November 13, 2000

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, 2nd Floor
P.O. Box 63
Lycoming, NY 13093

Subject: NINE MILE POINT NRC INSPECTION REPORT 05000220/2000-007 AND 05000410/2000-007

Dear Mr. Mueller:

On September 29, 2000, the NRC completed a team inspection of the Unit 1 containment spray and containment spray raw water systems and the Unit 2 service water system. The team also inspected evaluations of changes, tests, and experiments at both units. The enclosed report presents the results of that inspection. The results of this inspection were discussed with you and members of your staff on September 29, 2000.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

The team identified eight issues that were evaluated under the risk significance determination process (SDP) and were determined to be of very low safety significance (Green). These issues have been entered into your corrective action program and are discussed in the summary of findings and in the body of the attached inspection report. These issues involved multiple examples of non-cited violations of the design control requirements of 10 CFR 50 Appendix B, Criterion III, and two non-cited violations of the corrective action requirements of 10 CFR 50 Appendix B, Criterion XVI. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region I, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001, and the NRC Resident Inspector at the Nine Mile Point Nuclear Station.

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Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket Nos. 05000220, 05000410
License Nos. DPR-63, NPF-69

Enclosure: NRC Inspection Report 05000220/2000-007 and 05000410/2000-007

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REGION I

Docket Nos:	05000220 05000410
License Nos:	DPR-63 NPF-69
Report Nos:	05000220/2000-007 05000410/2000-007
Licensee:	Niagara Mohawk Power Corporation P. O. Box 63 Lycoming, NY 13093
Facility:	Nine Mile Point, Units 1 and 2
Location:	Scriba, New York
Dates:	September 11 - 29, 2000
Inspectors:	L. Scholl, Senior Reactor Inspector F. Bower, Resident Inspector (Calvert Cliffs NPS) C. Cahill, Reactor Inspector S. Chaudhary, Reactor Inspector L. Cline, Reactor Inspector D. Dempsey, Reactor Inspector M. Shlyamberg, Contractor K. Young, Reactor Inspector
Approved by:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 050000220-00-07; 050000410-00-07 on 9/11-15, 25-29, 2000; Niagara Mohawk Power Corporation (NMPC); Nine Mile Point Nuclear Station; Mitigating Systems; Other Activities (Cross-cutting/Corrective Actions).

The inspection was conducted by a region-based team of the Unit 1 containment spray and containment spray raw water systems and the Unit 2 service water system using NRC Baseline Inspection Procedure 71111.21, "Safety System Design and Performance Capability." The team also reviewed the NMPC's evaluation of changes, tests and experiments under the 10 CFR 50.59 process using NRC Baseline Inspection Procedure 71111.02, "Evaluations of Changes, Tests, or Experiments." The significance of issues is indicated by their color (Green, White, Yellow, or Red) and was determined by the Significance Determination Process (SDP) in Inspection Manual Chapter 0609 (see Attachment 2).

A. <u>Inspector Identified Findings</u>

Cornerstone: Mitigating Systems

- Green. The team identified that NMPC reclassified the safety function of the containment spray raw water system inter-tie check valves from active to passive components with only a pressure boundary safety function. This reclassification was used to provide the bases for removing the valves from the in-service test (IST) program. The team also found that several safety evaluations and calculations credited the valves with closing to prevent reverse flow from the containment spray and core spray systems into the containment spray raw water system. The failure to properly classify these valves was determined to be of very low risk significance (Green) by the SDP phase 1 screening. This conclusion was based on the finding that although the safety classification determination allowed removal of the valves from the IST program, the valves had not yet been removed from the test procedure and continued to receive a reverse flow closure test. Therefore, there were no actual consequences caused by this error. The failure to identify and translate the design basis requirements of the inter-tie check valves into the IST program is considered a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was entered in the NMPC corrective action program. (Section 1R21.1, Design-Mechanical, Electrical and Instrumentation and Control)

- Green. The team identified that the pump developed head acceptance criteria in the Unit 1 containment spray pump surveillance tests was non-conservative with respect to the design bases. The team determined this issue to be of very low risk significance (Green) by the SDP phase 1 screening. This conclusion was based the team's review of current surveillance test results which found the pumps had adequate margin between the measured values and the test acceptance criteria to account for the error introduced by the calculation. The failure to translate design basis assumptions into test procedure acceptance criteria is considered a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was entered in the NMPC corrective action program. (Section 1R21.1, Design-Mechanical, Electrical and Instrumentation and Control)

- Green. The team identified that NMPC did not consider the most limiting scenario when evaluating the adequacy of the net positive suction head (NPSH) for the containment spray system pumps. As a result, operating procedures were not consistent with the design bases. This issue was evaluated using the SDP phase 1 screening and was determined to be an issue of very low risk significance (Green). This conclusion was based on a consideration that system functionality would not be affected since pressure in the containment peaks, and then decreases to below 3.5 psig within approximately 12.5 minutes of a LOCA, resulting in the short term operation of two pumps on one suction strainer. Also, until containment pressure is reduced to 0 psig, at which time the pumps would be secured, the existing containment pressure serves to counteract the adverse effects of elevated torus water temperature and pressure drop across the strainer on NPSH. Additionally, if pump cavitation were to result from two pump operation on one strainer, it would be recognized by the operators and, once a cavitating pump was secured, additional NPSH would be available for the remaining operating pump. The failure to validate design assumptions is considered a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was entered in the NMPC corrective action program. (Section 1R21.1, Design-Mechanical, Electrical and Instrumentation and Control)

- Green. The team identified that, due to improper design assumptions, assumed margins in the Unit 2 service water system hydraulic performance analysis were incorrect. The analysis concluded that there was 20% margin to account for pump degradation and instrument uncertainty. However, as a result of the incorrect assumption the margin was not 20% and, if the pumps were assumed to be degraded by 10%, there would be no margin to account for instrument uncertainty. The team determined this issue was of very low safety significance (Green) by the SDP phase 1 screening. This conclusion was based on the team's review of the most recent surveillance procedure test results which were found to be acceptable and did not indicate significant loss of margin due to pump degradation. Therefore, the service water system was functional and the service water pumps remained operable. The failure to validate the design assumption is considered a non-cited violation of 10 CFR 50 Appendix B, Criterion III, Design Control. The issue was entered in the NMPC corrective action program. (Section 1R21.1, Design-Mechanical, Electrical and Instrumentation and Control)

- Green. The team identified that plant procedures did not include directions to start a containment spray raw water pump within 15 minutes following a loss-of-coolant accident (LOCA). This issue was evaluated using the SDP phase 1 screening and was determined to be of very low risk significance (Green). This conclusion was based on a consideration that system functionality would not be impaired. Following an accident, the operators monitor key parameters that include containment and torus temperature. In the event of an adverse trend, and/or actuation of the torus high temperature annunciator, plant procedures would result in the starting of a CSRW pump. However, the failure to translate design basis information into operating procedures is considered a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was entered in the NMPC corrective action program. (Section 1R21.2, Operations, Maintenance and Testing)

- Green. The team reviewed Unit 2 Nuclear Engineering Report NER-2M-037, which provides the requirements for taking reactor building unit coolers out of service for testing. The team identified that these requirements had not been incorporated in the service water operating procedure. The team determined this issue was of very low risk significance (Green) based on SDP phase 1 screening. This conclusion was based on the observation that, while not specified in the procedure, the current practice was to route all work orders that take a reactor building unit coolers out of service to engineering for approval. Also, the team did not identify any instances where the reactor building coolers were removed from service without the engineering requirements being met. The failure to properly translate design basis information into the operating procedure is considered a non-cited violation of 10 CFR 50 Appendix B, Criterion III, Design Control. The issue was entered in the NMPC corrective action program. (Section 1R21.2, Operations, Maintenance and Testing)

Cornerstone: Cross-Cutting Issues

Green. The team identified that NMPC had determined that the containment spray raw water system radiation monitors could alarm due to background radiation levels following a LOCA. However, the associated alarm response procedures had not been revised to alert operators to this potential and to provide appropriate response actions. The team determined this issue to be of very low risk significance (Green) by the SDP phase 1 screening. The team determined this issue to be of very low risk significance (Green) by the SDP phase 1 screening. This conclusion was based on a recognition that the simultaneous actuation of all four alarms following a LOCA, without a corresponding indication on the downstream service water system radiation monitor, would be sufficient information for the operators to recognize the alarms as spurious due to background radiation. The failure to implement corrective actions to correct the affected procedure is considered a non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Action. This item was entered into the NMPC corrective action program. (Section 4OA1, Identification and Resolution of Problems)

Green. The team identified that an Emergency Operating Procedure (EOP) attachment, intended to be a standalone procedure, did not contain all the directions necessary to perform the procedure without referring to other operations procedures. The team determined this issue to be of very low risk significance (Green) by the SDP phase 1 screening. This conclusion was based on the fact that the system operating procedure included the appropriate guidance, as did operator training, and the issue did not affect the operability of the service water system. The failure of NMPC to implement appropriate corrective actions to ensure adequacy of operating procedures is considered a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. This item was entered into the NMPC corrective action program. (Section 4OA1, Identification and Resolution of Problems)

Report Details

1. REACTOR SAFETY
Cornerstone: Mitigating Systems

1R21 Safety System Design and Performance Capability (IP 71111.21)

Introduction

The containment spray/containment spray raw water system (Nine Mile Point Unit 1) and service water system (Nine Mile Point Unit 2) were reviewed using Inspection Procedure 71111, Attachment 21, "Safety System Design and Performance Capability." The objective of this review is to verify that the design bases have been correctly implemented to ensure the systems can meet their functional requirements. The Unit 1 containment spray system (including the containment spray raw water subsystem) was selected because it is a risk significant mitigating system which provides containment pressure control and heat removal following accidents. The Unit 2 service water system was selected because it is a risk significant mitigating system which provides cooling water to various safety-related components and systems necessary for safe shutdown and accident mitigation.

