluation Summary:**

The evaluation/analysis of this change considered:

1) The specific proposed containment spray cross-over tie valve modifications.

2) The adequacy of the use of the containment spray system to provide an internal, self-actuated and maintained water seal feature to the subject valves.

3) The possible use of the raw water service water system intertie as a backup external, indirect water seal source option to the subject valves.

**Safety Evaluation No.:** 89-013, Rev. 1 (Continued)

**Safety Evaluation Summary:** (Continued)

4) The necessary operator actions that would be expected to establish, maintain and monitor the subject water seal operation.

The evaluation considered impact on documentation, plant operations and system performance. This evaluation demonstrates that the subject proposed modifications and changes are acceptable and that no unreviewed safety question exists.

By letter dated March 20, 1990, the NRC issued a safety evaluation which concluded that the proposed procedure for establishing the containment spray water seal was acceptable.

| | |
|---|---|
| **Safety Evaluation No.:** | 89-014 |
| **Implementation Document No.:** | Mod. N1-89-900 |
| **UFSAR Affected Pages:** | VIII-28, Fig. VIII-6 |
| **System:** | Neutron Monitoring System - Source Range Monitors (SRM) |
| **Title of Change:** | SRM Control Rod Withdrawal Permissive Setpoint |

**Description of Change:**

The purpose of the change is to clarify page VIII-28 of the FSAR. Page VIII-28 of the FSAR indicates that control rods can be withdrawn as long as the count rate is above $10^3$ cps. In fact, the count rate should be 100 cps.

**Safety Evaluation Summary:**

The setpoint described in the FSAR is considered a typographical error. The count rate of 100 cps has been the setpoint used in past operation per the Technical Specifications and calibration procedures. As such, this setpoint change does not pose an unreviewed safety question.

**Safety Evaluation No.:** 89-015

**Implementation Document No.:** N/A

**UFSAR Affected Pages:** N/A

**System:** Fuel

**Title of Change:** Full-Core Reload with a Complete Set of Blade Guides and Coincident Reactor Protection System (RPS) Logic

**Description of Change:**

This safety evaluation addresses an alternate method considered acceptable for Nine Mile Point Unit 1 reload. This method (coincident RPS logic with a complete set of blade guides) eliminates spurious scram concerns, but would deviate from General Electric's recommendation in ways that are judged not to reduce the margin of safety.

**Safety Evaluation Summary:**

This method of reload would not have full scrams initiated on single SRM, IRM or APRM trips since coincident RPS logic is in place. This method inserts all blades with the help of blade guides. A full core control rod withdrawal block will be inserted while fuel load is taking place. Coincident RPS logic requires two signals in different RPS channels of either IRMs or APRMs tripped simultaneously to initiate a full reactor scram. Coincident RPS logic (normal logic) does not provide scram protection for the partially loaded core in the early stages where two channels of IRMs are not adjacent to or surrounded by fuel. To rectify this, all blades will be inserted and a control rod withdrawal block will be implemented manually to ensure all rods stay at the full-in position. No one-rod-out refuel interlock jumpers need to be used in this method since all of the rods are full-in during the entire core load; i.e., the interlocks are fully operable.

Based on the safety evaluation, this modification does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 89-016, Rev. 1

**Implementation Document No.:** N/A

**UFSAR Affected Pages:** N/A

**System:** Radwaste

**Title of Change:** Storage of Radwastes on Elevation 225' of Waste Disposal Building

**Description of Change:**

During a plant start-up in July 1981 following an extended refuel and maintenance outage, problems were observed in the Reactor Water Clean-up System heat exchanger. Investigation of this problem resulted in a perturbation in the Reactor Building Closed Loop Cooling (RBCLC) System requiring the removal of the waste concentrator from service (the waste concentrator is cooled by the RBCLC System). The removal of the concentrator from service limited the ability to process high conductivity water and this in turn resulted in a substantial water inventory in the Waste Building. Concurrently, difficulties occurred in the processing of low conductivity water further compounding the water inventory problems. A piping failure in the Waste Building allowed high conductivity water to infiltrate the Low Conductivity System. The use of the Low Conductivity System required frequent filter change-outs and demineralizer regeneration, which further contributed to the water inventory. This necessitated the use of the lower elevation of the Waste Building for water and filter sludge inventory storage until the waste processing systems could be repaired and returned to normal service.

**Safety Evaluation Summary:**

This safety evaluation addresses the use of the drum storage area of the 225' elevation of the Waste Disposal Building for storage of liquid/spilled radwastes. The evaluation incorporates future storage, including storage until such time as elevation 225' decontamination is completed.

Allowing the 225' elevation to be used for the storage of liquid/spilled radwastes will not adversely affect the public health and safety. This conclusion is based on the following factors: a) the buildings' features (e.g., essentially no permeability through walls and floor due to thickness, floor topping and waterstops); and b) the lack of any indication of leakage out of the building as indicated by the grab samples taken from the storm sewer system started in 1979.

**Safety Evaluation No.:** 89-016, Rev. 1 (Continued)

**Safety Evaluation Summary:** (Continued)

No equipment is being added, deleted or modified in conjunction with this evaluation. ALARA concepts are being incorporated into the maintenance of the 225' elevation. Thus, safe operation or shutdown of Nine Mile Point Unit 1 will not be adversely affected.

Based on this evaluation, this change does not involve an unreviewed safety question.

**Safety Evaluation No.:** 89-017

**Implementation Document No.:** N/A

**UFSAR Affected Pages:** N/A

**System:** Shutdown Cooling System

**Title of Change:** Change in Design Basis for Shutdown Cooling System Isolation Valve 38-02

**Description of Change:**

Plant Technical Specifications require valve 38-02 to close within 40 seconds, but the specifications do not address the operational conditions other than to state that closure times are not expected to differ appreciably from accident conditions. While this is true for AC-powered valves, DC-powered valves must consider lower voltage at the valve motor than the rated voltage of the plant battery. Recent calculations show that for valve 38-02, the motor terminal voltage is 82 VDC (due to cable losses) with an initial battery voltage of 105 VDC. With 82 VDC at the motor terminals, valve 38-02 is calculated to close in 61 seconds with a differential pressure of 1250 psig across the valve. If the differential pressure is changed to 140 psig (120 psi reactor pressure plus static water head), the calculated applied DC voltage to the motor terminals increases because of lower line losses, and the valve will close in 38.3 seconds (calculated value) because the motor speed is increased as the applied voltage to the motor terminal increases.

