Enclosure A to
NMP1L 1482

IDENTIFICATION OF CHANGES, REASONS AND BASES FOR NMPC-QATR-1 (UFSAR APPENDIX B)

ENCLOSURE A

IDENTIFICATION OF CHANGES, REASONS, AND BASES FOR QA PROGRAM DESCRIPTION CHANGES (UNIT 1 UFSAR APPENDIX B)

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.0-1, Before Introduction	Add the following explanation: Technical Specifications Reference: Throughout the QATR are references to Technical Specifications (TS) requirements. Where a specific TS is referenced it is noted as either "CTS" or "ITS". "CTS" refers to the current Technical Specifications for Units 1 and 2. "ITS" refers to the Unit 2 Technical Specifications in the "Improved Technical Specifications" format. This reference is valid following NRC approval of Niagara Mohawk Power Corporation's (NMPC) ITS submittal and implementation of the associated License Amendment. In summary, "CTS" always relates to a Unit 1 current TS reference, while "CTS" relates to a Unit 2 TS reference only before NRC approval of the ITS submittal, after which "ITS" identifies the appropriate Unit 2 TS reference.	This paragraph provides an explanation for references to the Technical Specifications. This is needed because the revised QATR will need to be applicable under both the CTS and, after approval by the NRC, the ITS. This note also makes the QATR consistent with the Unit 1 TS, regardless of whether or when the Administrative Section of the Unit 1 TS is revised to be consistent with the Unit 2 ITS format.	As part of the ITS program, TS are being moved into the QATR. These requirements will remain the same. Therefore, the program will continue to satisfy the requirements of 10CFR50 Appendix B as previously approved by the NRC.
Page B.1-1, Section B.1.2.1 First paragraph	Delete "President" (both places) and replace with "Chief Executive Officer". Delete "corporate officers" and replace with "Chief Nuclear Officer".	Organization change – NMPC will become a wholly-owned subsidiary of a new holding company. The President will become the Chief Executive Officer, and the Chief Nuclear Officer will function for the corporate officers.	A NRC Order approving application regarding restructuring of NMPC by establishment of a holding company (Niagara Mohawk Holdings, Inc.) was issued 12/11/98. The restructuring will not affect NMPC's position, responsibility or commitment as owner and operator of the facilities.

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.1-2, Section B.1.2.1.1, First paragraph	a. Delete "President" and replace with "Chief Executive Officer".	a. Organization change - NMPC will become a wholly-owned subsidiary of a new holding company. The President will become the Chief Executive Officer.	a. A NRC Order approving application regarding restructuring of NMPC by establishment of a holding company (Niagara Mohawk Holdings, Inc.) was issued 12/11/98. The restructuring will not affect NMPC's position, responsibility or commitment as owner and operator of the facilities.
	b. Delete "under the Vice President and General Manager".	b. Organization change - The position of Vice President and General Manager - Nuclear has been deleted. The Chief Nuclear Officer will assume corporate and TS responsibility for overall plant nuclear safety. This restores the responsibilities and authority to the levels that existed prior to 2/20/96. The Vice President Nuclear Generation will have oversight responsibility for the Unit 1 and Unit 2 operations, radiation protection, maintenance, chemistry, technical support, and outage management functions to assure safe, orderly, and efficient plant operation through direct reporting of both Unit Plant Managers.	b. The changes are not to the Quality Assurance (QA) organization. For TS Amendments 162 and 83, the NRC reviewed whether adverse changes to QA reporting might result from the upper level management changes. QA will continue to report through the division of safety assessment, independent of the line organization for operations, maintenance, and engineering. This independent reporting arrangement is in accordance with SRP 13.4 and was considered acceptable by the NRC. Refer to Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendments Nos. 162 and 83.

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Page B.1-2, Section B.1.2.1.1.1	a. Insert new position. The Vice President Nuclear Generation reports to the Chief Nuclear Officer, and has overall divisional responsibility for plant operation to assure safe, orderly, and efficient plant operation is achieved. Additionally, the Vice President Nuclear Generation has oversight responsibility for the Assessment and Corrective Action Group. The Plant Managers and the Director – Assessment and Corrective Action report directly to this Vice President.	a. This position is intended to improve efficiency and enhance senior management oversight of NMPC's nuclear generation facilities.	a. The authority and duties of the Vice President Nuclear Generation are clearly established and delineated in writing.
	b. Delete "and General Manager" in two places and insert "Generation" after Nuclear. Delete "and engineering". Delete "Vice President Nuclear Engineering and".	b. Organization change – The position of Vice President and General Manager – Nuclear has been deleted. The Chief Nuclear Officer will assume corporate and TS responsibility for overall plant nuclear safety. This restores the responsibilities and authority to the levels that existed prior to 2/20/96. The Vice President Nuclear Generation will have oversight responsibility for the Unit 1 and Unit 2 operations, radiation protection, maintenance, chemistry, technical support and outage management functions to assure safe, orderly, and efficient plant operation through direct reporting of both Unit Plant Managers.	b. The changes are not to the QA organization. For TS Amendments 162 and 83, the NRC reviewed whether adverse changes to QA reporting might result from the upper level management changes. QA will continue to report through the division of safety assessment, independent of the line organization for operations, maintenance, and engineering. This independent reporting arrangement is in accordance with SRP 13.4 and was considered acceptable by the NRC. Refer to Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendments Nos. 162 and 83.
	c. Under Item "a," added "certain" between "of" and "procedures".	c. Clarification – The Vice President Nuclear Generation is responsible for the procedures listed in Table B-1.	c. The responsibilities for procedures are reflected in Table B-1.

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Page B.1-2, Section B.1.2.1.1.1 (cont'd.)	d. Added Item "e" to describe activities associated with overview of assessment of industry and in-plant operating experience.	d. To describe established responsibility.	d. The authority and duties of the Vice President Nuclear Generation are clearly established and delineated in writing.
Page B.1-3, Section B.1.2.1.1.2	Add "The Vice President Nuclear Engineering reports to the Chief Nuclear Officer."	Organization change - The Vice President Engineering will report directly to the Chief Nuclear Officer.	The changes are not to the Quality Assurance organization. For TS Amendments 162 and 83, the NRC reviewed whether adverse changes to QA reporting might result from the upper level management changes. QA will continue to report through the division of safety assessment, independent of the line organization for operations, maintenance, and engineering. This independent reporting arrangement is in accordance with SRP 13.4 and was considered acceptable by the NRC. Refer to Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendments Nos. 162 and 83.
Page B.1-3, Section B.1.2.1.1.3.a Page B.1-4	Add the TS Section 6.2.3 description at the end of this section.	The ISEG requirements currently in the Unit 2 TS are to be relocated to the QA Program. These requirements are being relocated to this section because ISEG reports to the Vice President Nuclear Safety Assessment and Support.	As part of the ITS program, the Unit 2 TS are being relocated to the QATR. The ISEG requirements from the Unit 2 TS have been directly transferred to the QATR with the following exceptions: 1. Headings and section numbers have been altered to reflect the QATR format system. 2. Unit 2 has been appropriately inserted in the description of the ISEG function as the ISEG function is only required for Unit 2. These requirements will remain the same. Therefore, the program will continue to satisfy the requirements of 10CFR50 Appendix B previously approved by the NRC.

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.2-5, Section B.2.2.15	Delete "presidential or".	Organization change - NMPC will become a wholly-owned subsidiary of a new holding company. The President will become the Chief Executive Officer.	A NRC Order approving application regarding restructuring of NMPC by establishment of a holding company (Niagara Mohawk Holdings, Inc.) was issued 12/11/98. The restructuring will not affect NMPC's position, responsibility or commitment as owner and operator of the facilities.
Page B.2-5, Section B.2.2.15.2	Delete "NMPC" and replace with "Niagara Mohawk Holdings, Inc."	Organization change - NMPC will become a wholly-owned subsidiary of a new holding company named Niagara Mohawk Holdings, Inc.	A NRC Order approving application regarding restructuring of NMPC by establishment of a holding company (Niagara Mohawk Holdings, Inc.) was issued 12/11/98. The restructuring will not affect NMPC's position, responsibility or commitment as owner and operator of the facilities.

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program *Continues to Satisfy* 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.2-6, Section B.2.2.16	Replace current section with TS Section 6.5.3 and make minor editorial changes.	SRAB requirements currently in the Unit 2 TS are to be relocated to the QA Program. Except as noted, the SRAB requirements were directly incorporated.	SRAB requirements currently in the Unit 2 TS are to be relocated to the QA Program. Except as noted below, the SRAB requirements were directly incorporated. 1. Headings and section numbers have been altered to reflect the QATR format system. 2. Paragraph 7.d has been revised to "the Operating License" rather than "this Operating License." Since the section is no longer in a document related to the Unit 2 operating license, this terminology is more accurate. 3. Paragraph 10.b has been revised to add item 7.f to a list of topics which require a report to the Chief Nuclear Officer. This was done to make the QATR consistent with Unit 1 requirements, and does not reduce Unit 2 CTS requirements. Since SRAB meetings cover both units and currently report their activities to the Chief Nuclear Officer within 14 days following each meeting, this additional requirement has no practical impact on SRAB current practices. Although there are some minor editorial differences, the change generally reflects the Unit 1 TS requirements except: 1. Section 6.5.3.10.a requires SRAB minutes to be issued within 30 days. The changes reflect the Unit 2 requirement for a 14-day report. This is not a reduction of the Unit 1 CTS requirement and, as discussed above, has no practical impact on current SRAB practices.

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.2-6, Section B.2.2.16 (cont'd.)			2. Section 6.5.3.10.c requires audit reports to be forwarded to the Chief Nuclear Officer within 90 days after review. The change reflects the Unit 2 requirement to forward audit reports within 30 days after audit completion. This is not a reduction of the Unit 1 CTS requirement, and reflects current QA practice and procedural requirements. 3. Paragraph 10.b of the revised QATR section requires item 7.b to provide a report to the Chief Nuclear Officer within 14 days. This requirement is not currently in the Unit 1 CTS; however, since SRAB meetings cover both units and since it currently reports its activities to the Chief Nuclear Officer within 14 days following each meeting, this additional requirement has no practical impact on SRAB current practices. The QA program will continue to implement the same or more conservative requirements. Therefore, the program will continue to satisfy the requirements of 10CFR50 Appendix B as previously approved by the NRC.

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.2-10, Section B.2.2.17	Replace current section with TS Section 6.5.1 and make minor editorial changes.	SORC requirements currently in the Unit 2 TS are to be relocated to the QA Program. Except as noted, the SORC requirements were directly incorporated.	SORC requirements currently in the Unit 2 TS are to be relocated to the QA Program. Except as noted below, the SORC requirements are directly incorporated. 1. Headings and section numbers have been altered to reflect the QATR format system. 2. A sentence has been added at the start of the section to reference the USAR section which addresses SORC. The Unit 1 UFSAR reference is also included. The addition of the USAR reference is required by the ITS submittal package. 3. An addition to new QATR paragraph B.2.2.17.6.b incorporates Unit 2 TS Section 6.6.b into the QATR as required by the ITS submittal package. 4. The end of the first sentence of Section 6.5.1.8 has been revised in the QATR revision to read, "...the Technical Specifications and this section" instead of "these technical specifications." Since the information is no longer uniquely in the TS, this phrasing is more logically accurate. Although there are some minor editorial differences, the changes generally reflect the Unit 1 TS except that revised QATR paragraph B.2.2.17.6.e is not required by the Unit 1 CTS. This paragraph is noted as applying to Unit 2. Therefore, the program will continue to satisfy the requirements of 10CFR50 Appendix B as previously approved by the NRC.
Page B.2-11, Section B.2.2.17.6.f	Add the following statement as item 6.f: Review of Licensee-initiated changes to the ODCM prior to implementation. Changes become effective upon acceptance by SORC.	This change adds a new QATR paragraph, B.2.2.17.6.f, which incorporates Unit 2 CTS requirement 6.14.2.b into the QATR as required by the ITS submittal package.	As part of the ITS program, Technical Specifications are being moved into the QATR. These requirements will remain the same. Therefore, the program will continue to satisfy the requirements of 10CFR50 Appendix B.

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Page B.2-11, Section B.2.2.17.7	Add the following statement as item 7.c: Review Safety Limit Violation Reports, submit Safety Limit Violation Report to SRAB and Vice President Nuclear Generation within 14 days of the violation, and notify the SRAB and Vice President Nuclear Generation within 24 hours in the event a Safety Limit is violated.	This change adds a new QATR paragraph, B.2.2.17.7.c, which incorporates Unit 2 CTS requirements of Sections 6.7.a, b, and c into the QATR as required by the ITS submittal package.	As part of the ITS program, the TS are being moved into the QATR. These requirements will remain the same. Therefore, the program will continue to satisfy the requirements of 10CFR50 Appendix B.

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.5-2, Section B.5.2.7 Page B.5-3, Section B.5.2.8 Page B.5-3, Section B.5.2.9	Add TS Section 6.8.2 as Section B.5.2.7, TS Section 6.8.3 as Section B.5.2.8, and TS Section 6.5.2 as Section B.5.2.9, and make minor editorial changes.	Procedural and review requirements currently in Sections 6.5.2, 6.8.2, and 6.8.3 of the Unit 2 TS are to be relocated to the QA Program. The change places a description of these requirements in the appropriate location within the QATR by adding new QATR Sections B.5.2.7 (CTS 6.8.2), B.5.2.8 (CTS 6.8.3) and B.5.2.9 (CTS 6.5.2). Except as noted, the procedure and review requirements were directly incorporated.	As part of the ITS program, TS are being moved into the QATR. Procedural and review requirements currently in Sections 6.5.2, 6.8.2, and 6.8.3 of the Unit 2 TS are to be relocated to the QA Program. Except as noted below, the changes directly incorporate the procedure and review requirements from the Unit 2 TS. Headings and section numbers have been altered to reflect the QATR format system. In general, these changes reflect the Unit 1 TS requirements except as follows: 1. Unit 1 CTS Section 6.5.2 contains a requirement not included in the Unit 2 CTS and not incorporated in the revised QATR. Specifically, there is a requirement that the Plant Manager assure the performance of an annual review of the Fire Protection Program and implementing procedures. This has not been added to the QATR at this time, but may be in the future pending the manner of the Unit 1 TS Administrative Section revision to be consistent with the Unit 2 ITS format. 2. Unit 1 CTS Sections 6.8.2 and 6.8.3 do not specifically note, as does the Unit 2 CTS, that reviews should be in accordance with TS Section 6.5.2; however, Unit 1 CTS Section 6.5.2 is essentially identical (except as noted above) to the Unit 2 section. Therefore, there is no practical impact on current activities and the requirements for the procedure and review process will remain the same. The QA Program will continue to satisfy the requirements of 10CFR50 Appendix B as previously approved by the NRC.

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Page B.17-1, Section B.17.2.2	Delete the last sentence and replace with TS Section 6.10, "Record Retention," and make some minor editorial changes.	Record requirements currently in the Unit 2 TS were relocated to the QA Program. This change places a description of record requirements in the appropriate location in the QATR, i.e., present Section B.2.17.2.2. Except as noted, the record retention requirements are directly incorporated.	As part of the ITS program, the TS are being moved into the QATR. Except as noted below, the changes directly incorporate the record requirements from the Unit 2 TS. 1. Headings and section numbers have been altered to reflect the QATR format system. 2. Revised QATR Section B.2.17.2.2.2.e is worded to reflect both the Unit 1 and Unit 2 record requirements, which are stated slightly differently in the respective CTS. Although there are some minor editorial differences, the changes generally reflect the Unit 1 TS requirements except: 1. Revised QATR Section B.2.17.2.2.2. "l" only applies to Unit 2 and has been so noted. 2. Unit 2 requires permanent retention of records related to reactor tests and experiments, while Unit 1 requires only a 5-year retention. This has no practical impact on the current procedural requirements for records related to reactor tests and experiments. These requirements for record retention will remain essentially the same. Therefore, the program will continue to satisfy the requirements of 10CFR50 Appendix B as previously approved by the NRC.
Page B.18-2, Section B.18.2.12	Delete "Unit 1, Sections 6.5.3.8.g, 6.13.1 and 6.13.2; and Unit 2, Sections 6.5.3.8.k, 6.5.3.8.1 and 6.5.3.8.m."	References to the TS are being deleted, as the requirements for both units will eventually be removed from the TS and reside only in the QATR.	The program will remain the same except that the requirements will reside in the QATR instead of the TS. As part of the ITS program, the TS are being moved into the QATR. These requirements will remain the same. Therefore, the program will continue to satisfy the requirements of 10CFR50 Appendix B as previously approved by the NRC.