.1 Mechanical, Electrical, and Instrumentation and Controls System Design

a. Inspection Scope

The team reviewed the Unit 1containment spray/containment raw water system and the Unit 2 service water system design and licensing basis documents, including the Updated Final Safety Analysis Report (UFSAR), plant Technical Specifications (TS), and system design basis documents (SDBDs) to determine the system and component functional requirements during normal operation and for accident mitigation. The team then reviewed additional documents, performed plant walkdowns and interviewed plant personnel to verify that the design bases functional requirements were properly implemented. The additional documents reviewed included selected items from the following: engineering analyses, calculations, plant modifications, piping and instrumentation drawings (P&IDs), electrical schematics, instrumentation and control drawings, logic diagrams, and instrument set points.

In performing this review, the team verified that the assumptions used in engineering analyses and calculations were appropriate, that proper methods and models were used, and that there was an adequate technical basis to support the conclusions. Where possible, the team performed independent calculations to evaluate the document's adequacy. In reviewing modifications, the team verified that the ability of the systems to perform their design functions was not adversely affected by the change and, for selected modifications, verified the adequacy of supporting engineering documents and post-modification testing. The plant walkdowns were performed to verify the physical installation was consistent with design bases document assumptions, design drawings and installation specifications. The specific items examined during walkdowns included pumps, piping, piping supports, motor operated valves (MOVs), air operated valves (AOVs), system instrumentation, valve positions, operator aids, and electrical switchgear.

The team also reviewed the UFSAR, TS, and SDBDs for interfacing support systems such as instrument air, process radiation monitoring, and control room annunciators. These systems were examined to assess their capability to support the operation of the selected Unit 1 and 2 systems.

b. Issues and Findings

Unanalyzed Leak Path From Unit 1

The team identified that there was a potential unanalyzed leakage path that would result in the release of radiologically contaminated torus water and/or the post accident design basis source term to the ultimate heat sink (i.e., Lake Ontario). This potential exists as a result of system inter-tie connections between the containment spray raw water (CSRW) system and the containment spray and core spray systems. CSRW loops 112 and 121 each have a 12 inch piping inter-tie connection to the containment spray system. The isolation between the containment spray raw water and the containment spray loops is provided by a check valve and a normally closed motor operated valve. Leakage past these valves could result in the leakage of torus water from the containment spray system into the containment spray raw water system which discharges to the lake via the service water system. This leakage would be prevented when the CSRW pumps are operating since the raw water system pressure is higher than the containment spray system pressure. However, following an accident there are conditions when containment spray pumps will be operating without the associated CSRW pumps operating. For example, following an accident, the containment spray pumps automatically start and the CSRW pumps are manually started later from the control room. Also, the failure of a CSRW pump to start or the securing of operating CSRW pumps per plant procedures could also result in the potential for leakage.

Containment spray raw water loops 111 and 122 have similar inter-tie connections with the core spray system. In this case, the operating pressure of the core spray system is higher than the CSRW system such that leakage of torus water into the raw water system could occur whenever a core spray pump is operating and would not be prevented by operation of the CSRW pumps.

The team also reviewed the maintenance history for the inter-tie check valves and MOVs and found that there were frequent failures of leak tests and quarterly functional tests. NMPC subsequently determined that leak testing of the valves was no longer required and as a result, current leak rate information is not available. NMPC entered this issue into the corrective action program as DER 1-2000-3154. This DER documents that there is no retrievable radiological assessment of the consequences of the leakage and that a more rigorous test program appears to be warranted for the inter-tie valves. NMPC’s initial assessment was that the systems remained operable since the source terms utilized in their 10 CFR 100 analyses result in very large margins between actual offsite doses and regulatory limits.

During the initial review of this issue NMPC indicated that any leakage into the raw water system would be detected by a downstream service water system radiation monitor. The team questioned whether any of the piping configurations would result in leakage flowing towards the raw water pumps instead of towards the discharge line past the radiation monitor if the associated CSRW pump was not operating. Additional review by NMPC identified a potential unmonitored release pathway for the inter-tie connection for core spray system loop 12. DER 1-2000-3323 was initiated to evaluate this issue. NMPC's initial assessment of this issue included a review of the torus water activity to ensure any leakage would not result in exceeding 10 CFR 20 limits for release to unrestricted waters. Additional interim actions include taking measures to prevent any unmonitored releases during core spray surveillance testing. These issues are unresolved pending NRC review of the results of the NMPC evaluation of the issue and an assessment of the risk significance of leakage through this flow path. **(UNR 05000220/2000-007-01)**

Safety Classification Determination

The team reviewed Safety Classification Determination 93-039 and found that containment spray raw water system inter-tie check valves 93-62, 93-60, 93-58 and 93-54 had been reclassified from active to passive components with only a pressure boundary safety function. The determination provided the basis for discontinuing the quarterly reverse direction exercise tests that had been performed under the in-service test (IST) program. However, safety evaluation 89-13, "Containment Spray Post DBA LOCA Appendix J Water Seal," stated that valves 93-60 and 93-62 must close to ensure that the Appendix J water seal is maintained in the event of an inadvertent opening of the downstream MOVs (93-72/93-73), and that they are reverse flow exercise tested in accordance with the IST program. Safety Evaluation 94-036, "Evaluation of BWROG-EPG Drywell Spray Limitations On The Containment Spray System Appendix J Water Seal," states that the containment spray raw water system inter-tie check valves are assumed to be leak tight (the MOVs are assumed to leak), and that this function is validated by routine surveillance testing and integrated leak rate testing. The evaluation further stated that "surveillance testing is required to ensure these assumptions (i.e., "minimal cross-tie leakage) remain valid." Calculation SO-AppJ-M003, "Containment Spray System Leakage Assessment," also assumed that the inter-tie check valves do not leak.

The failure to properly classify these valves was determined to be very low risk significance (Green) by the SDP phase 1 screening. This conclusion was based on the finding that although the safety classification determination allowed removal of the valves from the IST program, the valves had not yet been removed from the test procedure and continue to receive reverse direction exercise tests. Therefore, the valves remained operable and the system function was not adversely affected. The question of whether the valves require quantitative leak tests will be reviewed further as discussed in the unresolved item discussed above.

Failure to identify and translate the design basis requirements of the inter-tie check valves into the in-service test program is considered to be a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. **(NCV 05000220/2000-007-02)** The violation is being treated as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). The issue is in the NMPC corrective action program as DER 1-2000-3154.

Unit 1 Containment Spray System Heat Exchanger Fouling Factors

Section 3.2.1.2 of SDBD-203, “Torus Water Heat Removal Performance,” states that the primary containment heat transfer model was based on a combined containment spray raw water heat exchanger fouling factor of 0.0006 hr-°F-ft^2/BTU. This value is consistent with the manufacturer’s design specification, and was assumed in various design calculations including S14-93-HX07, “Determine HX K Value Input to Revised Containment Analysis,” and SOTORUSM009, “NMP-1 Pool Heatup Analysis.”

The fouling factor used in the design calculations was based on a new, clean heat exchanger and does not appear to account for service conditions. For example, a typical fouling factor established by the Tubular Exchanger Manufacture’s Association is 0.001 hr-°F-ft^2/BTU for city or well water such as the Great Lakes, as discussed in EPRI NP-7552, “Heat Exchanger Performance Monitoring Guidelines,” dated December 1991. NMPC’s bases for using the design fouling factor included: (1) the heat exchangers are maintained in a dry lay-up condition except during quarterly pump testing, (2) the heat exchangers are cleaned and inspected each operating cycle (24 months), and (3) confirmatory heat exchanger performance tests were conducted in 1990 and 1991.