**Safety Evaluation Summary:**

The change in the design basis differential pressure to 120 psig reactor pressure (plus static head) will not have a significant impact on the valve's ability to perform its required function of closing an an automatic or manual closure signal. The change will not adversely affect the safe operation or shutdown of Nine Mile Point Unit 1 and does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 89-018

**Implementation Document No.:** Mod. N1-88-121

**UFSAR Affected Pages:** X-3, Fig. X-1

**System:** Shutdown Cooling System, System #38

**Title of Change:** SDC Pump Pressure Switch Removal

**Description of Change:**

Problem Report #206 states that whenever the Shutdown Cooling (SDC) System is placed into service, the pump suction pressure switches (PS/RV12 A, B & C) are found to be inoperable. This is thought to be caused by over-ranging of the switches due to pressure leakage through system isolation valves which allows the switches to be subjected to reactor pressure (>1000 psig).

Experience has shown that isolating the switches until the system is placed in service does not work, as small amounts of leakage over a long period of time pressurize the switches anyway. Additionally, there is not a replacement switch with sufficient range (>1000 psig) that still has the sensitivity necessary to maintain a setpoint of 4 psig (decreasing).

This modification will permanently remove the pressure switches (RV12 A, B, & C) from the suction lines of Pumps 11, 12 & 13 of the Shutdown Cooling (SDC) System. The piping connections for the pressure switches will be capped. Additionally, the electrical contacts will be jumpered, the relays will be spared, and computer points (C199, D000 & D001) will be removed.

**Safety Evaluation Summary:**

The removal of the switches will not impact pump performance or system operation. The suction pressure at the pumps will be above 4 psi. This positive suction is due to elevation differences between the reactor and the pumps. An interlock in each pump's control circuitry prevents the pump from being started with any of its system isolation valves closed and will trip a running pump if any of its isolation valves leave the open position. Therefore, the pump has protection, other than the low suction pressure switch, from being operated on an isolated system. Based upon these and other considerations, removal of the pressure switches will not diminish pump or plant performance.

Based on this evaluation, this change does not involve an unreviewed safety question.

| | |
|---|---|
| **Safety Evaluation No.:** | 89-021 |
| **Implementation Document No.:** | LDCN U-N38 |
| **UFSAR Affected Pages:** | XI-10, XI-11, XV-52 |
| **System:** | Fuel |
| **Title of Change:** | Explicit Statement of Turbine Bypass Valve Capacity |

**Description of Change:**

FSAR Sections XI (page 10) and XV (page 11) currently describe the bypass system capacity as 40 percent of the turbine steam flow at the 1850 thermal megawatt initial power level, or that the turbine bypass valves are designed to pass up to 40 percent of the control valves wide open turbine steam flow. To avoid any possible confusion, and more accurately reflect design, the FSAR should be revised to describe the bypass system capacity as 40 percent of the rated steam flow at the 1850 thermal megawatt initial power level, and that the turbine bypass valves are designed to pass up to 2,901,000 lbm/hr at rated condition. FSAR Section XI Page 11 states that the condenser is designed to accommodate a load rejection of 40 percent of the turbine valves wide open load. This should be described as: 40 percent of the rated steam flow.

**Safety Evaluation Summary:**

This change will not affect any other activity or document other than the OPL-3 form submitted to General Electric for Reload Licensing. Since NMPC currently does not take credit in GE's transient analyses for the bypass valve capacity, this change will have no effect. Based on the evaluation, this change does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 89-022, Rev. 1

**Implementation Document No.:** Mod. N1-88-062, Rev. 1

**UFSAR Affected Pages:** Fig. VIII-2

**System:** Reactor Instrumentation

**Title of Change:** EOP, SOP - Isolation Bypass Jumpers to Perform Specified Steps of EOP - 3, 4, 4.1, and SOP - 3. Modify Reactor and Turbine Building Differential Pressure Gauge Scales

**Description of Change:**

Modification Request N1-88-062 was initiated to install Spade Lugs/Banana Jacks on terminals identified in the various EOPs and SOPs to enhance human factors. In addition, the scale for reactor building differential pressure was also changed.

**Safety Evaluation Summary:**

This modification could potentially affect nuclear safety in a way not previously evaluated in the Unit 1 FSAR if unauthorized use were to occur. This modification does not reduce the margin of safety as described in the FSAR due to existing administrative controls. These administrative controls include:

1) Restricted access to the control room.

2) Prior SSS approval for entry into the control panels where this sub-panel is located.

3) Procedural compliance including the fact that jumper use is only authorized by EOP and SOP directed actions.

**Safety Evaluation No.:** 89-022, Rev. 1 (Continued)

**Safety Evaluation Summary:** (Continued)

Since it was deemed that the ease with which the RPS could be bypassed could potentially increase the probability of occurrence of an accident, additional administrative control would be placed on this jumper panel. This will consist of a Plexiglas shield, locked over the panel. This would then require procedural action to remove the shield and insert the jumpers.

This evaluation demonstrates that the subject modifications are acceptable with the current design and operating basis and will not result in an unreviewed safety question. The addition of this shield panel is an additional administrative control that will offset the potential for an increase in the probability of an accident due to the facilitation of jumpering the RPS.

---

**Safety Evaluation No.:** 89-023

**Implementation Document No.:** Mod. 85-092

**UFSAR Affected Pages:** N/A

**System:** Automatic Depressurization System - System 66

**Title of Change:** 125 VDC System Redesign (ERV Upgrade)

**Description of Change:**

This modification will separate the ERV solenoid power supply from the logic control power to reduce the voltage drop by installing the new dedicated ERV solenoid power circuit.

This modification consists of 1) splitting of the power and control circuit functions via interposing relays, 2) changing the physical location of the power and control interface points to minimize the power cable lengths and to increase the power circuit cable sizes where practical, and 3) addition of DC power monitoring alarm.

**Safety Evaluation Summary:**

This change to the ERV circuitry is to ensure that required voltage can be delivered to the solenoid, that the required cable gauge for the power circuit limits the voltage drop due to line loss to within acceptable EQ voltages, and that annunciation is provided on loss of DC power. Based on the analysis and evaluation performed, it is concluded that this modification does not involve an unreviewed safety question.