UFSAR Appendix B Page/Section	Identification of Change	Reason for Change	Basis for Concluding that the Revised Program Continues to Satisfy 10CFR50 Appendix B and Commitments Previously Approved by the NRC
Table B-1, All Sheets	Delete VPGM-N heading.	Organization change – Position was deleted.	The TS change deletes that position and restores the table to the previous format.
Table B-1, Sheet 1, Item VIII	Add section "8" to the NQA-1 column.	Editorial. Was inadvertently left off table when it was reformatted.	The QA Program implements Section 8 of NQA-1. This was inadvertently left off the table when it was reformatted. The program continued to implement the requirement.
Table B-1, Sheet 1, Item IX	Add an "X" to the NG column.	QA requirements for special processes assigned to Nuclear Generation have been implemented. However, the "X" for Nuclear Generation was inappropriately removed from Table B-1 when it was reformatted.	The QA Program implements the requirements of 10CFR50 Appendix B, BTP 9.5-1 Appendix A, NQA-1, and ANS-3.2 for special processes including welding, heat treating, chemical cleaning, and special coatings. Nuclear Generation is responsible for these areas. Nuclear Generation procedures have been and continue to be in place for these special processes.
Table B-1, Sheet 2, Item XVII	Add an "X" to the NT, NE, and NG columns.	QA requirements for Quality Assurance records assigned to Nuclear Training, Nuclear Engineering, and Nuclear Generation have been implemented. However, the table was inappropriately updated to remove those organizations when it was reformatted.	The QA Program implements the requirements of 10CFR50 Appendix B, BTP 9.5-1 Appendix A, NQA-1, and ANS-3.2 for the records generated by these organizations. Procedures have and continue to require Quality Assurance records generated by these organizations to be processed in accordance with NMPC's Records Management Program (NIP-RMG-01).
Table B-3, Sheet 2, Items e. & g.	Add the following to the end of the last sentence: ", and Section B.5 of the QATR."	A reference was added to items "e" and "g" as some of the applicable requirements are being removed from the TS and relocated to QATR Section B.5.	As part of the ITS program, the TS are being moved into the QATR. These requirements will remain the same. Therefore, the program will continue to satisfy the requirements of 10CFR50 Appendix B as previously approved by the NRC.
Table B-4, Sheet 4	Add the following to the end of the last sentence: ", and Section B.5 of the QATR."	A reference was added to item "t" as some of the applicable requirements are being removed from the TS and relocated to QATR Section B.5.	As part of the ITS program, the TS are being moved into the QATR. These requirements will remain the same. Therefore, the program will continue to satisfy the requirements of 10CFR50 Appendix B as previously approved by the NRC.

Enclosure B to
NMP1L 1482

NINE MILE POINT – UNIT 1

SAFETY EVALUATION SUMMARY REPORT

1999

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

Safety Evaluation No.:	90-007 Rev. 0 & 1
Implementation Document No.:	Calculations S10-202-HV01, S10-202-HV02
UFSAR Affected Pages:	VII-37
System:	Reactor Building Emergency Ventilation
Title of Change:	Increase in Allowable Humidity

Description of Change:

The original design of the Reactor Building emergency ventilation system provided for the start of one pre-selected fan upon receipt of an initiation signal. In 1971 a modification was performed such that both fans automatically started in order to meet single failure criteria (only one filter train is necessary to limit doses within the guidelines of 10CFR100).

To ensure 99 percent efficiency for the charcoal filters, a duct heater is installed upstream to reduce the relative humidity of the air stream from 100 percent down to 70 percent. Calculation S10-202-HV01 shows that the 10-kW heater is adequate for flow resulting from operation with one 1600-cfm fan to reduce the relative humidity to 70 percent. Since both fans are now auto initiated, the duct heater can only limit the relative humidity to 80 percent or less (Calculation S10-202-HV02) until the operator shuts one fan down.

The UFSAR has been revised to indicate that relative humidity may be 80 percent for a period of up to 30 minutes following system initiation.

Safety Evaluation Summary:

ANSI N510 requires that testing be performed in accordance with ASTM D3803. This test loads the charcoal with humidity and radioactive iodine simultaneously. The test conditions are 80°C and 95-percent relative humidity for a period of 60 minutes. Therefore, this would bound the conditions of 80-percent relative humidity for 30 minutes.

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

Safety Evaluation No.:	90-024
Implementation Document No.:	Torsional Screening Test N1-PAT-12-1
UFSAR Affected Pages:	N/A
System:	Main Turbine - Generator
Title of Change:	Main Turbine - Generation Rotor System Torsional Screening Test (N1-PAT-12-1)

Description of Change:

As a result of analyses developed during investigation of the Maanshan Nuclear Power Station incident, where severe torsional resonance problems resulted in extensive turbine generator damage, General Electric Company (GE) calculated the torsional natural frequencies for similar LP turbine installations in the U.S. In some cases, GE calculated torsional natural frequencies which are too close to 120 Hz for unrestricted operation. GE's Technical Information Letter TIL-1012-2 recommended tests to confirm the calculation and to obtain sufficient data to modify the turbine generator if the torsional natural frequencies are measured very close to 120 Hz.

This safety evaluation evaluated performance of the "Turbine-Generator Rotor System Torsional Screening Test."

Safety Evaluation Summary:

The performance of the "Turbine-Generator Rotor System Torsional Screening Test" will not affect the safe operation or shutdown of the plant and does not increase the probability of an accident or malfunction of equipment important to safety. Turbine missile analysis indicated that the integrity of the reactor coolant boundary is preserved and the shutdown capability of the plant would be maintained. Synchronizing the unit by 10 degrees out of phase is within the design capability of the generator.

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

Safety Evaluation No.:	93-004
Implementation Document No.:	Calculation S22.4-201-V001
UFSAR Affected Pages:	N/A
System:	Condensate Makeup
Title of Change:	Utilization of Valve 58.1-04 as a Primary Containment Isolation Valve to Allow Performance of the Local Leak Rate Testing in the Power Operating Condition

Description of Change:

This safety evaluation evaluated the qualification of valve 58.1-04 as a primary containment isolation valve. In the past, the local leak rate tests (LLRT) have always been performed with the plant in cold shutdown. It became desirable to be able to perform the LLRT of blank flange 58.1-07 while the plant is in the power operating condition. Since the valve is tested in the reverse direction, it was necessary to demonstrate that a leak rate test performed in the reverse direction on valve 58.1-04 will be representative of the leakage the valve would experience during an accident.

Safety Evaluation Summary:

Changing the timing or configuration of the LLRT on this penetration will not impact the probability of occurrence of an accident or malfunction of equipment important to safety since none of the analyzed transients involve either the torus makeup line, the torus makeup valve 58.1-01, or blank flange 58.1-07. By maintaining valve 58.1-04 in the closed position, primary containment integrity is maintained. During performance of the LLRT, the total leakage of the penetration will be attributed to valve 58.1-04.

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

Safety Evaluation No.: 93-008

Implementation Document No.: Calculation S22.4-201-M004

UFSAR Affected Pages: N/A

System: Primary Containment

Title of Change: Local Leak Rate Testing of Double-Gasketed Primary Containment Penetrations and Air Locks

Description of Change:

Some double-gasketed air locks and penetrations were local leak rate tested during power operating condition when the containment pressure was at 1-1.5 psig and, consequently, the inner seals of these penetrations and air locks may not have been subjected to the required test pressure.

This safety evaluation evaluated continued operation with the inner seals of these penetrations tested at the differential pressure of 1-1.5 psig.

Safety Evaluation Summary:

This safety evaluation concluded that all the affected penetrations are operable, and performing a local leak rate test on these penetrations and air locks will not have any impact on the probability of occurrence of an accident or malfunction of equipment important to safety. In addition, these tests will not violate the Plant Technical Specifications.

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

Safety Evaluation No.: 94-009 Rev. 0 & 1

Implementation Document No.: Temporary Mod. 94-022

UFSAR Affected Pages: N/A

System: Radioactive Liquid Waste

Title of Change: Installation of Thermex with an Ozone Generator

Description of Change:

High conductivity liquid radioactive waste was temporarily processed and reclaimed for plant use through a series of equipment modules called Thermex. These modules include: 1) phase separator; 2) charcoal filtration; 3) process feed tank; 4) dissolved solids separator (DSS) which uses reverse osmosis; 5) dissolved solids polisher (DSP) which electrostatically captures ionic constituents; 6) ultra-violet light to degrade petroleum-based (organic) contaminants; and 7) mixed bed demineralizer. This temporary change includes the addition of a bag filter downstream of the charcoal beds to remove small, often highly-radioactive particles prior to the DSS; and the addition of an ozone generator downstream of the DSP to oxidize, and thereby ionize, organic compounds and enable removal of the organic derivatives in the mixed bed demineralizer.

Safety Evaluation Summary:

The Thermex system installation and operation has been evaluated and will be controlled by approved procedures. The addition of ozone generator to this system provides the ability to reduce organic contaminants in wastewater to within acceptable guidelines. The addition of an ozone generator presents no adverse consequences.

Processing of liquid radwaste has no impact on any component, system or structure which is important to nuclear safety. The addition of a 5g/hr ozone generator using atmospheric oxygen or bottled oxygen represents no flammable, explosive, or corrosive hazard to any equipment. The Thermex system is not located near any equipment important to safety, and the system's power supply does not provide power to any components or systems important to safety.

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

Safety Evaluation No.: 94-078 Rev. 0 & 1

Implementation Document No.: Procedure GAP-POL-01

UFSAR Affected Pages: XIII-6, XIII-7

System: N/A

Title of Change: Changes to the Maintenance and Radiation Protection Manager Responsibilities per GAP-POL-01

Description of Change:

Procedure GAP-POL-01 defines the composition and responsibilities of the Nuclear Generation organization, and reassigns the responsibilities for plant housekeeping and decontamination at Unit 1 from the Manager of Maintenance to the Radiation Protection Manager. Housekeeping and decontamination work more efficiently with the Utility Mechanics reporting directly to the branch that prioritizes decontamination efforts and evaluates the results of those efforts.

Safety Evaluation Summary:

The reorganization to change the reporting structure of the Utility Mechanics from the Maintenance branch to the Radiation Protection branch, as described in GAP-POL-01, meets the organizational criteria specified in the Technical Specifications as well as the Standard Review Plan (NUREG-0800). Clear lines of authority from the Plant Manager to the Manager Radiation Protection are maintained. Housekeeping and decontamination do not involve any responsibilities for activities important to the safe operation of the facility. The Manager Radiation Protection's independence from operating pressures is not affected by this change.

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

Safety Evaluation No.:	95-109 Rev. 0 & 1
Implementation Document No.:	DDC 1F00187
UFSAR Affected Pages:	XII-4, XII-5, 10A-119, 10A-121; Figures III-4, XII-1
System:	Radioactive Waste Disposal
Title of Change:	Thermex Leased Modular Equipment to Process Liquid Waste

Description of Change:

This safety evaluation evaluated conversion of Temporary Modification 94-022 to permanent plant equipment. Revision 1 of this safety evaluation changed installation strategy from "Phase 1" and "Phase 2" to "Stage 1" and "Stage 2" and specified Operations acceptance would occur after the chemical hoses were properly supported.

This safety evaluation adopted a concept of flexible and multiple strategies to normally process the plant's low purity, high conductivity (floor drain collector) sump water. The multiple strategy was derived from a series of equipment pieces, each designed to efficiently extract a type of contaminant rather than using one piece of equipment to perform all extractions.

Thermex is a modular system leased from a nuclear waste vendor. Its components are arranged on Elevation 261 of the Waste Building Extension. Thermex uses charcoal for gross filtration, bad filtration, reverse osmosis and electrolytic separation to accomplish "ultra filtration," and chemical oxidation and/or photo-oxidation to attack organics. None of these strategies uses a large amount of consumable media. This water is finally treated by ionic exchange through demineralizers, which enjoy a much lengthened useful life because the "heavy lifting" has been accomplished by other means. Depending on the quality of radioactive waste water, several of the extraction strategies may be unnecessary and may be bypassed.

Safety Evaluation Summary:

Thermex has doubled the processing capacity of the concentrator, and together with the concentrator gives Unit 1 three times the floor drain water processing capability than previously available. This factor greatly reduces the probability of a controlled liquid waste discharge in accordance with Technical Specification

Safety Evaluation No.: 95-109 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

3.6.15. Since the components of Thermex are wholly housed in the Waste Building and the Waste Building Extension, the potential for an uncontrolled discharge resulting from equipment failure is not credible.

Other benefits include a significant improvement in processed water quality, a ten-fold reduction in solid waste effluents, no liquid waste effluent, and a huge energy reduction (from 2000 kW to less than 60 kW).

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

Safety Evaluation No.:	95-110
Implementation Document No.:	DDC 1F00025
UFSAR Affected Pages:	III-15, III-18; Figure XII-1
System:	Radioactive Waste Disposal
Title of Change:	Retire Power to Obsolete Liquid and Solid Waste Systems' Components

Description of Change:

Power supplies for obsolete equipment, the No. 11 waste concentrator and its associated electric boiler, the DOW system polymer pumps and tank agitators, and the drum storage equipment in the "old" Radwaste Building, have been disconnected at the electrical circuit breakers and reserved for future use.

Safety Evaluation Summary:

Processing of gaseous, liquid, and solid radioactive wastes at Unit 1 is accomplished in conformance with the Process Control Program, Technical Specifications 3/4.6.15 and 16, and 10CFR50.36(a). Radioactive waste processing in accordance with the licensing basis has continued for years without using the obsolete equipment identified above, and will continue unimpeded with this equipment de-energized.

De-energizing the subject obsolete equipment by disconnecting the power supplied from the circuit breakers can be accomplished while remaining within the licensing basis for radioactive waste effluents, unplanned radioactive releases, and chemical control for water quality.

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

Safety Evaluation No.:	96-006
Implementation Document No.:	Procedure NSAS-POL-01
UFSAR Affected Pages:	XIII-3; F XIII-4
System:	N/A
Title of Change:	Reorganization; Changes to NSAS-POL-01 to Reflect Consolidation of the Maintenance and Technical Training Groups

Description of Change:

Procedure NSAS-POL-01 has been revised to reorganize the functions of the Technical Training group. The Technical Training group is comprised of General Employee Training (GET), Emergency Plan (EP), Radiation Protection (RP), and Chemistry training. These programs are currently under the direction of the General Supervisor of Technical Training.

This organization change will better utilize resources by eliminating the General Supervisor of Technical Training position and distributing the responsibilities between the General Supervisor Training Services/Engineering Training and the General Supervisor Maintenance Training.

The GET and EP training programs have been placed under the direction of the General Supervisor Training Services/Engineering Training. The RP and Chemistry training has been combined with the Maintenance Training program. The Maintenance Training group will be renamed "Technical Training" and the General Supervisor Maintenance Training title changes to "General Supervisor Technical Training."

Safety Evaluation Summary:

The proposed change to procedure NSAS-POL-01 establishes clear departmental responsibilities and lines of authority and communication for the Nuclear Training organization. The proposed organization structure satisfies the criteria of SRP 13.1.1 and conforms with the requirements of Section 6.2.1 of the Unit 1 and Unit 2 Technical Specifications.

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

Safety Evaluation No.:	96-102
Implementation Document No.:	Temporary Mod. 96-019
UFSAR Affected Pages:	N/A
System:	Roof & Floor Drains
Title of Change:	Turbine Bldg. 305' "Green Area" for Breaks and Lunches

Description of Change:

This change provided a temporary Turbine Building Green Area (TBGA) on Elevation 305', NW corner, within columns D-G, 3-4, where personnel working on turbine generator maintenance on Elevation 300' could eat, drink, and use sanitary facilities (portable toilets and hand wash stations) during breaks and lunches. The TBGA was equipped with air conditioning, tables and chairs, refrigerator, microwave oven, telephones, Gai-tronics, and portable sanitary facilities. This area was not used as an egress point from the radiologically-controlled area (RCA), nor was it used to release tools and equipment from the RCA.

Safety Evaluation Summary:

Based on the existing fire protection detection, Fire Brigade response, large area, and relatively small amount of additional combustibles, there is reasonable assurance that the TBGA does not have an adverse effect on the fire protection program or its defense-in-depth capability. The ability of Unit 1 to achieve and maintain safe shutdown in the event of a fire in this area will not be adversely affected. Use of the TBGA will be in compliance with the radiation shielding and access control design basis as detailed in Design Criteria Document 404, "Radioactive Source Terms and Shielding Design," 10CFR19 and 20, and the ALARA philosophy. There are no structural, seismic, flooding or electrical power board loading or breaker coordination concerns with installation of the Kelly Building and electrical support systems. The TBGA will not significantly affect other plant systems in the area. The TBGA does not affect nuclear safety.

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

Safety Evaluation No.:	97-011
Implementation Document No.:	Procedure N1-OP-16
UFSAR Affected Pages:	XI-10
System:	Condensate Pumps Discharge
Title of Change:	Condensate Demineralizer Operating/Design Flow

Description of Change:

This change revised the UFSAR, consistent with Operating Procedure N1-OP-16, to indicate the demineralizer tanks are sized for a nominal flow rate of 50 gpm per square foot of bed surface area when six demineralizers are in service at full power (1850 MWt). When it is necessary (due to ultrasonic resin cleaning or regeneration) to take one unit out of service, the flow is approximately 58 gpm per square foot. The maximum nominal design flow is 64 gpm per square foot of bed surface area.