The team identified the following during its review of the NMPC inspections and performance tests:

- The 1990 performance test results were documented in Problem Report (PR) 2140 and evaluated in calculation S14-93-HX05, “Containment Spray Loop 111 Test Results.” The documents indicated that the heat exchangers were fouled. For example, the calculation documented a combined average fouling factor of 0.00066 hr-°F-ft^2/BTU. However, while instrument error can have a significant effect on the results of the calculation, the test was performed using installed system instruments rather than more accurate test instrumentation. The calculation did not take instrument accuracy and related uncertainties into account.

- The heat exchangers were cleaned and in 1991 were performance tested again. The results of the test were inconclusive in that fouling factors between 0.00172 and (-) 0.00311 hr-°F-ft^2/BTU were calculated. NMPC attributed this result to inadequate instrumentation and the fact that the heat exchangers are not insulated. The effect of instrument uncertainty was not addressed in the calculation.

- Slight amounts of sludge or scale buildup on heat exchanger tubes can significantly greatly reduce thermal performance. The heat exchanger preventive maintenance work orders for the 1999 heat exchanger inspection indicated the presence of varying degrees of rust, grit, scale, and partial tube blockage that was not consistent with the design basis of the heat exchangers. The preventive maintenance procedure did not provide acceptance criteria for determining the acceptability of the as-found or as-left condition of the heat exchanger tubes.

Based on these findings, NMPC initiated DER 1-2000-3242 and performed an operability evaluation for the heat exchangers. NMPC concluded that the heat exchangers were degraded but operable provided that lake water temperature remained below 74 °F. The team concluded that the operability determination was acceptable. NMPC's review of past heat exchanger operability was ongoing at the conclusion of the inspection.

This item is unresolved pending NRC review the results of the NMPC evaluation of past containment spray system operability and an assessment of the risk significance of any periods when the heat exchangers may have been inoperable. **(UNR 05000220/2000-007-05)**

Unit 1 Containment Spray Pump Tests

The team reviewed calculation S14-80-F030, "Containment Spray System Design Basis Hydraulic Analysis," which established the design basis for the hydraulic performance of containment spray system. The team noted that the prediction of the performance of the system with degraded pumps was based on a pump performance curve from calculation S14-80-F10,"Containment Spray Flow Rates." However, this calculation used data for only one pump and did not evaluate if this was a “strong”, “weak”, or “average” performing pump. The team also found that the test data provided in calculation S14-80-F014 indicated that the as-tested performance of pumps 111 and 112 was below the design pump curve used in calculation S14-80-F030. As a result, the acceptance criteria in the containment spray pump surveillance tests was non-conservative with respect to the design bases. The team determined this issue to be of very low risk significance (Green) by the SDP phase I screening. This conclusion was based the team's a review of current surveillance test results which found the pumps had adequate margin between the measured values and the test acceptance criteria to account for the error introduced by the calculation. The failure to properly translate design bases assumptions into procedure acceptance criteria is considered an additional example of the non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. **(NCV 05000220/2000-007-02)** This item is in the NMPC corrective action program as DER 1-2000-3143.

Unit 1 Containment Spray Pump Net Positive Suction Head

The UFSAR identified the minimum net positive suction head (NPSH) margin (the margin between required and available NPSH) as 1.9 feet at 143°F. The NPSH margin was calculated based on scenario 4 of calculation S14-80-F003, "Containment Spray Pump NPSH Available Versus Required," which evaluates the effects of a LOCA coincident with a LOOP. During this scenario, all four containment spray pumps are assumed to start automatically and results in two containment spray pumps drawing suction from the same strainer assembly. This condition was considered bounding for NPSH concerns because it results in the highest strainer flow rate and corresponding highest debris buildup. The torus temperature assumed in this scenario was 143°F. This is the calculated temperature of the torus at 15 minutes following the accident. The reason for considering this temperature as bounding was that the analysis also assumes that at 15 minutes following the accident one of the containment spray pumps would be secured to permit starting a containment spray water pump without exceeding the rating of the emergency diesel generator (EDG).

The team noted that for a scenario involving a LOCA with off-site power available, the four containment spray pumps would automatically start and could continue to run greater than 15 minutes since there would be no EDG loading concerns and there were no specific procedure steps to secure the pumps until containment pressure was reduced sufficiently. As a result, there could be two pumps taking a suction on the same strainer at temperatures up to 159°F which is the calculated maximum torus temperature. Calculation S14STRAINERM001 concluded that long-term operation of two containment spray pumps through the same strainer results in unacceptable head loss. DER 1-2000-3333 was initiated to address this issue and was categorized as an outage restraint requiring resolution (procedure changes) prior to start-up from the in-progress maintenance outage.

This issue was evaluated using the SDP phase 1 screening and was determined to be an issue of very low risk significance (Green). This conclusion was based on a consideration that system functionality would not be affected since pressure in the containment peaks, and then decreases to below 3.5 psig, within approximately 12.5 minutes of a LOCA, resulting in only short term operation of two pumps on one suction strainer. Also, until containment pressure is reduced to 0 psig, at which time the pumps would be secured, the existing containment pressure serves to counteract the adverse effects of elevated torus water temperature and pressure drop across the strainer on NPSH. Additionally, if pump cavitation were to result from two pump operation on one strainer, it would be recognized by the operators and, once a cavitating pump was secured, additional NPSH would be available for the remaining operating pump.

The failure to validate design basis assumptions is an additional example of the non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. (**NCV 05000220/2000-007-02**) This item is in the NMPC corrective action program as discussed above.

Unit 2 Pump Performance Curves and IST Tests Data

The team reviewed calculation A10.1-N-341, "Three SWP Pumps - LOCA Under Degraded Conditions," which established the design basis for the hydraulic performance of the service water system. The team noted that the modeling of the system performance was done by using the original vendor performance curve and degrading it such that the degraded pump curve would intersect the TS operating point (8,000 gpm and 70 psid). At the constant total dynamic head (TDH) of 70 psid, the difference between the flows from the original certified pump curve and the degraded curve was about 20%. NMPC made an assumption (Assumption 6.1) that the 20% degradation provides margin for both the IST allowables and the instrument uncertainty. Since the in-service testing (IST) allows for up to 10% pump degradation, the 20% degradation assumption provided for an ample margin for the instrument uncertainty.

The team also reviewed the actual IST data and acceptance criteria and found that the baseline reference values for all six service water pumps was below the vendor certified curve. The largest deviation (10%) was between the baseline for pump 2SWP*P1A and the vendor certified curve. Since the IST procedure allows 10% allowance for pump degradation, there is no remaining margin in the calculation to account for instrument uncertainty should pump performance degrade to the acceptance criteria value. The team determined this issue was of very low safety significance (Green) by the SDP phase 1 screening. This conclusion was based on the team's review of the current surveillance procedure test results which were found to be acceptable and above the 10% degradation limit. The failure to validate the design assumption is considered to be an additional example of a non-cited violation of 10CFR 50 Appendix B, Criterion III, Design Control. **(NCV 05000410/2000-007-02)** This issue has been entered into the NMPC corrective action program as DER 2-2000-3352.

Unit 2 Service Water Model

The team reviewed calculation A10.1-N-340, "PROTO -FLO SWP Base Hydraulic Model - Normal Operation," which developed the hydraulic model for the service water system. The team also reviewed the use of this model in predicting the post-accident service water flow rates in calculation A10.1-N-341, " Three SWP Pump LOCA analysis."

The team found that the model was based on the results of the initial flow balance data from testing performed in 1985. Since that time the service water system has experienced erosion and corrosion problems. To address these problems NMPC performed chemical cleaning and selective small bore piping modifications to assure the delivery of required flows to the safety related loads. By the review of calculations and discussions NMPC personnel, the team found that the extent of the model validation against the as-built conditions was limited to an informal and qualitative comparison of the indicated flows.