**Safety Evaluation No.:** 89-025

**Implementation Document No.:** LDCNs U-N41, U-N42

**UFSAR Affected Pages:** IX-17, X-74

**System:** Fire Protection

**Title of Change:** Fire Protection for Cables in Trays which Run Through Hazardous Areas

**Description of Change:**

This safety evaluation addresses a change to the Nine Mile Point Unit 1 Fire Protection Program. Externally applied fire-resistant material (Flamemastic) has been applied to cables at Nine Mile Point Unit 1, in accordance with particular engineering designs, for both fire barrier penetrations and cable trays. Flamemastic has been applied to cables in trays to reduce exposed combustibles in Appendix R fire break zones to establish fire breaks, as deemed necessary by the Fire Hazards Analysis, and to protect cables in hazardous areas.

**Safety Evaluation Summary:**

Through this safety evaluation, NMPC has demonstrated that continuing to utilize Flamemastic for the protection of new cables run through trays in hazardous areas such as the diesel generator rooms, or in trays which run past motor control centers, powerboards and other equipment that will support a fire, is no longer necessary to ensure an adequate level of fire protection. The use of IEEE-383 qualified cable and the Appendix R evaluation for each modification provides a level of fire protection equivalent to or better than the level achieved when the original Flamemastic coating commitments were made. Present commitments and procedures will ensure that existing Flamemastic is maintained. The 20-ft. fire break zones in the reactor building and other fire breaks in the plant will also be maintained.

**Safety Evaluation No.:** 89-027, Rev. 1

**Implementation Document No.:** PR-1623

**UFSAR Affected Pages:** N/A

**System:** 96 (Diesel Generator Starting Air)

**Title of Change:** EDG Air Start System

**Description of Change:**

Portable air bottles will be used to recharge the diesel generator starting air system (as described in Damage Repair Procedures N1-EMP-DRP-005 and N1-EMP-DRP-008) to satisfy Appendix R safe shutdown requirements. This backup air supply system will be used only for the Appendix R fire event and is not intended to supplement and/or replace the diesel generator starting air system air compressors for any condition other than the postulated Appendix R event. The temporary air source is used to recharge one of the diesel generator air start system tanks to 165 psig.

**Safety Evaluation Summary:**

The Appendix R safe shutdown analyses demonstrate the successful mitigation of the Appendix R fire. The use of repair procedures to satisfy Nine Mile Point Unit 1 Appendix R cold shutdown requirements is permitted under 10CFR50 Appendix R Section III.G.1.b. As described in the FSAR, a minimum of five air bottles stored in the Nine Mile Point Unit 1 Storeroom will provide the capability to perform five successful air starts. This will not be compromised. Therefore, the diesel generator air start system will not function in any abnormal or unanalyzed condition. All diesel generator air start functional requirements will be satisfied and are assured since the resulting air capacity will satisfy the design basis requirements. Consequently, the diesel generator air start system will operate and satisfy all FSAR Section IX.B.4.1 operational requirements.

This Safety Evaluation has concluded that the use of portable air bottles to recharge the diesel generator starting air system to satisfy the Nine Mile Point Unit 1 Appendix R safe shutdown requirements, as described in the Damage Repair Procedures, N1-EMP-DRP-005 and N1-EMP-DRP-008, does not constitute an unreviewed safety question.

| | |
|---|---|
| **Safety Evaluation No.:** | 89-028 |
| **Implementation Document No.:** | Mod. N1-89-134 |
| **UFSAR Affected Pages:** | N/A |
| **System:** | Core Spray |
| **Title of Change:** | Core Spray Drag Valve Internal Replacement |

**Description of Change:**

Each of the core spray loops has a test line routed from between the isolation check valves and the outer motor-operated isolation valves back to the suppression chamber. These test lines are used to perform flow testing of the core spray pumps and topping pumps in accordance with Surveillance Test Procedure N1-ST-Q1. Each line has a pressure control valve (drag valve) located immediately downstream of the test line isolation valves. These drag valves are used to throttle the test line flow to the appropriate level. During testing, the drag valves are throttled back to 2200 gpm due to excessive pipe vibration at higher flows. This vibration is caused by cavitation in these valves.

To eliminate the problems and concerns that exist with the current surveillance procedure, the original valve manufacturer was asked to manufacture a new disk stack that would be able to better handle the high pressure drops and flow rates without cavitating.

**Safety Evaluation Summary:**

The parts used in the replacement of internals (disk stack and seat ring) in the drag valves (81-85 and 81-86) are nonpressure retention parts and will be fabricated in accordance with manufacturer's standard procedures and practices since there are no special code requirements that govern special parts such as these.

**Safety Evaluation No.:** 89-028 (Continued)

**Safety Evaluation Summary:** (Continued)

This modification does not constitute an unreviewed safety question. The function of the system remains unchanged. The modification only increases core spray system test line flows using new parts of similar design and materials.

---

**Safety Evaluation No.:** 89-030

**Implementation Document No.:** Mod. N1-89-215

**UFSAR Affected Pages:** N/A

**System:** 125 VDC Battery Boards

**Title of Change:** Installation of Class 1E Fuses at 125 VDC Battery Boards #11 & #12

**Description of Change:**

This modification upgrades the equipment which feeds the loads powered from the 125 VDC Battery Boards #11 & #12. Presently, these loads are fed from Battery Boards #11 and #12 through circuit breakers. However, based on short circuit calculations for the 125 VDC system, it was determined that the majority of the presently installed breakers do not provide sufficient short circuit interrupting capability. The reason for this condition is that the breaker interrupting ratings are less than the available short circuit currents, which means that the breakers do not operate reliably at these higher currents. In order to correct this condition, fuses will be added to replace the breakers. These fuses will be sized to clear the fault for the maximum available short circuit current.

**Safety Evaluation Summary:**

The deletion of the breakers and the addition of fuses will not increase the probability of occurrence of an accident previously analyzed in the SAR, nor will it decrease the margin of safety at Nine Mile Point Unit 1. Based on this evaluation, this change does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 89-031

**Implementation Document No.:** LDCN U-N45

**UFSAR Affected Pages:** X-62

**System:** Fire Suppression

**Title of Change:** Starting Air for Diesel Fire Pump

**Description of Change:**

Starting air for diesel engine 100-01 (for diesel fire pump 100-02) is supplied by two air receivers. Originally, these receivers were supplied with service air. In order to provide a reliable supply source, the air system is modified to supply these receivers with instrument air and provide service air as a backup.

**Safety Evaluation Summary:**

The transients and accidents analyzed in the FSAR are not affected either directly or indirectly by this change. This modification improves the fire protection capability of the plant by supplying starting air from a more reliable source.

Based on the analyses and evaluations performed, it is concluded that this change does not involve an unreviewed safety question.