Safety Evaluation Summary:

As described in the Illinois Water Treatment letter (1969), the operating experience shows the condensate demineralizers will maintain the reactor coolant pressure boundary integrity at 1850 MWt with five condensate demineralizers in service with no adverse consequences, and still provide acceptable effluent quality. The plant normally operates with a flow of 49 gpm with six demineralizers in service at full power, and approximately 58 gpm per square foot of demineralizer surface area when at full power (1850 MWt) with five demineralizers in service. The demineralizers, including the laterals and effluent strainers, are designed for a 64 gpm per square foot capability, and the resin manufacturer recommends a flow of 45-60 gpm per square foot for condensate polishing. The proposed change clarifies and makes the UFSAR consistent with the design and operation. Therefore, the proposed change does not increase the probability of occurrence of an accident previously analyzed in the UFSAR.

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

Safety Evaluation No.: 97-021

Implementation Document No.: Procedure N1-OP-11

UFSAR Affected Pages: X-20, X-23

System: Reactor Building Closed Loop Cooling (RBCLC), Turbine Building Closed Loop Cooling (TBCLC)

Title of Change: CLC System Leakage Detection

Description of Change:

A discrepancy was identified in the UFSAR regarding the TBCLC and RBCLC systems. The UFSAR stated that "...leakage out of the system is noted by an integrating flow meter in the system's makeup line." The actual design has no integrating flow instrumentation in either the makeup lines to the RBCLC or TBCLC systems. The actual design does have flow switches and corresponding flow alarms in the Control Room for both RBCLC and TBCLC systems. These flow switches are located in the 1 1/2" makeup supply lines to the respective CLC systems. This safety evaluation considered the plant as-built condition.

Safety Evaluation Summary:

There are no accidents evaluated in the UFSAR affected by the described changes. The existing CLC systems instrumentation, with the high flow alarm instead of the integrated flow meter, meets GDC requirements and provides adequate means for evaluating makeup rate/losses from the RBCLC system. Therefore, the proposed changes do not increase the probability of occurrence of an accident previously evaluated in the UFSAR.

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

Safety Evaluation No.:	97-024
Implementation Document No.:	DER 1-96-2896
UFSAR Affected Pages:	10A-85
System:	Fire Protection Water - Hose Stations
Title of Change:	Documentation of Deviation from NFPA 14

Description of Change:

This change documents a deviation from the NFPA standard which requires an approved device to be installed at outlets of standpipe systems to reduce the pressure at the outlet to 100 psig. The code requirement as stated in NFPA 14 is based on potential use of the hose station by building occupants without specific training in handling high-pressure hose. Due to the intended use of this system at Unit 1, deviating from this requirement has no adverse effect on system operability or use.

This change allows a permanent deviation from NFPA 14 requirements for the lack of a pressure-reducing device at hose station outlets to limit pressure to 100 psig.

Safety Evaluation Summary:

This evaluation examines the requirement for pressure-limiting devices on fire hose systems installed in the plant. Based on the use of fire hose exclusively by Fire Brigade personnel who receive training in the use and handling of high-pressure hose through hands-on exercises at the Niagara Mohawk Fire Training School, the need for pressure-limiting devices does not exist. The lack of pressure-limiting devices on fire hose stations has no effect on the ability to achieve and maintain safe shutdown.

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

Safety Evaluation No.: 97-105

Implementation Document No.: GE Field Disposition Report DDR-0229

UFSAR Affected Pages: VII-18

System: Liquid Poison

Title of Change: Liquid Poison Sparger

Description of Change:

This change revised the UFSAR to indicate the liquid poison sparger is a 360° sparger with 10 equally spaced ¼" holes around the bottom of the sparger; and to indicate adequate mixing is provided by reactor recirculation. This safety evaluation considers the "as-built" configuration of the liquid poison sparger.

Safety Evaluation Summary:

In the event of a failure to scram, the proposed change has no effect on the liquid poison system's ability to bring the reactor to a cold, xenon-free shutdown condition. The system has been evaluated to perform the safety function even with a gross failure of the sparger. The proposed change makes the UFSAR consistent with the "as-built" design. Therefore, the proposed change does not increase the probability of occurrence of an accident previously analyzed in the UFSAR.

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

Safety Evaluation No.: 97-106

Implementation Document No.: DDC 2S11022

UFSAR Affected Pages: Figure III-1

System: N/A

Title of Change: Removal of Substation "Q"

Description of Change:

This change physically removed substation "Q" and its associated transformer, switches, and fencing. Substation "Q" was originally installed to facilitate the construction of NMP2 and is no longer required to be in service.

Safety Evaluation Summary:

Substation "Q" is unrelated to plant systems or structures and its removal will have no impact on safe operation of the plant. The work activities associated with the removal of the substation will be performed in accordance with applicable procedures.

Removal of the substation has no impact on the probable maximum precipitation flood analysis. The location of removal activities is adequately separated from systems and structures important to safety which will preclude any adverse impact.

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

Safety Evaluation No.: 97-109

Implementation Document No.: DDC 1F00137

UFSAR Affected Pages: Figure VI-23

System: Reactor Building & Waste Building Closed Loop Cooling

Title of Change: Retirement of Solenoid Blocking Valves Used with Drywell Air Coolers

Description of Change:

Solenoid blocking valves BV-70-118, BV-70-120, BV-70-122, BV-70-124, BV-70-126, and BV-70-128 were "electrically retired in place" because water leakage was causing the solenoids to short out. The solenoid actuators for these valves were installed incorrectly and pulling the fuses for the valves alleviated the problem.

Safety Evaluation Summary:

The drywell air coolers' solenoid-operated blocking valves being electrically retired in place are designed to "spring to open/fail open" position due to a loss of electrical power. In the event of an accident, these valves would be in an open position and would still supply a flow path to the unit coolers. The unit coolers are not required for an accident condition. The safety-related function of the subject valves is to maintain the pressure boundary of the RBCLC system, and valve position is not critical. Technical Specification Sections 3.2.5.a and 3.2.5.b refer to unidentified leakage. Both of these sections have been previously evaluated to avoid a loss-of-coolant accident. Because reactor coolant will not be affected by this proposal, the margin of safety as defined in the basis for this Technical Specification will not be reduced by this proposal.

Since the solenoid-operated blocking valves are designed to "spring to open/fail open" position, their condition of being "electrically retired in place" will not have any adverse effect on the maintenance of pressure boundary integrity. Also, the mechanics of the solenoid-operated blocking valves are set up in such a way that with these valves electrically retired in place, the RBCLC system will still be capable of providing cooling flow during any plant condition.

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

Safety Evaluation No.:	97-110
Implementation Document No.:	Calculation S20.1-38-V001
UFSAR Affected Pages:	10B-20, 10B-31
System:	Shutdown Cooling
Title of Change:	Spurious Operation of Shutdown Cooling Valves IV-38-01 and IV-38-13

Description of Change:

NRC Information Notice (IN) 92-18 addressed concerns related to a specific type of Appendix R fire-induced hot short in the control circuits of motor-operated valves. A review of the implications of the application of the hot short to existing safe shutdown valves credited in the Appendix R analysis for control complex fire scenarios identified four valves as potentially vulnerable to the IN described short. These valves are the shutdown cooling isolation valves IV-38-01, IV-38-02, and IV-38-13, and the loop 12 pump suction blocking valve BV-38-04. Whereas the IN assumes that overload heater availability precludes valve damage, an Engineering assessment anticipated potential inability to open valves IV-38-01 and IV-38-13 if stalled closed due to valve damage occurring prior to the thermal overload relay trip. Valves IV-38-02 and BV-38-04 were found to be capable of withstanding the effects of the stall thrust. The original Safe Shutdown Analysis (SSA) had anticipated hot short vulnerability to the control circuits of the valves; however, it did not anticipate precluding spurious operation to avoid valve mechanical damage. Maintaining de-energization of the valves by opening the motor control center breakers, while already being provided the administrative controls of Technical Specification Amendment 154, was not invoked by the SSA, as would be required by Appendix R for anticipated fire damage effects which would challenge credited safe shutdown equipment.

The UFSAR (SSA) has been revised to include the requirement for maintaining de-energization of valves IV-38-01 and IV-38-13 by locking their breakers in the open position to preclude spurious operation and potential valve damage.

Safety Evaluation Summary:

The applicable accident initiators are a loss-of-coolant accident resulting in a low-low reactor water level and containment isolation, and a shutdown cooling system

Safety Evaluation No.: 97-110 (cont'd.)

Safety Evaluation Summary: (cont'd.)

line break also resulting in an isolation signal to the shutdown cooling isolation valves IV-38-01 and IV-38-13. The isolation valves are currently maintained closed during normal power operations to assure integrity of the containment upon receipt of an isolation signal. Therefore, removal of the power to the valves will not degrade containment isolation capability, and the probability and consequences of an accident are not increased. The configuration status of maintaining the breakers open during normal power operations will not impact operational or design requirements of the valves. The reliability of the shutdown cooling system and the breakers associated with the isolation valves are unaffected by the proposed breaker positioning. The proposed change is within the current licensing basis as described in the Bases for Technical Specifications 3/4.2.7 and 3/4.6.2b. With the valve alignment normally closed and electrically de-energized, the margin of safety is unaffected by the proposed change.

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

Safety Evaluation No.: 97-113

Implementation Document No.: Test Procedure N1-ST-Q12

UFSAR Affected Pages: X-32

System: Spent Fuel Pool Makeup

Title of Change: Spent Fuel Pool Makeup Rate

Description of Change:

This safety evaluation analyzed a change to the UFSAR to correct wording describing maximum available makeup rate to the spent fuel pool (SFP). The UFSAR previously stated that the maximum possible rate of makeup by the condensate transfer system to the SFP was 75 gpm. Actual test results show that the condensate transfer system can provide a significantly higher makeup rate of about 105 gpm.

Safety Evaluation Summary:

The change to the UFSAR is conservative. There is no requirement limiting the maximum available makeup rate to the SFP from the condensate transfer system. Since the minimum makeup flow rate remains greater than the calculated maximum required flow rate, the requirement for keeping the stored spent fuel covered with water is not adversely impacted.

The condensate transfer system provides makeup water to several users. The makeup requirement to most of these users is intermittent and nonconcurrent. The actual total makeup to the SFP is controlled by level control switches. This change does not increase the total makeup requirement to the SFP and, hence, the total integrated makeup requirement from the condensate transfer system is not impacted.

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

Safety Evaluation No.: 97-114

Implementation Document No.: P&ID C-18009-C, Sht. 1, Procedure N1-OP-3

UFSAR Affected Pages: V-7, X-3; Figure X-2

System: Reactor Water Cleanup (RWCU)

Title of Change: RWCU System UFSAR Changes

Description of Change:

The RWCU system schematic in UFSAR Figure X-2 has been revised to be consistent with the RWCU design piping and instrumentation drawing and the operating procedure. The revisions included a change in the position of valves BV-37-07R and BV-33-30 from normally closed to normally open; the removal of a relief valve from the UFSAR figure (the relief valve was never installed); and the correction of the drawing to show globe valves in place of gate valves.

Safety Evaluation Summary:

This safety evaluation evaluated changes to the RWCU system piping and instrumentation as described in UFSAR Figure X-2. The changes were made to make the UFSAR description consistent with the as-built configuration of the plant, design drawings and plant operating procedures. This evaluation demonstrated that the proposed changes were in conformance with all of the applicable criteria for the cleanup system. The evaluation concluded that the reactor drain valve and blowdown line orifice bypass valve can be maintained in the open position with no adverse effect on nuclear safety. The evaluation concluded that the relief valve on the common line between the demineralizers and cleanup pumps was never installed because adequate overpressure protection exists without it. The evaluation also concluded that the use of globe valves was more desirable for the purposes of regulating flow than the gate valves shown in the UFSAR figure.

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

Safety Evaluation No.:	97-115
Implementation Document No.:	Procedures N1-OP-17, N1-OP-22, N1-OP-15A
UFSAR Affected Pages:	X-27, X-28, X-29, XI-10, XI-11, XI-14, XI-15, XII-6, XII-8; Table XII-2 Sh 1 & 2; Figures XI-6, XII-1
System:	Resin Transfer and Cleaning, Makeup Demineralizer, Acid/Caustic
Title of Change:	Delete Resin Regeneration for Condensate Demineralizers and the Makeup Demineralizer Systems

Description of Change:

This safety evaluation evaluated changes to the condensate demineralizer, makeup demineralizer, and acid and caustic systems to: 1) correct UFSAR discrepancies describing differential pressure alarms associated with each of the condensate demineralizers; 2) evaluate the makeup demineralizer system as a batch makeup system which uses service water as its primary source, and to correct discrepancies in the described makeup demineralizer effluent water quality; and 3) address changes to the acid and caustic (treatment) system and associated resin regeneration activities, no longer possible because the acid and caustic pumps and tanks have been removed.

Safety Evaluation Summary:

The systems affected by these changes do not initiate any of the accidents previously evaluated in the UFSAR, nor do these changes affect any accident precursors. By maintaining reactor coolant quality within Technical Specification (TS) limitations, the probability of a large break loss-of-coolant accident (LOCA) is not increased. Also, maintaining coolant makeup quality in accordance with the TS minimizes crack propagation and does not increase the probability of a reactor internal component failure or misalignment. Finally, maintaining sufficient coolant makeup capability in accordance with the TS to address the small break LOCA does not increase the probability of such a break.

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

Safety Evaluation No.: 97-117

Implementation Document No.: Simple Design Change SC1-0067-94

UFSAR Affected Pages: Figure VII-1

System: Core Spray

Title of Change: Core Spray IV Pressure Binding Relief

Description of Change:

This simple design change added a pressure binding relief path to core spray isolation valves IV 40-02 and IV 40-12 to ensure the valves will open under all postulated conditions. An instrument line for these valves was connected to the existing ¼" tap for the lantern gland stuffing box and connected to the reactor side drain valves. The drain valves were locked open to provide the relief path to the reactor side process piping. The valve packing was modified to allow pressure in the bonnet to relieve through the ¼" tap.

Safety Evaluation Summary:

The core spray system is required to operate to prevent overheating of the fuel following a postulated loss-of-coolant accident. The outside IVs are required to open, if closed, when reactor pressure is 365 psig or less. The addition of a pressure relief path to these valves will prevent the potential for pressure locking in the redundant core spray loops.

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

Safety Evaluation No.: 97-118

Implementation Document No.: Evaluation FPEE 0-98-001

UFSAR Affected Pages: 10A-28

System: N/A

Title of Change: Sealed-Beam Portable Hand Lights

Description of Change:

This safety evaluation evaluated the elimination of the reference to "sealed-beam" battery-powered hand lights in the UFSAR.

Safety Evaluation Summary:

Appendix A to BTP 9.5-1 Section D-5(b), "General Guidelines for Plant Protection," and NUREG-0800, Section C.5.g(2), "Lighting and Communication," recommend that "suitable sealed beam battery powered hand lights be provided for emergency use...". The portable hand lights used by the Fire Brigade, Control Room and damage repair personnel are not the sealed-beam type. However, 10CFR50 Appendix R, Section III.H, does not specify what type or design the portable lights must be, only that they be available.

The nonsealed-beam portable lights used by the Fire Brigade, Operators and damage repair personnel are suitable for their application; they are inherently reliable and are routinely inspected and tested. Since 10CFR50 Appendix R does not specifically require the use of sealed-beam portable hand lights and the portable hand lights provided are effective and reliable for their intended use, the elimination of the specific requirement that they be sealed beam has no adverse effect on any structure, system or component important to safety. This change also does not adversely affect the plant's ability to achieve and maintain safe shutdown.

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

Safety Evaluation No.: 97-119

Implementation Document No.: DER 1-96-3194

UFSAR Affected Pages: I-7

System: Containment

Title of Change: UFSAR Section I.0 Inaccurate

Description of Change:

This safety evaluation analyzed the impact of revising the UFSAR to clarify how containment integrity is maintained at the drywell airlocks and the reactor building access doors. The statement was misleading because it inferred that the security keycard and alarm system ensures containment integrity when, in fact, the security keycard and alarm system is for the purpose of access control and intrusion detection.

Safety Evaluation Summary:

Containment integrity is maintained through procedural compliance, compliance with design basis, and adherence to Technical Specifications. There has never been a keycard or remote alarm located at the drywell airlock doors. The only remote alarm for the drywell airlock is an alarm installed at a radiation protection gate leading to it. This alarm is for the purpose of alarming the high radiation area outside the primary containment near the drywell airlock doors. If a local alarm is given that its alternate is open, procedural controls and training forbid personnel from opening an airlock door during periods where primary containment is required to be intact. Mechanical interlocks prevent both doors from being opened at the same time.

Access control and containment integrity at the Turbine Building to Reactor Building access points are maintained through use of procedural controls and alarms. Security keycard readers and remote alarms are installed in the outside door for the purpose of access control and intrusion detection at the Turbine Building entrances to the Reactor Building, per the requirements of 10CFR73.55, and serve no purpose when it comes to maintaining containment integrity.