The team also noted that, since the modeling did not account for the effects of instrument uncertainty on the predicted flow rates to the safety related components, actual flows could be less than the predicted flows. In addition, calculation A10.1-N-341 identified that the predicted flows and/or pressures had little or no margin to the design limits. NMPC initiated DER 2-2000-3380 to further evaluate this issue. This issue is unresolved pending NRC review of the findings of the NMPC evaluation and an assessment of the risk significance of any identified deficiencies. **(UNR 05000410/2000-007-03)**

.2 Operations, Maintenance and Testing

a. Inspection Scope

The team reviewed various documents and plant procedures to verify that the selected systems were being operated and maintained consistent with the design and licensing bases. The operational readiness and material condition of the selected systems were assessed by conducting system walkdowns and by the review of specific documentation that included operating procedures, maintenance history records, preventive maintenance program and records, surveillance test procedures and results, system health reports, operator logs, vendor documents, and calibration records. Additionally, the team interviewed NMPC personnel including plant operators, system engineers, maintenance personnel, and instrumentation and control personnel regarding performance and operation of the selected systems. Various emergency operating procedures and support procedure N1-EOP-1, “NMP1 EOP Support Procedure,” (including applicable attachments) were also reviewed to verify that design and licensing basis information had been appropriately included.

b. Issues and Findings

Operation of Unit 1 CSRW Pumps

The team noted that the suppression pool heat-up analysis assumes that a CSRW pump will be started to provide cooling to the containment spray system heat exchanger within 15 minutes following a LOCA. The team reviewed EOPs, including Attachment 16, “Torus Cooling,” and Attachment 17, “Auto or Manual Initiation of Containment Spray,” and found that they did not include procedural steps to ensure the CSRW pumps would be started within the design bases time of 15 minutes. This issue was evaluated using the SDP phase 1 screening and was determined to be of very low risk significance (Green). This conclusion was based on a consideration that system functionality would not be impaired. Following an accident, the operators monitor key parameters that include containment and torus temperature. In the event of an adverse trend, and/or actuation of the torus high temperature annunciator, plant procedures would result in the starting of a CSRW pump. The failure to translate design information into operating procedures is considered an additional example of a non-cited violation of 10 CFR 50 Appendix B, Criterion III, Design Control. **(NCV 05000220/2000-007-02)** This item is in the NMPC corrective action program as DER 1-2000-3161.

Unit 2 Reactor Building Unit Coolers

The team reviewed Nuclear Engineering Report NER-2M-037 which provides acceptance criteria for reactor building unit cooler testing. This report provides the following requirements for taking unit coolers out of service for testing. “When taking unit coolers out of service for testing, the service water temperature must be 70°F or lower. This will insure that sufficient cooling will be available to other coolers to support the design heat loads. No more than one unit cooler should be taken out of service at a time”. The team noted that this limitation was not incorporated into the service water operating procedure, N2-OP-52. The team determined this issue was of very low risk significance based on SDP phase 1 screening. This conclusion was based on the observation that, while not specified in the procedure, the current practice was to route all work orders that take a reactor building unit coolers out of service to engineering for approval. Also, the team did not identify any instances where reactor building coolers were improperly removed from service. The failure to translate design basis information into the operating procedure is considered an additional example of a non-cited violation of 10 CFR 50 Appendix B, Criterion III, Design Control. **(NCV 05000410/2000-007-02)** This issued is in the NMPC corrective action program as DER 2-2000-3353.

.3 (Closed) Licensee Event Report (LER) 50-410/2000-07: NMPC identified that a single failure of the high pressure core spray (HPCS) Division III diesel generator to start following a loss of offsite power (LOOP), could result in a low flow demand for service water. The low system flow could then cause the operating pumps to trip if flow dropped below the low flow trip set point. This is contrary to 10 CFR 50, Appendix A, Criterion 44, which requires the heat load of structures, systems, and components to be transferred to an ultimate heat sink under normal operating and accident conditions assuming a single failure.

NMPC performed Safety and Availability Assessment, SAS-00-035, "NMP2 HPCS Single Failure Due to SW Problems, DER2-2000-1452," Revision 0, and concluded that the risk associated with this failure was very low.

NMPC installed a temporary modification to establish service water flow through the residual heat removal (RHR) heat exchangers to prevent the inadvertent low flow trip of the service water pumps and subsequently installed a permanent modification to remove the low flow trips from the service water pumps.

The team reviewed safety evaluation number 00-048, "Removal of the Service Water Pumps' Low Flow Trip," Revision 0, and noted that it stated under a LOOP scenario with the isolation of all non-essential loads and the failure of the HPCS diesel, all operating service water pumps would trip due to low flow. The team also noted that in SAS-00-035 and the in LER, NMPC incorrectly assumed that Division II service water pumps would remain available to provide cooling flow. NMPC initiated DER 2-2000-3381 document these discrepancies and reassess the issue.

The team and the Region I senior reactor analyst subsequently reviewed the NMPC reassessment of this issue, SAS-00-035, "Re-analysis of NMP2 HPCS Single Failure Due to SW Problem, DER 2-2000-1452, LER-007," Revision 1. The re-analysis included a NMPC review of actual calibration data for service water pump low flow trip instrumentation from December 16, 1992, through February 17, 2000. Based on this review, NMPC did not identify any instance where the service water pumps would have tripped on low flow if the HPCS diesel failed to start. Therefore, although the original assumptions utilized in the NMPC assessment were incorrect, the problem documented in the LER would not have had a significant impact on core damage frequency (CDF).

The team concluded that the corrective actions for this issue were appropriate and the risk significance of this issue was very low. This LER is closed.

1RO2 Evaluations of Changes, Tests, and Experiments (IP 71111.02)

a. Inspection Scope

The team reviewed safety evaluations performed by NMPC to implement changes to the facilities and/or procedures as described in the UFSAR, and tests and experiments not described in the UFSAR to verify that they were reviewed and documented in accordance with 10 CFR 50.59 and that any safety issues pertinent to the changes, test, and/or experiments had been properly resolved. The team also verified that NMPC conclusions that the changes, tests, or experiments did not require prior approval by NRC or a license amendment were appropriate. The inspection included discussions with cognizant engineering personnel as well as the review of supporting technical information such as calculations, engineering analyses, and industry recommendations. The scope of this review nine safety evaluations and twenty-five applicability reviews (ARs).

Additionally, the team verified that NMPC was identifying problems associated with the implementation of the 50.59 safety evaluation program and that the problems were entered into the corrective action program. A sample of DERs associated with problems with the safety evaluation program were reviewed to ensure the issues were properly evaluated and appropriate corrective actions were implemented.

b. Issues and Findings

There were no findings identified.

4. OTHER ACTIVITIES (OA)

4OA1 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed NMPC activities associated with the identification and resolution of problems associated with the Unit 1 containment spray and containment spray raw water systems and the Unit 2 service water system. The team reviewed a sample of DERs associated with the selected systems to evaluate the adequacy of the corrective actions that were identified and also assessed the timeliness of the completion of the corrective actions. For selected DERs the team also reviewed associated operability evaluations and/or verified the completion of corrective actions.

b. Issues and Findings

DER 1-1998-0644 - CSRW Radiation Monitors

This DER identified problems associated with the containment spray raw water system radiation monitors which were installed to provide indication of gross heat exchanger tube leaks. This DER was initiated to re-evaluate a previous recommendation (DER 1-92-4287) to retire these monitors due to their inability to detect tube leakage during normal operations and because they could provide erroneous indication by alarming during a design basis event due to high background radiation.

In November 1998, engineering services completed the re-evaluation, and as a result, design engineering was directed to generate a Design Change/Configuration Change, including an Applicability Review/Safety Evaluation, to retire the radiation monitors. At the time of this inspection the safety evaluation had been performed but was not yet approved by plant management.

The team also noted that DER 1-1998-0644 stated that the annunciator response procedure (ARP) associated with these radiation monitors would inappropriately direct operators to secure the containment spray system following an accident. However, the team found that affected procedure, N1-ARP-K2, Rev. 4, “Control Room Panel K2 Alarm Response,” had not been revised to resolve this problem. The team determined this issue to be of very low risk significance (Green) by the SDP phase 1 screening. This conclusion was based on a recognition that the simultaneous actuation of all four alarms following a LOCA, without a subsequent indication on the downstream service water system radiation monitor, would be sufficient information for the operators to recognize these as spurious alarms due to background radiation. The failure to implement corrective actions to correct the affected procedure is considered to be a non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Action. **(NCV 05000220/2000-007-04)** This item is in the corrective action program as DER 1-2000-3293.

<u>DER No. 2-1998-3093 - Service Water System Operation</u>

NMPC initiated DER 2-1998-3093 to address operational concerns associated with the assumptions and results of calculation No. A10.1-N-34, “Three SWP Pumps - LOCA Under Degraded Conditions.” This calculation was performed to support TS changes associated with the service water system. TS LCO 3.7.1.1 was changed to require that in operational conditions 1, 2, and 3, four service water pumps shall be operating and the divisional cross connect valves shall be open. In this alignment, the most limiting design bases event is a Loss of Coolant Accident (LOCA) followed by the single failure of one of the four operating service water pumps.

NMPC determined that when operating a residual heat removal system (RHR) heat exchanger at full service water flow following a LOCA, operator action was required to start an additional pump or secure non-essential service water loads if no additional service water pumps are available. This action is necessary to prevent operation of the system at flow rates which have not been analyzed.