---

**Safety Evaluation No.:** 89-033

**Implementation Document No.:** LDCN U-N49

**UFSAR Affected Pages:** Table VI-3a

**System:** Control Rod Drive (CRD) Scram Discharge Volume (SDV)

**Title of Change:** Air Operated SDV Vent and Drain Valves

**Description of Change:**

The Control Rod Drive (CRD) Scram Discharge Volume (SDV) is equipped with a pair of vent valves (IV 44.2-15 and -16) and a pair of drain valves (IV 44.2-17 and -18).

The closing times for these vent and drain valves are specified in Technical Specifications Table 3.2.7. The FSAR also specifies the closure times for the same valves in Table VI-3A. There is, however, a discrepancy in the closing times (for the vent valves) specified in these two documents. For vent valves IV 44.2-15 and -16, Table 3.2.7 in the Technical Specifications specifies 10-second closure time (using scram exhaust path), whereas Table VI-3A in the FSAR specifies 18-second closing time.

**Safety Evaluation Summary:**

In August 1984, Niagara Mohawk submitted an application for license amendment. Included in this application was an addition of the SDV vent and drain valves in Table 3.2.7 because they were inadvertently excluded in a previous revision to this table; i.e., Amendment No. 44, dated May 19, 1981. In the subject application, however, the closure time proposed for the SDV vent valves was erroneously changed to 18 seconds (in Table 3.2.7), whereas that for the drain valves remained 10 seconds.

This change to the FSAR is administrative in nature to make the FSAR consistent with the Technical Specifications, and does not affect the safe operation or shutdown capability of Nine Mile Point Unit 1.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

**Safety Evaluation No.:** 89-036, Rev. 1

**Implementation Document No.:** Special Operating Procedure (SOP) 5

**UFSAR Affected Pages:** N/A

**System:** Reactor Protection System Bus #11 and #12

**Title of Change:** Alternate RPS Instrumentation for MG Set 162 & 172 Load Shed to Support Appendix R Scenario for Reload with Existing Station Batteries.

**Description of Change:**

The Restart Action Plan (RAP) for NMP1 contains Specific Issue 18: 125 VDC Systems concerns. In addressing the operability and functional capabilities of the 125 VDC Systems, the Appendix R Fire Scenario was verified against design basis requirements and assumptions. The verification concluded that the batteries will not be able to perform their design function if drained by the MG Set 162 and 172 loads. To this end, it is necessary to load shed the Reactor Protection System (RPS) Motor-Generator (MG) Sets 162 and 172 within thirty minutes of the event initiation. This procedure change will load shed at 30 minutes into the event.

**Safety Evaluation Summary:**

This safety evaluation has concluded that the load shed of the RPS MG Sets 162 and 172 assures the Appendix R vital plant parameters are accounted for and that no unreviewed safety question exists. The 10CFR50 Appendix R safe shutdown capability requires monitoring of vital plant parameters for safe shutdown:

a. Scram Verification (not required for cold shutdown and refuel modes).

b. Reactor Vessel Coolant Level (achieved by alternate/substitute instrumentation tygon hose and/or pressure gauge).

**Safety Evaluation No.:** 89-036, Rev. 1 (Continued)

**Safety Evaluation Summary:** (Continued)

c. Reactor Vessel Pressure (not required for cold shutdown and refuel modes).

d. Torus Water Temperature (not required for cold shutdown and refuel modes).

The substitute/alternate instrumentation for reactor vessel coolant level is acceptable and has not degraded the Appendix R effectiveness for operation in the cold shutdown and refuel modes.

These conditions determine that no unreviewed safety question exists and that this change (procedural) will accomplish full compliance with Appendix R for Nine Mile Point Unit 1 reload.

---

**Safety Evaluation No.:** 89-038, Rev. 1

**Implementation Document No.:** Mod. N1-89-229

**UFSAR Affected Pages:** IX-26, IX-29

**System:** 125 VDC Batteries

**Title of Change:** Installation of Class 1E Batteries in 125 VDC Battery Rooms #11 & #12

**Description of Change:**

The objective of this modification is to upgrade the 125 VDC system to provide the required 125 VDC capacity with margin to support the Appendix R Load Scenario for Nine Mile Point Unit 1. The Appendix R Scenario is considered to be the worst case with respect to battery loading. Presently, Bus #11 and #12 loads are individually powered by C & D Type LCR-21 Batteries with a 1500 ampere-hour rating. In order to support the Appendix R load scenarios for Nine Mile Point Unit 1, the existing C & D Type LCR-21 batteries will be replaced with C & D Type LCR-33 batteries and racks.

**Safety Evaluation Summary:**

The replacement of the batteries will not increase the probability of occurrence of an accident previously analyzed in the SAR, nor will it decrease the margin of safety at Nine Mile Point Unit 1. This modification replaces existing batteries to provide sufficient power to support emergency loads on a loss of all AC power due to a fire (Appendix R scenario).

Based on this evaluation, the modification does not constitute an unreviewed safety question.

---

**Safety Evaluation No.:** 89-039

**Implementation Document No.:** LDCN U-N54

**UFSAR Affected Pages:** Fig. IX-6

**System:** Emergency Diesel Generators

**Title of Change:** Revise Figure IX-6 in NMP1 FSAR

Description of Change:

Figure IX-6 is being revised to eliminate the 3-second time delay in starting Core Spray Pumps 111 and 121, and to change the emergency diesel generator load profile between 10 minutes and 30 minutes to reflect the answer provided to the question on Page V-15 of the First Supplement to the FSAR.

Safety Evaluation Summary:

The 3-second time delay on Core Spray Pumps 111 and 121 was removed by a design modification in 1971. This results in faster pump acceleration and operation since it starts immediately after the diesel generator circuit breaker closes.

Figure IX-6 is also being revised to show the manual tripping of a containment spray pump prior to manual starting of a containment spray raw water pump at approximately 30 minutes to reflect the intent of the answer provided to the question on page V-15 of the First Supplement to the FSAR. Presently, analyses are not readily available that would permit manual tripping of the core spray topping pumps at approximately 10 to 30 minutes into a loss-of-coolant accident. Therefore, Figure IX-6 is being revised to show manual tripping of a redundant containment spray pump only.

Based on the analyses and evaluations performed, it is concluded that this change does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 89-040

**Implementation Document No.:** Mod. N1-89-900

**UFSAR Affected Pages:** X-62

**System:** Fire Suppression (100)

**Title of Change:** Fire Pump Start Settings

**Description of Change:**

Devices 100-156A and 100-157A are pressure switches which monitor the fire header and are associated with the starting of the electric pump and the diesel-driven pump respectively. Presently, both of these devices are set for contact closure at 100 psig. The proposed settings for devices 100-156A and 100-157A are 110 psig and 100 psig respectively.