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

Safety Evaluation No.: 97-120

Implementation Document No.: Procedure NIP-DES-04

UFSAR Affected Pages: I-6, VI-20, VI-21; Table VI-3b Sh 3

System: Reactor Core Spray, Containment Spray, Core Spray

Title of Change: Revise UFSAR Table VI-3b and NIP-DES-04, Attachment 7, Primary Containment Isolation Valves

Description of Change:

The core spray high point vent valves (40-30, 40-31, 40-32, and 40-33) and the core spray condensate supply (keep fill) valves (40-20, 40-21, 40-22, and 40-23) were inadvertently included in UFSAR Table VI-3b. The subject lines do not enter the free space of the containment; therefore, these valves do not belong in these tables. The lines do connect to the core spray system which communicates directly with the reactor coolant system. The subject valves are reactor coolant isolation valves and are appropriately included in UFSAR Table VI-3a.

Safety Evaluation Summary:

The subject valves were inadvertently included on the primary containment isolation valve (lines entering the free space of the containment) list. The proposed change revises the UFSAR to accurately depict the actual plant configuration. No requirements of the valve design, maintenance, surveillance, or operation will be changed. Nothing which could initiate an accident is affected by this change. Further, the valves will continue to be subject to Technical Specification 3.2.7, which is the more limiting applicable Technical Specification. The proposed change to the valve tables will not increase the probability of occurrence of an accident previously evaluated in the UFSAR. The additional changes which revise incorrect statements in the UFSAR and Technical Specification Bases do not affect the probability of occurrence of an accident. Redundancy of primary containment isolation valves has no bearing on the probability of occurrence of an accident.

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

Safety Evaluation No.:	97-122
Implementation Document No.:	Procedures GAP-POL-01, NEP-POL-01, NIP-TQS-01
UFSAR Affected Pages:	XIII-5, XIII-6; Figures XIII-2, XIII-3
System:	N/A
Title of Change:	Organization Change - Combine Unit 1 and Unit 2 Plant Process Computer Support Personnel into a Single Organization

Description of Change:

This safety evaluation evaluated the impact to the nuclear organization resulting from combining design, maintenance, and technical support into a single onsite department that reports to the Engineering Organization.

Safety Evaluation Summary:

The current nuclear organization includes personnel responsible for design, maintenance, and technical support associated with plant process computers. Each service area is currently the responsibility of individual departments including Design Engineering, Unit 1/Unit 2 Technical Support and Unit 1 Instrument and Control. The proposed organization change combines respective personnel into a single organization. The criteria applied to evaluating the change is primarily based on the Unit 1 UFSAR and Unit 2 USAR descriptions of the NMPC Nuclear Quality Assurance Program (Appendix B) and Conduct of Operations (Chapter 13) and Technical Specifications Section 6. The proposed organization change is analyzed against Technical Specification requirements for organization lines of authority, responsibility, and staff qualifications. Evaluation of the proposed change against the applicable criteria indicates continued conformance with all criteria.

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

Safety Evaluation No.:	97-123
Implementation Document No.:	DER 1-96-2956
UFSAR Affected Pages:	XI-15
System:	Main Turbine and Auxiliary Annunciator
Title of Change:	Revision to UFSAR to Reflect Actual Alarm Function (Low Pressure Alarm in Place of Listed High Pressure Alarm) for the Turbine Exhaust Hood Spray

Description of Change:

This safety evaluation analyzed a change to the UFSAR to reflect the turbine exhaust hood spray alarm on low pressure, which is consistent with the as-installed plant condition. The alarm is actually an alarm for turbine exhaust hood spray low pressure.

Safety Evaluation Summary:

The design intent of the turbine exhaust hood spray alarm is to provide indication to the Operators of a problem in the exhaust hood water spray system. The intent of the statements in the UFSAR was to indicate that there is an alarm available for indication of abnormal conditions. The low-pressure alarm provides this function.

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

Safety Evaluation No.:	97-126
Implementation Document No.:	Calculation S10-210HV13
UFSAR Affected Pages:	III-12
System:	Control Room Vent Chilled Water (CRAC)
Title of Change:	Failed TCV for Control Room Chilled Water System

Description of Change:

This safety evaluation analyzed the Control Room ventilation air cooling function with temperature control valve TCV-210.1-56 inoperable for an extended period. Due to a degraded motor/controller 210-90, this three-way control valve is in the failed open position, allowing full flow of chilled water through cooling coils #11 and #12. A bypass valve (210.1-16) installed around the TCV is also kept in the full open position as a safety measure to ensure adequate chilled water is supplied through the cooling coils in the event the TCV failed closed. The UFSAR Control Room ventilation system drawing, Figure III-14, shows a motor-operated TCV. The motor-operated portion of the TCV was not operable as shown. Also, the bypass valve remained open, which was contrary to its normal position (closed) as shown in drawing C-18047-C.

Safety Evaluation Summary:

With both valves open, there is no impact to the system operation and the system remains within the existing design basis evaluation. The increase in cooling water flow with both valves open is within the permissible range set by design calculations. No safety concerns exist with the above field configurations.

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

Safety Evaluation No.:	97-128 Rev. 0 & 1
Implementation Document No.:	Test Procedure N1-IST-HYD-003
UFSAR Affected Pages:	N/A
System:	Emergency Cooling
Title of Change:	Non-ASME Hydrostatic Leakage Test of Emergency Cooling System Loop #12 Steam Piping

Description of Change:

This safety evaluation evaluated a hydrostatic test configuration of the emergency cooling system loop #12 which helped to ascertain whether a tube leak existed in the 122 emergency condenser (EC). There was evidence that such a leak may have existed, but the same evidence may have been indicative of other plant/system circumstances. This test/troubleshooting configuration provided data to determine whether a tube leak existed.

A hydrostatic pump was connected to the outboard drain line of the air-operated EC return line IV-39-06. This was equipped with a throttle valve and a high-capacity relief valve set at about 600 psig. The EC system steam side was vented and filled and pressurized up to about 500 psig. Leakage at the condenser 122 tubes was measured by observing condenser shell side level increase. Maintaining test pressure at less than 600 psig ensured that no test media flowed toward the reactor vessel thus creating power or hydraulic transients.

Safety Evaluation Summary:

The piping subject to the hydrostatic test is adequately isolated from the rest of the plant systems to ensure that the operating plant is not adversely affected by the test and that the test is not adversely affected by the operating plant.

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

Safety Evaluation No.: 97-129

Implementation Document No.: Calculation S14-54-HX08

UFSAR Affected Pages: X-41, X-44

System: Fuel Handling

Title of Change: Spent Fuel Storage Pool Filtration and Cooling System

Description of Change:

This safety evaluation analyzed changes to the UFSAR regarding the fuel handling system as follows:

1. The UFSAR stated that the spent fuel pool canal to the reactor was of the same depth as the canal from the reactor to the internals storage pit. The depth of the spent fuel pool canal is 22 ft. 9 in., while the internals storage pit canal depth is 19 ft. 6 in.

2. The UFSAR implied that the steel-lined concrete shield plugs atop the reactor head cavity and its two transfer canals vary in thickness between 4 ft. and 5 ft. 5 in. The plugs above the reactor head cavity are of a uniform 5 ft. 6 in. (including the steel lining), while the plugs which fill the transfer canals are 4 ft. 9.75 in. thick in the refuel canal and, although concave, at least 4 ft. thick in the equipment transfer canal.

3. The UFSAR implied that even in the event of a failure to maintain spent fuel pool water level within the control band, station personnel could readily access the refuel floor. During normal circumstances station personnel can readily access the refuel floor, but if the spent fuel pool were inadvertently drained, radiation dose rates would increase exponentially, and access to the refuel floor would be controlled by station radiological procedures.

Safety Evaluation Summary:

This change corrects the description of the facility as described in the UFSAR. This change has no effect on station equipment or operation, nor does it have any effect on the cask drop protection system or types of casks and how they are handled. This proposal more accurately describes the design of the reactor cavity and its associated canals, as well as the shielding provided for the reactor cavity during power operation. It also more accurately describes the expected

Safety Evaluation No.: 97-129 (cont'd.)

Safety Evaluation Summary: (cont'd.)

radiological conditions on the refuel floor if the ability to maintain water shielding over irradiated fuel is compromised.

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

Safety Evaluation No.:	97-131
Implementation Document No.:	Calculation S14-41-M002
UFSAR Affected Pages:	VII-17; Figure VII-6
System:	Liquid Poison
Title of Change:	Revision to UFSAR to Reflect Removal of Low Level (Temperature Actuated) Immersion Heater Cut-off Switch

Description of Change:

During a review of the UFSAR, a paragraph was identified that did not correctly describe the actual plant configuration. The UFSAR stated, "...a low level (temperature actuated) cut-off switch prevents heater damage when it is uncovered."

This low level cut-off switch was removed in 1988 per resolution to Problem Report #210. Therefore, the UFSAR has been revised to reflect the actual plant configuration by removing the description of this cut-off switch.

UFSAR Figure VII-6 was revised to show a level alarm off the level transmitter. This alarm was erroneously removed in a previous UFSAR revision. Figure VII-6 has also been revised to show that a temperature switch is used for temperature alarm functions instead of the temperature indicating switch which was shown.

Safety Evaluation Summary:

The UFSAR is being revised to be consistent with the current configuration of the liquid poison system and is not implementing any physical changes to the plant. The physical changes were implemented previously.

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

Safety Evaluation No.: 97-132

Implementation Document No.: DDC 2S11038

UFSAR Affected Pages: Figure III-1

System: N/A

Title of Change: Demolition of Temporary Structures East of the Unit 2 Structures - M&TE and Paint Storage Buildings

Description of Change:

This safety evaluation evaluated the demolition of the paint storage silos and M&TE buildings located east of the NMP2 plant structures.

These buildings were built for use as temporary buildings during the construction of NMP2. The paint storage buildings were being used for the storage of miscellaneous items by the Buildings and Grounds Department. The majority of the items stored in these buildings were moved to investment recovery and the rest were moved into space available in the remaining buildings. The M&TE building was empty.

Safety Evaluation Summary:

All the buildings to be demolished are located in an area that was not used as a flow channel for the probable maximum precipitation analysis. Removal of these buildings and the consequent reduction in the run-off coefficient would make the analysis more conservative. These buildings to be demolished have no impact on the previously calculated X/Q values. The design margins for the Control Room fresh air intakes are not compromised. Demolition of these temporary buildings will reduce the demand on plant systems such as fire protection, domestic water, sanitary sewer, storm sewer, and plant communications. Power to these buildings is from the temporary power loop. Eliminating these buildings will eliminate the need to maintain significant portions of the temporary power loop. Location of the demolition activities is adequately separated from safety-related systems and structures to preclude any adverse impact from construction activities.

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

Safety Evaluation No.: 97-133

Implementation Document No.: Mod. N1-97-032

UFSAR Affected Pages: III-24, 10A-42, 10A-50, 10A-59, 10A-70, 10A-86, 10A-101, 10A-110, 10A-112, 10B-77, 10B-78, 10B-136, 10B-141, 10B-146, 10B-151; Table XVI-27 Sh 2; Figures 10A-3B, 10A-4B, 10A-5B

System: Fire Suppression

Title of Change: Elimination of Fixed-Foam System

Description of Change:

This modification eliminated the fixed-foam injection from the deluge water spray and hose station systems assigned to the fire areas for 1) the turbine generator under Elevation 300'-0"; 2) the turbine generator oil reservoir tanks; and 3) the hydrogen seal oil unit.

Safety Evaluation Summary:

The only accident that the proposed change could affect is the turbine generator fire. The Safe Shutdown Analysis and the Fire Hazards Analysis evaluate the effects of the turbine generator lube oil fire after occurrence. Once the fire has occurred, the analyses postulate a loss of all equipment within the fire zones affected by the proposed change. No equipment required to achieve or maintain safe shutdown is located within the fire zones affected by the change. The fixed-foam systems are not initiators, precursors, or preventors of a fire. The fixed-foam systems are suppression systems which act to mitigate the effects of a fire. The proposed change cannot exacerbate the analyzed fire or the effects of the fire, since the accident occurs and all equipment in the fire zone is lost regardless of how or if the fire is controlled.

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

Safety Evaluation No.:	97-134 Rev. 1
Implementation Document No.:	Procedures NDD-OPS, GAP-RMP-01, GAP-POL-01, GAP-OPS-01
UFSAR Affected Pages:	XIII-6, XIII-7, XIII-10; Figure XIII-2
System:	N/A
Title of Change:	Reorganization to Change the Reporting Structure of the Unit 1 Radwaste Section to the Radiation Protection Branch

Description of Change:

The administrative reporting structure of the Unit 1 Radwaste section has been changed from the Unit 1 Operations branch to the Unit 1 Radiation Protection branch. This was done to provide single-point accountability for radioactive waste shipping and to consolidate similar technical activities under one Branch Manager.

Safety Evaluation Summary:

The proposed change in the reporting structure of the Unit 1 Radwaste Operations section from the Unit 1 Operations branch to the Unit 1 Radiation Protection branch meets the organizational criteria specified in the Technical Specifications, as well as ANSI N18.7-1972 and the Standard Review Plan (NUREG-0800). Clear lines of authority from the Plant Manager through the Manager Radiation Protection to the Supervisor Radwaste Operations are maintained. Responsibilities for activities important to safe operation (i.e., radwaste processing, interim storage, packaging and shipping) are maintained and the distinct functional area of radiation protection remains separately supervised and managed. The Manager Radiation Protection's independence from operating pressures is not affected by this change.

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

Safety Evaluation No.: 97-135 Rev. 0 & 1

Implementation Document No.: Calculation S14-54-HX10
Procedure N1-STP-062, Rev. 0

UFSAR Affected Pages: N/A

System: Service Water (SW), Reactor Building Closed Loop Cooling (RBCLC), Spent Fuel Pool Cooling (SFPC), Reactor Shutdown Cooling (SDC), Fire Water

Title of Change: Alternate Service Water Configuration

Description of Change:

Maintenance of the SW system was performed during Forced Outage 97-07 due to a valve failure. Procedure N1-STP-062 established alternate service water cooling water flow to/from the RBCLC heat exchangers, isolated service water to the Turbine Building, and isolated service water to the Turbine Building closed loop cooling heat exchangers. The alternate SW system configuration was performed to ensure spent fuel pool and reactor decay heat removal requirements were met during the maintenance activity.

Safety Evaluation Summary:

The SW system alternate configuration is taking place with the plant in cold shutdown, the reactor vessel head installed, no refueling operations in progress and ≥ 40 days after reactor shutdown. Using the heat loads for SFPC, SDC, instrument air, and drywell cooling, engineering calculation S14-54-HX10 showed that SFP temperature would not exceed 125°F and reactor water temperature would not exceed 212°F. Other systems or components supported by RBCLC due to plant condition provided no added heat load, or heat load added is minimal and operation of these systems/components is supported by calculations.

Revision 1 of the safety evaluation removes the temperature criteria for turbine closed loop cooling. The temperature limit of maintaining the system above 85°F is based on normal plant operation and the plant is in cold shutdown for this special test.

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

Safety Evaluation No.: 97-136

Implementation Document No.: Design Change N1-97-037

UFSAR Affected Pages: V-20, XIV-10; Table V-1 Sh 3

System: Emergency Cooling

Title of Change: Replace Emergency Condenser Tube Bundles

Description of Change:

This design change replaced the original emergency condenser tube bundles with new bundles manufactured to be equivalent or better in fit, form, and function, utilizing upgraded materials to be more resistant to intergranular stress corrosion cracking, and utilizing more recent ASME Code Edition(s). A common mode tube failure mechanism was identified in the emergency condensers causing cracking which had propagated through-wall. The shells of the emergency condensers communicate directly with the outside atmosphere. The tubes of the emergency condenser communicate directly with the reactor coolant system. A tube leak in an emergency condenser could provide a direct path for reactor coolant to the environment.

Changes in equipment specifications included: 1) a change in material from Type 304 stainless steel to Type 316 stainless steel (with a special low carbon content); 2) moderately thicker tube walls; and 3) modified arrangement of lifting lugs and rigging points. Also, the replacement bundles are designed and fabricated to ASME III, Subsection NC-1986, where the original bundles were designed to ASME III, Class A, 1965.

Safety Evaluation Summary:

The pressure integrity of the new bundles is ensured and the specified heat removal capacity of the bundles is maintained. The emergency cooling system will continue to be capable of performing its safety function to remove decay heat for all analyzed conditions for which the design basis requires it to perform, and no additional mechanism which could create the possibility of an accident or malfunction or increase the consequences of an accident or malfunction is introduced. There is no reduction in the margin of safety as defined in the UFSAR or Technical Specification Bases. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 97-137

Implementation Document No.: Procedure N1-OP-4

UFSAR Affected Pages: VII-5, X-2

System: Shutdown Cooling (SDC)

Title of Change: Revision to UFSAR to Clarify Statements Concerning the Feed Breaker Position and Control Logic Interlocks Associated With the SDC Isolation Valves (IV) 38-01, 38-02, and 38-13

Description of Change:

The UFSAR stated that SDC IVs will be closed and their circuit breakers locked in the "off" position during normal plant operation. This is not true for IV 38-02, which no longer has a breaker. Electrical protection for SDC IV 38-02 is performed by fuses. The UFSAR has been revised to indicate that power is removed from the valves.