The team noted that following an accident the EOPs direct placing both RHR heat exchangers in service at 7400 gpm (full) flow each when operating in the suppression pool cooling and/or containment spray modes. Also, the team found that procedure N2-EOP-PC, “Primary Containment Control,” directed the initiation of suppression pool cooling but did not provide any specific direction for aligning the RHR or SW systems for this mode. NMPC operations personnel indicated that procedure N2-OP-14, “Residual Heat Removal System,” would be used to perform the alignment and section H.12 of that procedure provides the appropriate direction for the system operation. The team also noted that when initiated containment spray, step PCP 5 of EOP N2-EOP-PC refers the operator to procedure N2-EOP-6, “NMP2 EOP Support Procedure,” Attachment 22. However, this procedure does not provide specific direction for aligning the RHR and SWP systems.

NMPC considers the EOP attachments to be “stand-alone” documents which should be capable of being performed without utilizing additional procedures. As such, performance of the steps as stated could result in operating the plant in a high flow condition that had not been evaluated for potential concerns such as pump runout and/or motor overload. The team determined this issue to be of very low risk significance (Green) by the SDP phase 1 screening. This conclusion was based on the fact that the system operating procedure includes the appropriate guidance, as does operator training, and the issue does not affected the operability of the service water system.

The failure of NMPC to implement appropriate corrective actions to ensure adequacy of operating procedures is considered an additional example of the non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Action. **(NCV 05000410/2000-007-04)** NMPC entered this issue into the corrective action program as DER 2-2000-3351.

4OA5 Management Meetings

The team presented the preliminary inspection findings to Mr. J. Mueller and other members of NMPC management on September 29, 2000, who acknowledged the findings presented. An additional phone discussion to discuss inspection findings was conducted on November 5, 2000.

PARTIAL LIST OF PERSONS CONTACTED

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L. Doerflein	Chief, DRS Systems Branch
G. Hunegs	Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000220/2000-007-01	UNR	Potential Unanalyzed Leak Path.
05000220/2000-007-03	UNR	Unit 2 service water model accuracy.
05000220/2000-007-05	UNR	Unit 1heat exchanger fouling factors.

Opened/Closed

05000220/2000-007-02 05000410/2000-007-02	NCV	Inadequate design inputs and failure to translate design into specifications and procedures.
05000220/2000-007-04 05000410/2000-007-04	NCV	Failure to properly implement corrective actions.

Closed

05000410/2000-007-00	LER	Plant Outside Design Basis due to Single Failure Susceptibility of Service Water and Emergency Core Cooling Systems

LIST OF ACRONYMS USED

AOV	Air Operated Valve
ARP	Annunciator Response Procedure
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
DP	Differential Pressure
DER	Deviation Event Report
ECCS	Emergency Core Cooling System
EOP	Emergency Operation Procedure
FCV	Flow Control Valve
gpm	gallons per minute
HPCS	High Pressure Core Spray
IST	In-service Testing
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MOV	Motor-Operated Valve
NRC	Nuclear Regulatory Commission
OP	Operating Procedure
PCE	Procedure Change Evaluation
P&ID	Piping & Instrument Diagram
psid	pounds per square inch differential
RHR	Residual Heat Removal
RPS	Reactor Protection System
SDP	Significance Determination Process
SDBD	System Design Basis Document
SE	Safety Evaluation
SRA	Senior Reactor Analyst
SSC	Structures, Systems, and Components
ST	Surveillance Test
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report

ATTACHMENT 1
LIST OF DOCUMENTS REVIEWED

Calculations

Unit 1

NER-10-016, Revision 0,	Evaluation of Drywell to Wetwell Bypass Flow Path on Peak Torus Pressure
S14-80-F002, Revision 1,	Containment Spray Suction Froude Numbers
S14-80-F003, Revision 3,	Containment Spray Pump NPSH Available Versus Required
S14-80F006, Revision 0,	Hydraulic Resistance of Containment Spray System 112
S14-80F007, Revision 0,	Hydraulic Resistance of Containment Spray System 111
S14-80F010, Revision 0,	Containment Spray Flow Rates
S14-80-F014, Revision 4,	Containment Spray System IST Approved Pump Curves and Design Bases
S14-80F030, Revision 0,	Containment Spray System Design Basis Hydraulic Analysis
S14-93-F003, Revision 2,	IST Approved Pump Curves - Containment Spray Raw Water Pumps, including Calculation Disposition 1 to Revision 1
S14-93F007, Revision 3,	Containment Spray Raw Water Required Pressure and Total Design Head
S14-93HT01, Revision 1,	Containment Spray Raw Water Heat Exchanger
S14-93-HT02, Revision 0,	Thermal Hydraulic Performance Analysis of Containment Spray Heat Exchangers
S14-93-HT03, Revision 0,	Containment Heat Exchanger at 3400 gpm and 81°F
S14-93HX01, Revision 1,	CSRW HX Thermal Performance Evaluation
S14-93-HX04, Revision 0,	CSRW Heat Exchanger Thermal Performance Evaluation
S14-93HX05, Revision 0,	Containment Spray Loop 111 Test Results
S14-93HX07, Revision 0,	Determine HX K Value Input to Revised Containment Analysis, including Calculation Disposition 00A
S14-93LBP0006A, Rev. 0,	Containment Spray Raw Water System
S14STRAINERM002, Rev. 1,	ECCS System Strainer Air/Steam Ingestion Analysis
S0TORUSM09, Revision 1,	Disposition 01D, Maximum Torus Temperature assuming as found containment spray heat exchangers fouling.
ELMSAC-DEGVOLT-STUDY, Revision 0,	Degraded Voltage Calculation for Nine Mile Point - Unit 1 Final Report
4160VAC-PB102 & 103-PD, Revision 0,	Coordination & Protection Study for PB102 & 103
EC-136, Revision 4,	Degraded Voltage Relay Set point
EC-196, Revision 1,	Degraded Grid Relay, Under Voltage Relay & Associated Timer Relay Set point Calculation
CS-SWP*01, Revision 5,	Set point Calculation for Bistables 2SWP*FSL96A, B, C, D, E, F Service Water Pump - Low Flow Trip
12177-CS-SWP*01,	Set point and Reset Setting for SWP Discharge Flow Switches 2SWP*FLS 96A, B, C, D, E, F
SO-APPJ-M003,	Containment Spray Leakage Assessment
1H-013, Revision 1,	Containment Spray Leakage
SOTORUSM009, Rev. 1,	Nine Mile Point Unit 1 Pool Heatup Analysis, dated February 23, 2000

ATTACHMENT 1- List of Documents Reviewed (Cont.)

Unit 2

A10.1-N-097, Revision 2,	Disposition 2B, SWP Steady State Analysis LOCA + LOOP, Div I, III, >10 minutes
A10.1-N-098, Revision 2,	Disposition 2B, SWP Steady State Analysis LOCA + LOOP, Div II, III, >10 minutes
A10.1-N-099, Revision 2,	Disposition 2B, SWP Steady State Analysis LOCA + LOOP, Div II, III, <10 minutes
A10.1-N-340, Revision 0,	PROTO -FLO SWP Base Hydraulic Model - Normal Operation, April 15, 2000
A10.1-N-341, Revision 0,	Three SWP Pumps - LOCA Under Degraded Conditions
CS-HVR*06, Revision 1,	Reactor Building Emergency Ventilation Recirculation Unit Cooler Auto-Start Time Delay Relays, dated June 28, 1998
HVC-064, Revision 03,	Disposition 03A, Heat Gain and Cooling Requirements for Standby Switchgear Rooms-Control Building Elevation 261 Ft
HVC-075, Revision 00,	Dispositions 00C and 00D, Evaluate Duty and Flow Thru 2HVC*UC102

Design Bases Documents

SDBD-203, Revision 1,	System Design Basis Document for Containment Spray
SDBD Change Notices	-N1-94-001-LS111, N1-91-009-LS004, N1-91-009-LS005, 1F00117A, 1S00112, 1M00042, 1F01067A, 1E00459, 1M00689, 1E00604
SDBD-803, Revision 2,	AC Electrical Distribution System
SDBD-SWP, Revision 0,	Service Water System Design Basis Document

Deviation/Event Reports

DER 1-1991-1424	Containment Spray Heat Exchanger Vents
DER 1-1991-1616	Containment Spray Cooling Water PSV Set point
DER 1-1992-4195	Programmatic Controls For Pump Motor Current Test Data
DER 1-1994-1974	Containment Spray Heat Exchanger
DER 1-1998-0644	Containment Spray Raw Water Rad Monitor Problems not resolved
DER 1-1998-2285	Valve 93-71 Dual Indication
DER 1-1998-2349	Excessive Sludge Found In Raw Water Piping
DER 1-1998-2403	Containment Spray Loop 121 Drain Time Excessive
DER 1-1998-2826	CSP 121 Low Suction Pressure
DER 1-1998-3175	Unclear LCO Requirement for CTN Spray Water Seal
DER 1-1998-3460	Containment Spray System LCO clarity SE 89-13
DER 1-1999-2688	Valve 80-46 Failed to Open
DER 1-1999-0831	Inappropriate Design Input
DER 1-1999-4063	K1-1-7 Received when Securing Containment Spray
DER 1-1999-4212	Containment Spray System 121 Declared Inoperable
DER 2-1992-2281	2SWP*V17 and V32 - Bypass Valves for Rx and Turb.
DER 2-1996-0680	Motor Running Current Exceeds Acceptance Criteria
DER 2-1997-0804	Inadequate Documentation Basis for the Selection of Set points
DER 2-1997-2714	Zebra Mussel Bio Box Skid Connection to Service Water Not Controlled

ATTACHMENT 1- List of Documents Reviewed (Cont.)