Devices 100-260 and 100-261 are pressure switches which monitor air tanks 100-10A and 100-10B respectively, and automatically start the diesel on low air pressure. Presently, both of these switches are set at 75 psig. The proposed setting for devices 100-260 and 100-261 continues to be 75 psig.

**Safety Evaluation Summary:**

The proposed setpoint of 110 psig header pressure will activate the motor-driven fire pump when the header pressure drops to this level. If the flow demand for the system increases and the motor-driven pump cannot maintain the header pressure, the diesel fire pump will automatically start when the header pressure drops to 100 psig. This will ensure that there is sufficient water supply available for the expected fire scenario. Therefore, the existing margin of safety has not been reduced.

As a result of reviewing vendor information for the diesel air starter, a pressure of 50 to 150 psig inlet pressure is required. Therefore, the minimum pressure to start the diesel is 50 psig and the maximum available pressure is 100 psig. Thus a setting of 75 psig, which is what the switches are being presently set at, provides sufficient margin for the instruments to prevent spurious actuation and yet ensure sufficient air to start.

Based on the analyses and evaluations performed, it is concluded that this change does not constitute an unreviewed safety question.

| | |
|---|---|
| **Safety Evaluation No.:** | 90-004 |
| **Implementation Document No.:** | LDCN U-N61 |
| **UFSAR Affected Pages:** | VII-4 |
| **System:** | Core Spray |
| **Title of Change:** | Core Spray Strainer Acceptable Blockage |

**Description of Change:**

Currently, the FSAR does not indicate an acceptable clogging limit for the core spray strainers. Design calculations indicate that flow blockage of the strainers up to 50% only reduces the core spray flow into the reactor vessel by 1 to 2 percent. Thus, the core spray flow is relatively insensitive to flow blockages of up to 50 percent in the strainer.

The proposed change would define an acceptable clogging limit at which point design core spray flow would still be available.

**Safety Evaluation Summary:**

Clarification of an acceptable blocked strainer limit does not impact the current Appendix K analysis. The current GE LOCA analysis indicates that 4158 gpm of core spray flow is delivered to the vessel at a reactor pressure of 0 psig. Core spray flow to the vessel with 50% blocked strainers is 4790 gpm, which is greater than the 4158 gpm necessary for the LOCA analysis.

Based on the analysis, revision of the FSAR to define an acceptable clogging limit for the core spray strainers does not constitute an unreviewed safety question.

| | |
|---|---|
| **Safety Evaluation No.:** | 90-005 |
| **Implementation Document No.:** | LDCN U-N55 |
| **UFSAR Affected Pages:** | XIII-4 |
| **System:** | Quality Assurance Topical Report |
| **Title of Change:** | Revision 5 to Quality Assurance Topical Report (QATR-1) |

**Description of Change:**

Revision 5 to the Topical Report is a general update and clarification including changes requested by corporate organizations since the issue of Revision 4. There are no basic changes in the Quality Assurance Program. Organizational changes reflect the changes in Nuclear Division Staff and other changes, addition of the Manager, Independent Assessment Group, including changes in the Technical Superintendent's responsibilities, the Vice President of Quality Assurance reporting responsibility, changes to the Quality Assurance organization's responsibilities, a new organization chart, a new responsibility matrix, an addition of a section explaining when contractors not on the Qualified Contractor List are authorized to provide products or services, an updated procedure appendix, and revised nonconformance and corrective action sections. Editorial changes were made where necessary.

**Safety Evaluation Summary:**

Editorial changes to the Topical Report do not reduce any previous commitments; therefore, the effectiveness of the QA Program remains unchanged.

The organizational changes have been addressed by previous safety evaluations.

An exception was added to Appendix B of the Topical Report, which reflects an upgrade in NFPA code year from 1975 to 1986 as delineated in NQA-1, 1989. This does not reduce the effectiveness of the QA Program.

The QATR-1 revision number has been deleted from the Nine Mile Point Unit 1 FSAR to prevent any inconsistencies between the two documents. Since the QATR-1 is updated annually, this does not reduce the effectiveness of the QA Program.

| | |
|---|---|
| **Safety Evaluation No.:** | 90-009 |
| **Implementation Document No.:** | LDCN U-N63 |
| **UFSAR Affected Pages:** | IX-29, IX-33 |
| **System:** | 125 VDC |
| **Title of Change:** | Update FSAR Description of 125 VDC Battery Chargers: LDCN U-N63 |

**Description of Change:**

The purpose of this FSAR change is to enhance and clarify the description of battery charger capabilities. This change stems from corrective action 12.B.2 of the 1989 Restart Action Plan (RAP), which requires updating the FSAR based on findings of the design basis report recently formulated for the 125 VDC battery chargers.

**Safety Evaluation Summary:**

The proposed change to enhance the FSAR description of battery charger capabilities does not involve an unreviewed safety question. There is no physical change or operational change involved. The description of performance characteristics to be placed in the FSAR is taken from a design basis report, written according to the official format of the Design Basis Reconstitution Program (Engineering Program Integration), and the report has been approved by Design Engineering Management. The materials reviewed to formulate the report include all pertinent industry standards, regulatory guides, and NMPC design documents relating to the battery chargers.

**Safety Evaluation No.:** 90-010.

**Implementation Document No.:** LDCN U-N40

**UFSAR Affected Pages:** Sections IV, VI, VII, XV

**System:** Core Spray and Fuel

**Title of Change:** Core Spray System and LOCA Analysis

**Description of Change:**

Several concerns associated with the core spray system resulted from a NRC Safety System Functional Inspection conducted during the September 1988 through October 1988 time period. This evaluation addresses the effects on the Loss of Coolant Accident (LOCA) Analysis due to reduction in core spray flow, due to system resistance and diversion of flow, as stated in Unresolved Item 88-201-02 of their February 1, 1989 letter.

This safety evaluation also addressed FSAR Section XVI, page 164, which assumes that pipe whip resulting from a recirculation line break could damage a core spray line and result in single sparger operation.