The UFSAR stated that the SDC IVs are "...interlocked so that only one valve can be exercised at a time with reactor pressure greater than 120 psig." SDC inboard and outboard suction IVs 38-01 and 38-02 are interlocked in this manner. The outboard SDC discharge IV 38-12 is a check valve. Therefore, control circuit interlocks between the inboard and outboard discharge IVs 38-13 and 38-12 are not required. The UFSAR has been revised to clarify which SDC IVs are interlocked when the reactor pressure is above 120 psig.

The UFSAR stated that a SDC "...system isolation also occurs if the reactor pressure increases to greater than or equal to 120 psig." A reactor pressure of 120 psig or less is a permissive to open the valves, but does not initiate an automatic SDC system isolation. The SDC system has manual controls for initiation and isolation of the system. The UFSAR has been revised to remove the incorrect reference to automatic system isolation on 120 psig reactor pressure.

Safety Evaluation Summary:

The proposed changes do not alter the performance or intended function of the SDC system isolation valves or any other equipment important to safety. The changes are consistent with the design basis of the SDC system and will not affect existing failure modes nor introduce any new failure modes to any structure, system or component. The UFSAR will be revised to be consistent with the actual

Safety Evaluation No.: 97-137 (cont'd.)

Safety Evaluation Summary: (cont'd.)

plant configuration with respect to the description and operation of the SDC system IVs 38-01, 38-02, and 38-13.

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

Safety Evaluation No.:	97-138 Rev. 0, 1, 2, & 3
Implementation Document No.:	Mod. N1-97-038
UFSAR Affected Pages:	V-17, V-18, X-6; Figures V-1, X-3
System:	Emergency Condenser, Control Rod Drive
Title of Change:	Emergency Condenser Keep-Full Modification

Description of Change:

This modification added a keep-full system which uses part of the flow from the control rod drive (CRD) system and injects it into the emergency condenser (EC) return line. The keep-full function is designed to prevent the possibility of premature failure of the EC tubes (due to thermal cycling) by maintaining the level of condensate in the EC stem inlet leg above the tubes while the plant is in normal operation with the ECs in standby.

Safety Evaluation Summary:

The pressure integrity of the EC condensate return piping is maintained. The emergency cooling system will continue to perform its safety function to remove decay heat for all analyzed conditions. The CRD system will continue to perform its function to inject makeup water to the reactor vessel in case of a small line break. No additional mechanism which could create the possibility for an accident or malfunction or increase the consequences of an accident or malfunction is introduced. There is no reduction in the margin of safety as defined in the UFSAR or Technical Specification Bases.

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

Safety Evaluation No.: 97-139

Implementation Document No.: GE Evaluation KF-9748

UFSAR Affected Pages: IV-20

System: Control Rod Drive (CRD)

Title of Change: UFSAR Update for Control Rod Withdrawal Speed - Cycle 13

Description of Change:

This change updated the UFSAR to provide the cycle 13 specific maximum control rod withdrawal rate. This change allowed operation with withdrawal speeds up to 6.0 inches per second, corresponding to a 24-second stroke time.

Safety Evaluation Summary:

An analysis by General Electric concluded that such operation is bounded by the assumptions used in the rod withdrawal error (RWE) analysis and the minimum critical power ratio safety limit analyses.

Addition of the bases used in the RWE for maximum control rod withdrawal time provides information which can be used to determine operability of a control rod if the stroke time is found out of specification.

The original design and function of the CRD system are unchanged; the ability of the CRD to function as described in the UFSAR is not affected; and the performance requirements as defined in the Technical Specifications are not affected by the proposed change.

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

Safety Evaluation No.: 98-001

Implementation Document No.: Procedures N1-SOP-9, N1-SOP-9.1

UFSAR Affected Pages: 10B-11, 10B-21, 10B-22, 10B-23, 10B-26, 10B-27, 10B-28, 10B-43, 10B-44, 10B-45, 10B-46, 10B-47, 10B-48, 10B-198

System: Emergency Cooling, Shutdown Cooling, Reactor Water Cleanup, Main Steam, Remote Shutdown

Title of Change: Revise Appendix R Safe Shutdown Analysis

Description of Change:

This change revised the Unit 1 Safe Shutdown Analysis by deleting unnecessary Operator actions to counter the effects of spurious operations. Specifically, this change:

1. Deleted all Operator actions requiring verification of valve positions for IV-39-11R, IV-39-12R, IV-39-13R, IV-39-14R, IV-05-01R, IV-05-11, IV-05-04R, IV-05-12, IV-39-07R, IV-39-08R, IV-33-04, BV-33-41, IV-38-02, IV-01-03 and IV-01-04.

2. Deleted Operator actions to manually close valves FCV-39-15, FCV-39-16, IV-39-09R, IV-39-10R, VLV-33-07, VLV-33-08 and IV-38-01.

3. Changes tripping of the Main Turbine to disconnection of the air to IV-01-03 and IV-01-04 in order to isolate the main steam letdown path.

4. Deleted Operator actions to pull cycling electromatic relief valve fuses in addition to monitoring reactor level and pressure parameters in the east/west instrument rooms.

5. Made editorial/format changes to clarify and eliminate inconsistencies regarding the design basis for spurious operations.

Safety Evaluation Summary:

The combined results of both Revisions 2 and 3 to FPEE 1-90-014 were evaluated. The results indicate that:

Safety Evaluation No.: 98-001 (cont'd.)

Safety Evaluation Summary: (cont'd.)

- Verification of valve positions is not required for the Appendix R Safe Shutdown Analysis because alternate actions to manually operate redundant valves, where necessary, were provided in the UFSAR and subsequently incorporated into the post-fire procedures. Therefore, 8-hour emergency battery lighting is not required to support valve position verification actions.

- Operator actions to manually close valves, in lieu of verifying valve positions, are also not required for 10CFR50 Appendix R compliance where spurious operation of non-high/low pressure interface valves is not credible when an excess of two hot shorts is required to cause the spurious operation of equipment. Thus, 8-hour emergency battery lighting to support unnecessary Operator actions is not required.

With the exception of two Operator actions, the proposed changes eliminate the need for accessing the Reactor Building to perform time critical hot shutdown actions. Operation of emergency cooling vent manual isolation valves located on Reactor Building Elevation 340', IV-05-31 (formerly identified as EC-309) and IV-05-32 (formerly identified as EC-310), is still required for fires in fire areas FA6, FA9, FA10 and FA11.

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

Safety Evaluation No.: 98-002 Rev. 0 & 1

Implementation Document No.: Procedure NEP-POL-01

UFSAR Affected Pages: 10A-8, 10A-11, 10A-13, XIII-2; Figure XIII-3

System: N/A

Title of Change: Nuclear Engineering Organization Changes

Description of Change:

This change integrates the Unit 2 Project Management and Plant Evaluation sections into one group. In addition, it creates a new Engineering branch, "Engineering Services," with two new sections, "Engineering Programs" and "Engineering Assurance."

Safety Evaluation Summary:

The proposed organization will continue to perform the functions described in the UFSAR. The intent is to provide greater focus and control of Engineering programs while creating greater programmatic consistency and administrative efficiency. The organizational structure will continue to satisfy the acceptance criteria of Standard Review Plan 13.1.

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

Safety Evaluation No.: 98-004

Implementation Document No.: Procedure N1-OP-6

UFSAR Affected Pages: X-31

System: Spent Fuel Pool Cooling (SFPC)

Title of Change: Spent Fuel Storage Pool Filtration and Cooling System

Description of Change:

This safety evaluation evaluated operating the SFPC system with the flow controller in the automatic or manual mode, thereby providing a constant recirculation flow of spent fuel pool water, and a constant heat removal capability for a given Reactor Building closed loop cooling flow.

Safety Evaluation Summary:

From the time of system stabilization after offload until the next refueling outage, the heat load (decay heat from the spent fuel) is always decreasing. Operator surveillance of the system occurs on a shift basis and is sufficiently frequent to address the vagaries of ambient temperature and evaporative cooling by adjusting the manual setpoint of the flow controller, if necessary.

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

Safety Evaluation No.:	98-005
Implementation Document No.:	Temporary Mod. 98-004
UFSAR Affected Pages:	N/A
System:	Control Room Emergency Ventilation (CREVS)
Title of Change:	De-energize Control Room Ventilation System 15-kW Heater

Description of Change:

The existing plant design for the CREVS contains a 15-kW duct heater associated with the emergency air inlet. Outside air flows through this heater. This heater is energized only if the relative humidity is greater than the setpoint associated with the heater's controller. This temporary modification de-energized the 15-kW duct heater and its associated controls for the CREVS.

Safety Evaluation Summary:

The CREVS, as configured and analyzed, meets the design and licensing basis commitments as defined in the UFSAR and other design and licensing basis documents.

The proposed change does not impact the ability of the CREVS to perform its safety function. The design bases efficiency of the CREVS filters does not depend on the 15-kW duct heater. Therefore, Control Room habitability remains unaffected.

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

Safety Evaluation No.: 98-006

Implementation Document No.: Nuclear Division Policy (NDD-POL)

UFSAR Affected Pages: XIII-1, XIII-2, XIII-4, XIII-5, XIII-8, B.1-2; Figures XIII-1, XIII-2, XIII-5

System: N/A

Title of Change: Reorganization: Change to Nuclear Division Policy to Reflect Establishment of Position, "Vice President Nuclear Generation"

Description of Change:

The Nuclear Division Policy, "POL," has been revised to reflect the new Nuclear Organizational Structure, resulting from the Niagara Mohawk corporate changes approved by the Board of Directors in a restructuring plan filed with the New York State Public Service Commission on February 27, 1998.

The new Nuclear Organizational Structure entails reinstating the position of Vice President Nuclear Generation (previously detailed in NDD-POL, Revision 08). The Vice President Nuclear Generation reports to the Vice President and General Manager - Nuclear, and has oversight responsibility to assure safe, orderly, and efficient plant operation of both units on site; and is responsible for Operations, Radiation Protection, Maintenance, Chemistry, Technical Support, and Outage Management for both units on site. The Plant Managers report directly to this Vice President.

The new Nuclear Organizational Structure also entails creation of the position of Vice President Special Projects. The Vice President Special Projects reports to the Chief Nuclear Officer, and has overall responsibility for issues related to the New York Nuclear Operating Company (NYNOC) and assigned special projects.

Safety Evaluation Summary:

The proposed upper management organizational structure satisfies applicable acceptance criteria and does not impact accident or malfunction initiation or consequences.

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

Safety Evaluation No.:	98-008
Implementation Document No.:	Evaluation FPEE 1-90-015
UFSAR Affected Pages:	Figure 10A-7
System:	Thermal Shield Walls
Title of Change:	Upgrade of Reheater Rooms Walls to Thermal Shield Walls

Description of Change:

This safety evaluation analyzed the fire protection features of the turbine generator condenser area, Fire Zone T1, and determined the possible fire propagation paths outside of this fire zone. The findings and recommendations of FPEE 1-90-015 were used to support the adequacy of cable and component separation criteria for the Unit 1 Safe Shutdown Analysis. UFSAR Figure 10A-7 has been revised to identify the reheater rooms walls as thermal shield walls.

Safety Evaluation Summary:

FPEE 1-90-015 is an analysis of the potential fire propagation paths from a worst-case lube oil fire in the turbine condenser area.

The current configuration of the reheater rooms walls (except the walls facing the turbine) satisfies the Unit 1 design basis requirements for thermal shield walls as defined by the UFSAR and FPEE 1-89-004. The walls are poured concrete and are equivalent to a 3-hour rating. All penetrations in the walls are sealed with 3-hour rated penetration seal assemblies.

This change improves the fire protection of the affected fire zones because the thermal shield walls limit fire spread and damage associated with a worst-case lube oil fire. Reclassifying these walls in the UFSAR also incorporates them into the fire barrier surveillance and breach permit programs. The administrative controls imposed by these programs ensure the configuration and fire barrier integrity of these walls is maintained.

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

Safety Evaluation No.:	98-009
Implementation Document No.:	DDC 1E00331
UFSAR Affected Pages:	III-11; Figure III-14
System:	Control Room Ventilation
Title of Change:	Abandon in Place (4) NSR Control Room Emergency Ventilation System 575 W Heating Pads

Description of Change:

This configuration change permanently abandoned in place the 575-W heating pads. The plant design for the control room emergency ventilation system (CREVS) contains four 575-W heating pads located on the outside of the charcoal filter metal enclosures. There are two heaters located on top of the filter housing and two on the bottom, all designated as Component ID H-210-19R. Heating pads were originally installed around 1979 to assure that the iodine removal efficiency of the charcoal filters was sufficient to meet Technical Specification requirements as documented in a safety evaluation (SE) dated February 23, 1979. The SE stated that the heaters would prevent excessive moisture buildup which decreases the charcoal filter adsorption efficiency. The function of the heating pads to prevent moisture buildup was classified as nonsafety related because the heating pads were only intended to add additional conservatism in the design for moisture removal. A field walkdown of the charcoal filter housing revealed that the power to the heaters was disconnected. Furthermore, the temperature switches which control the operation of the heaters are not capable by design to be calibrated because the sensing elements are not accessible. The short-term solution was to disconnect the power to the heating pads, and the long-term solution was to implement a plant modification to permanently abandon the heating pads in place.

Safety Evaluation Summary:

The CREVS, as configured and analyzed, meets the design and licensing basis commitments as defined in the UFSAR and other design and licensing basis documents. The proposed change does not impact the ability of the CREVS to perform its safety function. The design bases efficiency of the CREVS filters does not depend on the four 575-W heating pads. Therefore, Control Room habitability remains unaffected.

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

Safety Evaluation No.:	98-010
Implementation Document No.:	Mod. N1-98-009
UFSAR Affected Pages:	III-9, III-10, III-11, III-12, VIII-4, VIII-7; Table VIII-4 Sh 1, 2, 3
System:	Control Room Heating, Ventilating and Air Conditioning
Title of Change:	Change Control Room Air Treatment System Intake Duct High Radiation Monitor Setpoint and Add Loss-of-Coolant Accident Signal and Main Steam Line Break Signal to Control Room Emergency Ventilation System Initiation Logic

Description of Change:

This modification added a spare relay contact from reactor protection system logic circuits to initiate the Control Room Emergency Ventilation System (CREVS) on a main steam line break signal as well as a loss-of-coolant accident (LOCA) signal. The CREVS will auto-initiate on main steam line high flow and main steam line tunnel high temperature and on high drywell pressure or lo-lo reactor vessel water level. This change was installed and tested with the unit in cold shutdown. NRC approval was obtained prior to declaring the Control Room Air Treatment (CRAT) system operable with these new changes installed.

In addition, this change also provided for a change to the actuation setpoint for the CREVS intake duct high radiation monitors to be consistent with NMP1's Control Room habitability study and in compliance with the Technical Specifications. This proposal involved decreasing the radiation monitor trip setpoint for RAM-210-42A and RAM-210-43A from ≤800 cpm to ≤193 cpm.

Safety Evaluation Summary:

The proposed change to the CRAT system ventilation intake duct radiation monitor setpoint does not impact the UFSAR or Technical Specifications. Its implementation may take place with the unit in any reactor mode of operation.

With the unit in cold shutdown, the installation of the new Control Room emergency actuation logic will not adversely impact the CRAT system. System initiation on main steam line break will not occur and with the unit in cold shutdown, the probability of an automatic actuation of the CREVS will not be increased. Installation and post-installation testing will be controlled using

Safety Evaluation No.: 98-010 (cont'd.)

Safety Evaluation Summary: (cont'd.)

approved procedures to ensure that the LOCA signals do not disable the core spray, reactor recirculation or emergency core cooling systems.

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

Safety Evaluation No.:	98-011
Implementation Document No.:	Procedures N1-FHP-27A, N1-OP-34, N1-OP-6
UFSAR Affected Pages:	X-30, X-31
System:	Spent Fuel Pool Cooling (SFPC)
Title of Change:	Increasing the Maximum Allowable Bulk Spent Fuel Pool Water Temperature With a Single Failure to 140°F

Description of Change:

In the recent past, refueling outages at NMP1 have begun during the month of February. During the month of February, the normal water temperature of Lake Ontario is between 30°F and 40°F. Refueling Outage 15 (RFO15) commenced in Spring 1999, when Lake Ontario temperatures were anticipated to be less than 81°F, the Technical Specification maximum for lake water temperature. This safety evaluation evaluated an increase in the SFPC system maximum bulk pool temperature to 140°F from 125°F, in the event of a single active failure. It evaluated using one or both SFPC trains to maintain the spent fuel pool temperature ≤125°F, with a maximum pool temperature of ≤140°F in the event of a single active failure.

Safety Evaluation Summary:

NMP1 was evaluated against the criteria in Standard Review Plan (SRP) 9.1.3 for Technical Specification Amendment 54; therefore, it is appropriate to evaluate NMP1 against the criteria in SRP 9.1.3 when increasing the SFPC system maximum bulk pool temperature to 140°F from 125°F. Both types of spent fuel racks in the SFP (flux trap and Boraflex) have been analyzed (thermal/hydraulic and criticality) to temperatures higher than 140°F and found to be acceptable. The pool structure has been analyzed to 212°F and the system piping design temperature is 170°F. The maximum temperature for the anion resin used in Epifloc 21-H is 140°F; therefore, the SFPC system's ability to remove corrosion products, impurities and radioactive materials from the pool water will not be affected. Increasing the pool water temperature to 140°F will not cause nucleate boiling to occur.