DER 2-1998-0475	Impeller on 2SWP*P1C Needs Replacement After 2 Years Per Vendor
DER 2-1998-1391	Suction Line to 2SWP*RV3B Full of Mud and Sand
DER 2-1998-1858	Significant Corrosion Buildup on Service Water Valves
DER 2-1998-2642	The Number of SWP Bar Rack Heater Required by TS 4.7.1.1.2 is Inadequate to Support Calculation Assumptions for A10.1-N-303
DER 2-1998-2892	Tech Spec SR 4.7.1.1.2.B is Non-conservative
DER 2-1998-3093	WA-Outstanding Operational Questions and Concerns Regarding Calculation A10.1-N-341
DER 2-1999-0293	Incorrect Operating Coil of 120VAC Discovered in Starter/Contactor for 2SWP*MOV1E at 2EHS*MCC101-3A
DER 2-1999-0794	USAR Table 9B.8-3, SWP Bar Rack Heaters Not Included IN Control Room Fire Analysis
DER 2-1999-1491	Controlled LCR IL2SWP-009 and SPDS 2SWP*FSL534 Contain Inaccurate Information
DER 2-1999-1898	Possible Incorrect Termination on Motor Leads
DER 2-1999-1961	Multiple Safety Class Determinations Associated with One Component
DER 2-1999-2044	Over-greasing of 2SWP-P3 Leads to Hot Bearings.
DER 2-1999-2523	Inlet Line to 2SWP*RV80E Found 15-50% Plugged.
DER 2-1999-2765	Low Suction Pressure on 2SWP*P1F
DER 2-1999-3719	Wiring Error on Div II Bar Rack Heater Control Circuit
DER 2-1999-3804	2HVP*UC2 Failed Thermal Performance Evaluation Test
DER 2-2000-0072	Flow Through Unit Cooler 2HVR*UC407D exceeds Design Flow.
DER 2-2000-0086	NER-2M-008 States Incorrect Flow Direction
DER 2-2000-0429	Potential Degradation of Unit Cooler Air Flows
DER 2-2000-0931	QI-Below Min Wall Readings Found on E/C Component NP0063
DER 2-2000-1034	Loss of Division II Service Water
DER 2-2000-1120	Fuse Blown While Landing Jumper
DER 2-2000-1124	Division II Service Water Pump Non Essentials Valve 2SWP*MOV19B Failure to Stay Open
DER 2-2000-1245	Service Water System May not Meet Single Failure Requirements
DER 2-2000-1334	SWP Non-Essential Header Isolations Went Closed
DER 2-2000-1444	Foreign Material on Optical Isolator Output Card
DER 2-2000-1452	Postulated Single Failure of HPCS Diesel Under DBA LOCA/LOOP Results in Potential Loss of Division 1
DER 2-2000-1493	Entered EOP-SCC (Floor Drain Sump Water High-High Alarm)
DER 2-2000-1585	Service Water Pump Strainer Allowable Pressure Drop per TS Table Exceeds Analysis Inputs.
DER 2-2000-1592	2SWP*V260 Failed Reverse Flow Test
DER 2-2000-1818	OE-Service Water Pump Expansion Joint Water Hammer.
DER 2-2000-2315	Potential Trip of 2SWP*MOV95A, 95B
DER 2-2000-2986	2SWP*V260 Failed Reverse Flow Exercise Test of N2-OSP-SWP-Q005

ATTACHMENT 1- List of Documents Reviewed (Cont.)

Electrical Schematics & Logic Diagrams

C-19409-C, Sheet 8, Revision 44, One Line Diagram Aux. System 600 Volt Power Boards 16, 161A & 161B
C-19409-C, Sheet 9, Revision 39, One Line Diagram Aux. System 600 Volt Power Boards 17, 171A & 171B
C-19410-C, Sheet 2, Revision 20, Elementary Wiring Diagram 4.16KV Emergency Power Boards & Diesel Generators (#102 & #103 Power Circuits)
C-19437-C, Sheet 2, Revision 40, Sheet 5, Revision 23, Sheet 11, Revision 10, Elementary Wiring Diagram 600 Volt Power Board 161B Power Circuits
C-19438-C, Sheet 1, Revision 29, Sheet 9, Revision 5, Elementary Wiring Diagram 600 Volt Power Board 167 Power Circuits
C-19440-C, Sheet 2, Revision 39, Sheet 5, Revision 22, Sheet 12, Revision 12, Elementary Wiring Program 600 Volt Power Board 171B Power Circuits
C-19951-C, Sheet 5, Revision 12, Elementary Wiring Diagram Miscellaneous Solenoid Valves
12177-ESK-5SWP01, Sheet 1, Revision 13, Sheet 2, Revision 14, DC Elementary Diagram - 4.16KV Switchgear Circuit Service Water Pump
12177-ESK-5SWP07, Sheet 1, Revision 11, Sheet 2, Revision 10, DC Elementary Diagram 4.16KV Switchgear Circuit Service Water Pump 1A
12177-ESK-6SWP01, Revision 13, AC Elementary Diagram- 600 V MCC Circuits Service Water to Turbine Building MOV’s
12177-ESK-6SWP11, Revision 13, AC Elementary Diagram - 600 V MCC Circuits Service Water Pump make up to CW Flow Valve HYDR Pump
12177-ESK-6SWP14, Sheet 1, Revision 14, Sheet 2, Revision 1, AC Elementary Diagram - 600 V MCC Circuit Service Water Pump HDR Isolation MOV’s
12177-ESK-6SWP18, Sheet 1, Revision 9, Sheet 2, Revision 9, AC Elementary Diagram - 600 V MCC Circuit Service Water Pump Discharge MOV’s
12177-ESK-6SWP21, Sheet 1, Revision 12, Sheet 2, Revision 13, AC Elementary Diagram - 600 V MCC Circuits Service Water Backwash Valves
12177-ESK-6SWP11, Revision 13, AC Elementary Diagram - 600 V MCC Circuit Make up to CW Flow Valve HYDR Pump - Unit 2
12177-ESK-6SWP26, Revision 9, AC Elementary Diagram 600 V MCC Circuit SWP MOV’s from Diesel Generator 2EGS*EG2 CLR
12117-ESK-7SWP21, Revision 6, AC Elementary Diagram 120 VAC Circuits - Unit 2
12177-ESK-6SWP42, Revision 7, AC Elementary Diagram 600 V MCC Circuit SWP Makeup to CW Flow Valve HYDR Pump - Unit 2

12177-ESK-7SWP17, Revision 12, DC Elementary Diagram Miscellaneous DC Circuits Service Water Loss of Offsite Power Control

ATTACHMENT 1- List of Documents Reviewed (Cont.)

12177-ESK-7SWP27, Revision 8,	AC Diagram 120 VAC Circuits SWP Strainer Aux. Control Circuits
12177-ESK-7SWP05, Revision 10,	AC Elementary Diagram - Miscellaneous AC Circuits Service Water Pump Unit Cooler Valves
12177-ESK-7SWP20, Revision 5,	AC Elementary Diagram - Miscellaneous AC Circuits Service Water Pump Unit Cooler Valves
EE-M01A, Revision 14,	Plant Master One Line Diagram Normal Power Distribution
EE-M01B, Revision 7,	Plant Master One Line Diagram Emergency Power Distribution
EE-1Q, Revision 16,	4160V One Line Diagram - 2ENS*SWG101 (-G) Emergency Bus
EE-1R, Revision 15,	4160V One Line Diagram 2ENS*SWG103 (-Y) Emergency Bus
LSK-9-10A, Revision 15	Logic Diagram - Service Water
LSK-9-10B, Revision 15	
LSK-9-10C, Revision 15	
LSK-9-10D, Revision 14	
LSK-9-10E, Revision 14	
LSK-9-10F, Revision 18	
LSK-9-10G, Revision 17	
LSK-9-10H, Revision 17	
LSK-9-10K, Revision 18	
LSK-9-10Q, Revision 15	
LSK-9-10Y, Revision 13	
LSK-9-10AE, Revision 2	