**Safety Evaluation Summary:**

A major review of the core spray system capability to inject water into the reactor vessel under accident conditions has been completed. This review identified sources of internal flow loss within the core spray system that reduced the available flow to the reactor vessel. In addition, the available flow for analysis purposes was further reduced to provide margin between the pump flow capability and the credited flow to the reactor vessel. This margin provides allowance for surveillance test data, otherwise a low pump surveillance test result would require taking equipment out of service for inspection and possible repair. These reductions in available core spray flow to the reactor vessel reduce the minimum flow to the "hot bundle", which is the bundle that establishes the MAPLHGR limits. Also, credit for flow from only one sparger was assumed to account for potential recirculation line pipe whip. New fuel limits (MAPLHGR) for the Type 277, 299 and 321 fuel bundles have been established so that the calculated plant response to a design basis LOCA remains within the limits of 10CFR50.46. Therefore, this change does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 90-011

**Implementation Document No.:** LDCN U-N69

**UFSAR Affected Pages:** VII-9

**System:** Reactor Core Spray

**Title of Change:** Change of Core Spray Isolation Valve Maximum Acceptable Stroke Time Limits

**Description of Change:**

The new Appendix K analysis results specify a change in the stroke time limit for the pump discharge to the reactor vessel valves (IV 40-10, 40-11, 40-12/40-01, 40-02, 40-09). The change increases their maximum opening time from the 20 seconds currently shown in the FSAR to a requirement of 22.5 seconds.

**Safety Evaluation Summary:**

This Safety Evaluation analyzes the impact on safety of revising the Nine Mile Point Unit 1 Updated FSAR to reflect the change in the maximum acceptable stroke time limit.

The revised maximum core spray pump discharge valve stroke time is consistent with the core spray initiation time assumed in the latest 10CFR50 Appendix K Loss of Coolant Accident analysis, for which the calculated plant response remains within the limits of 10CFR50.46. Therefore, based on the analyses and evaluations performed, it is concluded that the change in the stroke time limit does not involve an unreviewed safety question.

| | |
|---|---|
| **Safety Evaluation No.:** | 90-012 |
| **Implementation Document No.:** | Mod. N1-89-900 |
| **UFSAR Affected Pages:** | VIII-39, VIII-40 |
| **System:** | APRM Flow Unit |
| **Title of Change:** | Flow Unit Upscale/Comparator Trip Settings |

**Description of Change:**

The proposed settings for Devices RI03A and RI03B are 103% ± 1% for the upscale trip and 6% ± 1% for the comparator trip setting. These values resulted from a General Electric Report (EDE-134-0889). With NMPC supplying calibration information, General Electric was asked to supply NMPC with the analytical limits and trip setpoints for the flow units. Their response stated that the analytical limits for the upscale trip and comparator trip are 107.1% and 10% respectively along with describing the trip setpoints at 104% and 7% respectively. The tolerance of 1% for the upscale trip was established using past values used in previous surveillance procedures. The 1% tolerance for the comparator trip was established by Electrical Engineering and the I&C Department.

**Safety Evaluation Summary:**

In comparing the old upscale setting (100% + 1%) to the proposed setting (103% + 1%), the old setting was more restrictive than the proposed setting and, therefore, should not present any operational concerns. In comparing the old comparator trip setting (14% + 2%) to the proposed setting (6% + 1%), the new setting is more restrictive. The new setpoints are within the analytical limits and, therefore, are acceptable.

The proposed setpoint changes will not increase the probability of occurrence of an accident previously analyzed or reduce the margin of safety. The proposed setpoints do not create the potential for a new accident or malfunction, but are additions to the existing design basis. It is concluded that this change does not constitute an unreviewed safety question.

| | |
|---|---|
| **Safety Evaluation No.:** | 90-018, Rev. 1 |
| **Implementation Document No.:** | LDCN U-N77 |
| **UFSAR Affected Pages:** | Section XV |
| **System:** | Nuclear Fuel |
| **Title of Change:** | Correction of FSAR Text Concerning Number of Blades Withdrawn, Control Rod Drop Accident, and Bundle Drop Accident |

**Description of Change:**

The changes to Page XV-138 and the top half of page XV-139 modify the FSAR to reflect Technical Specification Amendment 27.

The paragraph added to Section 3.2 on Page XV-139 states that the existing Bundle Drop Accident analysis bounds the effects of the new GE8X8EB fuel.

The changes to XV-149 through XV-152 state that General Electric submitted generic analyses justifying deletion of cycle-specific Control Rod Drop Accident analyses. Pages XV-169 and XV-170 contain a similar change for the Fuel Loading Error Analysis. Page XV-171 changes the safety limit CPR from 1.07 to 1.04, as approved in Safety Evaluation 89-012.

**Safety Evaluation Summary:**

The existing analysis of the Bundle Drop Accident will bound the effects of the new GE8X8EB fuel because of its lighter weight and essentially-identical structure. The lighter weight would yield less impact force if the bundle were dropped.

The changes on pages XV-149 through XV-171 were justified generically for GE-BWRs with BPWs in General Electric document NEDE-24011 US Supplement, Rev 09, "GE Standard Application for Reload Fuel."

Based on the analyses and evaluations performed, it is concluded that this change does not constitute an unreviewed safety question.

**Safety Evaluation No.:** 90-020

**Implementation Document No.:** LDCN U-N81

**UFSAR Affected Pages:** VII-61a

**System:** Feedwater (HPCI)

**Title of Change:** FSAR Change

**Description of Change:**

Revise FSAR Section VII, Paragraph I.3.0 to eliminate a statement indicating High Pressure Coolant Injection (HPCI) is necessary to prevent fuel clad temperatures from exceeding allowable limits during Loss of Coolant Accidents (LOCA). HPCI is not credited in the LOCA analysis for Nine Mile Point Unit 1.

This change was requested by the NRC Staff as a result of staff review of a proposed Technical Specification amendment changing HPCI operability requirements.

**Safety Evaluation Summary:**

Supplement 1 to the original FSAR described the performance of the Emergency Core Cooling System (ECCS) and included an evaluation of HPCI capability in response to an NRC staff question. This evaluation was the initial source for the claim that the HPCI would be available in the event of a LOCA and it has been carried over in the updated FSAR. At the time of the original evaluation, loss of off-site power was considered a remote occurrence and not a mandatory assumption for the LOCA analysis.

Removal of the statement in the FSAR corrects a false impression that the HPCI system is required to mitigate the consequences of a LOCA. Although off-site power may be available and consequently feedwater flow would be available, it is assumed not to be available for the purposes of the LOCA analysis.