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

Safety Evaluation No.: 98-014

Implementation Document No.: Mod. N1-98-015

UFSAR Affected Pages: III-11; Figure III-14

System: Control Room Ventilation

Title of Change: Control Room Ventilation Damper Control

Description of Change:

This modification removed power from the outside air and recirculation air damper control motors. Dampers are now physically adjusted in the field in lieu of using the remote manual controller in the Control Room. This change does not affect the damper position, only the method by which the damper is correctly positioned.

This design change: 1) allows the recirculation air damper to be adjusted without affecting the position of the outside air damper; 2) allows independent adjustment of the outside air damper to control the amount of outside air flow; and 3) eliminates the possibility of damper movement due to inadvertent movement of the control switch or a single electrical failure of the damper motors or associated controls.

Safety Evaluation Summary:

The Control Room Emergency Ventilation System (CREVS) is not an initiator of any accidents previously evaluated in the UFSAR. The CREVS is utilized in maintaining Control Room habitability during and after a loss-of-coolant accident, main steam line break, fuel handling accident, and the control rod drop accident. This change will not impact the ability of the CREVS to perform its safety function. The change in method by which the dampers are positioned will not cause a change to any system interface in a way that would increase the likelihood of an accident. Therefore, the proposed change does not increase the probability of occurrence of an accident previously evaluated in the UFSAR.

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

Safety Evaluation No.:	98-015 Rev. 0 & 1
Implementation Document No.:	Mod. N1-98-016
UFSAR Affected Pages:	III-11, III-12, III-45, XV-25, XV-26, XV-82; Figure III-14
System:	Control Room Ventilation
Title of Change:	Reconcile Design and Licensing Basis Air Flow Rates for Control Room Air Treatment System

Description of Change:

This modification revised both the filtered and unfiltered design air flow rates for the Control Room Air Treatment (CRAT) system.

Several existing system parameters were verified as appropriate or reestablished for the subject design change. Included were maximum outside air temperature and relative humidity, the maximum ambient Control Room envelope temperature under normal and emergency conditions, and minimum and maximum outside air flow rates. The CRAT system was modeled, analyzed and tested. The analyses and testing demonstrated that the system meets design basis and regulatory requirements under all normal and emergency conditions.

Safety Evaluation Summary:

The CRAT system is not an initiator of any accidents previously evaluated in the UFSAR. The CRAT system is utilized in maintaining Control Room habitability during and after a loss-of-coolant accident, main steam line break, fuel handling accident, control rod drop accident and small line break. The proposed changes will not cause a change to any system interface in a way that would increase the likelihood of an accident. Therefore, the proposed changes do not increase the probability of occurrence of an accident previously evaluated in the UFSAR.

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

Safety Evaluation No.: 98-016

Implementation Document No.: DDC 1E00342

UFSAR Affected Pages: Figure IX-6

System: Diesel Generator

Title of Change: Control Room Fire Dampers - DG Loading

Description of Change:

This change enables power to be restored to the fire dampers in Smoke Zone 5, Auxiliary Control Room, following a loss of offsite power (LOOP) by adding an automatic reset circuit to these fire dampers. The automatic reset will actuate following a LOOP and diesel generator loading. This new design feature will restore the Auxiliary Control Room ventilation flow path and allow the Control Room Emergency Ventilation System to function following a design basis accident without the need for Operator action.

Safety Evaluation Summary:

This change will result in additional continuous loading on the diesel generators; however, calculated loading on each diesel generator is below the design load limit corresponding to the 2000 hour/year rating of the units. This has been analyzed in relation to design and licensing requirements found in the UFSAR, Technical Specifications, and design basis documents, and is acceptable.

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

Safety Evaluation No.: 98-019

Implementation Document No.: DDC 2S11106

UFSAR Affected Pages: Figure III-1

System: N/A

Title of Change: Demolition of Steel Fab Shop

Description of Change:

This safety evaluation evaluated the demolition of the Steel Fab Shop located east of the NMP2 plant structures. This building was built for use as a temporary building during the construction of NMP2 and was demolished as part of the Facilities Improvement Program.

Safety Evaluation Summary:

The building being demolished is located in an area that was not used as a flow channel for the probable maximum precipitation analysis. Removal of this building and the consequent reduction in the run-off coefficient would make the analysis more conservative. The building has no impact on the previously calculated X/Q values. The design margins for the Control Room fresh air intakes are not compromised. The location of the demolition activities is adequately separated from safety-related systems and structures to preclude any adverse impact from construction activities.

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

Safety Evaluation No.: 98-020

Implementation Document No.: DDC 1S00276

UFSAR Affected Pages: Figures III-4, 10A-4, 10A-5

System: N/A

Title of Change: Replace Existing Gates G-6 and G-7 with New 12'-0" High Out-Swing Gates

Description of Change:

The 4'-0" high gates previously located on Turbine Building El. 261' at doors D-40 and D-41 prevented personnel from exiting/entering the Turbine Building. New gates have been installed which are 12'-0" high and swing out instead of swinging inward. The added height is due to a NRC OSRE inspection finding that requires additional height, when doors D-40 and D-41 are open, to prevent access to the Turbine Building.

Safety Evaluation Summary:

The 12'-0" high gates provide the same functional requirements as the existing gates by ensuring that unauthorized access or unsanctioned exiting from the radiologically-controlled area is maintained and controlled. In addition, the replacement has no effect on plant systems, structures or components.

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

Safety Evaluation No.: 98-021

Implementation Document No.: Evaluation FPEE 1-90-014 Rev. 4

UFSAR Affected Pages: 10B-24, 10B-32, 10B-33, 10B-207

System: Core Spray

Title of Change: Appendix R High/Low Pressure and Inventory Loss Flow Path Isolation for the Core Spray System Injection High Point Vent Lines

Description of Change:

The NMP1 Appendix R Safe Shutdown Analysis (SSA) considers the core spray system injection high point vent lines as potential leak paths. The analysis took credit for normally closed manual valves 40-26 (loop 11) and 40-27 (loop 12) for the high/low pressure and inventory loss interface isolation function of the vent lines. This was documented in Fire Protection Engineering Evaluation FPEE 1-90-014, Revisions 1, 2, and 3. During the design verification process of Revision 3 of this FPEE, it was discovered that the normal position of valves 40-26 and 40-27 was changed to locked open, throttled, to preclude post-LOCA thermal overpressurization concerns in the high point vent line piping between valves IV-40-30 and IV-40-32 in loop 11, and IV-40-31 and IV-40-33 in loop 12. The thermal overpressurization concern was identified in Generic Letter 96-06.

This change resulted in a discrepancy between the SSA and existing plant configuration. As a result of implementation of one of the corrective actions, Appendix R high/low pressure and inventory loss interface isolation function for the core spray high point vent line will now credit the normally closed motor-operated valves 40-30 and 40-31. To prevent spurious operation of these valves, the associated power supply breakers are locked open.

Safety Evaluation Summary:

Motor-operated valves 40-30 and 40-31 are normally closed. If open during administratively-controlled surveillance testing, they close automatically by either a RPS signal or interlock with the associated core spray injection valves. They can also be closed manually by operation of a hand switch located in the Control Room. The safety-related function of these valves during and following a design

Safety Evaluation No.: 98-021 (cont'd.)

Safety Evaluation Summary: (cont'd.)

basis accident is to close and remain closed to maintain pressure boundary integrity of the core spray system and primary containment. Locking open of the supply circuit breaker for these valves during normal plant operation will not prevent them from performing their safety-related function(s). The proposed change is consistent with current Unit 1 design and licensing bases for compliance with 10CFR50 Appendix R. It establishes a high/low pressure interface and inventory loss flow path boundary equal to the boundary previously evaluated for the core spray system high point vent. This proposed change will not affect the safety-related function of the core spray system high point vent components and will maintain the SSA safe shutdown capability.

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

Safety Evaluation No.:	98-022
Implementation Document No.:	DDC 1M00627
UFSAR Affected Pages:	XI-7
System:	Offgas
Title of Change:	Offgas Chiller Discharge Temperature Setpoint Change

Description of Change:

The offgas chiller outlet temperature was changed from -4°F to 20°F to eliminate the frequent cycling of the chiller in the summer and to improve the reliability of the system.

Safety Evaluation Summary:

UFSAR Section XI-B.3 states that the chillers are provided to remove moisture from the gas, through cooling, prior to the gas entering the charcoal columns. After two hours of cool cycle, if a chiller outlet temperature exceeds -4°F for longer than a preset time, another chiller will automatically start. An evaluation was performed and results of the evaluation demonstrate that the chiller outlet temperature can be raised to 20°F. The holdout time of the charcoal delay system is still within the license and design basis limit. This change does not involve change in the function of the offgas system and does not change the design inputs used in any transient or accident analysis or reduce the margin of safety of any Technical Specification.

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

Safety Evaluation No.:	98-023 Rev. 0 & 1
Implementation Document No.:	Calculation S11-01F004
UFSAR Affected Pages:	XV-30
System:	Main Steam
Title of Change:	NMP1 UFSAR Correction and Technical Specification Bases Change

Description of Change:

This change corrects the UFSAR which previously stated, "The valves are set so that initiation of closure occurs with a venturi pressure loss of 20 psi." The UFSAR statement was incorrect based on the results derived in main steam venturi flow calculation S11-01F004 and is not consistent with the "...unrecovered differential pressure consistently about 10 percent..." conclusion in the UFSAR. Calculation S11-01F004 validates anticipated differential pressure and flow performance characteristics associated with main steam line (MSL) flow venturi(s) (FE-01-98, FE-01-101) during a postulated large MSL break event. Also for clarification, this change revised the UFSAR main steam line critical flow statements to be in terms of venturi differential pressure instead of venturi pressure loss.

Safety Evaluation Summary:

This proposal changes an incorrect statement in the UFSAR which is not consistent with the results derived in main steam venturi flow calculation S11-01F004 or with conclusions in UFSAR Section VII.F. This change also provides clarification to the UFSAR by discussing venturi critical flow parameters in terms of differential pressure instead of in terms of pressure loss. This change will have no impact on the safe operation or shutdown of the plant. This proposal will have no impact on any Technical Specification (TS) required actions or response times associated with MSL isolation or main steam isolation valve closure times during a postulated MSL guillotine break event. There is no reduction in the margin of safety, as defined in any Technical Specification Basis, for MSL high flow isolation because the increase in the calculated setpoint allowable value is insignificant compared to the magnitude of the calculated margin. This change does not impact any previously calculated site MSL break accident dose rates in the UFSAR or impact any 10CFR100 limits. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 98-057

Implementation Document No.: Procedures GAP-POL-01, NIP-TQS-01

UFSAR Affected Pages: XIII-7

System: N/A

Title of Change: Radiation Protection Department Organization Change

Description of Change:

The position titled "ALARA Supervisor" has been changed to "Lead Engineer - ALARA/Radiological Engineering" and the responsibilities of the position titled "Internal and External Dosimetry Supervisor" have been absorbed by this position. All other duties and functions of the revised titled position remain unchanged. The position titled "Internal and External Dosimetry Supervisor" has been eliminated, with all duties and functions absorbed by the current "Generation Specialist" position, with reporting lines to the Lead Engineer - ALARA/Radiological Engineering.

Safety Evaluation Summary:

The proposed Radiation Protection Department organizational changes conform to the NMP2 Technical Specifications Section 6.2.1 and do not impact initiation of accidents or a malfunction of equipment important to safety or radiological consequences.

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

Safety Evaluation No.: 98-100

Implementation Document No.: Temporary Mod. 98-025

UFSAR Affected Pages: N/A

System: N/A

Title of Change: Temporary Removal of Turbine Building West Wall Section

Description of Change:

This temporary modification installed a noncombustible enclosure over the hatch on Elevation 291'-0". The enclosure had roll-up doors installed at the west and east ends. There was a space of approximately 2 ft. between the enclosure wall and the inside of the Turbine Building west wall which was sealed all around using flashing on both sides and on the top of the enclosure. After the enclosure was installed and sealed, a section of panels was removed from the Turbine Building west wall exposing the west side of the enclosure to the outside to allow the feedwater to travel in/out of the Turbine Building. When the feedwater heater replacement was completed, the wall panel was installed, followed by the disassembly of the enclosure, returning the Turbine Building to its original condition.

Safety Evaluation Summary:

This modification will provide an adequate entry/exit for the existing and replacement feedwater heaters without compromising any of the functions of the Turbine Building. The temporary modification will provide the same function as the existing wall and adhere to all of the same standards. It will have no impact on the safe operation or shutdown of the plant.

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

Safety Evaluation No.:	98-101
Implementation Document No.:	NMP1 Emergency Operating Procedures, Severe Accident Procedures, Plant-Specific Technical Guidelines and Plant-Specific Severe Accident Guidelines
UFSAR Affected Pages:	VI-24, VIII-44, XIII-12, XVIII-3; Table VIII-1
System:	Various
Title of Change:	Changes to the NMP1 Emergency Operating Procedures and Establishment of the NMP1 Severe Accident Procedures

Description of Change:

This revision of the Emergency Operating Procedures (EOP) and the establishment of the Severe Accident Procedures (SAP) implement the new guidelines established by the BWR Owners' Group (BWROG). The SAPs, together with the modified EOPs, form an integrated set of symptomatic instructions that attempt to cover all possible mechanistic accident sequences. This new revision adopts a new strategy for coping with emergency conditions which degrade into severe accidents. The EOPs contain strategies applicable prior to the transition to a severe accident, and the SAPs contain strategies applicable after the transition.

Safety Evaluation Summary:

This safety evaluation addresses the use of the BWROG Emergency Procedure Guidelines and Severe Accident Guidelines (EPG/SAG) and Appendices as the basis documents for general revision of the NMP1 Plant-Specific Technical Guidelines (PSTG) and SAGs, as well as the EOPs and SAPs.

Changes to the EPGs have been made to remove strategies from the EPGs that are applicable only to the SAGs, ensure a smooth transition from the EPGs into the SAGs, and incorporate recommended EPG changes identified during the severe accident mitigation development effort.

This safety evaluation concluded that implementation of the changes from Revision 4 of the EPGs to Revision 1 of the EPGs/SAGs is warranted as it applied to the NMP1 EOPs/SAPs. It was also concluded that the exceptions taken from Revision 1 of the EPGs/SAGs, as depicted in the NMP1 PSTGs/PSSAGs and EOPs/SAPs, are appropriate and warranted.

Safety Evaluation No.: 98-101 (cont'd.)

Safety Evaluation Summary: (cont'd.)

The proposed changes do not increase the probability of occurrence or consequences of any accident previously evaluated in the UFSAR. It was also concluded that the proposed changes do not increase the probability of occurrence or consequences of a malfunction of equipment important to safety evaluated previously in the UFSAR. This was determined even though the EOPs/SAPs do prescribe actions that may authorize operation of equipment beyond their normal design parameters and may prescribe the defeat of design interlocks. Finally, it was determined that the proposed changes do not create the possibility of an accident or malfunction of equipment important to safety of a different type than any evaluated previously in the UFSAR, and do not reduce the margin of safety as defined in the basis for any Technical Specification.

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

Safety Evaluation No.: 98-103 Rev. 0 & 1

Implementation Document No.: Mod. N1-97-033

UFSAR Affected Pages: IV-26, IV-30, IV-32, XVI-12, XVI-21, XVI-21a, XVI-124; Table XVI-9a Sh 1 & 2; Figures XVI-12c, XVI-12d

System: Reactor Vessel

Title of Change: Core Shroud Vertical Weld Repair Clamp

Description of Change:

The Unit 1 core shroud vertical weld repair addressed the cracking of vertical welds V4, V9, and V10. The repair consisted of a clamp with a plate with attached pins which were inserted into holes which were machined by the electric discharge machining process on either side of the flawed vertical weld. The clamp bridges across the flawed vertical weld and transmits the pressure load normally transmitted through the vertical weld. Two clamps were used for the V9 weld, two clamps for the V10 weld, and one clamp was used for the shorter V4 weld. Prior to this repair being utilized as a structural replacement for the welds, NRC approval was received.

Safety Evaluation Summary:

The repair weld clamps transmit the shroud hoop pressure force which would be transmitted through the shroud vertical weld. The structural load path is from the shroud through a bayonet eccentric/threaded pin to the clamp plate, and through the clamp plate and other bayonet eccentric/threaded pin assembly back to the shroud. The repair and the shroud attachment satisfy the design by analysis stress and fatigue criteria of the ASME Boiler & Pressure Vessel Code, Section III, Subsection NG, for all core shroud design basis load combinations, and the repair maintains the core shroud allowable leakage to within design basis allowables for normal emergency and accident conditions. This repair does not provide any new accident initiators as described based on the analysis performed.