Modifications

N1-91-009, Revision 1,	Replace Operators on Containment Spray Inter-tie Valves EBN 80-40 & 80-45
N1-96-005,	ECCS Suction Strainer Replacement
DDC 1E00459,	New Alarm Set point for ECCS Pump Low Suction Pressure Switch
DDC 1E00385A, Revision A	Support Removal of Differential Pressure Switches
DDC 1M00628,	Support Installation of Large Capacity Suction Strainers, the Core & Containment Spray Pump Suction Lines
DCC 2E12083,	Add a Division I Loss of Off-site Power Contact
DCC 2E12084,	Add a Division II Loss of Off-site Power Contact
2M11792,	Reduce Maximum Allowable IST Stroke Times for SWP Valves
PC2-8350-50,	Service Water Hammer Related Modifications
PC2-9014-50,	Service Water Chemical Treatment
PC2-0083-91,	2SWP*STR4A/B/C/D/E/F Coupling Pin Improvement
PC2-0062-97,	Removal of Trip Function of Chiller Low Service Water Flow 2SWP*FSL29A and FSL29B
PC2-0034-97, 2E11307 -	Damaged Cable Replacement 2SWPAYC709
PC2-0135-92,	Internal Wiring Changes in 2SWP-PNL170 to Eliminate Sneak Circuit
PC2-0141-99, 2A00035 -	Revise Pump Acceptance Criteria in M2-0006

PC2-0121-91,	Lower Set point for 2SWP*CAB23B, A
PC2-0125-94,	Service Water Strainer Alignment Modification
PC2-8259-50,	Service Water Strainer Improvement
PC2-0031-94,	IST - SWP Check Valve Internal Removal
PC2-0096-30,	Removal of Service Water Check Valves 2SWP*V201A and *V201B

Nuclear Engineering Reports

NER-2M-026,	Service Water System Operational Guideline Modes 4 and 5
NER-2A-002-SWP,	In-service Test Program Bases Document For IST Program Attachment Service Water (SWP)
NER-2M-041,	SWP Pump Operability Status Assessment For Inoperable SWP Valves (2SWP*MOV66A/B)

Piping & Instrument Drawings (P&ID)

C-18012-C, Sheet 1, Revision 22,	Reactor Containment Spray Raw Water System
C-18012-C, Sheet 2, Revision 43,	Reactor Containment Spray System
C-18012-C, Sheets 1,2,3,	Drywell and Torus Isolation Valves

PID-11A-15, Revision 15,	Piping & Instrumentation Diagram Service Water System
PID-11B-15, Revision 15	
PID-11C-16, Revision 16	
PID-11D-10, Revision 10	
PID-11E-11, Revision 11	
PID-11F-22, Revision 22	
PID-11G-16, Revision 16	
PID-11H-26, Revision 26	
PID-11J-17, Revision 17	
PID-11L-19, Revision 19	
PID-11M-14, Revision 14	
PID-11N-9, Revision 9	
PID-11P-25, Revision 25	
PID-11Q-8, Revision 8	

Procedures

N1-ARP-H1, Rev. 3, Control Room Panel H1 Alarm Response Procedures
N1-ARP-F1, Rev. 4, Control Room Panel F1 Alarm Response Procedure
N1-ARP-K1, Rev. 4, Control Room Panel K1 Alarm Response Procedure
N1-ARP-K2, Rev. 4, Control Room Panel K2 Alarm Response Procedures
N1-EOP-4, Rev. 8, Primary Containment Control Emergency Operating Procedure
N1-EOP-4.2, Rev. 1, Hydrogen Control Emergency Operating Procedure
N1-EOP-1, NMP1 EOP Support Procedure Rev. 3
N1-EPM-GEN-150, Revision 5, 4.16KV Breaker & Motor Inspection
N1-IPM-093-001, Containment Spray Raw Water System Flow
N1-MPM-080-410, Containment Spray Heat Exchangers Preventive Maintenance (80-13R, 80-

14R, 80-33R, 80-34R)
N1-ODP-PRO-0305, Rev. 0, EOP/SAP Technical Basis
N1-OP-2, Rev. 28, Core Spray System Operation Procedure
N1-OP-14, Rev. 41, Containment Spray System Operating Procedures
N1-ST-C4, Rev. 07, Containment Spray Air Flow For Spray Headers And Nozzles Test
N1-ST-Q6A, Rev. 06, Containment Spray System Loop 111 Quarterly Operability Test
N1-ST-Q6B, Rev. 06, Containment Spray System Loop 121 Quarterly Operability Test
N1-ST-Q6C, Rev. 06, Containment Spray System Loop 112 Quarterly Operability Test
N1-ST-Q6D, Rev. 06, Containment Spray System Loop 122 Quarterly Operability Test
N1-ST-Q28, Rev. 07, Containment Spray Raw Water Inter-Tie Check Valve Quarterly Operability Test
N1-ST-R2, Revision 23, LOCA & EDG Simulated Auto Initiation Test
N1-ST-SO, Rev. 20, Shift Checks Surveillance Test Procedure
N1-TSP-201-001, Integrated Diesel Generator Load Testing
N2-ARP-01, Rev. 0, 2CEC*PNL852 Series 100 Alarm Response Procedure
N2-ARP-01, Service Water Pump 1A/1C/1E Discharge Flow Low
N2-EOP-PCH, Rev. 0, Hydrogen Control Emergency Operating Procedure
N2-EOP-PC, Rev. 9, Primary Containment Control Emergency Operating Procedure
N2-EOP-6, Rev. 5, NMP2 EOP Support Procedure
N2-EPM-GEN-5Y550, Rev. 5, GE 4.16KV Magne-Blast Breakers & Associated Motors
N2-ESP-SWP-R791, Rev. 3, Refueling Cycle SW Heater Test
N2-ESP-SWP-W790, Rev. 7, Weekly SW Heater Resistance Test
N2-IPM-SWP-A101, Rev. 1, Service Water Pump Discharge Flow Instrument Calibration
N2-MAP-SAT-0101, Rev. 06, Check Valve Inspection
N2-OP-11, Rev. 8, Service Water System Operating Procedure, dated March 25, 2000
N2-OP-13, Rev. 6, Reactor Building Closed Loop Cooling System Operating Procedure
N2-OP-14, Rev. 4, Turbine Building Closed Loop Cooling System Operating Procedure
N2-OP-31, Rev. 14, Residual Heat Removal System Operating Procedure
N2-OP-52, Rev. 6, Reactor Building Ventilation Operating Procedure
N2-OP-100B, Rev. 7, HPCS Diesel Generator Operating Procedure
N2-OP-100A, Rev. 8, Standby Diesel Generators Operating Procedure
N2-OSP-EGS-R004, Revision 4, Operating Cycle Diesel Generator Simulated Loss of Off Site Power with ECCS Div I & II
N2-OSP-SWP-@001, Service Water Pump Curve Validation Test
N2-OSP-SWP-CS001, Service Water Valve Operability Test
N2-OSP-SWP-CS002, Service Water Check Valves Forward and Reverse Flow Exercise Test
N2-OSP-SWP-Q@001, Division 1 Service Water Operability Test and ASME XI Pressure Test
N2-OSP-SWP-Q@003, Control Building Chiller Condensing Water Pump Operability Test and ASME XI Pressure Test
N2-OSP-SWP-Q002, Service Water Pump and Valve Operability Test
N2-OSP-SWP-Q004, Division 2 Service Water Operability Test
N2-OSP-SWP-Q005, Rev. 01, Division 3 Service Water Operability Test
N2-OSP-SWP-R001, Rev. 04, Service Water Actuation Test
N2-OSP-SWP-R002, Rev. 03, Service Water Valve Position Indication Operability Test
N2-OSP-SWP-R003, Rev. 04, Diesel Generator Loss of Offsite Power with No ECCS Division I & II
N2-RCPM-GEN-V070, Rev. 00, Protective/Auxiliary Relays and Timers
N2-SOP-03, Rev. 03, Loss of AC Power

N2-SOP-11, Rev. 00, Loss of Service Water Special Operating Procedure
N2-TTP-HVC-@101, Rev. 03, Performance Evaluation Test For Unit Cooler 2HVC*UC101A and B
N2-TTP-HVC-@103, Rev. 00, Performance Evaluation Test For Unit Cooler 2HVC*UC103A and B
N2-TTP-HVC-@106, Rev. 02, Performance Evaluation Test For Unit Cooler 2HVC*UC106
N2-TTP-SWP-012, Rev. 00, Validation of Flow After the Removal of Service Water Check Valves 2SWP*V201A and 2SWPV201B
N2-TTP-THR-V001, Rev. 0, Thermography of Service Water Pumps
TE-80-50, Temperature Loop Calibration Containment Heat Exchanger 111 Inlet Temp