**Safety Evaluation No.:** 90-021

**Implementation Document No.:** LDCN U-N83

**UFSAR Affected Pages:** XV-70, XV-71

**System:** Main Steam

**Title of Change:** FSAR Revision - Main Steam

**Description of Change:**

The NRC requested additional information in support of a proposed Technical Specification Amendment submitted with Letter NMP1L 0178. In developing the response which required information about the Main Steam Line (MSL) Break Analysis, it was noted that the FSAR text differed from Technical Specification requirements. The Technical Specifications require the MSL isolation valves to close within 10 seconds, whereas the FSAR gave a MSL isolation valve closure time of 8 seconds. The FSAR text also stated that partial core uncovery occurred, which conflicts with FSAR Figure XV-25 and the "Technical Supplement - Petition to Increase Power Level".

**Safety Evaluation Summary:**

The FSAR description of the MSL break accident scenario is revised to be consistent with the Technical Specification allowable isolation valve closure time of 10 seconds. The total time of the accident scenario remains unchanged at 11 seconds. The reactor coolant mass loss, radiological releases and off-site doses are not changed. An inconsistency within the FSAR text states that the reactor core is partially uncovered during the MSL break scenario. FSAR Figure XV-25 shows that MSL isolation occurs prior to losing enough reactor coolant that would cause core uncovery. In addition, the Technical Supplement to the Petition to Increase Power Level and Technical Specification bases for Section 3.2.7/4.2.7 "Reactor Coolant System Isolation Valves" both state that no core uncovery occurs during the MSL break accident. The total mass of reactor coolant above the top of the active core is shown as 144000 lb in the original (1966) MSL break accident calculations, which confirms FSAR Figure XV-25 is correct and that no core uncovery occurs during the accident scenario. The total net coolant mass loss during the 11-second accident scenario is 101000 lbs.

Based on the evaluation performed, it is concluded that this change does not constitute an unreviewed safety question.

| | |
|---|---|
| **Safety Evaluation No.:** | 90-025 |
| **Implementation Document No.:** | LDCNs U-N21, U-N32, U-N34 and U-N88 |
| **UFSAR Affected Pages:** | Sections V, IX, X |
| **System:** | N/A |
| **Title of Change:** | FSAR Update - 1990 |

**Description of Change:**

Update UFSAR as follows:

1) Revise page X-45 to change check valve and automatic closure of solenoid valve from 85 psig or less to 80 psig or less.

2) Revise page X-47 to change trip open pressure of cross-tie valve from 75 psig to 90 psig.

3) Revise UFSAR Figure X-3 to show rod over piston area vented to CRD exhaust header.

4) Revise UFSAR page IX-25 by deleting the KW limiter (overload) as a diesel engine protection device.

5) Revise UFSAR Figure V-2 to reflect as-built function of Nozzle N7L.

**Safety Evaluation Summary:**

The changes to the FSAR from the above-mentioned LDCNs are to update the FSAR to the as-built conditions of the plant. They do not change any systems' function. Engineering documents and operating procedures currently reflect these changes. There will be no physical changes to the plant as a result of these FSAR changes. It is concluded that these changes do not constitute an unresolved safety question.

**Safety Evaluation No.:** 90-031

**Implementation Document No.:** LDCN U-N86

**UFSAR Affected Pages:** XIII-8, Fig. XIII-1, Fig. XIII-2

**System:** Organization

**Title of Change:** Nine Mile Point Nuclear Division Organization as of May 30, 1990

**Description of Change:**

Section XIII of the FSAR describes the organization responsible for operation of Nine Mile Point Unit 1. In order to reflect the current organizational structure of the Nuclear Division, changes have been made in titles and additional positions included. Titles have been changed to be more job-specific. Additional positions were added to enhance the productivity of the Nuclear Division while easing the work load.

**Safety Evaluation Summary:**

These organization structures provide for the integrated management of activities that support the operation and maintenance of the facility.

These changes allow for:

1) Clear lines of authority to the General Superintendent.

2) Defined responsibility for activities important to the safe operation of the facility.

3) Distinct functional areas separately supervised and/or managed.

4) The reporting responsibility and authority of the functional areas of radiation protection, quality assurance and training are independent from operating pressures (Operations).

Based on the analyses and evaluations performed, it is concluded that this change does not constitute an unreviewed safety question.

## INSERTION INSTRUCTIONS

The following instructions are for the insertion of Revision 8 into the Nine Mile Point Unit 1 FSAR (Updated).

Remove pages listed in the REMOVE column and replace them with the pages listed in the INSERT column. Dashes (---) in either column indicate no action required.

The format for replacement pages has been changed. In previous revisions, every change incorporated on a page since the initial issuance of the document was indicated in the margin by a vertical bar and a corresponding revision number.

Beginning with Revision 8, the current revision and corresponding date will be shown at the foot of each replacement page and vertical bars placed in the margin to indicate only those changes made in the current revision.

A List of Effective Pages Volume is being issued to each setholder of the FSAR (updated). This volume lists each page in the FSAR (updated) and the current revision of the page. It should be used to verify that your FSAR (updated) is complete and current. It will be udpated each year with the FSAR (updated) revision and should be kept with your set of the FSAR (updated).

Please affix the stickers to the inside front cover of Volumes 1 and 2 of your FSAR (updated).

## INSERTION INSTRUCTIONS

### VOLUME I

| REMOVE | INSERT |
|---|---|
| Volume I Cover Sheet | Volume I Cover Sheet |
| vi | vi |
| viii | viii |
| ix | ix |
| xi | xi |
| xii | xii |
| xvi | xvi |
| xvii | xvii |
| xxi | xxi |
| xxii | xxii |
| xxv | xxv |
| xxvi | xxvi |
| xxvii | xxvii |
| xxviii | xxviii |
| xxix | xxix |
| xxx | xxx |
| xxxi | xxxi |
| xxxviii | xxxviii |
| xxxix | xxxix |
| xl | xl |
| xli | xli |
| xlii | xlii |
| xliii | xliii |
| xliv | xliv |
| xlv | xlv |
| xlvi | xlvi |
| xlvii | xlvii |
| xlviii | xlviii |
| il | il |
| l | l |
| li | li |
| lii | lii |
| liii | liii |
| liv | liv |
| lv | lv |