The special equipment and processes which are being used to minimize the in-vessel debris generation and the shroud repair installations have no impact on other work, and do not affect any systems which are initiators of an accident evaluated in the UFSAR. The requirements of NUREG-0612 will be met. The tooling for "heavy loads" has been designed and will be used in accordance with NUREG-0612. The Framatome Technologies Incorporated (FTI) auxiliary bridge has been accepted for use at Unit 1 via Safety Evaluation 99-007. The FTI core

Safety Evaluation No.: 98-103 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

cover has been evaluated to be acceptable for use at Unit 1 without fuel in the vessel in accordance with Procedure N1-FHP-52. Therefore, the core shroud repair installation process will not increase the probability of any initiators or precursors of the transients and accidents previously evaluated in the UFSAR.

Since the core shroud repair maintains the core shroud function within ASME Code criteria and within design basis criteria for all design basis load combinations, the core shroud functions assumed in the UFSAR are maintained and the probability of occurrence of an accident previously evaluated in the UFSAR is not increased.

Based on the evaluation performed, it is concluded that addition of the vertical weld repair does not involve an unreviewed safety question.

Safety Evaluation No.: 98-104 Rev. 0 & 1

Implementation Document No.: Mod. N1-96-005

UFSAR Affected Pages: VII-2, VII-3, VII-5, VII-7, VII-9 thru VII-13, XV-40, XV-82, XVI-52, XVI-52a; Figures VII-1, VII-3

System: Containment Spray, Core Spray

Title of Change: ECCS Suction Strainer Replacement

Description of Change:

This modification installed suppression pool suction strainers on the Unit 1 core spray and containment spray systems in response to NRC Bulletin 96-03. This modification also included the following: 1) removed retired-in-place H202 monitoring tubing in the torus; 2) removed the torus spray strainer baskets in STR-80-63 and STR-80-64; 3) provided new spectacle flanges at FLG-81-99 and FLG-81-100 with a flow side strainer opening; 4) removed the small opening strainer baskets and the pressure control valves and replaced them with orifices; 5) capped the differential pressure switches and removed the Control Room annunciators; and 6) the northwest torus hatch access ladder was removed and not reinstalled. The emergency core cooling system (ECCS) strainers installed by this modification are designed to accommodate the quantity of debris expected to be generated during pipe break accidents in accordance with the BWROG's URG for ECCS Suction Strainer Blockage. New ECCS suction strainers and their supporting assemblies were installed in a dual suction strainer design. Existing downstream strainers had their baskets removed and their differential pressure switches and alarms made obsolete.

Safety Evaluation Summary:

The replacement ECCS strainers were designed to meet the worst-case debris loading, while still meeting the core spray and containment spray pumps net positive suction head (NPSH) and filtration requirements. The new ECCS strainer design accounts for the generation, transport and accumulation of debris during accident mitigation. The newly added ECCS strainers and their supports are designed to the ASME Boiler and Pressure Vessel Code and are seismic Category 1 safety-related components.

NUREG-0661 was used in the evaluation of the suction strainer assemblies, as well as the Mark I Containment Program Load Definition Report, Plant-Unique Analysis Application Guide, and the NMP1 Plant-Unique Analysis Report. The new

Safety Evaluation No.: 98-104 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

ECCS strainers satisfy the design by analysis by using the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsections NC and NE, for all containment spray and core spray system design basis load combinations; and by verifying that core spray and containment spray pump NPSH available is greater than NPSH required. Strainer assemblies' installation movement will be in accordance with the heavy loads criteria of NUREG-0612. The design conforms to the guidance outlined in NRC Bulletin 96-03 and the Unit 1 existing licensing basis.

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

Safety Evaluation No.:	99-001
Implementation Document No.:	Mod. N1-98-038
UFSAR Affected Pages:	IV-18; Figures IV-7, IV-7a
System:	Control Rod Drive (CRD)
Title of Change:	Use of Modified BWR/6 Control Rod Drives at NMP1

Description of Change:

This safety evaluation addressed the use of modified General Electric BWR/6 original equipment CRDs as standard replacement CRDs at Unit 1. The BWR/6 CRD design incorporates a new hydraulic buffer configuration and higher-strength materials, and implements the latest design improvements. The CRDs were modified by replacing the uncoupling rod, cooling water orifice, o-ring spacer plate, and adding two additional holes to the ring flange to accept the existing position indicating probe. The modification was performed by the original equipment manufacturer, General Electric Company (GE). The modified BWR/6 CRDs have been successfully installed in several BWR/2-5 nuclear operating plants.

Safety Evaluation Summary:

The BWR/6 drives were designed to be fully compatible with the existing equipment interfaces. The drive functions the same as the existing drives, and the supporting hydraulic system requirements remain unchanged. The BWR/6 drive operating performance in a BWR/2-5 plant will remain the same as the existing drives. As applied at Nine Mile Point (BWR/2-5), the BWR/6 drive will be subjected to significantly lower operating loads than its inherent design capability. Furthermore, the BWR/6 drive design has been subjected to one of the most comprehensive evaluation and testing programs conducted by GE. As a result, the BWR/6 drive design has demonstrated its reliable performance throughout the expected operating conditions.

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

Safety Evaluation No.: 99-002 Rev. 0 & 1

Implementation Document No.: Procedures GAP-OPS-01, NDD-OPS

UFSAR Affected Pages: XIII-16

System: N/A

Title of Change: Change to Content of the Radwaste Log Book

Description of Change:

This change revised the UFSAR to reflect the updated content requirements for the Radwaste Log Book, which is maintained in the Radwaste Control Room by the on-shift Operator.

Safety Evaluation Summary:

The proposed change to the description of the Radwaste Log Book meets the recordkeeping requirements specified in Procedures NDD-OPS and GAP-OPS-01, as well as the guidelines in INPO OP-205. The responsibility for maintaining the log and the storage of completed logs is not affected by this change and nuclear safety is not adversely affected.

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

Safety Evaluation No.: 99-003

Implementation Document No.: Temporary Mod. 98-029

UFSAR Affected Pages: N/A

System: Structures, Turbine Building Shield Wall

Title of Change: Temporary Removal of Block Walls

Description of Change:

This temporary modification removed block walls from third-point feedwater heater rooms 113 and 133 to create a transit path to allow for the removal of heaters 113 (HTX-51-13) and 133 (HTX-51-15), and to allow for the subsequent installation of the new replacement heaters. The block walls were removed at power prior to shutdown. After the heater replacement was completed, the block wall was reconstructed to its original configuration.

Safety Evaluation Summary:

The major significance of this block removal relates to potential dose rate increases throughout the site. Areas of potential impact include the Control Room and those immediate areas surrounding the egress created by the block wall removal. A radiological calculation performed for the Control Room is not impacted because no direct shine path exists and, therefore, post-accident dose rates in the Control Room will not be increased. The installation of wall coverings for those openings created by the removal of the block walls will maintain access control for entry into these high radiation compartments. Additionally, for those areas immediately outside the temporary wall coverings that have dose rates of 5 mr/hr or greater, and are within the "radiation area" dose level, radiological postings and boundaries will be installed. Therefore, the resulting dose rates have been demonstrated to be within the UFSAR analyses and managed by plant processes and procedures.

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

Safety Evaluation No.: 99-004

Implementation Document No.: Procedure GAP-POL-01

UFSAR Affected Pages: XIII-1, XIII-6, XIII-7, XIII-8, XIII-20, B.1-2, B.1-3, B.1-5; Figure XIII-2

System: N/A

Title of Change: Establishment of the Assessment and Corrective Action Group

Description of Change:

This safety evaluation evaluated a revision to Procedure GAP-POL-01 to reflect establishment of the Nuclear Generation/Assessment and Corrective Action group, creation of the position of Director - Assessment and Corrective Action, and transfer of various functions to the Assessment and Corrective Action group. The Director - Assessment and Corrective Action reports directly to the Vice President Nuclear Generation and is responsible to ensure consistency in approach and application of the DER program, OE program, and branch self-assessments.

Functions to be transferred to the newly-formed Assessment and Corrective Action group will include DER trend data reporting, administration of the DER database, processing of industry operating experience/OE, and oversight of the branch self-assessment process.

Safety Evaluation Summary:

The proposed management organizational structure and transfer of functional responsibilities satisfies applicable acceptance criteria and does not impact accident or malfunction initiation or consequences, nor does it affect the design of structures, systems or components, the operation of plant equipment or systems, nor maintenance, modification, or testing activities.

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

Safety Evaluation No.:	99-006 Rev. 0 & 1
Implementation Document No.:	Mod. N1-97-017
UFSAR Affected Pages:	Figure VIII-2
System:	Feedwater, High-Pressure Coolant Injection (HPCI)
Title of Change:	Feedwater Control System Setpoint Setdown

Description of Change:

This modification added a Feedwater Control system setpoint setdown circuit to the Feedwater Control system coincident with a reactor scram. The modification also added a new blind controller with a reduced setpoint that provides a lower water level control input to flow control valve (FCV) #13 valve controller. The controller gets its initiation signal from a reactor pressure vessel (RPV) low water level, coincident with scram confirmatory logic taken from a set of Channel 11 and 12 reactor protection system scram relay contacts located in the Channel 11 reactor trip bus circuit.

Safety Evaluation Summary:

Lowering the setpoint demand on FCV #13 following an automatic or manual reactor scram coincident with a RPV low-level alarm will not prevent the Feedwater/HPCI system from performing any of its functions described in the UFSAR. This change meets the design, material and construction standards for the Feedwater/HPCI systems, and the new circuits are separated and isolated from the safety-related and "Q" functions of the existing circuits; therefore, the likelihood of an accident occurring because of failures in these systems remains the same. The failure modes of the new circuits are bounded by the failures of the existing feedwater control circuits; therefore, the probability of the loss of feedwater is not increased. The change enhances the ability of the Feedwater Control system to maintain acceptable water level following a scram, and avoid potential high RPV water level immediately following a scram; therefore, the setpoint setdown function is not an initiator or precursor to an accident or transient. This setpoint change does not affect any design basis setpoints currently used to evaluate an accident evaluated in the UFSAR. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 99-007

Implementation Document No.: Temporary Mod. 99-007

UFSAR Affected Pages: N/A

System: Reactor Vessel

Title of Change: Use of Auxiliary Platform

Description of Change:

This temporary modification utilized the auxiliary work platform to provide personnel with a work area from which the activities previously performed from the Refuel Bridge could be performed during the refuel outage. These activities included, but were not limited to, the following: 1) reactor vessel examinations, 2) shroud inspections, 3) in-vessel visual inspections/repairs, 4) tie rod inspections, and 5) shroud/core spray piping/sparger repair activities.

The auxiliary platform is equipped with two 2500-lb. capacity jib cranes that were designed per the requirements of NUREG-0612 and ANSI N14.6. The auxiliary platform is designed to be moved manually along the Refuel Bridge rail tracks as required for the above-referenced activities.

Safety Evaluation Summary:

The auxiliary platform is seismically designed such that it will not collapse, fall or leave the Refuel Bridge rails during a seismic event. The two jib cranes are designed per the requirements of NUREG-0612 and ANSI N14.6, and heavy load lifts performed with these cranes will be controlled via N1-MPP-GEN-914 and other applicable procedures. The addition and use of the auxiliary platform and its two jib cranes will facilitate vessel serving and inspection activities previously performed from the Refuel Bridge.

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

Safety Evaluation No.: 99-008

Implementation Document No.: Temporary Mod. 99-010

UFSAR Affected Pages: N/A

System: Emergency Cooling

Title of Change: Close Low Point Drain Valves VLV-39-112 and VLV-39-111

Description of Change:

The emergency cooling system low point drain valves VLV-39-112 and VLV-39-111 were temporarily closed to eliminate or reduce the steam leak from the packing gland of VLV-39-112. This temporary configuration was removed during Refueling Outage 15 when a new replacement valve was installed.

Safety Evaluation Summary:

The emergency cooling system will continue to be capable of performing its safety function to remove decay heat for all analyzed conditions for which the design basis requires it to perform, and no additional mechanism which could create the possibility for an accident or malfunction or increase the consequences of an accident or malfunction is introduced. The pressure integrity of the emergency cooling is ensured by isolating the steam leak, and the ability of the low point drain function will not be adversely impacted.

The drain lines for the steam supply piping to the loop 11 emergency condenser are not an initiator of any accidents previously evaluated in the UFSAR. The isolation of one of two parallel condensate drains from the steam supply piping to the loop 11 will have no effect on the ability of the drains to prevent water hammer, as the remaining drain will adequately drain the supply piping with the reactor at power. Therefore, this proposed temporary change does not affect the probability of an accident previously evaluated in the UFSAR.

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

Safety Evaluation No.: 99-009 Rev. 0 &1

Implementation Document No.: Mod. N1-98-014

UFSAR Affected Pages: VIII-8, VIII-55

System: Main Steam

Title of Change: Main Steam Line Electromatic Relief Valve Setpoint Tolerance Increase

Description of Change:

The as-found calibration of three of the six electromatic relief valve (ERV) pressure switches, PIS-36-90A, E and F, were found to be out of their calibration tolerance. The calibration tolerance for these setpoints is very tight at ±1%. Historically, in the industry it has been identified that it is extremely difficult to maintain relief valve setpoints within a ±1% tolerance band. Therefore, across the industry many safety valve setpoint tolerances have been revised up to ±3% in accordance with ANSI/ASME OM-1-1981; however, it does not address relief valves. Since the ERVs are not ASME safety valves, adherence to this criteria is not required. The revised allowable value will be ±2%.

The new design will change only one parameter from the existing design. The setpoint error will change from ±12 to an allowable value of ±24 psig, which is a revision of approximately ±1% to ±2%. The margin of error between the new analytical limits and the existing setpoints will be ±40 psig or approximately ±3.7%.

Safety Evaluation Summary:

This safety evaluation discusses the increase in the ERV setpoint tolerance (allowable value) from ±12 psig to ±24 psig. The setpoint tolerance has been analyzed for an analytical limit of ±40 psig. A thorough analysis has been made of all applicable transients and accidents and all critical parameters remain within design values. The change will have the restriction that the as-found value for at least two relief valves must be greater than the as-found high reactor pressure scram value.

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

Safety Evaluation No.: 99-011

Implementation Document No.: Procedure NEP-POL-01

UFSAR Affected Pages: Figure XIII-3

System: N/A

Title of Change: Nuclear Engineering Organization Changes

Description of Change:

The Engineering organization includes two unit-specific Engineering branches and a common Engineering Services branch. The Engineering Services branch has created a new group that encompasses the ASME Section XI program functions which were previously maintained by the Engineering Programs group. This reorganization strengthens these programs by providing more focus and dedicated supervisory oversight.

Responsibilities of the ASME Section XI Programs group include the development, maintenance, implementation, and assessment of the effectiveness of the following programs:

- In-service Inspection
- In-service Testing
- Containment Boundary Examinations
- Pressure Testing
- Appendix J
- Erosion/Corrosion

Safety Evaluation Summary:

The proposed organization change remains within the acceptance criteria utilized for the basis of the current Engineering organization, including NUREG-0800. It does not affect the Emergency Response functions of the Engineering organization, or the operation, testing and maintenance assumptions in the UFSAR. Implementation of the proposed organization change requires revision to existing policies and procedures that govern the applicable activities. Conformance with the procedures ensures that personnel qualification and training will be maintained in the new organization for Engineering personnel who evaluate plant conditions and accident scenarios. Primary areas affected by this change are the administrative, technical and maintenance procedures that describe functional responsibility and activities. This change does not alter the requirements that

Safety Evaluation No.: 99-011 (cont'd.)

Safety Evaluation Summary: (cont'd.)

govern the procedures or the activities described in the UFSAR. The change will maintain the required (Technical Specification 6.0) lines of authority, reporting requirements, procedural controls, administrative, and recordkeeping functions of the current Engineering organization.

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

Safety Evaluation No.: 99-012 Rev. 0 & 1

Implementation Document No.: N/A

UFSAR Affected Pages: I-10, I-15, IV-7, IV-12, IV-13, IV-32, V-21, XV-3, XV-5, XV-6, XV-7, XV-15, XV-68, XV-79; Tables V-1 Sh 2, XV-9 Sh 1 & 2, XV-9a

System: Various

Title of Change: Operation of NMP1 Reload 15/Cycle 14

Description of Change:

This change consisted of the addition of new fuel bundles and the establishment of a new core loading pattern for Reload 15/Cycle 14 operation of NMP1. One hundred seventy-six (176) new fuel bundles of the GE11 design were loaded. Various evaluations and analyses were performed to establish appropriate operating limits for the reload core. These cycle-specific limits were documented in the Core Operating Limits Report (COLR). Revision 1 of this safety evaluation implemented the following:

a. Clarification of changes to the analysis input which resulted in a revision to the UFSAR.

b. Revised evaluation of the margin to spring safety valve lift analysis using updated nominal scram time input assumptions.

c. Revised evaluation of pressure regulator out-of-service including changes to the COLR.