Program Plans

NMP2-IST-005, Rev. 0, “In-service Pump and Valve Testing Program Plan Second Ten-Year Interval
NMP1-IST-003, Rev. 3, In-service Pump and Valve Testing Program Plan Third Ten-Year Interval

Safety & Availability Assessments

SAS-00-018, Rev. 0, Loss of Division II Service Water
SAS-00-022, Rev. 0, Loss of Division II Service Water Single Failure
SAS-00-035, Rev. 0, NMP2 HPCS Single Failure Due to SW Problems
SAS-00-035, Rev. 1, Re-analysis of NMP2 HPCS Single Failure Due to SW Problem

Safety Evaluations and Applicability Reviews

AR37767, Replacement of Containment Spray Raw Water Pump and Components for 93-04 by Configuration Change 1F01067
SE 81-20, Torus Modifications
SE 84-25, Equipment Inspection and Replacement for Environmental Qualification - Maintenance Items Only
SE 84-37, Reactor Instrumentation, Emergency Condenser Core Spray and Emergency Ventilation System
SE 84-72, Containment Spray (80) Containment Spray Raw Water (93), Reactor Containment N2 Purge Fill (201.8) and Reactor Containment N2 Supply (201.9)
SE 84-76, Replace Solenoid Valves & Position Limit Switches on Valves 80-15, 16
SE 84-77, Replace Flow Transmitters 80-49A, 56A and 93-30A, 33A
SE 84-78, Relocate Cleanup IV’s 33-01, 02, from Power Board 167 to power Boards 161B & 171B & Containment Spray Valves 80-02, 21 from Power Boards 161B, 171B to Power Board 167
SE 84-91, Replace Position Limit Switches on Valves 80-35, 36, 40, 41, 44, & 45, 63-04 and 05
SE 84-93, Replace Flow Transmitters 80-71A, 76A, 93-32A, 34A, 81.1-02(RV-26B)
SE 84-95, Replace Solenoid Valves on Valves 80-15, 16, 35 & 36
SE 86-002, Containment Spray Heat Exchanger Replacement - Modification Number 85-52
1979 Modification to Reroute Line to Containment Spray Header in Torus
SE 89-008, Service Water System Divisional Separation Logic Associated with

2SWP*FV47A/B & FV54A/B
SE 89-013, Containment Spray Post DBA LOCA Appendix J Water Seal
SE 91-023, Heat Removal Capacity of the Containment Spray System based on the Design Basis Reconstitution LOCA Suppression Chamber Temperature Response Analysis
SE 92-041, Revise Logic for Service Water Valves MOV 95A/B and MOV 66A/B
SE 92-057, Provide Venting Capability to the Containment Spray Heat Exchangers
SE 92-067, Delete SWP/CWS Make-up Header Low Pressure Annunciator (Nuisance Alarm)
SE 93-014, Replace Bowl Assemblies and Shafting For The Containment Spray Raw Water Pumps - Modification Number SDC SC1-0014-93
SE 94-036, Evaluation of BWROG-EPG Drywell Spray Limitations on the Containment Spray System Appendix J Water Seal
SE 94-057, Replacing Operators on Containment Spray Inter-tie Valves 80-40 & 80-45
SE 95-047, USAR Table 3.9A-12 Update
SE 96-082, Removal of Service Water Check Valves 2SWP*V201A and *V201B
SE 97-065, Inlet Service Water Temperature Control for Control Building Chillers
SE 98-054, Service Water Lineup During Unit Outages
SE 99-093, Service Water System Lineup During Unit Outages
SE 2000-046, Service Water System Low Flow Trip Avoidance
SE 00-048, Removal of the Service Water Pumps Low Flow Trip

Specifications

MDC-11, Rev. 09, Nine Mile Point Unit 1 Pump Curves and Acceptance Criteria
M2-0005, Rev. 05, ASME Section XI Check Valve Acceptance Criteria
M2-0003, Rev. 05, In-service Test Program Plan Valve Stroke Time Limits in Safety Direction
M2-0006, Rev. 04, ASME OM In-service Test Pump Performance Acceptance Criteria

System Health Reports

Unit 1 Containment Spray and Containment Spray Raw Water
Unit 2 Service Water System

Training Documents

O2-OPS-001-276-2-00, Rev., Intake Structure and Service Water System (SWP)

Vendor Manuals

N1W31500PUMP001, Worthington Pump Bowl Assemblies
N1W31500PUMP002, Can Pumps Instruction Manual & Parts List
N2L20000VALV0P004, Rev. 01, Limitorque Type SMB Instruction & Maintenance Manual
N1001835HTEXCH001, Rev. 00, Installation, Operation, and Maintenance Manual for Shell and Tube Heat Exchangers

Work Orders

Work Order 1H3ND, Rev. 0, Medium Voltage Circuit Breaker Overhaul

Miscellaneous

Problem Report 2140, Problem report which documents the fouling condition of the heat exchanger prohibiting design performance.

Station Operations Review Committee Meeting Minutes, May 19, 1994

Problem Report PR 2179. Significant Buildup of Scale for Containment Spray Heat Exchangers.

Nuclear Engineering Procedure, NEP-DES-13, Preparation and Control of Design Basis Documents, Revision 1

Nuclear Engineering Procedure, NEP-DES-04, Design Document Changes, Revision 6

DOCUMENTS REVIEWED FOR IP 71111.02

Unit 1

Safety Evaluations

93-008,Rev. O, Local Leak Rate Testing of Double Gasketed Primary Containment Penetration and Air Locks
98-01, Rev. 0, Revised Appendix R Safe Shutdown Analysis for Emergency Cooling (EC), Reactor Water Cleanup (CU), Shutdown Cooling (SDC), Main Steam (MS), and Remote Shutdown (RDS)
98-10, Rrev, 0, Change Control Room Air Treatment System Intake Duct High Radiation Monitor Set Point, and Add Loss of Coolant Accident Signal and Main Steam Line Break Signal to Control Room Emergency Ventilation System Initiation Logic.

Applicability Reviews

Configuration Control Changes - AR-36574; AR-37594
Temporary Modifications - AR-36573; AR-24557; AR-37703; AR-37723; AR-37780; AR-37770; AR37768; AR-35057
Design Changes - AR-24411; AR-23527; AR-24460; AR-35515
Procedure Changes - AR-35193; AR-35195;

Deviation/Event Reports

DER 1-1999-1480	Safety Evaluation 98-104 Conformance Section Does Not Support Conclusion in the Safety Evaluation.
DER 1-1999-2515	Failure to Perform Safety Evaluation for Retirement of RBEV Humidity Indicators.

Unit 2

Safety Evaluations

98-058, Rev. 0, The HVK Makeup Water Supply Pressure Control Valve Upgrade and A Pressure Relief Valve addition.
99-066, Rev. 0, Incorrect Analytical Value Reported in UFSAR.
99-093, Rev. 0, Service Water Valve Lineup During Unit Outages.
2000-040, Rev. 0, Actuator Motor Change-out for 2ICS*MOV121.
2000-048, Rev. 00, Removal of the Service Water Pumps Low Flow Trip.
2000-080, Rev. 0, Parallel Operation of Reactor Reticulation and Shutdown Cooling Pumps

Applicability Reviews

Configuration Control - AR-42959; AR-43064; AR-15100; AR17580; AR-43047AR-26209;
Modifications - AR45482; AR-45414; AR-25698

Deviation/Event Reports

DER 2-1998-3746	Failure to Perform Safety Evaluation.
DER 2-1999-0184	LDCR Not Incorporated in USAR; Safety Evaluations Not Reported.
DER 2-1999-2976	Validity of Safety Evaluation Not Demonstrated.
DER 2-2000-0506	Safety Evaluation 99-088, Rev. 1 Rejected at SORC Meeting.
DER 2-2000-2762	Temporary Procedure Change to N2-op-15 without Required 50.59.

Miscellaneous Documents

Procedures
NC.NA-AP.ZZ-0059(Q), Service Water Pump Trouble Shooting
N1P-ECA-01, Attachment 5, Trend Codes

Letters: NMPIL 1265, 11/07/99 Submittal of Revision 15 to NMP-1 FSAR
NMPIL 1482, 11/05/99, Submittal of Revision 16 to NMP-1 FSAR
NMPIL 1838, 11/30/98, Submittal of Revision 10 to NMP2 FSAR including 50.59 Evaluations

ATTACHMENT 1- List of Documents Reviewed (Cont.)

Data sort Report of All U-2 DERs initiated between 01/01/95 and 09/10/2000.

Design Document Changes: SCI-0053-94, Modify CKV-80-38, Containment Spray Check Valve.

1S00112, Revise SDBD-203 and PSRS-80-1, Pipe Spec. Standards.

ATTACHMENT 2
NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues with low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, as described in the matrix. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix. More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.