## INSERTION INSTRUCTIONS

### VOLUME I

| REMOVE | INSERT |
|---|---|
| IV-1 | IV-1 |
| IV-5 | IV-5 |
| IV-11 | IV-11 |
| IV-12 | IV-12 |
| IV-19 | IV-19 |
| --- | IV-19a |
| IV-21 | IV-21 |
| IV-22 | IV-22 |
| V-4 | V-4 |
| V-5 | V-5 |
| V-11 | V-11 |
| V-13 | V-13 |
| V-18 | V-18 |
| VI-10 | VI-10 |
| VI-47 | VI-47 |
| --- | VI-47a |
| VI-48A | VI-48a |
| VI-63 | VI-63 |
| VII-2 | VII-2 |
| VII-4 | VII-4 |
| VII-5 | VII-5 |
| VII-6 | VII-6 |
| --- | VII-6a |
| VII-8 | VII-8 |
| VII-9 | VII-9 |
| VII-61a | VII-61a |
| VII-62 | VII-62 |
| VII-62a | VII-62a |
| VIII-8 | VIII-8 |
| VIII-26 | VIII-26 |
| VIII-28 | VIII-28 |
| VIII-38 | VIII-38 |
| VIII-39 | VIII-39 |
| VIII-40 | VIII-40 |
| --- | VIII-40a |
| IX-17 | IX-17 |
| --- | IX-17a |
| IX-24 | IX-24 |
| --- | IX-24a |
| IX-25 | IX-25 |
| IX-26 | IX-26 |
| IX-27 | IX-27 |
| IX-29 | IX-29 |
| --- | IX-29a |
| IX-33 | IX-33 |

## INSERTION INSTRUCTIONS

### VOLUME I

| REMOVE | INSERT |
|---|---|
| X-2 | X-2 |
| X-3 | X-3 |
| X-9 | X-9 |
| X-45 | X-45 |
| X-47 | X-47 |
| X-62 | X-62 |
| --- | X-62a |
| X-74 | X-74 |
| XI-10 | XI-10 |
| XI-11 | XI-11 |

## INSERTION INSTRUCTIONS

### VOLUME II

| REMOVE | INSERT |
|---|---|
| Volume II Cover Sheet | Volume II Cover Sheet |
| vi | vi |
| viii | viii |
| ix | ix |
| xi | xi |
| xii | xii |
| xvi | xvi |
| xvii | xvii |
| xxi | xxi |
| xxii | xxii |
| xxv | xxv |
| xxvi | xxvi |
| xxvii | xxvii |
| xxviii | xxviii |
| xxix | xxix |
| xxx | xxx |
| xxxi | xxxi |
| xxxviii | xxxviii |
| xxxix | xxxix |
| xl | xl |
| xli | xli |
| xlii | xlii |
| xliii | xliii |
| xliv | xliv |
| xlv | xlv |
| xlvi | xlvi |
| xlvii | xlvii |
| xlviii | xlviii |
| il | il |
| l | l |
| li | li |
| lii | lii |
| liii | liii |
| liv | liv |
| lv | lv |

# INSERTION INSTRUCTIONS

## VOLUME II

| REMOVE | INSERT |
| --- | --- |
| XIII-2 | XIII-2 |
| XIII-4 | XIII-4 |
| XIII-8 | XIII-8 |
| XIII-18 | XIII-18 |
| XV-7 | XV-7 |
| XV-11 | XV-11 |
| --- | XV-11a |
| XV-12 | XV-12 |
| XV-13 | XV-13 |
| XV-22 | XV-22 |
| XV-38 | XV-38 |
| XV-39 | XV-39 |
| XV-52 | XV-52 |
| --- | XV-52a |
| XV-60 | XV-60 |
| XV-70 | XV-70 |
| XV-71 | XV-71 |
| XV-79 | XV-79 |
| XV-80 | XV-80 |
| XV-81 | XV-81 |
| XV-81a | XV-81a |
| XV-82 | XV-82 |
| --- | XV-82a |
| --- | XV-82b |
| XV-83 | XV-83 |
| XV-84 | XV-84 |
| XV-85 | XV-85 |
| XV-86 | XV-86 |
| XV-87 | XV-87 |
| XV-88 | XV-88 |
| XV-89 | XV-89 |
| XV-90 | XV-90 |
| XV-91 | XV-91 |
| XV-92 | XV-92 |
| XV-93 | XV-93 |
| XV-94 | XV-94 |
| XV-95 | XV-95 |
| XV-96 | XV-96 |
| XV-97 | XV-97 |
| XV-98 | XV-98 |
| XV-99 | XV-99 |
| XV-100 | XV-100 |
| XV-101 | XV-101 |
| XV-102 | XV-102 |
| XV-103 | XV-103 |

## INSERTION INSTRUCTIONS

### VOLUME II

| REMOVE | INSERT |
|---|---|
| XV-104 | XV-104 |
| XV-105 | XV-105 |
| XV-106 | XV-106 |
| XV-107 | XV-107 |
| XV-108 | XV-108 |
| XV-109 | XV-109 |
| XV-110 | XV-110 |
| XV-111 | XV-111 |
| XV-112 | XV-112 |
| XV-113 | XV-113 |
| XV-114 | XV-114 |
| XV-115 | XV-115 |
| XV-116 | XV-116 |
| XV-117 | XV-117 |
| XV-118 | XV-118 |
| XV-119 | XV-119 |
| XV-120 | XV-120 |
| XV-121 | XV-121 |
| XV-122 | XV-122 |
| XV-123 | XV-123 |
| XV-124 | XV-124 |
| XV-125 | XV-125 |
| XV-126 | XV-126 |
| XV-127 | XV-127 |
| XV-128 | XV-128 |
| XV-129 | XV-129 |
| XV-130 | XV-130 |
| XV-131 | XV-131 |
| XV-132 | XV-132 |
| XV-133 | XV-133 |
| XV-134 | XV-134 |
| XV-135 | XV-135 |
| XV-136 | XV-136 |
| XV-137 | XV-137 |
| XV-137a | XV-137a |
| XV-137b | XV-137b |
| XV-137c | XV-137c |
| XV-137d | XV-137d |
| XV-137e | XV-137e |
| XV-137f | XV-137f |
| XV-137g | XV-137g |
| XV-137h | XV-137h |
| XV-137i | XV-137i |
| XV-137j | XV-137j |
| XV-137k | XV-137k |

NINE MILE POINT UNIT 1 FSAR (UPDATED)

INSERTION INSTRUCTIONS

VOLUME II

| REMOVE | INSERT |
| --- | --- |
| XV-138 | XV-138 |
| XV-139 | XV-139 |
| XV-149 | XV-149 |
| XV-150 | XV-150 |
| XV-151 | XV-151 |
| XV-152 | XV-152 |
| XV-169 | XV-169 |
| --- | XV-169a |
| XV-170 | XV-170 |
| XV-171 | XV-171 |