Safety Evaluation Summary:

The reload analyses and evaluations are performed based on the General Electric Standard Application for Reactor Fuel, NEDE 24011-P-A-13 and NEDE 24011-P-A-13-US (GESTAR II). This document describes the fuel licensing acceptance criteria; the fuel thermal-mechanical, nuclear, and thermal-hydraulic analyses bases; and the safety analysis methodology. For Reload 15, the evaluations included transients and accidents likely to limit operation because of minimum critical power ratio considerations; overpressurization events; loss-of-coolant accident; and stability analysis. Appropriate consideration of equipment out-of-service was included. Limits on plant operation were established to assure that

Safety Evaluation No.: 99-012 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

applicable fuel and reactor coolant system safety limits are not exceeded. Based on the evaluation performed, it is concluded that NMP1 can be safely operated during Reload 15/Cycle 14 and that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	99-013
Implementation Document No.:	ISI Program Plan, Test Procedure NMP1-PT-004 Rev. 1
UFSAR Affected Pages:	X-16
System:	Scram Discharge Volume (SDV)
Title of Change:	Post-Scram Walkdown of Scram Discharge Volume Header and Instrument Volume, or Hydrostatic Test in Accordance with ASME Code Section XI 1983 Edition to Satisfy GL 86-01 Recommendations

Description of Change:

The ISI Program Plan and test procedure have been revised to allow performance of either a post-scram reset walkdown of SDV header and instrument volume, or a hydrostatic test, in accordance with ASME Code Section XI 1983 Edition, Summer 1983 Addenda, IWA-5000 and IWC-5000, including a visual examination of the piping, once per refueling cycle.

Safety Evaluation Summary:

The post-scram reset walkdown would be conducted as soon as possible but not more than 30 minutes following the scram reset. This is only a visual inspection for leakage of the SDV header and instrument volume. The hydrostatic test, when performed in lieu of the post-scram reset walkdown, is performed during a refueling outage. Neither method of detecting SDV piping system leakage will have an effect on the principle fission product barriers (i.e., fuel cladding, reactor coolant pressure boundary, or containment), or on plant radiation shielding, or on the initiation or operation of any of the plant's engineered safety features. Accordingly, the proposed inspection and testing options could not increase the radiological consequences of an accident evaluated previously in the Unit 1 UFSAR.

As described in Generic Letter 86-01, the NRC staff has concluded that periodic leak tests, inspection and post-scram reset walkdown of SDV piping, as recommended in BWROG-8420, provide adequate capability for detection of leakage resulting from the staff postulated through-wall flaw in the SDV piping system. Performing a hydrostatic test each refueling outage in lieu of a post-scram walkdown, as a matter of preference at Unit 1, has been reviewed and accepted by the NRC staff.

Safety Evaluation No.: 99-013 (cont'd.)

Safety Evaluation Summary: (cont'd.)

Accordingly, performing either method of leakage detection will continue to satisfy the NRC staff position in Generic Letter 86-01 regarding the scram system piping integrity.

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

Safety Evaluation No.: 99-015

Implementation Document No.: Procedure S-MMP-GEN-014

UFSAR Affected Pages: N/A

System: Reactor Water Cleanup (RWCU)

Title of Change: Installation of Freeze Seal for IV-33-02R

Description of Change:

A freeze seal and mechanical plug were installed on a section of RWCU piping between the reactor vessel and valve IV-33-02R to facilitate Appendix J testing and the repair of IV-33-02R

Safety Evaluation Summary:

The freeze seal on IV-33-02R is necessary for reactor inventory isolation to maintain water level for shielding purposes to limit exposure below 10CFR20 limits. The entire core will be offloaded into the spent fuel pool (SFP) and movement of irradiated components outside of the SFP will only be allowed after review by Radiation Protection.

During this freeze seal process, normal refueling water level will be maintained in the reactor cavity and storage pit. Therefore, the spent fuel will be covered by water at all times to permit unrestricted access to the operating floor. Since the SFP is isolated from the center cavity area by the SFP gate, and the SFP water level is monitored from the Control Room, no postulated accident could occur to uncover the fuel as a result of performing this activity.

Based on the procedural limitations and administrative control imposed, as well as the reliability of the freeze seal and the mechanical plug, this method of isolating the reactor inventory in the vessel is acceptable.

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

Safety Evaluation No.:	99-018
Implementation Document No.:	GE Procedures GE-NMP1-2006, GE-NMP1-2007, GE-NMP1-2032VO
UFSAR Affected Pages:	N/A
System:	Reactor Pressure Vessel (RPV)
Title of Change:	Reactor Vessel Shell Weld Examination Using GERIS 2000

Description of Change:

The RPV shell welds were inspected using the GERIS 2000 ID inspection tool by mounting an ultrasonic test (UT) manipulator tool to the reactor vessel and core shroud with fuel in the reactor core. The GERIS tool was operated concurrent with core refueling operations. The manipulator was developed, fabricated, and seismically analyzed by MAN Energie in Nurnberg, Germany, for GE Nuclear Energy (GENE). The mechanical computer-controlled manipulator system, coupled with a GENE-designed UT system, has the capability to conduct UT inspection from the inner surface of the RPV above the core shroud support plate. The examinations of the RPV shell welds were performed to satisfy the requirements of 10CFR50.55a and ASME Section XI.

Safety Evaluation Summary:

This safety evaluation evaluated the use of the GERIS 2000 against seismic requirements for: Class I components whose failure could cause significant release of radioactivity; the potential for loose parts; and the requirements for the lifting of heavy loads over fuel. Seismic requirements were shown to be satisfied, preventive measures are in place to minimize the potential for loose parts, and lifting of heavy loads will meet NUREG-0612 requirements. Controls will be in place to ensure that the proposed inspection activity does not compromise safe refueling activities. Damage to core spray piping or other in-vessel internals was judged to be not credible. The Fire Protection Program and material compatibility criteria were also evaluated and the proposed activity was found to be acceptable with respect to these criteria.

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

Safety Evaluation No.:	99-020
Implementation Document No.:	Temporary Mod. 99-026
UFSAR Affected Pages:	N/A
System:	4160 VAC
Title of Change:	Temporary Power Supply to Powerboard #101, Cubicle 2-9 Loads

Description of Change:

During planned powerboard (PB) 101 maintenance work, PB 101 and the loads normally fed were de-energized. To reduce the plant impact due to this maintenance, 4160-V, 3-phase power was temporarily provided to loads normally fed from cubicle 2-9 in PB 101. Cubicle 2-9 on PB 101 supplies power to the Sewage Treatment Plant, the Energy Information Center, the Unit 1 Security Building and the Meteorological Computer Building. The remaining loads were evaluated for plant impacts and remained de-energized for the duration (approximately 36 hours) of the work. The source of the temporary 4160 VAC originated at an existing 13.2-kV Lake Road feeder line. The location of the Lake Road feeder is about 75 yards southwest of the Sewage Treatment Building. The temporary power distribution was provided from the Lake Road feeder through a portable (13.2 kV to 4.16 kV) ratio bank. Upon re-energization of PB 101, cubicle 2-9 was restored to normal service. Temporary power equipment was energized as needed while PB-101 was out of service.

Safety Evaluation Summary:

The analysis concludes that the proposed change has no impact on the function and performance of the normal AC distribution PB 101. The proposed temporary power installation will be completely disconnected from the AC electric distribution system (ACEDS) and powered by an alternate source from an existing 13.2-kV Lake Road feeder. After PB 101 maintenance is complete, normal power distribution will be restored through PB 101. There are no changes to ACEDS that will occur as a result of this change which would affect system description, performance, function or basis documentation. Electrical equipment used for providing the temporary power for the proposed change has been sized and installed in compliance with the National Electrical Safety Code. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	99-022
Implementation Document No.:	Procedure N1-EOP-3
UFSAR Affected Pages:	N/A
System:	Main Steam
Title of Change:	Incorporate a Fuel Zone Correction Curve in the Emergency Operating Procedures

Description of Change:

During anticipated transient without scram (ATWS) conditions, the Operators may be directed to lower reactor water level in order to reduce core flow caused by natural circulation, thereby reducing core thermal power. The reactor pressure vessel (RPV) water level must be controlled very low, but still ensure adequate core cooling. This low level ranges between the top of active fuel (-84 inches) and 0 inches. Unit 1 has a fuel zone water level instrument system which is used during an ATWS situation to determine the collapsed water level in the core after compensating for pressure. The level indicated by the system is used in N1-EOP-3. The algorithm used to convert the measured differential pressure to an indicated level assumes that all the liquid above the variable leg lower tap is at saturated conditions. A GE TRACG model provides a basis for correcting the fuel zone level indication bias based on thermal power conditions during an ATWS event. The Fuel Zone Correction Curve for ATWS Conditions has been added to N1-EOP-3.

Safety Evaluation Summary:

The Emergency Operating Procedures are used in response to accidents and transients and are not accident initiators.

Incorporating the Fuel Zone Correction Curve for ATWS Conditions into N1-EOP-3 assists the Operators in responding to an ATWS condition by having a more accurate RPV water level indication. The safety evaluation has determined that the proposed change will not increase the probability or consequences of a malfunction of equipment important to safety. It was also determined that the proposed change will not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR. Finally, it was determined that the proposed change will not reduce the margin of safety as defined in the basis for any Technical Specification. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.: 99-026

Implementation Document No.: DDC 1M00788

UFSAR Affected Pages: XVI-21; Table XVI-9a Sh 2

System: Reactor Vessel

Title of Change: Modification to the Core Shroud Repair Tie Rod Upper Spring Assemblies

Description of Change:

The core shroud tie rod assemblies were designed and installed during refueling outage (RFO) 13 in 1995 to replace shroud welds H1 through H7. Each tie rod assembly includes an upper spring which was attached to an upper spring bracket by (4) 3/8" cap screws. One of the cap screws was found to be broken during a visual inspection of the 166° tie rod following reinstallation of the upper spring assembly. The root cause was identified as intergranular stress corrosion cracking (IGSCC) in the alloy X-750 cap screw material in conjunction with large, sustained differential thermal expansion stress due to dissimilar materials at the interface. The upper spring and cap screws material is alloy X-750 and the upper spring bracket material is 300 series stainless steel. Results of the metallurgical evaluation confirmed IGSCC as the failure mechanism. Other tie rod dissimilar metal connections affected by this condition existed at a total of 24 locations [(4 tie rod upper springs) x (4 cap screws at the upper end + 2 cap screws at the lower end of each upper spring)]. A modification to the upper spring assembly was designed to eliminate this problem at all 24 locations.

The original design of the tie rod assemblies and subsequent modifications made to the tie rods during RFO14 were performed as an alternative to ASME Section XI as permitted by 10CFR50.55a(a)(3). Consequently, the original design and subsequent modifications made during RFO14 were submitted to and approved by the NRC, as documented in the NRC safety evaluations (SE). The new upper spring modification involved a change to the installed tie rod assemblies as described in NMPC's previous NRC submittals and the approved NRC SE for the original tie rod design. This safety evaluation documents the NMPC review of the upper spring modification, in accordance with the provisions of 10CFR50.59, prior to NRC submittal. The upper spring, upper wedge modification involved the addition of a clamp fabricated from 300 series stainless steel, which is secured to each existing upper spring bracket by two XM-19 bolts installed perpendicular to the plane of maximum thermal expansion. The two XM-19 bolts replaced the

Safety Evaluation No.: 99-026 (cont'd.)

Description of Change: (cont'd.)

function of the (4) existing 3/8" cap screws. The XM-19 threaded bolts are staked in place by locking pins. The clamp functions to prevent the existing 3/8" cap screws from becoming loose parts.

The stainless steel upper contacts on the upper springs were modified by a similar method. A clamp fabricated from 300 series stainless steel is secured to each upper contact and to the upper spring by one XM-19 bolt installed perpendicular to the plane of maximum thermal expansion. The XM-19 threaded bolt replaced the function of the (2) remaining 3/8" cap screws to secure the upper contact to the upper spring. The portion of the existing 3/8" cap screw not locked into the upper spring was removed prior to reinstallation of the upper spring. This assured that the part of the screw that could potentially crack was removed so that there is no longer a possibility of a loose part. The function of the new clamp is to secure the upper contact to the upper spring. The XM-19 threaded bolt is staked in place by locking pins.

Safety Evaluation Summary:

The applicable criteria and conformance for this safety evaluation were the same criteria as those used for the original tie rod repair safety evaluation, as applicable. The conformance sections specifically address the proposed modification to the upper springs. The analysis reviewed applicable criteria such as design life, material fabrication, stress analysis, thermal cycles, radiation effects, and the potential for loose parts. The evaluation concluded the upper spring modification met the requirements of the original tie rod design specification previously approved by the NRC. This modification replaces the function of the failed cap screws and ensures the upper spring maintains lateral support to the shroud. The evaluation demonstrates that the proposed modification can be implemented: 1) without an increase in the probability or consequences of an accident or malfunction previously evaluated, 2) without creating the possibility of an accident or malfunction of a new or different kind from any previously evaluated, and 3) without reducing the margin of safety in the NRC SE for the original shroud repair.

Based on the evaluation performed, it is concluded that installation of the upper spring modification does not involve an unreviewed safety question.

Safety Evaluation No.:	99-027
Implementation Document No.:	Temporary Mod. 99-032
UFSAR Affected Pages:	N/A
System:	Reactor Pressure Vessel Internals
Title of Change:	Temporary Removal of the Tie Rod Upper Lateral Springs During Cold Shutdown and Refueling Condition

Description of Change:

The core shroud tie rod assemblies were designed and installed to replace shroud welds H1 through H7. Each tie rod assembly includes a top (upper) spring. The core shroud tie rod upper springs had potentially-degraded 3/8" cap screws that attach the upper spring bracket to the upper spring. Each upper spring was temporarily removed to correct the potentially-degraded cap screw. The modification to the tie rod upper springs was evaluated in Safety Evaluation 99-026. The upper springs are designed to restrain lateral movement of the core shroud between shroud welds H1 and H2, the ring between H2 and H3, and the shell between H3 and H4. When the upper springs were removed to perform the modification for the cap screws, the tie rod assemblies no longer had the ability to restrain lateral movement. This safety evaluation evaluated the safety consequences associated with having one or more fuel bundles loaded in the core with one or more of the four upper tie rod springs removed temporarily from the tie rod assemblies until the modification to the upper springs was completed and the springs were reinstalled.

Safety Evaluation Summary:

The basic safety function of the shroud and tie rod assemblies to provide positioning and support for the fuel assemblies, control rods, in-core flux monitors, and other vessel internals will be unaffected during the period when the upper springs are removed from the tie rod assemblies and there is fuel in the core. In the cold shutdown condition all rods are typically inserted; therefore, the scram capability is not required, yet during the refueling condition the ability for control rod insertion is required. General Electric Company (GE) analysis demonstrates that the maximum horizontal displacement of the core shroud at the top guide elevation is 1.81 inches and the corresponding allowable displacement is 1.87 inches. Therefore, should control rod insertion be required, control rod movement will be available. The correct coolant distribution will be maintained for the cold

Safety Evaluation No.: 99-027 (cont'd.)

Safety Evaluation Summary: (cont'd.)

shutdown and refueling conditions. Since the compressive load to resist horizontal shroud weld separation is much larger than the lift load from recirculation system flow, horizontal weld separation in the vertical direction cannot occur for the assumed plant conditions; therefore, there will be no vertical displacement of the core spray piping. A structural evaluation of the core spray piping, taking into account the maximum horizontal displacement of 1.81 inches, determined that the maximum stress intensity on the core spray piping is within the stress allowables; therefore, the structural integrity of the core spray piping in both the vertical and horizontal directions would be maintained. GE analysis demonstrates that the maximum horizontal displacement of the core shroud at the top guide elevation is 1.81 inches, which is within the allowable displacement of 1.87 inches. Should control rod insertion be required, control rod movement will be available.

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

Safety Evaluation No.: 99-028

Implementation Document No.: Procedure N1-FHP-9R

UFSAR Affected Pages: X-43

System: Reactor Building Crane

Title of Change: Moving New Fuel Bundles into the Spent Fuel Pool from the New Fuel Storage Vault Using the Reactor Building 25-Ton Overhead Crane

Description of Change:

In the past, fresh fuel bundles have been transferred from the new fuel storage vault to the spent fuel pool (SFP) using the Reactor Building (RB) overhead crane 1000-lb. hoist. The movement of the new fuel bundles into the SFP previously occurred prior to the refueling outage. During Refuel Outage 15, the SFP storage racks were nearly full, requiring the new fuel to remain in the new fuel storage vault until core reload was in progress.

The RB overhead crane 1000-lb. hoist was not available and, as an alternative, the RB overhead crane 25-ton hoist was used to lift new fuel bundles from the new fuel storage vault to the SFP.

Safety Evaluation Summary:

The accident described in the UFSAR which may be impacted is the refueling (bundle drop) accident. The proposed change provides an equivalent alternative to the use of the RB overhead crane 1000-lb. hoist to move new fuel. The procedures to be used do not alter the sequence of off-load, do not change the intended location of irradiated fuel storage, and do not change the height to which irradiated fuel bundles are to be raised. The operation of the fuel handling equipment will remain within the assumptions of the fuel handling accident analysis. The proposed change does not create the potential for any new accident precursors or increase the probability or frequency of existing accidents or precursors. Therefore, the probability of occurrence of an accident previously evaluated in the UFSAR is not increased.

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